APPENDIX 3I REPORT ON ANALYSIS OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT

The information contained in this appendix was not revised, but has been extracted from the original FSAR and is provided for historical information.

APPENDIX 3I

REPORT ON

ANALYSES OF HIGH ENERGY LINE BREAKS

OUTSIDE CONTAINMENT

Prepared for

PUBLIC SERVICE COMPANY OF NEW HAMSPHIRE

SEABROOK STATION

Prepared by

United Engineers & Constructors
A Raytheon Company

Report No. 9763-006-S-N-2

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES
(Continued)

LIST OF FIGURES
(Continued)

SUMMARY

The environmental envelopes that the safety related Class 1E equipment will experience following postulated high energy line breaks outside containment have been determined. Systems containing high energy lines for which breaks have been evaluated include the Main Steam, Feedwater, Auxiliary Steam and Condensate, Chemical and Volume Control, Steam Generator Blowdown, and Hot Water Heating.

1.0 INTRODUCTION

It is necessary to demonstrate that equipment used to perform a required safety function for Seabrook Nuclear Station - Units 1 & 2 are capable of functioning properly in the normal, abnormal, or accident environmental conditions to which they could be exposed. As stated in NUREG-0588[1], among these environmental conditions are the elevated temperature, humidity, and/or pressure which could result from the postulated rupture of high energy lines which may be in the vicinity of this equipment. The purpose of this study is to evaluate the consequences of high energy line breaks outside containment and develop the environmental envelopes for Class 1E equipment.

2.0 METHOD OF ANALYSIS

Each of the high energy lines and all of the Class 1E equipment outside containment were identified and located. Based on this information, the various plant buildings were nodalized and the high energy line break (HELB) locations chosen in such a way as to provide an accurate representation of the environmental conditions that would result in the vicinity of the Class 1E equipment following a postulated HELB.

2.1 Mass and Energy Releases

Each high energy line was evaluated on the basis of the methods of Standard Review Plans 3.6.1 and 3.6.2[2] to determine the types, areas, and locations of postulated ruptures that would result in the most severe environmental conditions at each of the Class 1E equipment. The break releases were calculated using the Moody critical flow model[3] and accounting for physical restrictions within the system (e.g. flow and pressure control valves) and the frictional effects of the piping system.

These release rates were taken to be constant, i.e. no decay of the reservoir pressure was assumed, until isolation of the ruptured line was initiated or, as in the case of the closed Hot Water Heating Systems, until the piping inventory was depleted.

The methods and assumptions employed in calculating the mass and energy release rates for each high energy line are outlined in Table 2.1-1. As noted in this table, isolation of many of these lines will be accomplished by the use of redundant temperature detectors in various plant areas that, in the event of elevated temperatures, will send closure signals to redundant isolation valves present in the

high energy lines. The locations of these temperature detectors are provided in Figures 2.1-1, 2.1-2, and 2.1-3.

The mass and energy release rates used in evaluating the pressure, temperature, and humidity responses throughout the various plant areas are calculated and defined in References 6, 7, and 8.

2.2 Pressure/Temperature/Humidity Transients

The environmental conditions that result due to postulated high energy line ruptures were determined for the following areas:

1. Primary Auxiliary Building (PAB)
2. Containment Enclosure Area (CEA)
3. Fuel Storage Building (FSB)
4. Main Steam/Feedwater Pipe Chase
5. Tank Farm Area (TFA)
6. Waste Processing Building/Primary Auxiliary Building (WPB/PAB) Chase

For HELB other than Hot Water heating Line Breaks (HWHLB), the environmental Responses of the PAB, CEA, TFA, WPB/PAB Chase, and MS/FW Pipe Chase were calculated using the COMPRESS(4) computer program. Using the break mass and energy releases and the building nodalizations discussed previously, COMPRESS calculates the transient pressures, temperatures, and humidities that would occur throughout the plant building following these ruptures. The methods and assumptions used in these pressure/ temperature calculations agree with those of NUREG-0588(1).

Table 2.2-1 lists the ambient conditions, building initial conditions, and other pertinent design basis information used in analyzing these environmental transients. The ambient and initial conditions were

chosen so as to maximize the temperature response that would result from these postulated HELB. In addition, the Uchida condensing steam heat transfer correlation is used during the condensing mode while a convective heat transfer coefficient of 2.0 Btu/hr-ft^2-°F is used otherwise.

The environmental response of the PAB, CEA, and FSB to postulated HWHLB was calculated using a reasonable, yet still conservative, hand calculation method which accounted for mass and heat transfer between the hot water and the room air. Since the HWH subsystems are closed systems which will not be isolated and these plant areas are supplied with ventilation air by non-Class 1E systems, the maximum temperatures and humidities that result from HWHLB are calculated by releasing the total HWH subsystem fluid mass into the initial room air mass.

The building initial conditions were determined based on the historical distribution of ambient conditions which occur during the time of the year when the HWH system is in operation (September through May). These conditions are defined in Table 2.2-1.

3.0 HELB ANALYSES AND RESULTS

The environmental response of the plant buildings to postulated high energy line ruptures were calculated using the methods outlined in Section 2.0. The results of these HELB analyses (other than HWHLB) are presented in the following sections.

3.1 Primary Auxiliary Building

From an evaluation of each of the high energy lines in the PAB and their operating conditions, it was concluded that the break locations listed in Table 3.1-1 would provide environmental envelopes for the Class 1E equipment.

Figure 3.1-1 shows the layout of the PAB and the zone designations which were useful in defining the environmental parameters throughout the PAB. Zone 32A, which is not shown, represents the PAB below the the (-)6' elevation and includes the piping tunnels, Zone 32B represents the 2' and (-)6' elevations, and Zones 32 and 33C, 32 and 33D, and 32 and 33E represent the 7', 25', and 53' elevations, respectively. Zones 47 and 48 represent the Chemical and Volume Control System (CVCS) equipment vaults and contain no Class 1E equipment.

Table 3.1-2 summarizes the peak and enveloping temperatures and pressures that would occur in each of these zones for each postulated high energy line rupture. All areas can be taken to experience 100% relative humidity, condensing environments, however, air displacement and thus essentially pure steam environments would be expected to occur only in the general vicinity of the postulated breaks.

For each of the ruptures considered in these tables there follows a series of four figures, lettered A though D. The A series of these figures (e.g. Figure 3.1-2A, 3.1-3A) physically defines the nodal arrangement which was chosen to analyze the rupture's effect on the PAB environment. The B series provides the flow diagrams and physical parameters (volumes, heat sink areas, flow areas) for this nodal arrangement. Figures C and D provide the calculated temperature and pressure transients for each of the nodes defined in the A and B series figures.

3.2 Containment Enclosure Area

The Containment Enclosure Area contains several high energy (CVCS) lines, however, only the letdown line operates at an elevated temperature. Therefore, only a rupture of this line has been considered as stated in Table 3.2-1.

The layout of the Containment Encosure Area, which includes the Mechanical Penetration Area, the Charging Pump Cubicles, and the Residual Heat Removal (RHR), Safety Injection (SI), and Containment Spray (CBS) Vaults, is shown in Figure 3.2-1A, Sheets 1 and 2. These figures also show the nodal arrangement used, while Figure 3.2-1B provides the corresponding flow diagram and physical parameters. Table 3.2-2 summarizes the pressures and temperatures experienced in the various areas of the enclosure volume following a postulated CVCS letdown line break. Figures 3.2-1C and 3.2-1D show the transient temperatures and pressures in the CEA. By a variation of the assumed initial conditions (10% vs. 95% relative humidity), an additional investigation was made which determined the maximum pressure response of the CEA. This result is shown in Figure 3.2-1E. For the HELB

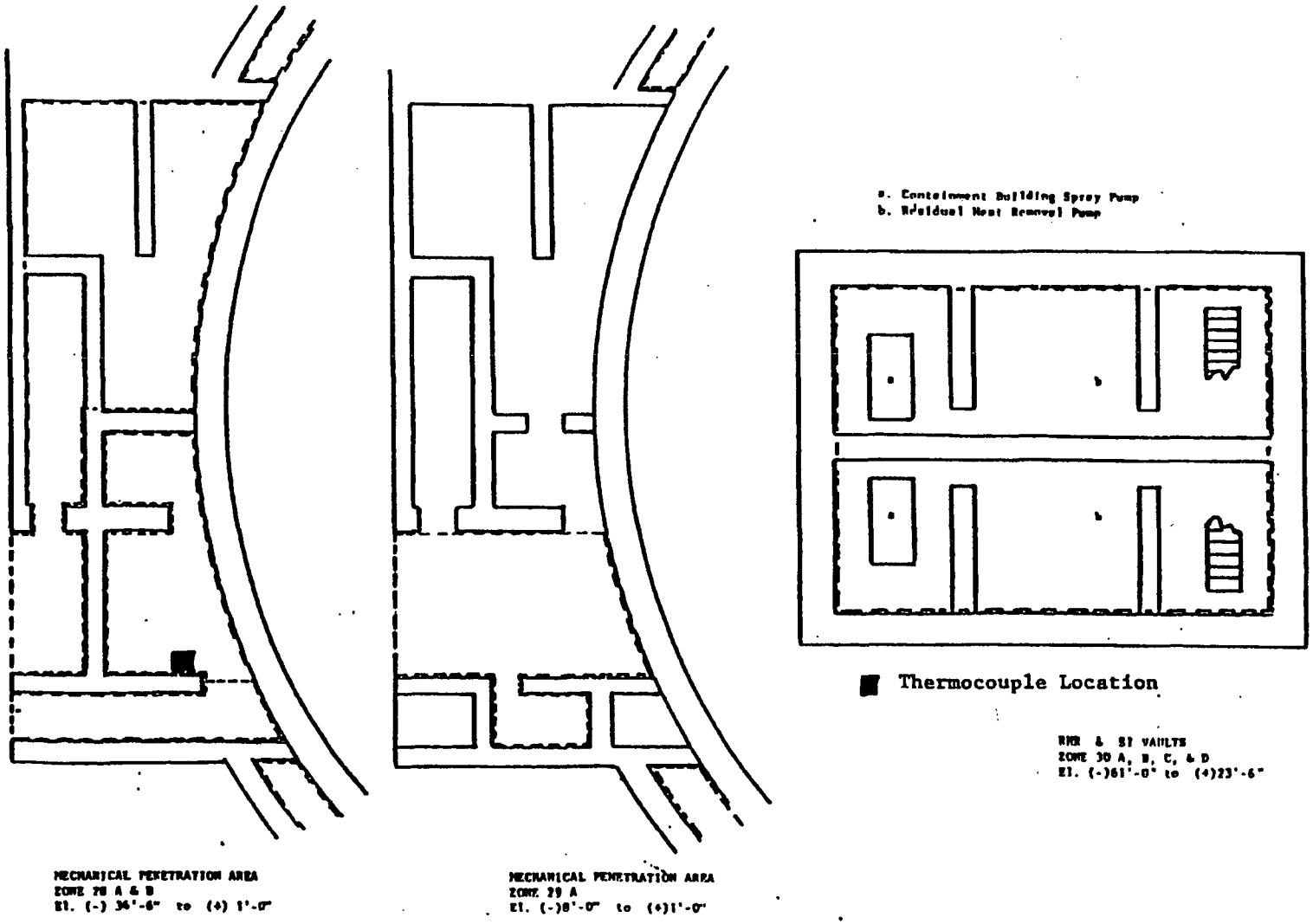
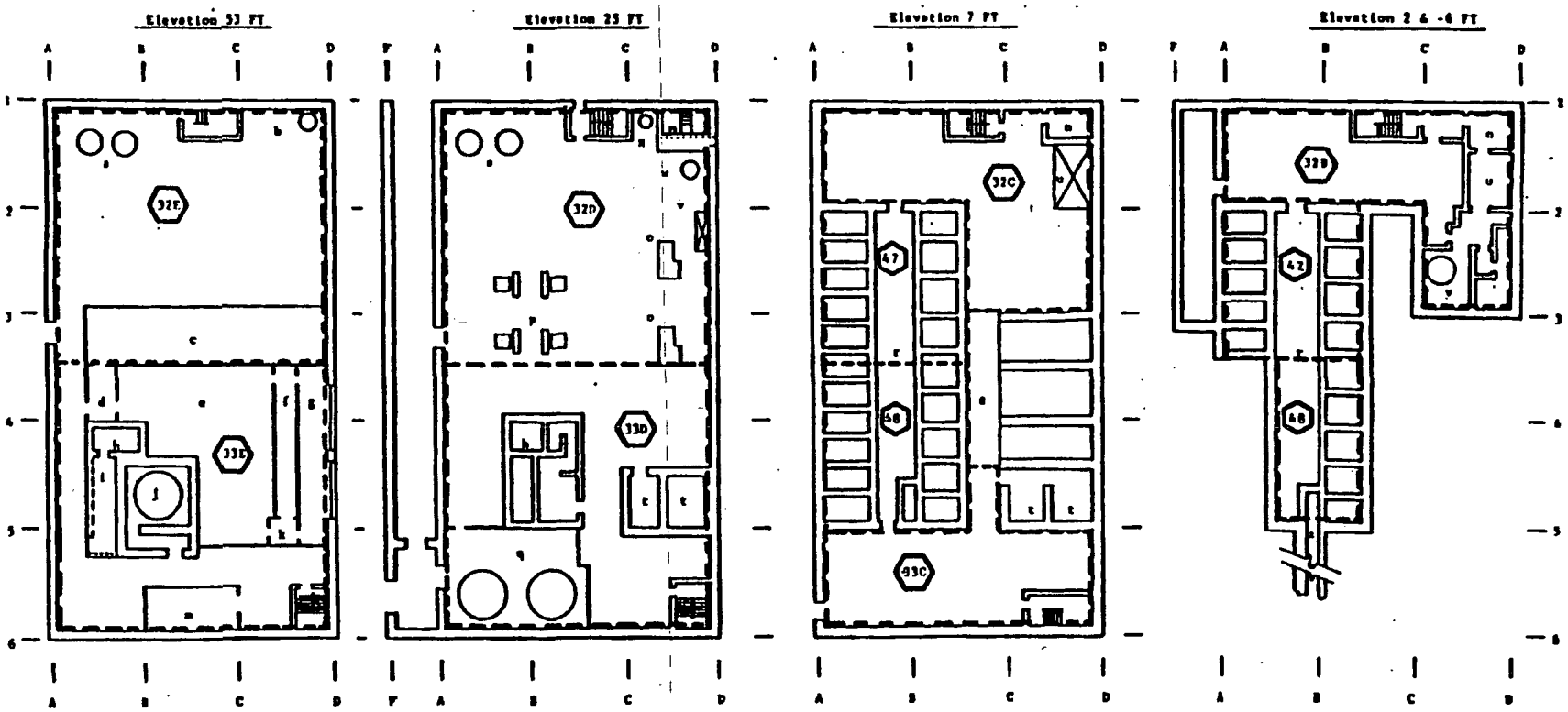
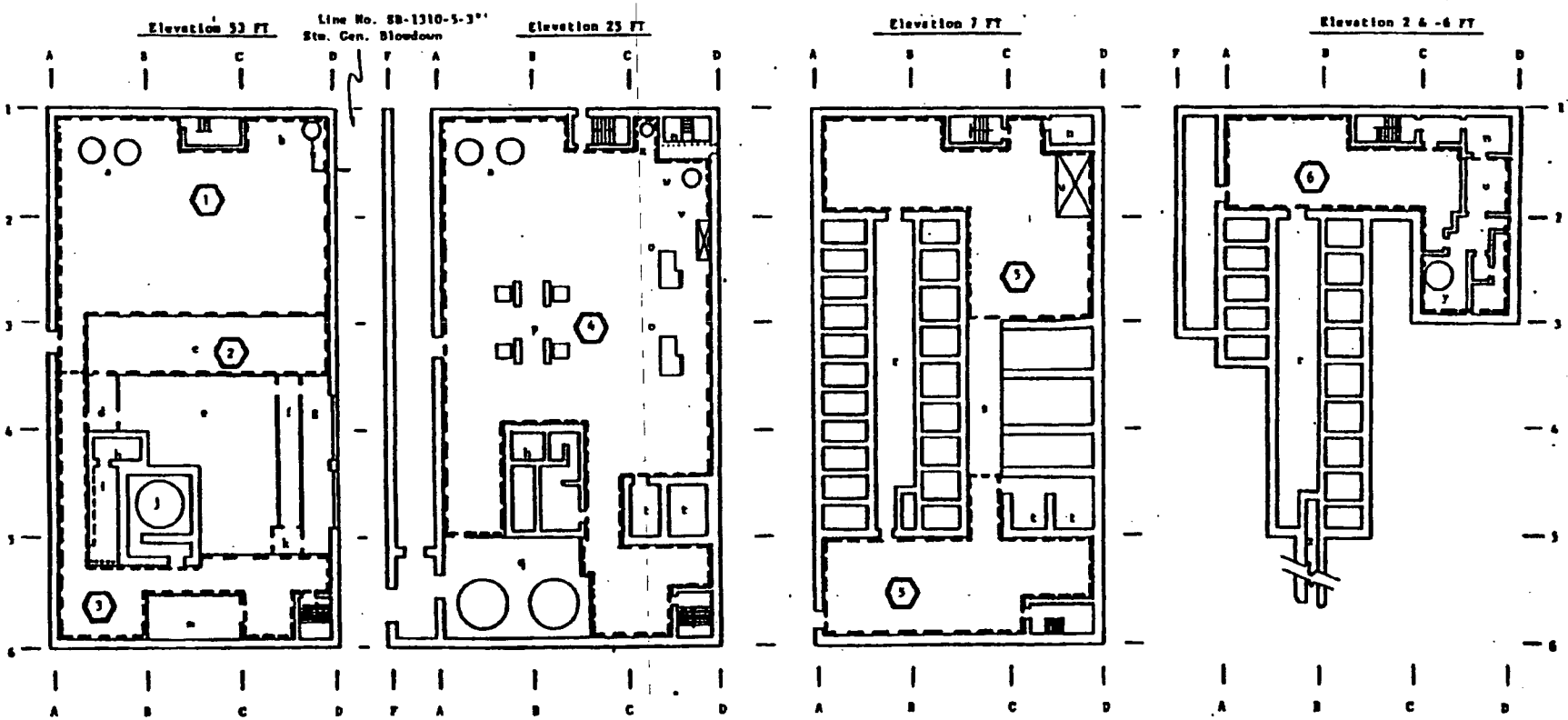


Figure 2.1-3: Containment Enclosure Area Showing Locations of HELB Temperature Detection Thermocouples



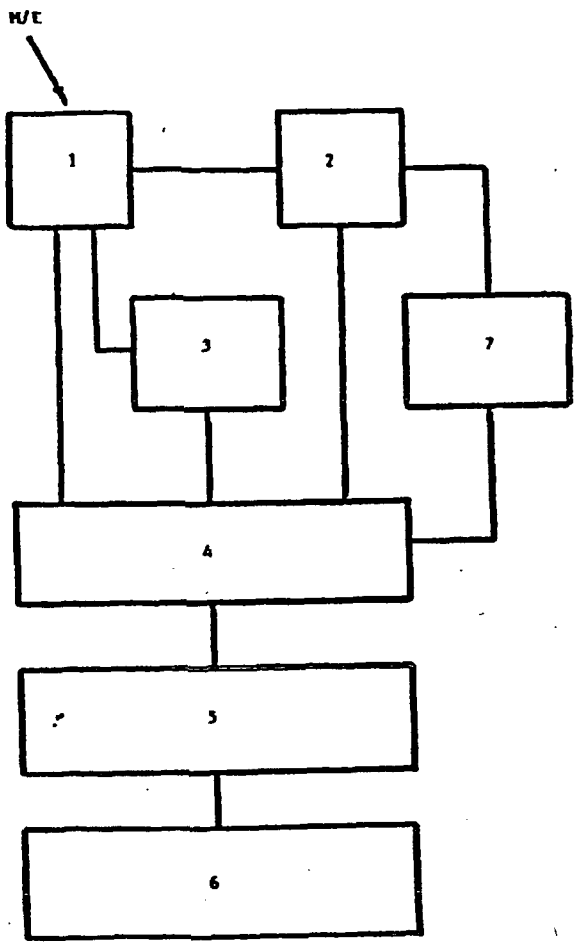
a. PCCW Heat Exchangers
b. Blowdown Flash Tank
c. PAB Exhaust Fan Room
d. Air Lock
e. PAB Inlet & Containment Purge Fans
f. Inlet Heaters
g. Inlet Filters
h. RAACT Pipe Chase
i. Valve Aisle
j. CVCS Tank
k. Air Lock
m. Storage Area
n. Electrical Chase
o. 480 v. Load Centers
p. PCCW Pumps
q. Boric Acid Tanks
r. CVCS Vaults
s. Charging Pump Vaults
t. Letdown Degassifier
u. Non-RAACT Pipe Chase
v. HVAC Opening
w. Boron Injection Tank
x. Chiller Surge Tank
y. Aux. Steam Condensate Tank
z. Pipe Trench

Figure 3.1-1: Zone Designations of Primary Auxiliary Building at Various Elevations



a. PCCW Heat Exchangers
b. Blowdown Flash Tank
c. PAB Exhaust Fan Room
d. Air Lock
e. PAB Inlet & Containment Purge Fans
f. Inlet Heaters
g. Inlet Filters
h. RAACT Pipe Chase
i. Valve Aisle
j. CVCS Tank
k. Air Lock
l. Storage Area
m. Electrical Chase
o. 480 v. Load Centers
p. PCCW Pumps
q. Boric Acid Tanks
r. CVCS Vaults
s. Charging Pump Vaults
t. Letdown Degassifier
u. Non-RAACT Pipe Chase
v. HVAC Opening
w. Boron Injection Tank
x. Chiller Surge Tank
y. Aux. Steam Condensate Tank
z. Pipe Trench

Figure 3.1-2A: Nodal Arrangement of Primary Auxiliary Building at Various Elevations for Steam Generator Blowdown Line Break Analysis



NODE	VOLUME (ft^3)	HEAT SINK AREA(ft^2)
1	95,490	18,000
2	23,520	560
3	53,930	8,000
4	243,400	42,670
5	108,070	18,500
6	38,235	15,900
7	ATMOSPHERE	

FLOW PATHS' CHARACTERISTICS

FROM NODE	TO NODE	AREA(ft^2)	INERTIA(ft^{-1})	K_c	LOSS FACTOR K_{exp}	LOSS FACTOR K_{fric}	K_{total}
1	2	15.0	.05	.78	1.0	.01	1.79
1	3	128.7	.45	.42	.85	.17	1.44
1	4	9.40	.88	.78	1.0	.20	1.98
2	4	31.5	.09	.78	1.0	.01	1.79
2	7	20.0	5.00	.78	1.0	3.50	5.28
3	4	10.6	5.40	.78	1.0	2.22	4.00
4	5	44.8	.80	.78	1.0	.30	2.08
4	7	20.0	.50	.78	1.0	1.60	3.38
5	6	5.9	8.50	.78	1.0	3.2	4.98

Figure 3.1-2B: Nodal Parameters of Primary Auxiliary Building for Steam Generator Blowdown Line Break Analysis

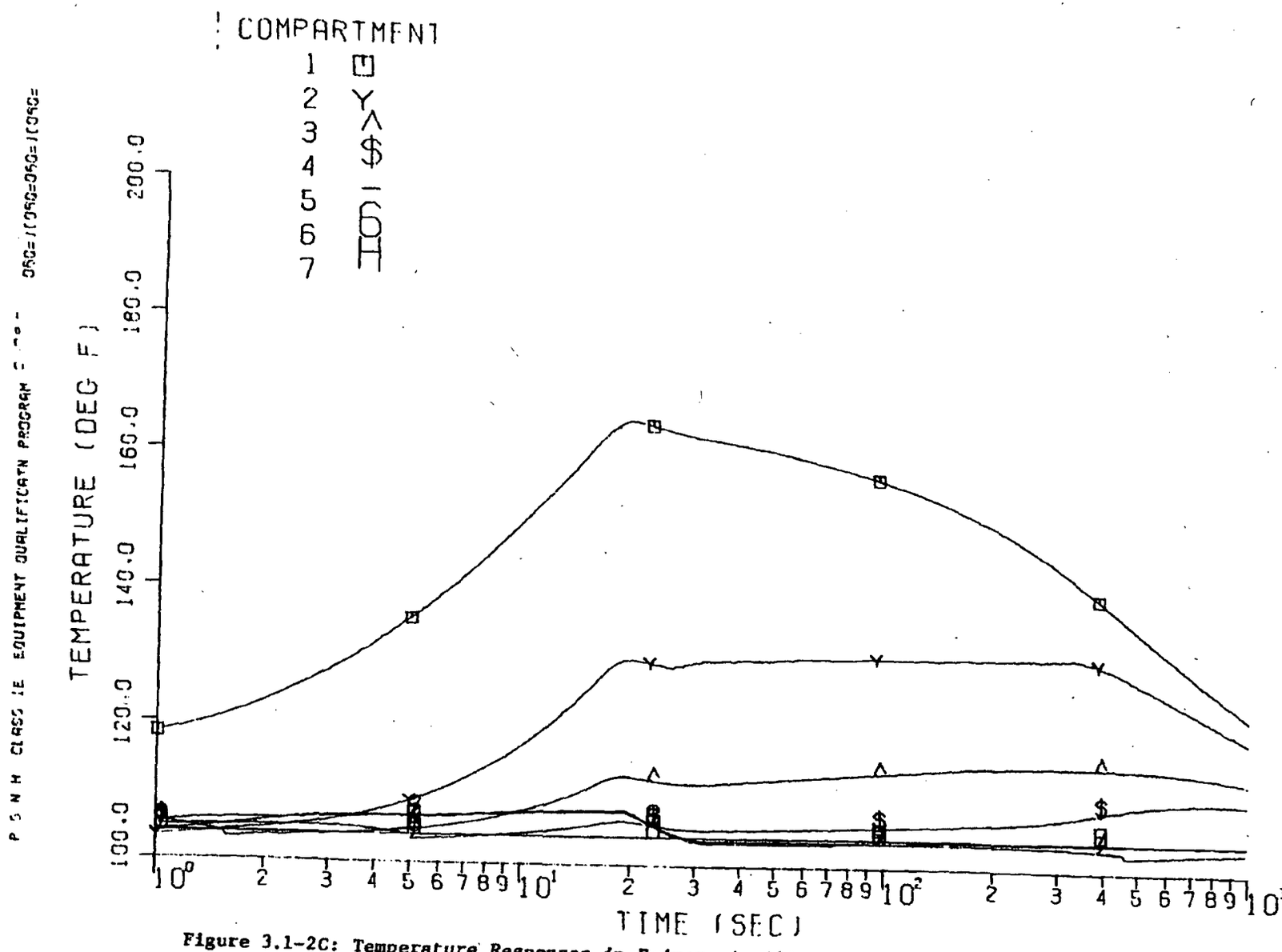


Figure 3.1-2C: Temperature Responses in Primary Auxiliary Building Following A Rupture of 3" Steam Generator Blowdown Line

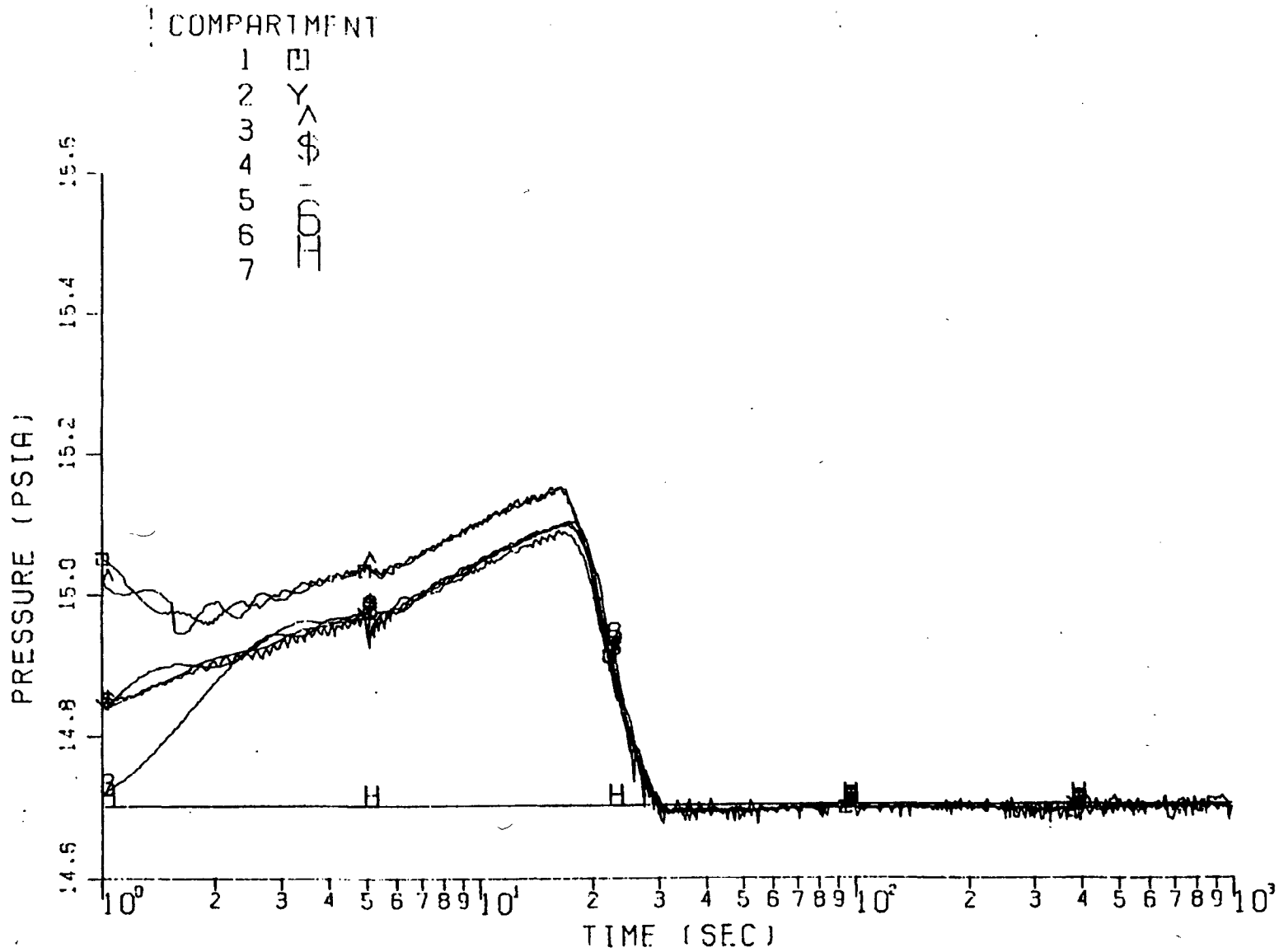
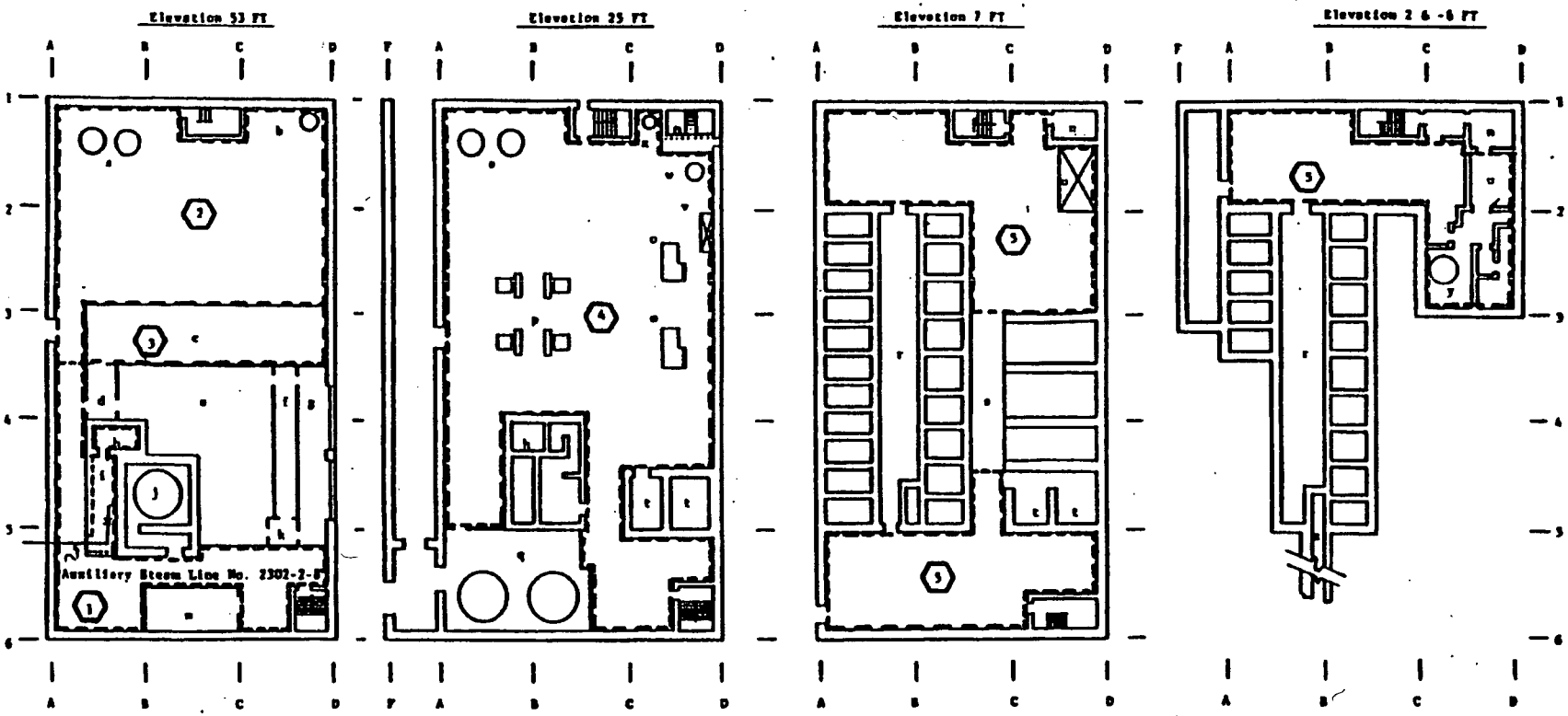
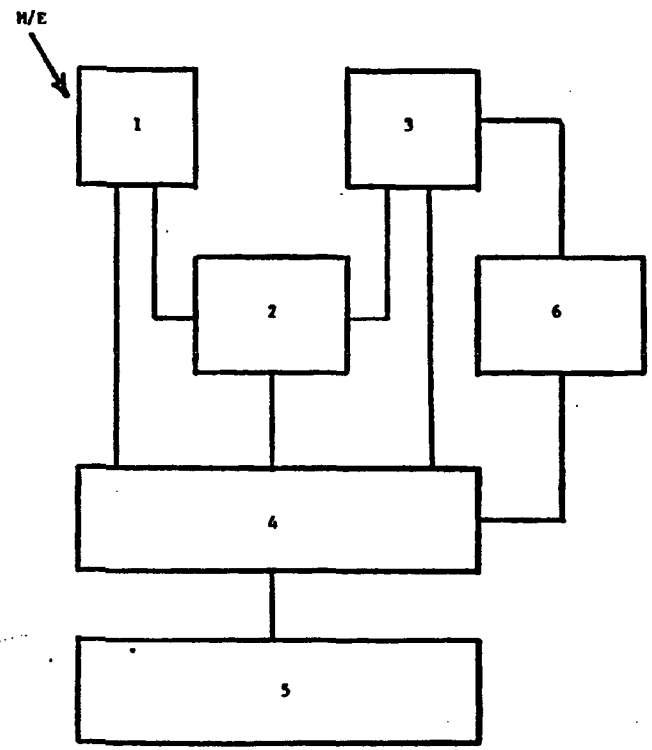


Figure 3.1-2D: Pressure Responses in Primary Auxiliary Building Following a Rupture of 3" Steam Generator Blowdown Line



a. PCCW Heat Exchangers
b. Blowdown Flash Tank
c. PAB Exhaust Fan Room
d. Air Lock
e. PAB Inlet & Containment Purge Fans
f. Inlet Heaters
g. Inlet Filters
h. RAACT Pipe Chase
i. Valve Aisle
j. CVCS Tank
k. Air Lock
l. Storage Area
m. Electrical Chase
o. 480 v. Load Centers
p. PCCW Pumps
q. Boric Acid Tanks
r. CVCS Vaults
s. Charging Pump Vaults
t. Letdown Degasifier
u. Non-RAACT Pipe Chase
v. HVAC Opening
w. Boron Injection Tank
x. Chiller Surge Tank
y. Aux. Steam Condensate Tank
z. Pipe Trench

Figure 3.1-3A: Nodal Arrangement of Primary Auxiliary Building at Various Elevations for Auxiliary Steam Line AS-2302-2-8" Break Analysis



NODE	VOLUME (ft^3)	HEAT SINK AREA(ft^2)
1	53,930	8,000
2	95,490	18,000
3	23,520	560
4	243,400	42,670
5	146,300	34,400
6	ATMOSPHERE	

FLOW PATHS CHARACTERISTICS

FROM NODE	TO NODE	AREA(ft^2)	INERTIA (ft^{-1})	LOSS FACTOR K_c	K_{exp}	K_{fric}	K_{total}
1	2	128.7	.45	.42	.85	.17	1.44
1	4	10.6	5.40	.78	1.0	2.22	4.00
2	3	15.0	.05	.78	1.0	.01	1.79
2	4	9.4	.88	.78	1.0	.20	1.98
3	4	31.5	.09	.78	1.0	.01	1.79
3	6	20.0	5.00	.78	1.0	3.50	5.28
4	5	44.8	.80	.78	1.0	.30	2.08
4	6	20.0	.50	.78	1.0	1.60	3.38

Figure 3.1-3B: Nodal Parameters of Primary Auxiliary Building for Auxiliary Steam Line AS-2302-2-8" Break Analysis

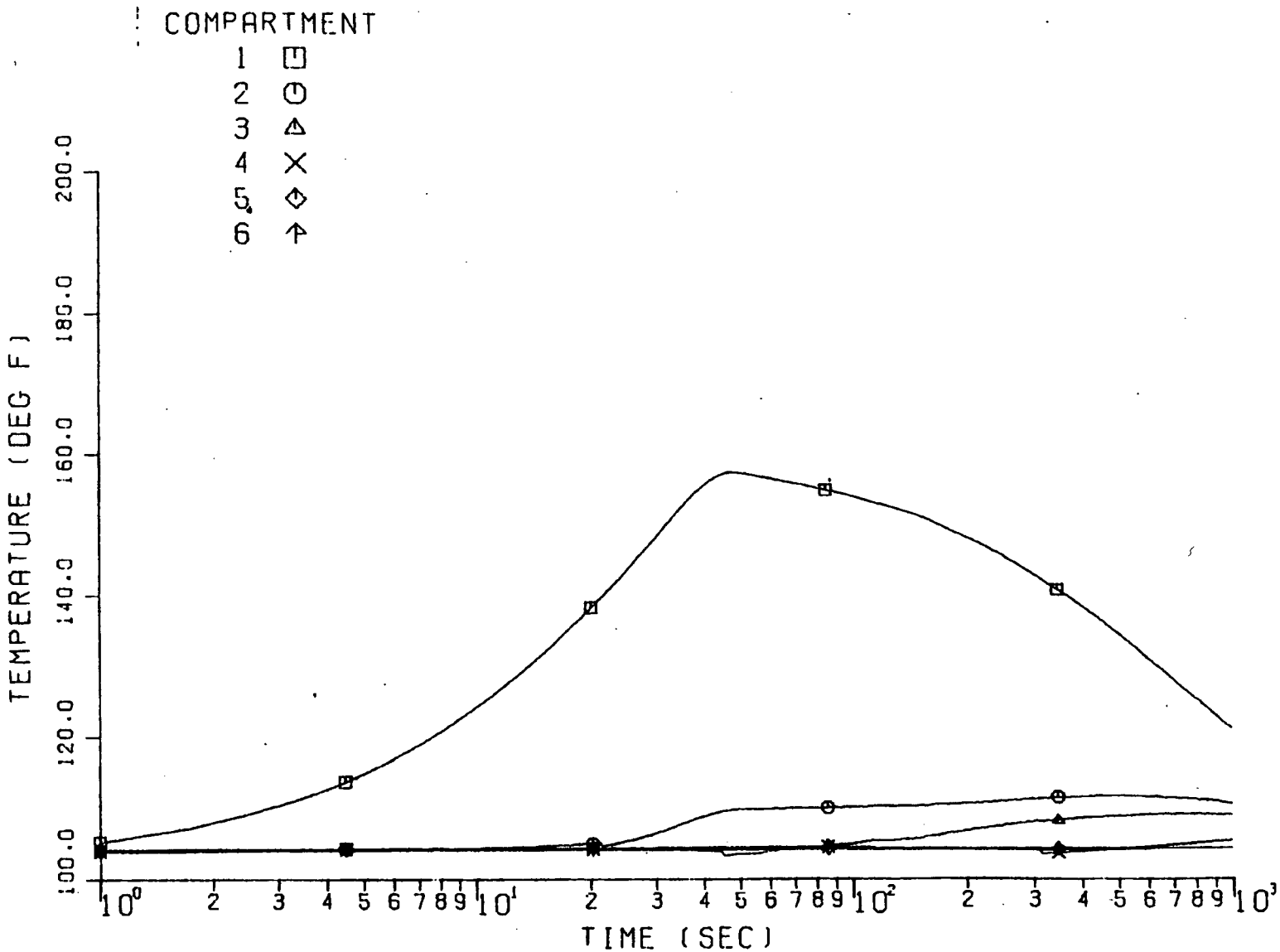


Figure 3.1-3C: Temperature Responses in Primary Auxiliary Building Following a Rupture of 8" Auxiliary Steam Line

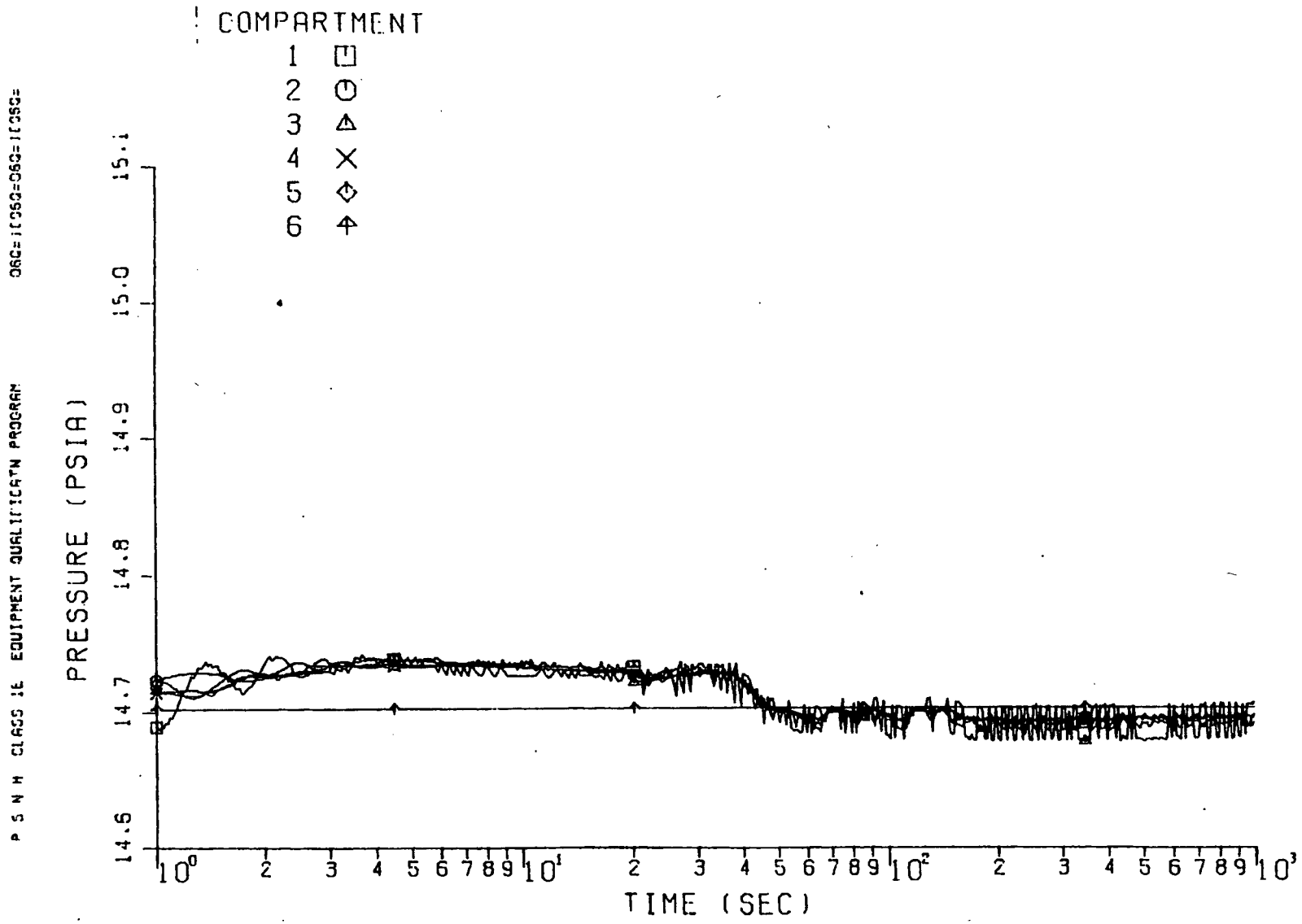
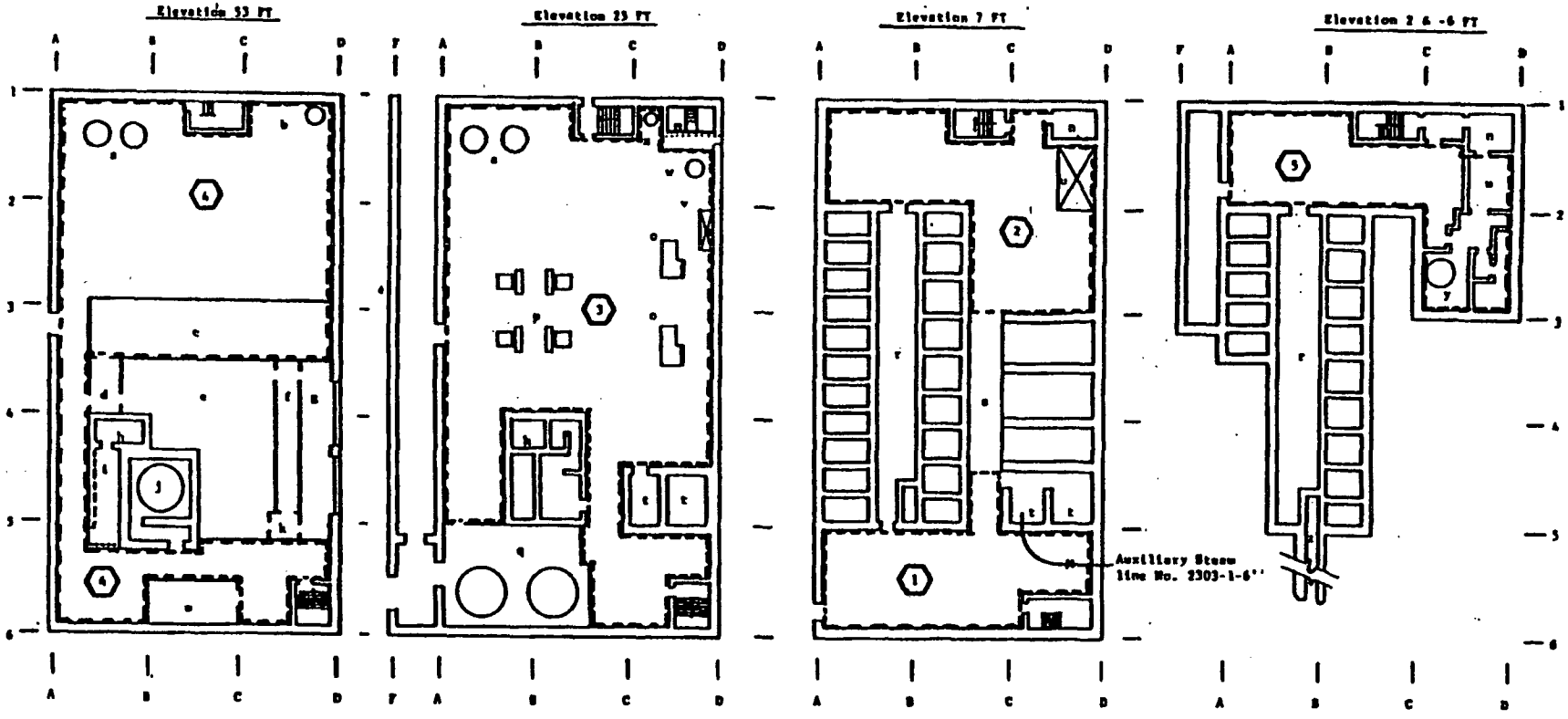
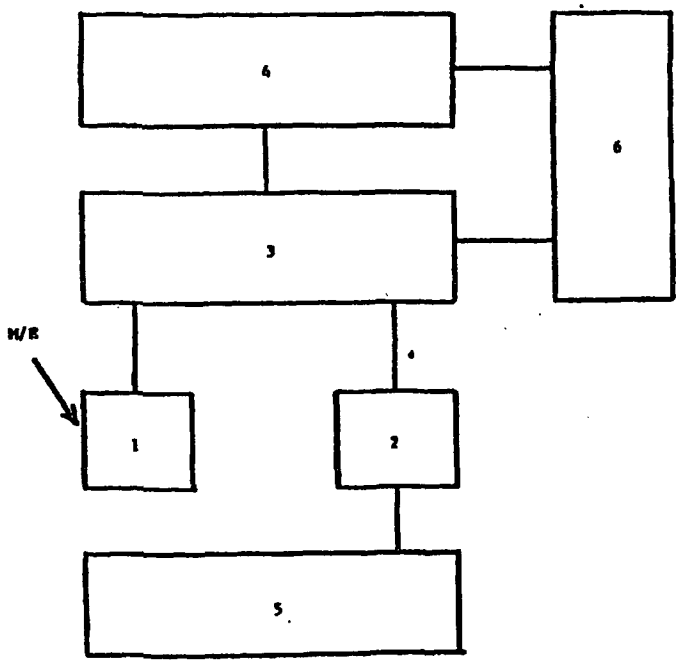


Figure 3.1-3D: Pressure Responses in Primary Auxiliary Building Following a Rupture of 8" Auxiliary Steam Line



a. PCCW Heat Exchangers
b. Blowdown Flash Tank
c. PAB Exhaust Fan Room
d. Air Lock
e. PAB Inlet & Containment Purge Fans
f. Inlet Heaters
g. Inlet Filters
h. RAACT Pipe Chase
i. Valve Aisle
j. CVCS Tank
k. Air Lock
m. Storage Area
n. Electrical Chase
o. 480 v. Load Centers
p. PCCW Pumps
q. Boric Acid Tanks
r. CVCS Vaults
s. Charging Pump Vaults
t. Letdown Degasifier
u. Non-RAACT Pipe Chase
v. HVAC Opening
w. Boron Injection Tank
x. Chiller Surge Tank
y. Aux. Steam Condensate Tank
z. Pipe Trench

Figure 3.1-4A: Nodal Arrangement of Primary Auxiliary Building at Various Elevations for Auxiliary Steam Line AS-2303-1-6" Break Analysis



NODE	VOLUME (ft^3)	HEAT SINK AREA(ft^2)
1	38,200	8,500
2	49,700	10,000
3	243,400	42,670
4	172,940	26,560
5	38,235	15,900
6	ATMOSPHERE	

FLOW PATHS CHARACTERISTICS

FROM NODE	TO NODE	AREA(ft^2)	INERTIA (ft^{-1})	LOSS FACTOR K_c	K_{exp}	K_{fric}	K_{total}
1	3	60.0	.04	.78	1.0	.02	1.80
2	3	44.8	.80	.78	1.0	.30	2.08
2	5	5.9	8.50	.78	1.0	3.20	4.98
3	4	51.5	1.33	.78	1.0	.33	2.11
3	6	20.0	.50	.78	1.0	1.60	3.38
4	6	20.0	5.00	.78	1.0	3.50	5.28

Figure 3.1-4B: Nodal Parameters of Primary Auxiliary Building for Auxiliary Steam Line AS-2303-1-6" Break Analysis

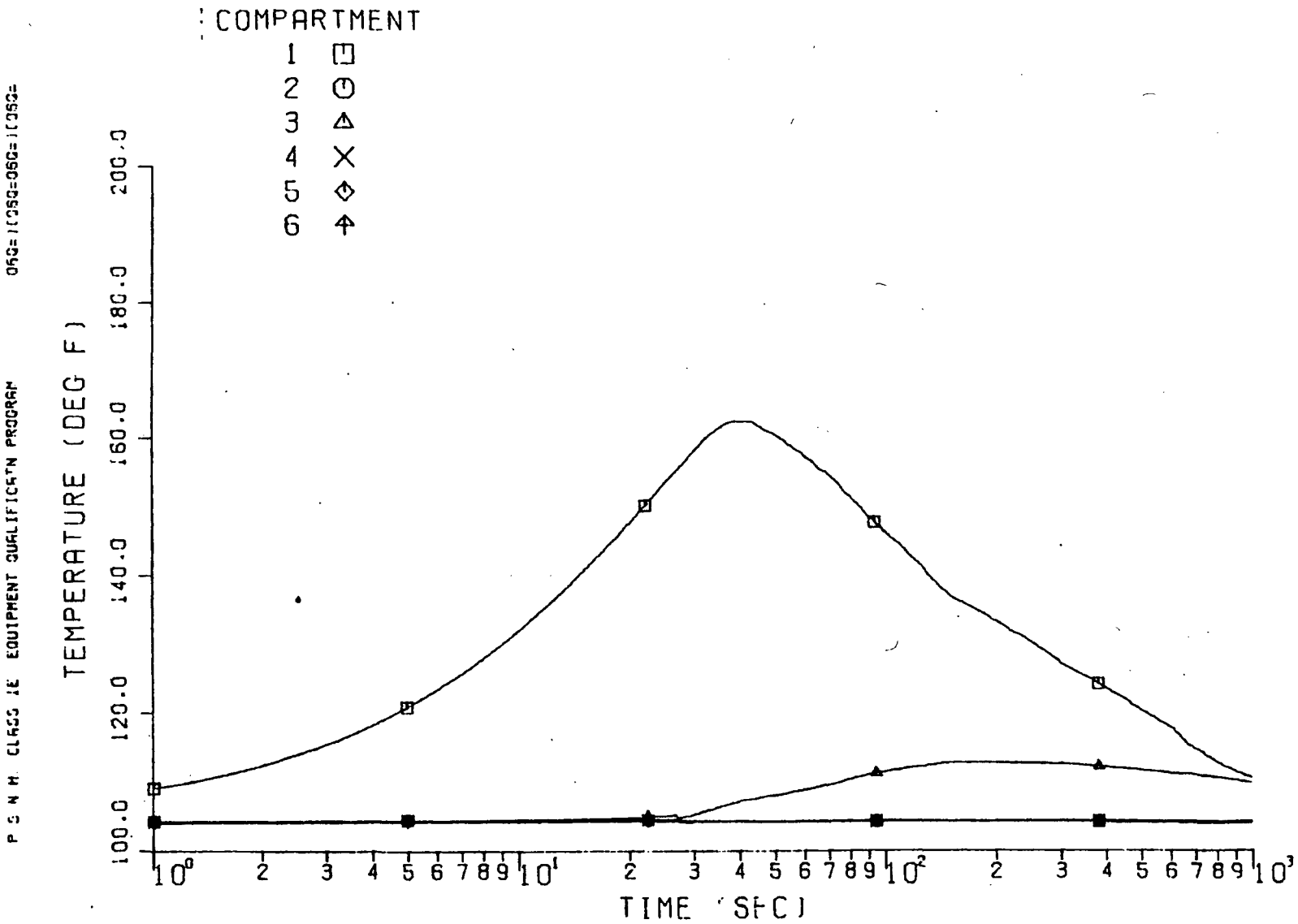


Figure 3.1-4C: Temperature Responses in Primary Auxiliary Building Following a Rupture of 6" Auxiliary Steam Line

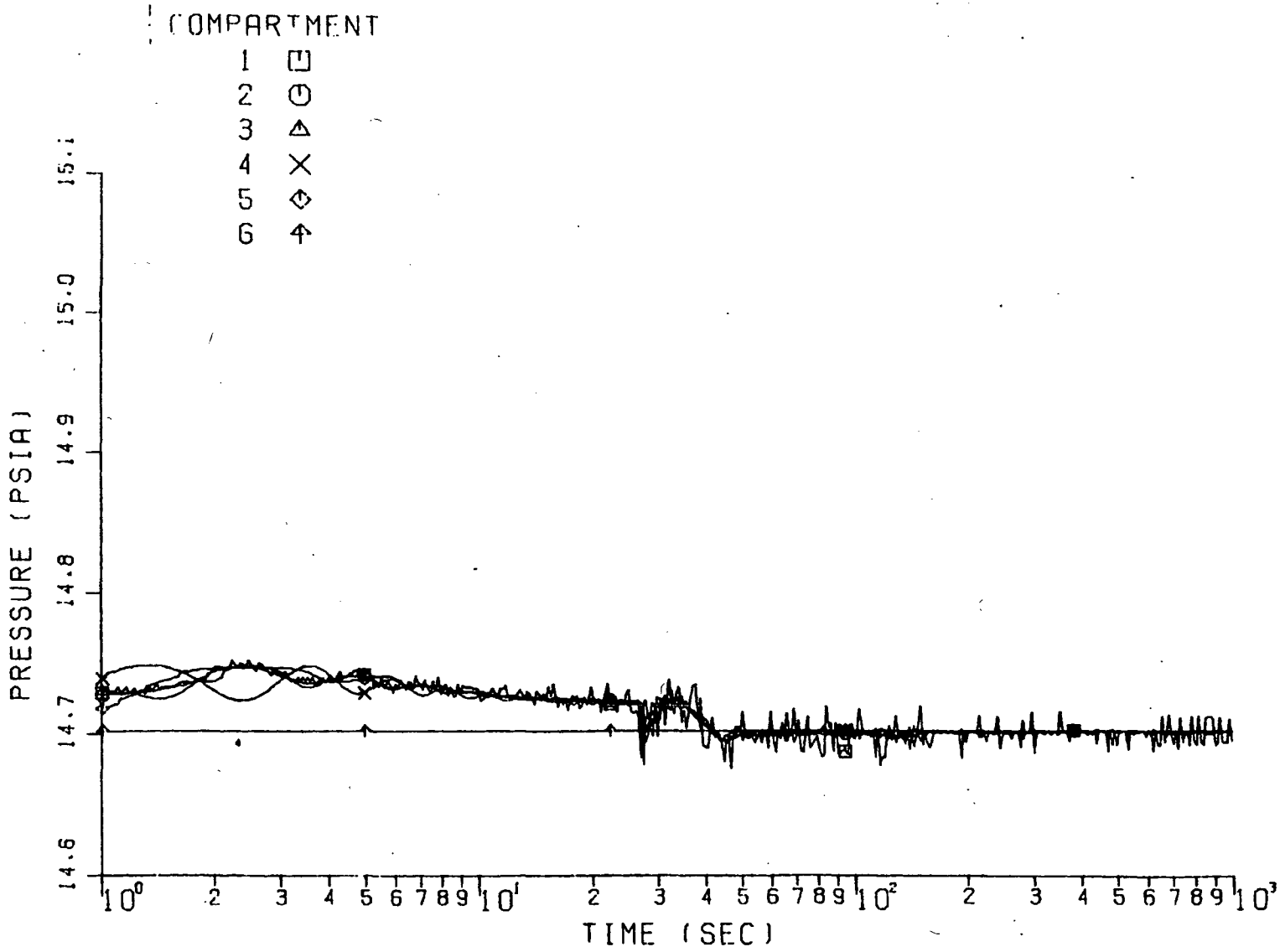
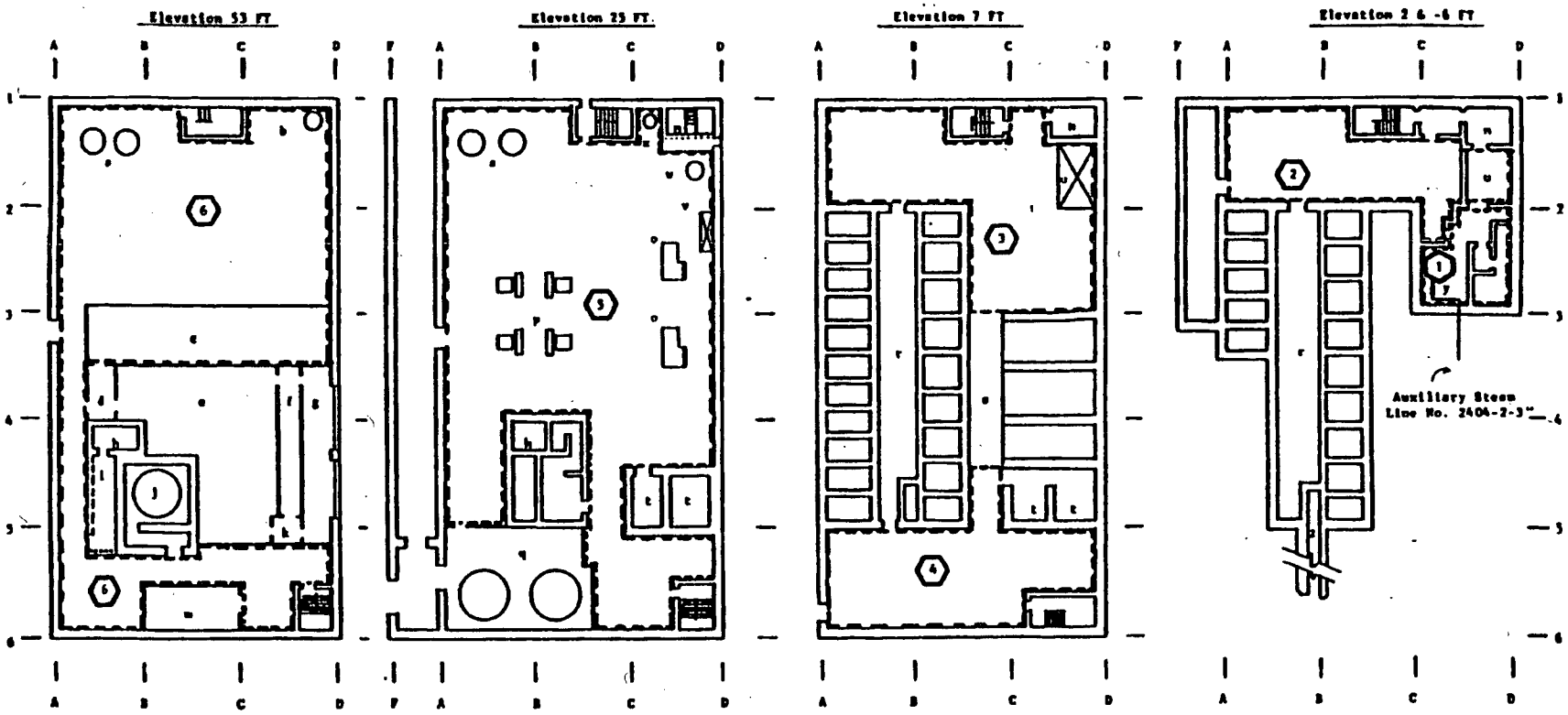
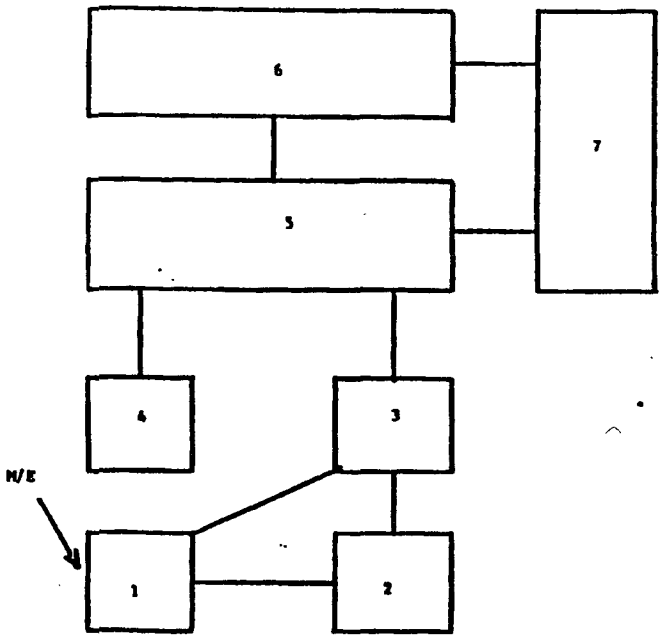


Figure 3.1-4D: Pressure Responses in Primary Auxiliary Building Following a Rupture of 6" Auxiliary Steam Line



a. PCCW Heat Exchangers
b. Blowdown Flash Tank
c. PAB Exhaust Fan Room
d. Air Lock
e. PAB Inlet & Containment Purge Fans
f. Inlet Heaters
g. Inlet Filters
h. RAACT Pipe Chase
i. Valve Aisle
j. CVCS Tank
k. Air Lock
m. Storage Area
n. Electrical Chase
o. 480 v. Load Centers
p. PCCW Pumps
q. Boric Acid Tanks
r. CVCS Vaults
s. Charging Pump Vaults
t. Letdown Degassifier
u. Non-RAACT Pipe Chase
v. HVAC Opening
w. Boron Injection Tank
x. Chiller Surge Tank
y. Aux. Steam Condensate Tank
z. Pipe Trench

Figure 3.1-5A: Nodal Arrangement of Primary Auxiliary Building at Various Elevations for Auxiliary Steam Condensate Line ASC-2404-2-3" Break Analysis



NODE	VOLUME (ft^3)	HEAT SINK AREA(ft^2)
1	8,645	4,180
2	29,700	11,700
3	49,700	10,000
4	38,200	8,500
5	243,400	42,670
6	172,940	26,560
7	ATMOSPHERE	

FLOW PATHS CHARACTERISTICS

FROM NODE	TO NODE	AREA(ft^2)	INERTIA (ft^{-1})	LOSS FACTOR K_c	K_{exp}	K_{fric}	K_{total}
1	2	2.1	15.0	.78	1.0	1.5	3.28
1	3	3.0	10.3	.78	1.0	1.4	3.18
2	3	5.9	8.5	.78	1.0	3.20	4.98
3	5	44.8	.80	.78	1.0	.30	2.08
4	5	6.3	.32	.78	1.0	.10	1.88
5	6	51.5	1.33	.78	1.0	.33	2.11
5	7	20.0	.50	.78	1.0	1.60	3.38
6	7	20.0	5.00	.78	1.0	3.50	5.28

Figure 3.1-5B: Nodal Parameters of Primary Auxiliary Building for Auxiliary Steam Condensate Line ASC-2404-2-3" Break Analysis

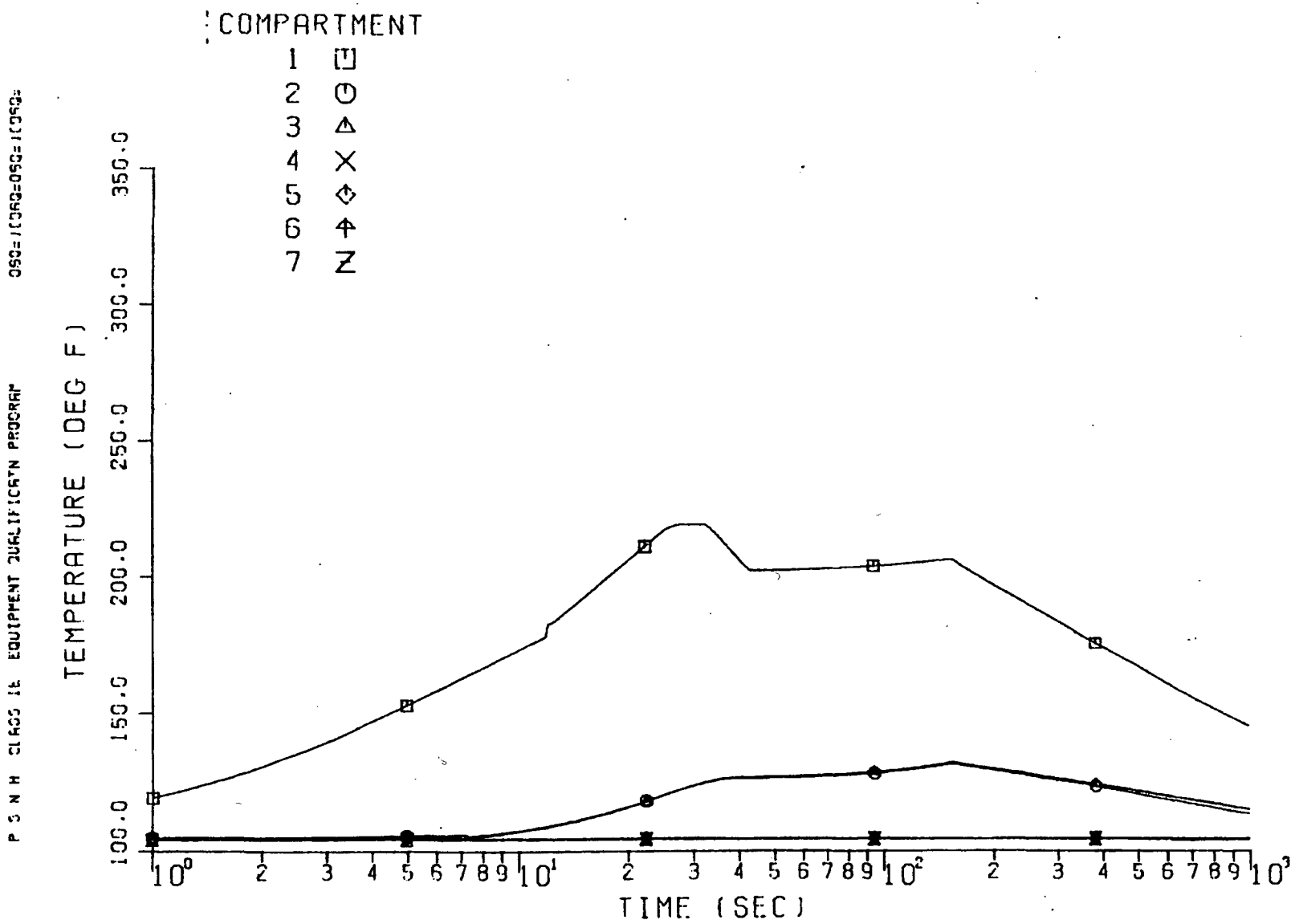


Figure 3.1-5C: Temperature Responses in Primary Auxiliary Building Following a Rupture of 3" Auxiliary Steam Condensate Line

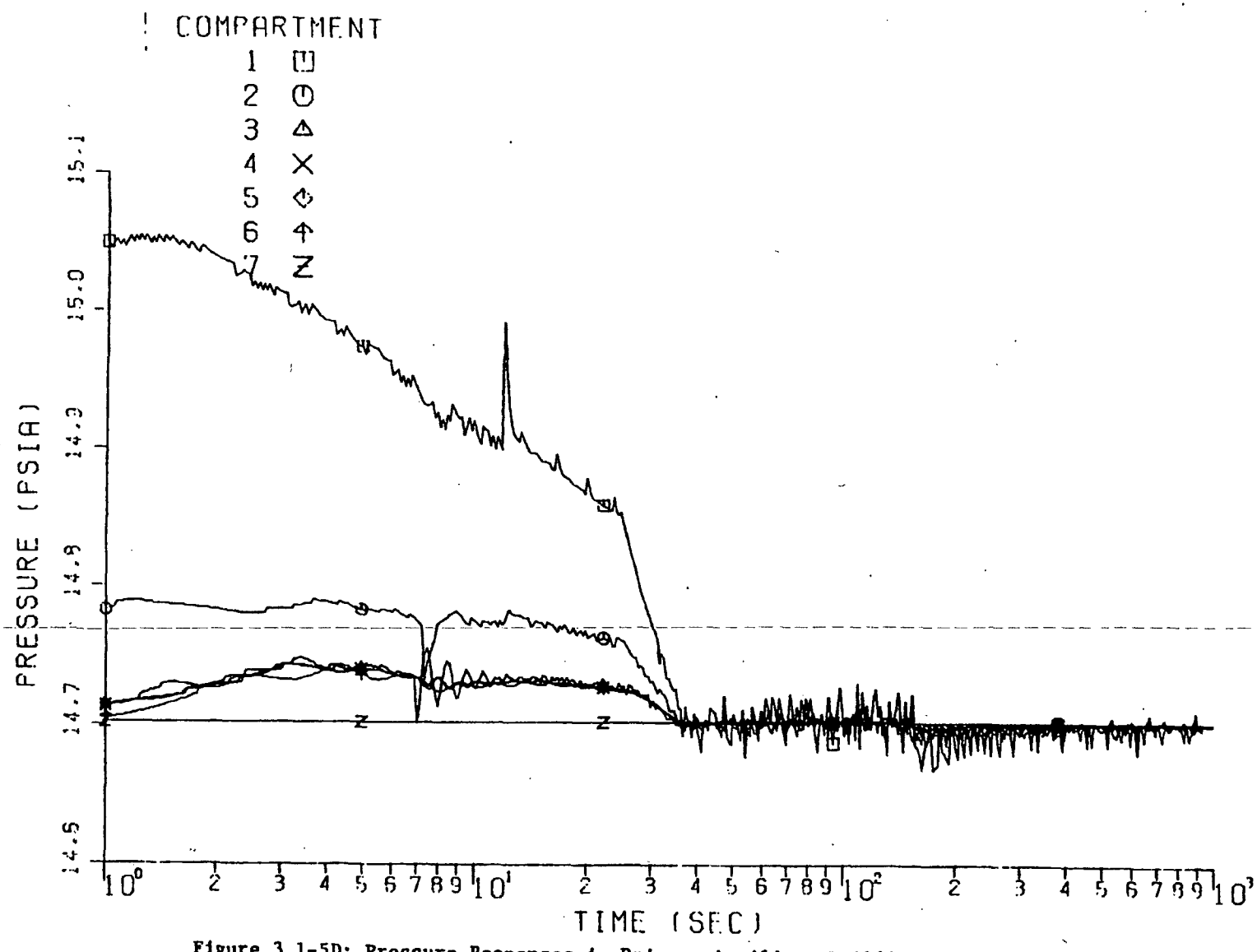
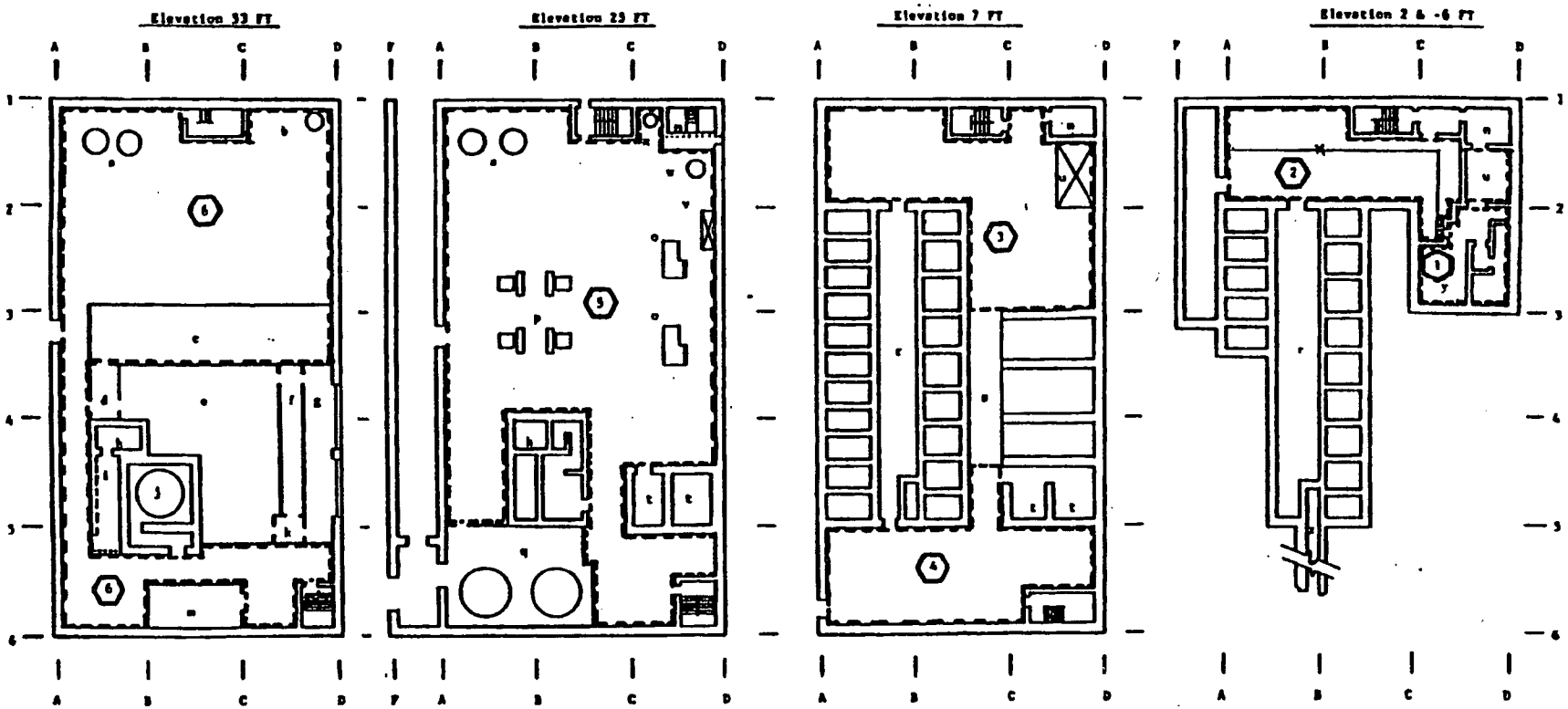
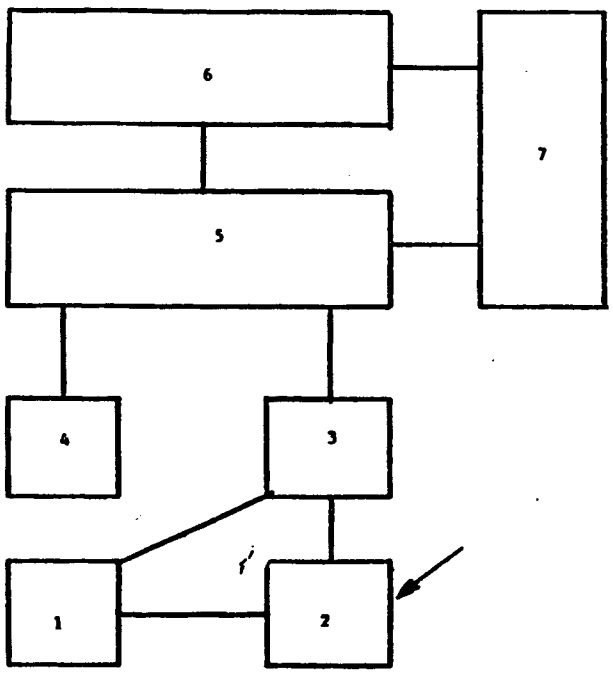


Figure 3.1-5D: Pressure Responses in Primary Auxiliary Building Following a Rupture of 3" Auxiliary Steam Condensate Line



a. PCCW Heat Exchangers
b. Blowdown Flash Tank
c. PAB Exhaust Fan Room
d. Air Lock
e. PAB Inlet & Containment Purge Fans
f. Inlet Heaters
g. Inlet Filters
h. RAACT Pipe Chase
i. Valve Aisle
j. CVCS Tank
k. Air Lock
l. Storage Area
m. Electrical Chase
n. 480 v. Load Centers
p. PCCW Pumps
q. Boric Acid Tanks
r. CVCS Vaults
s. Charging Pump Vaults
t. Letdown Degassifier
u. Non-RAACT Pipe Chase
v. HVAC Opening
w. Boron Injection Tank
x. Chiller Surge Tank
y. Aux. Steam Condensate Tank
z. Pipe Trench

Figure 3.1-6A: Nodal Arrangement of Primary Auxiliary Building at Various Elevations for Auxiliary Steam Condensate Line ASC-2406-1-4" Break Analysis



NODE	VOLUME (ft^3)	HEAT SINK AREA(ft^2)
1	8,645	4,180
2	29,700	11,700
3	49,700	10,000
4	38,200	8,500
5	243,400	42,670
6	172,940	26,560
7	ATMOSPHERE	

FLOW PATHS CHARACTERISTICS

FROM NODE	TO NODE	AREA(ft^2)	INERTIA (ft^{-1})	K_c	LOSS FACTOR K_{exp}	K_{fric}	K_{total}
1	2	2.1	15.0	.78	1.0	1.5	3.28
1	3	3.0	10.3	.78	1.0	1.4	3.18
2	3	5.9	8.5	.78	1.0	3.20	4.98
3	5	44.8	.80	.78	1.0	.30	2.08
4	5	6.3	.32	.78	1.0	.10	1.88
5	6	51.5	1.33	.78	1.0	.33	2.11
5	7	20.0	.50	.78	1.0	1.60	3.38
6	7	20.0	5.00	.78	1.0	3.50	5.28

Figure 3.1-6B: Nodal Parameters of Primary Auxiliary Building for Auxiliary Steam Condensate Line ASC-2406-1-4" Break Analysis

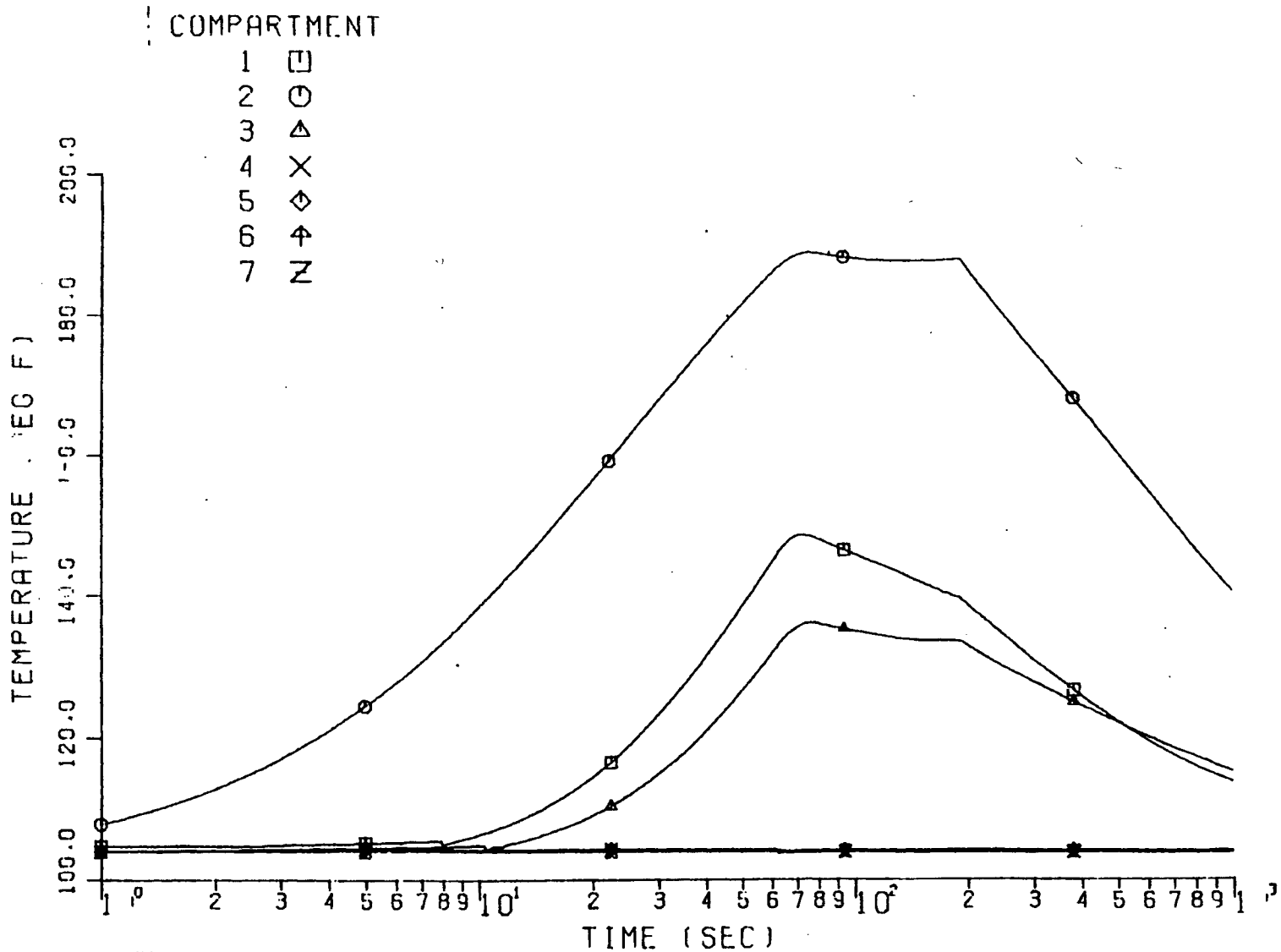


Figure 3.1-6C: Temperature Responses in Primary Auxiliary Building Following a Rupture of 4" Auxiliary Steam Condensate Line

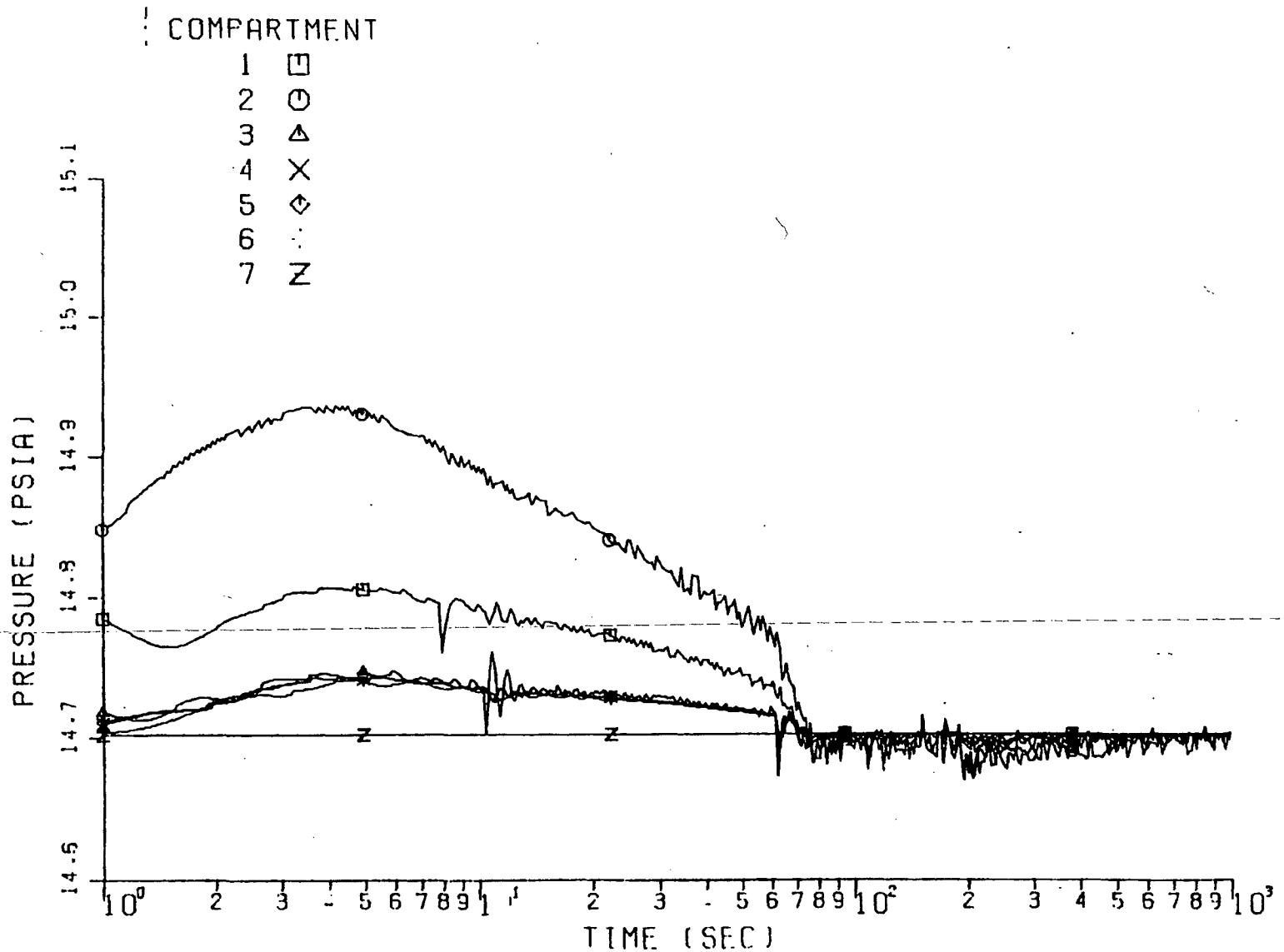
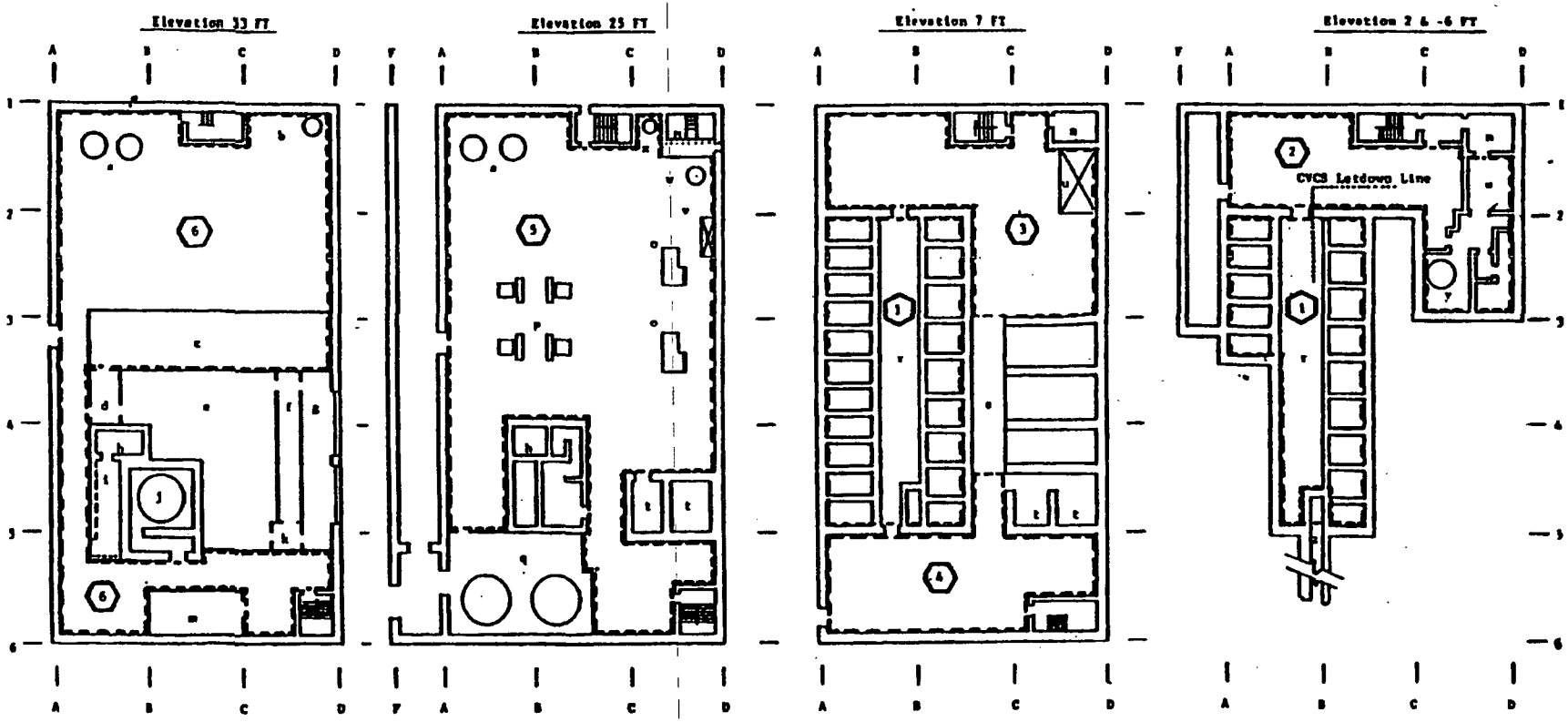
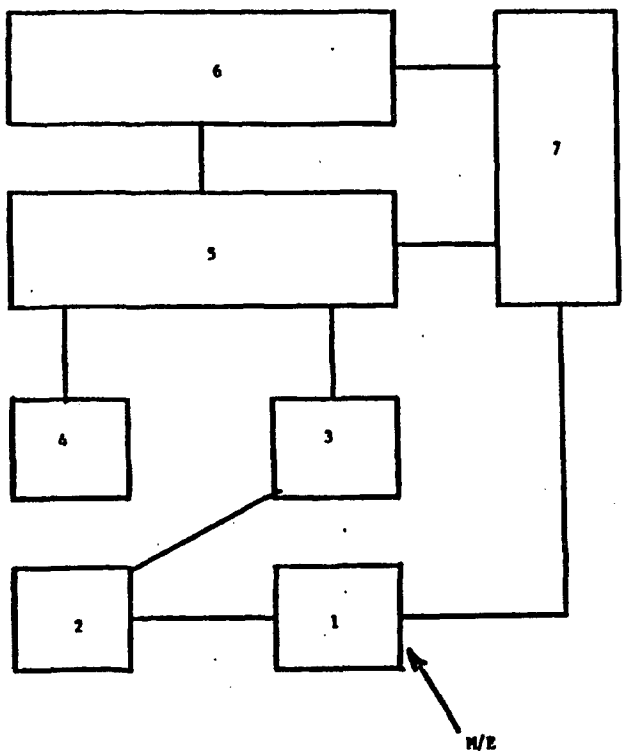


Figure 3.1-6D: Pressure Responses in Primary Auxiliary Building Following a Rupture of 4" Auxiliary Steam Condensate Line



a. PCCW Heat Exchangers
b. Blowdown Flash Tank
c. PAB Exhaust Fan Room
d. Air Lock
e. PAB Inlet & Containment Purge Fans
f. Inlet Heaters
g. Inlet Filters
h. RAACT Pipe Chase
i. Valve Aisle
j. CVCS Tank
k. Air Lock
l. Storage Area
m. Electrical Chase
o. 480 v. Load Centers
p. PCCW Pumps
q. Boric Acid Tanks
r. CVCS Vaults
s. Charging Pump Vaults
t. Letdown Degassifier
u. Non-RAACT Pipe Chase
v. HVAC Opening
w. Boron Injection Tank
x. Chiller Surge Tank
y. Aux. Steam Condensate Tank
z. Pipe Trench

Figure 3.1-7A: Nodal Arrangement of Primary Auxiliary Building at Various Elevations for CVCS Letdown Line Break Analysis



NODE	VOLUME (ft^3)	HEAT SINK AREA(ft^2)
1	47,100	23,000
2	38,235	15,900
3	49,700	10,000
4	38,200	8,500
5	243,400	42,670
6	172,940	26,560
7	ATMOSPHERE	

FLOW PATHS CHARACTERISTICS

FROM NODE	TO NODE	AREA (ft^2)	INERTIA (ft^{-1})	LOSS FACTOR K_c	K_{exp}	K_{fric}	K_{total}
1	2	.80	45.0	.11	.11	.22	.44
1	7	1.4	37.5	.34	.20	.67	1.21
2	3	2.3	47.0	.78	1.0	1.80	3.58
3	5	44.8	.80	.78	1.0	.30	2.08
4	5	6.3	.32	.78	1.0	.10	1.88
5	6	51.5	1.33	.78	1.0	.33	2.11
5	7	20.0	.50	.78	1.0	1.6	3.38
6	7	20.0	5.00	.78	1.0	3.5	5.28

Figure 3.1-7B: Nodal Parameters of Primary Auxiliary Building for CVCS Letdown Line Break Analysis

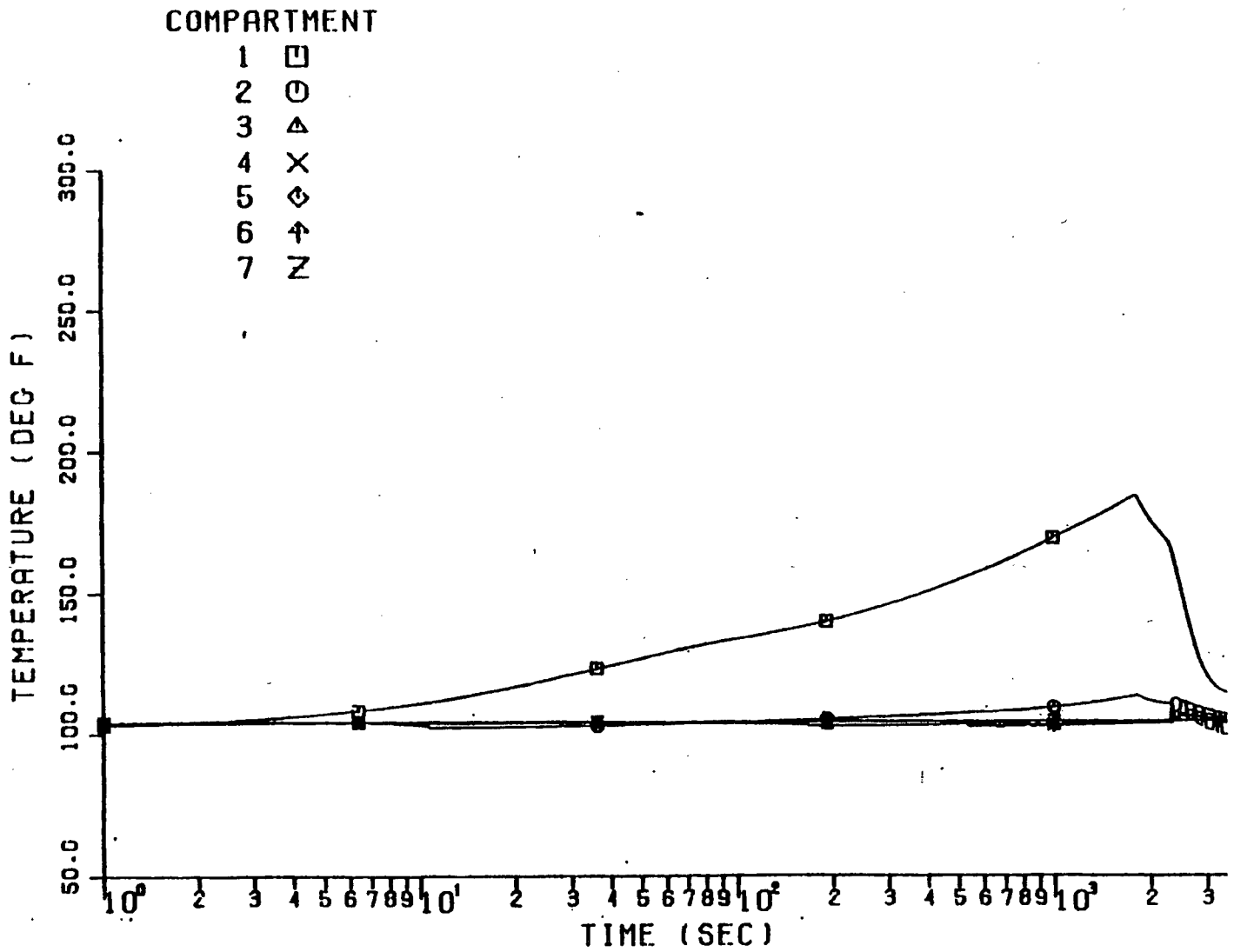


Figure 3.1-7C: Temperature Responses in Primary Auxiliary Building Following a Rupture of CVCS Letdown Line

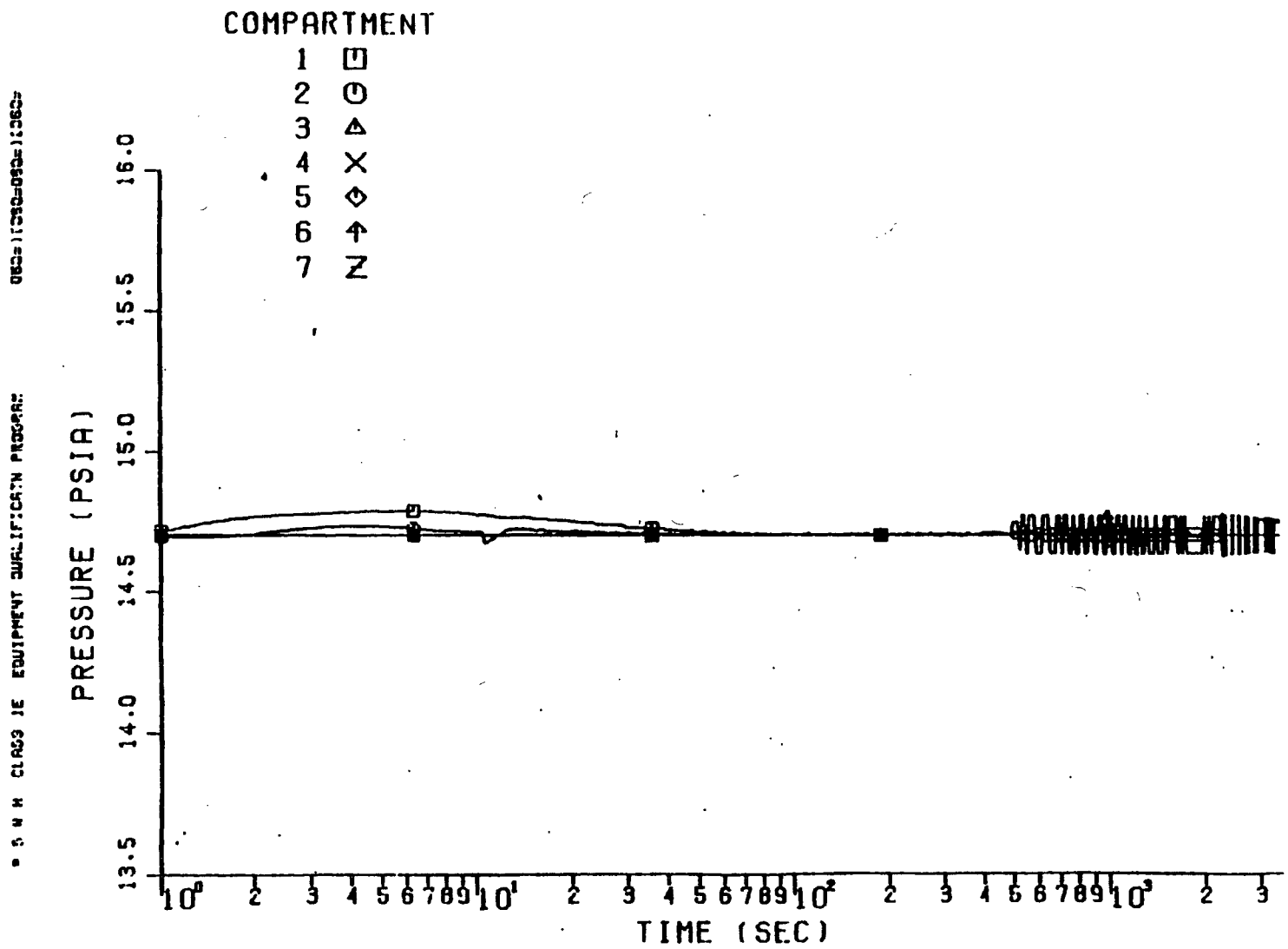


Figure 3.1-7D: Pressure Responses in Primary Auxiliary Building Following a Rupture of CVCS Letdown Line

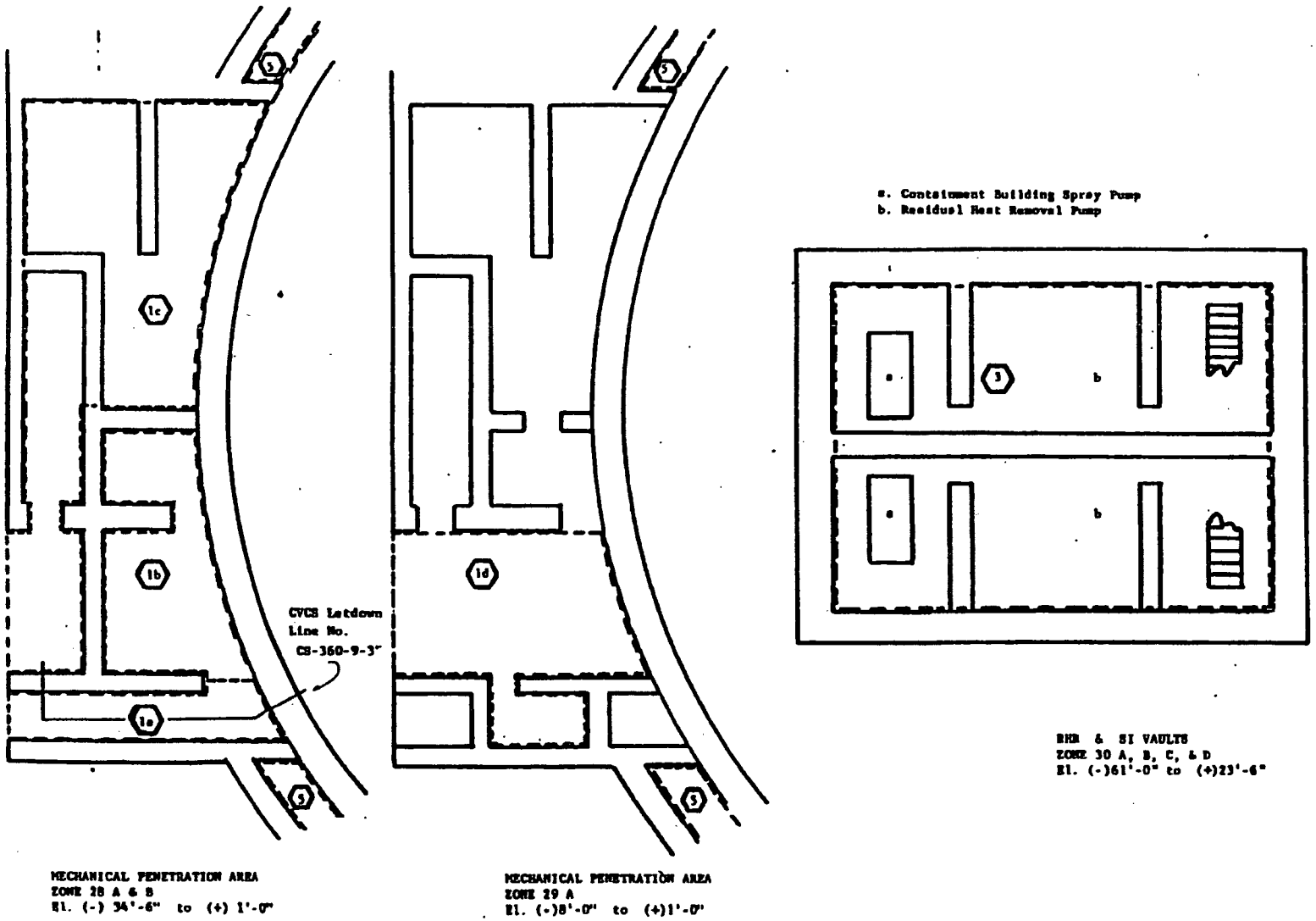


Figure 3.2-1A: Containment Enclosure Area Showing Nodal Arrangement
(Sheet 1 of 2) for CVCS Letdown Line Break Analysis

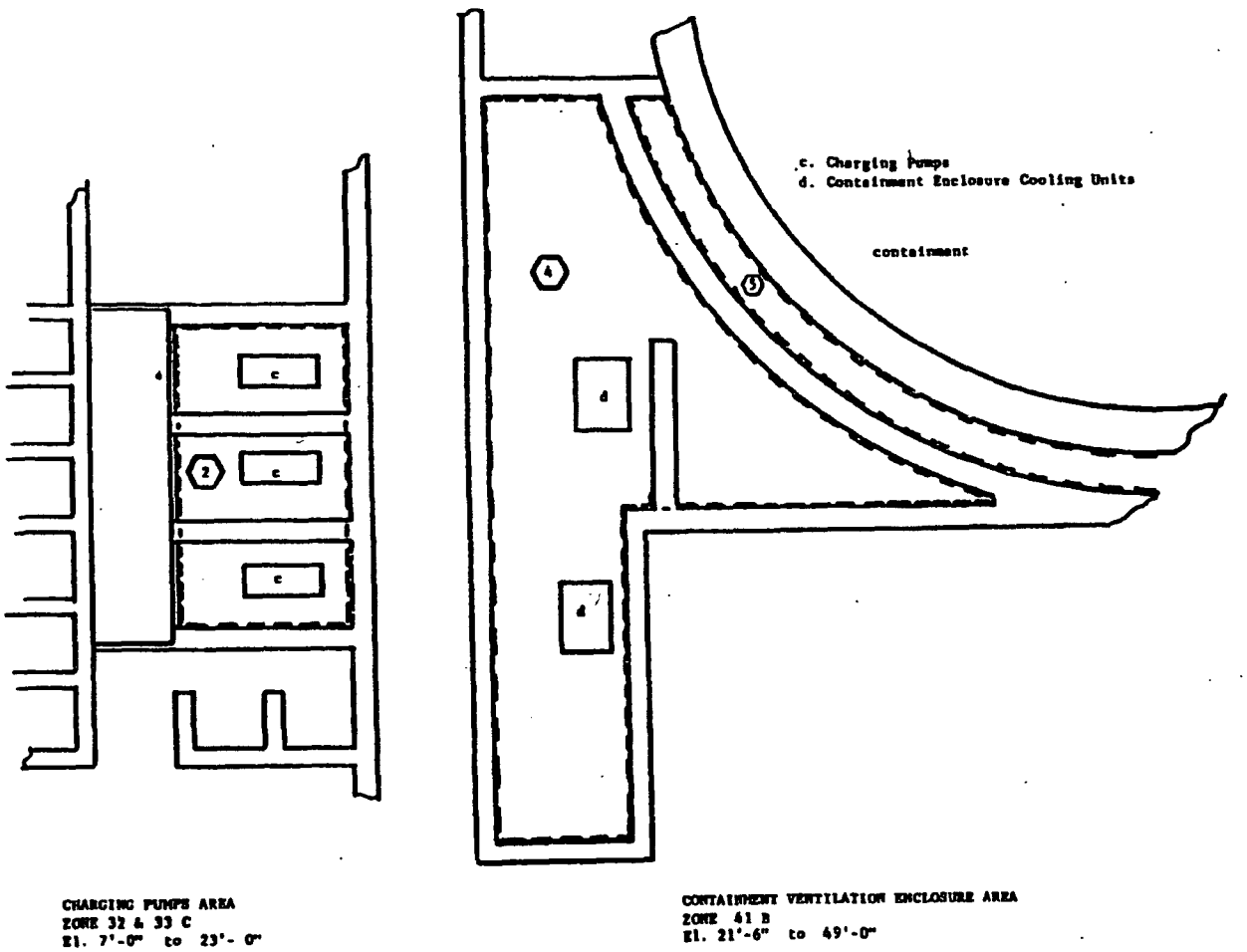
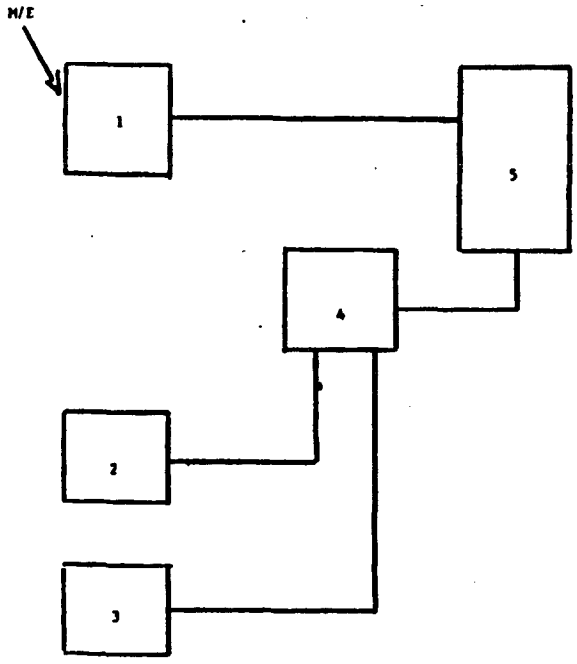


Figure 3.2-1A : Containment Enclosure Area Showing Nodal
(Sheet 2 of 2) Arrangement for CVCS Letdown Line Break Analysis



NODE	VOLUME (ft^3)	HEAT SINK AREA(ft^2)
1	66,355	21,400
2	12,000	4,050
3	144,000	61,170
4	92,570	15,000
5	524,350	145,000

FLOW PATHS CHARACTERISTICS

FROM NODE	TO NODE	AREA (ft^2)	INERTIA (ft^{-1})	LOSS FACTORS K_c	K_{exp}	K_{fric}	K_{total}
1	5	20.0	.20	.78	1.0	.02	1.80
2	4	8.0	5.50	.78	1.0	.74	2.52
3	4	18.9	4.07	.78	1.0	7.1	8.88
4	5	28.0	.18	.78	1.0	.10	1.88

Figure 3.2-1B: Nodal Parameters of Containment Enclosure Area for CVCS Letdown Line Break Analysis

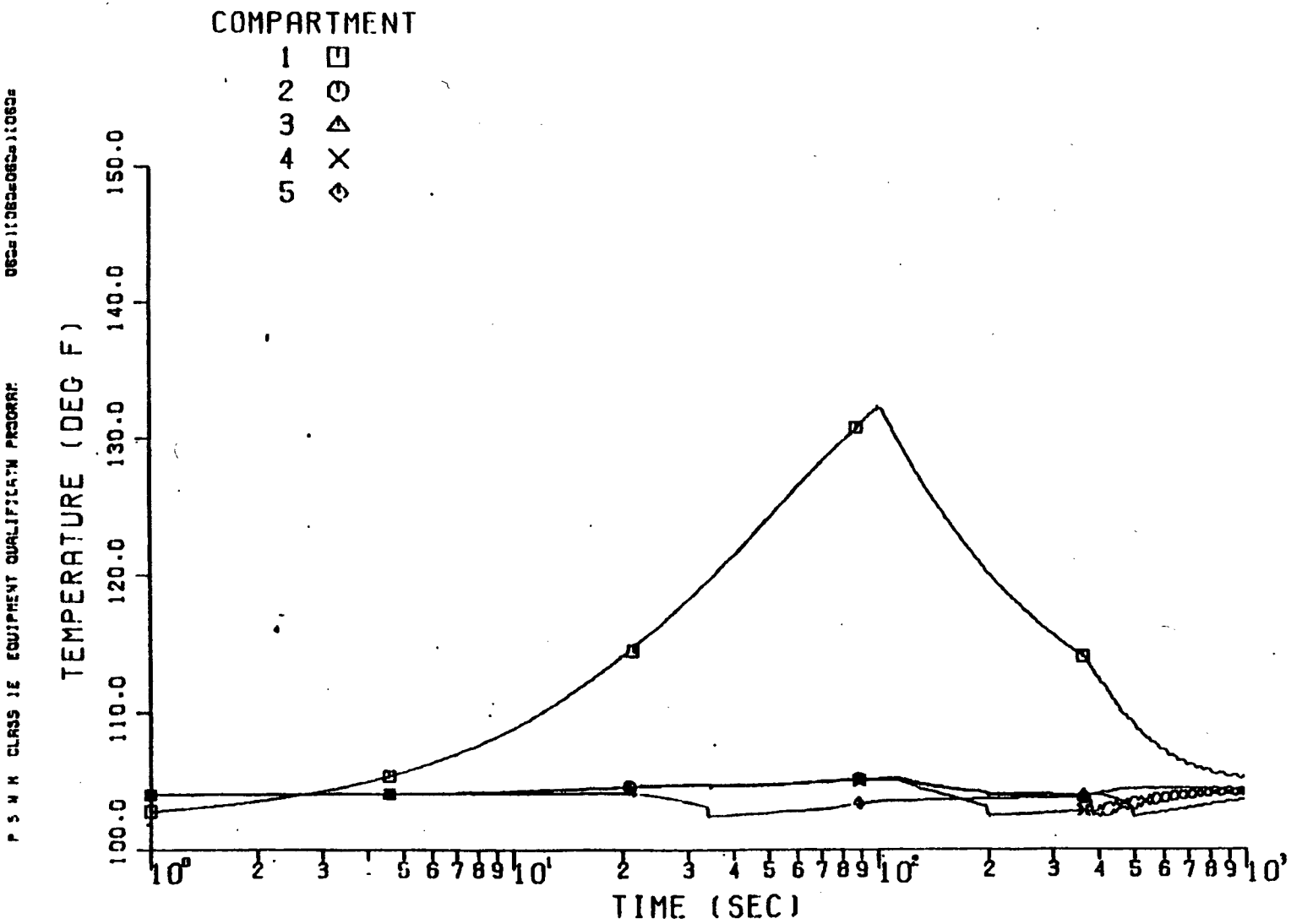


Figure 3.2-1C: Temperature Responses in Containment Enclosure Area Following a Rupture of 3" CVCS Letdown Line

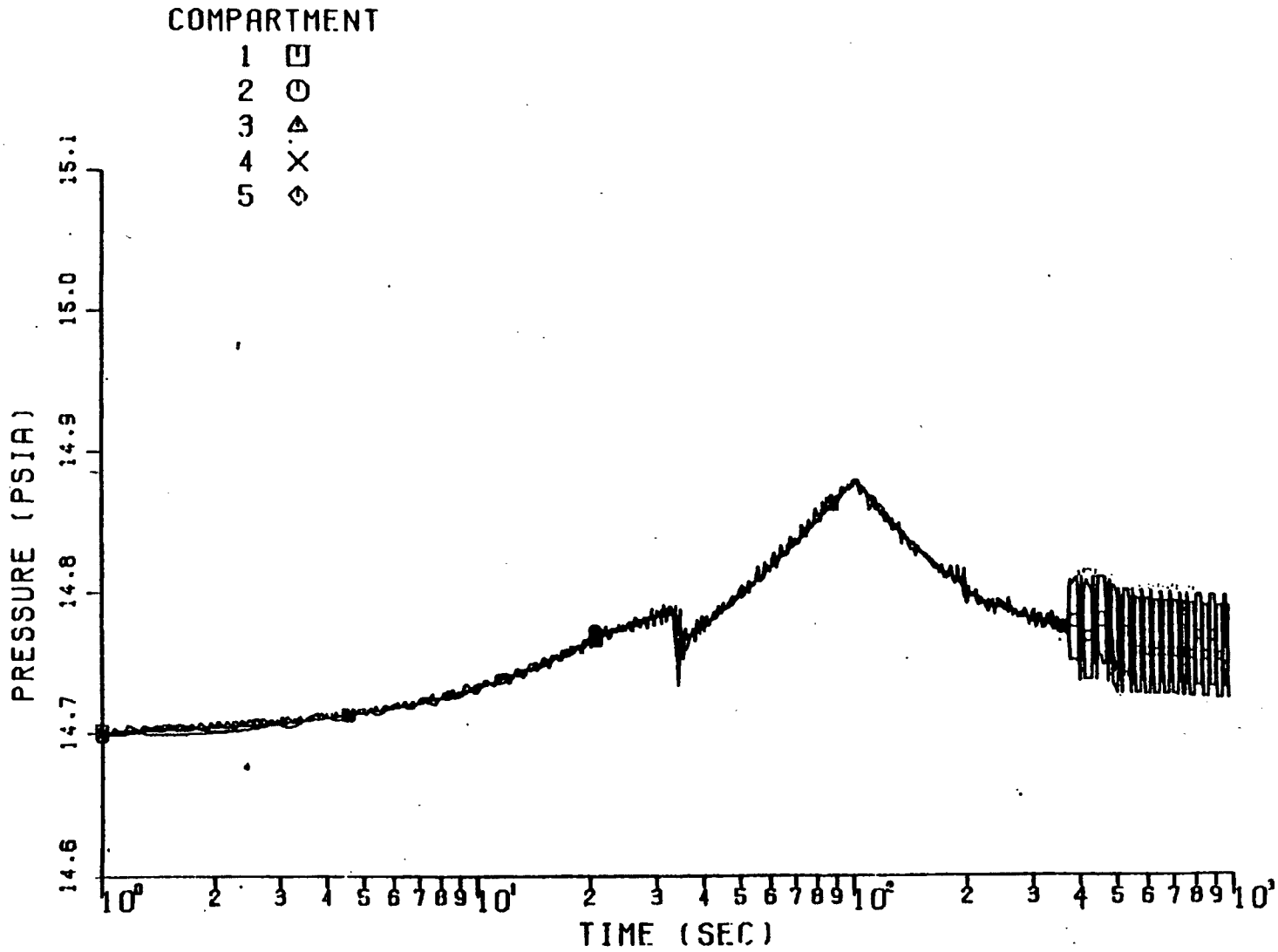


Figure 3.2-1D: Pressure Responses in Containment Enclosure Area Following a Rupture of 3" CVCS Letdown Line

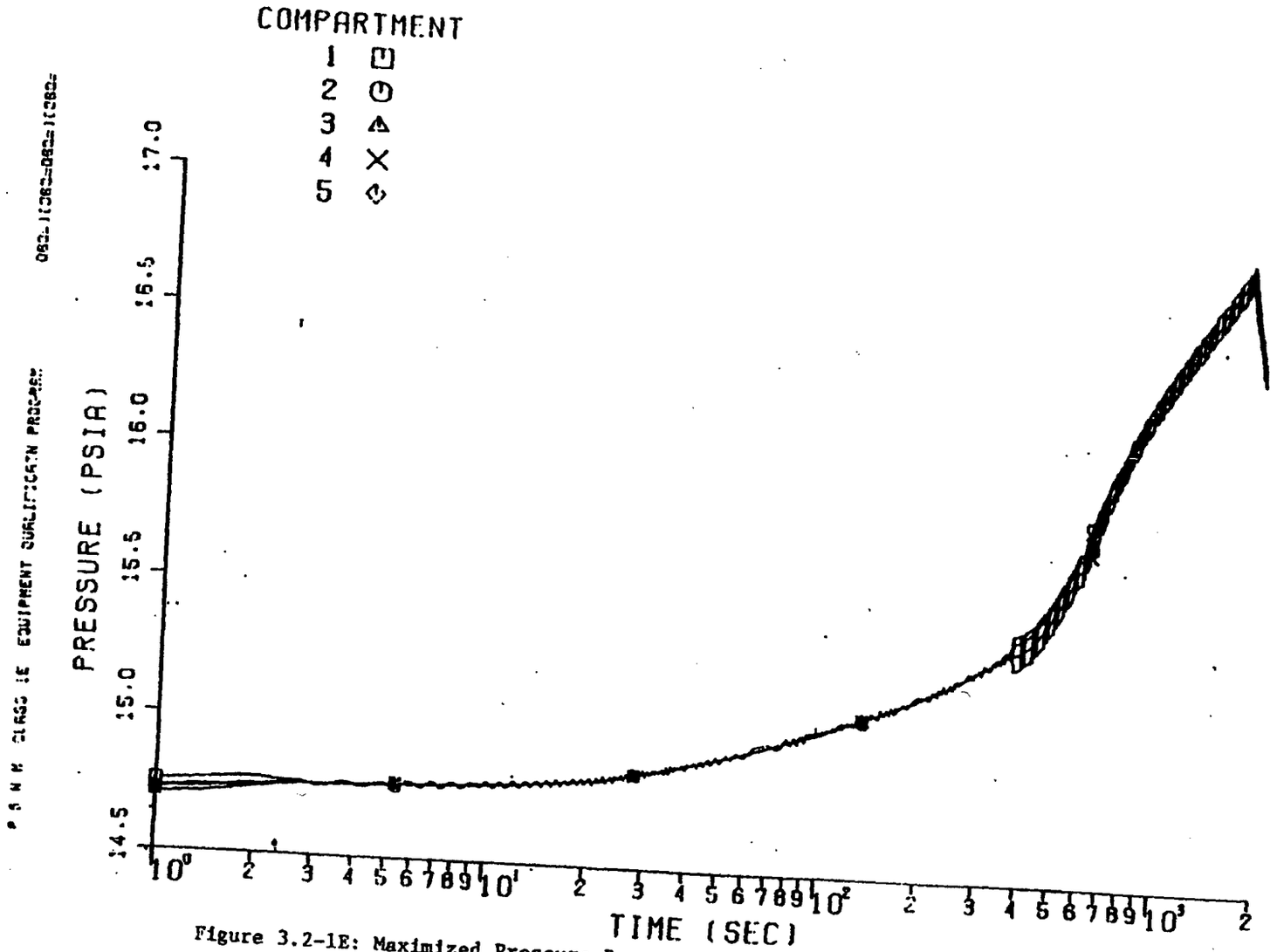


Figure 3.2-1E: Maximized Pressure Responses in Containment Enclosure Area Following a Rupture of 3" CVCS Letdown Line

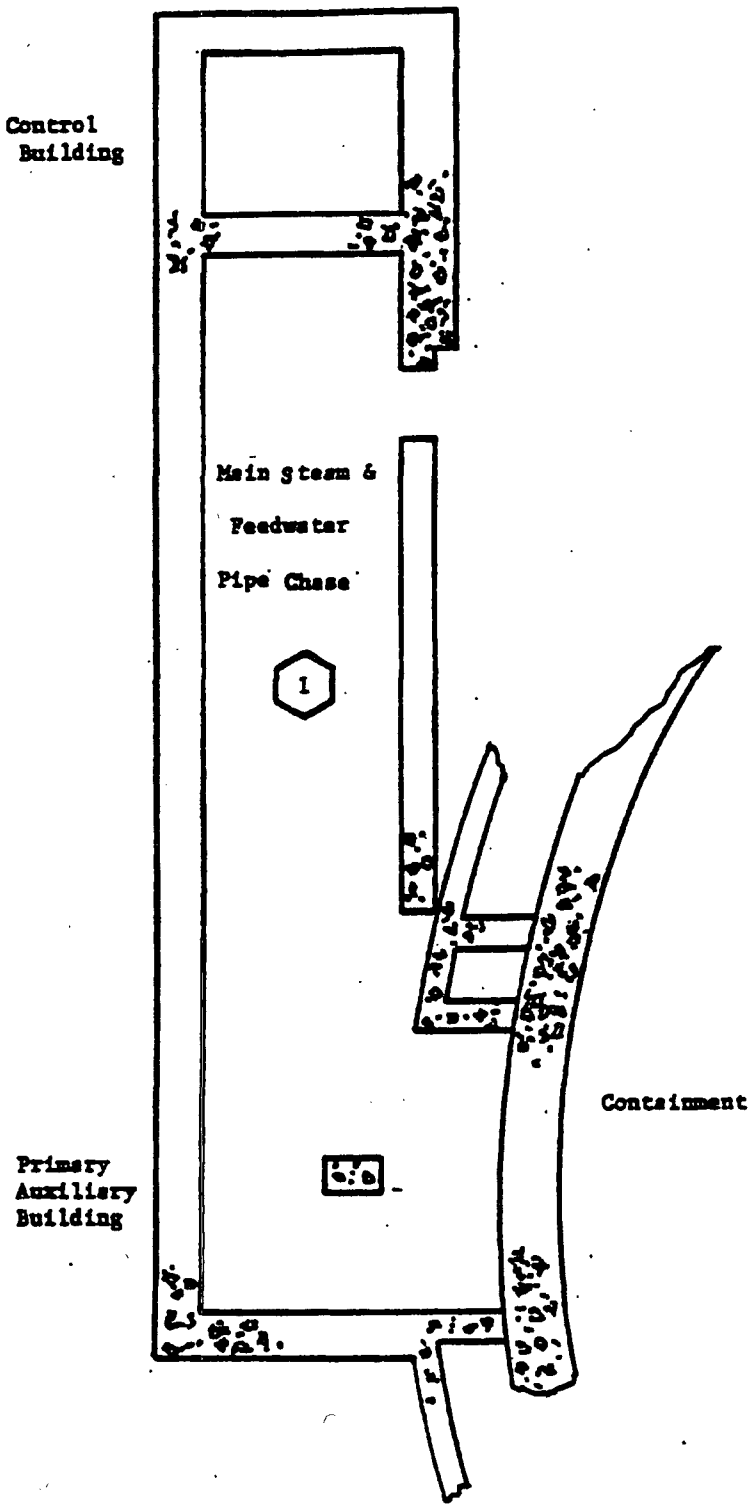
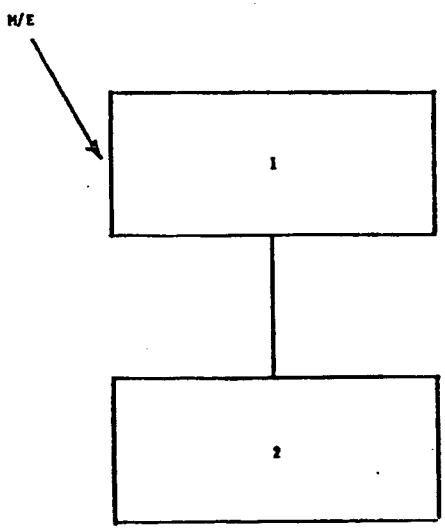


Figure 3.3-1A: Nodal Arrangement of Main Steam/
Feedwater Pipe Chase



NODE NODE	VOLUME (ft^3)	HEAT SINK Area (ft^2)
1	69,270	16,930
2	Atmosphere	

FLOW PATH CHARACTERISTICS

FROM NODE	TO NODE	AREA (ft^2)	INERTIA (ft^{-1})	LOSS FACTORS K_c	K_{exp}	K_{fric}	K_{total}
1	2	856	.01	.78	1.0	.10	1.88

Figure 3.3-1B: Nodal Parameters of Main Steam/Feedwater Pipe Chase for Main Steam Line Break Analysis

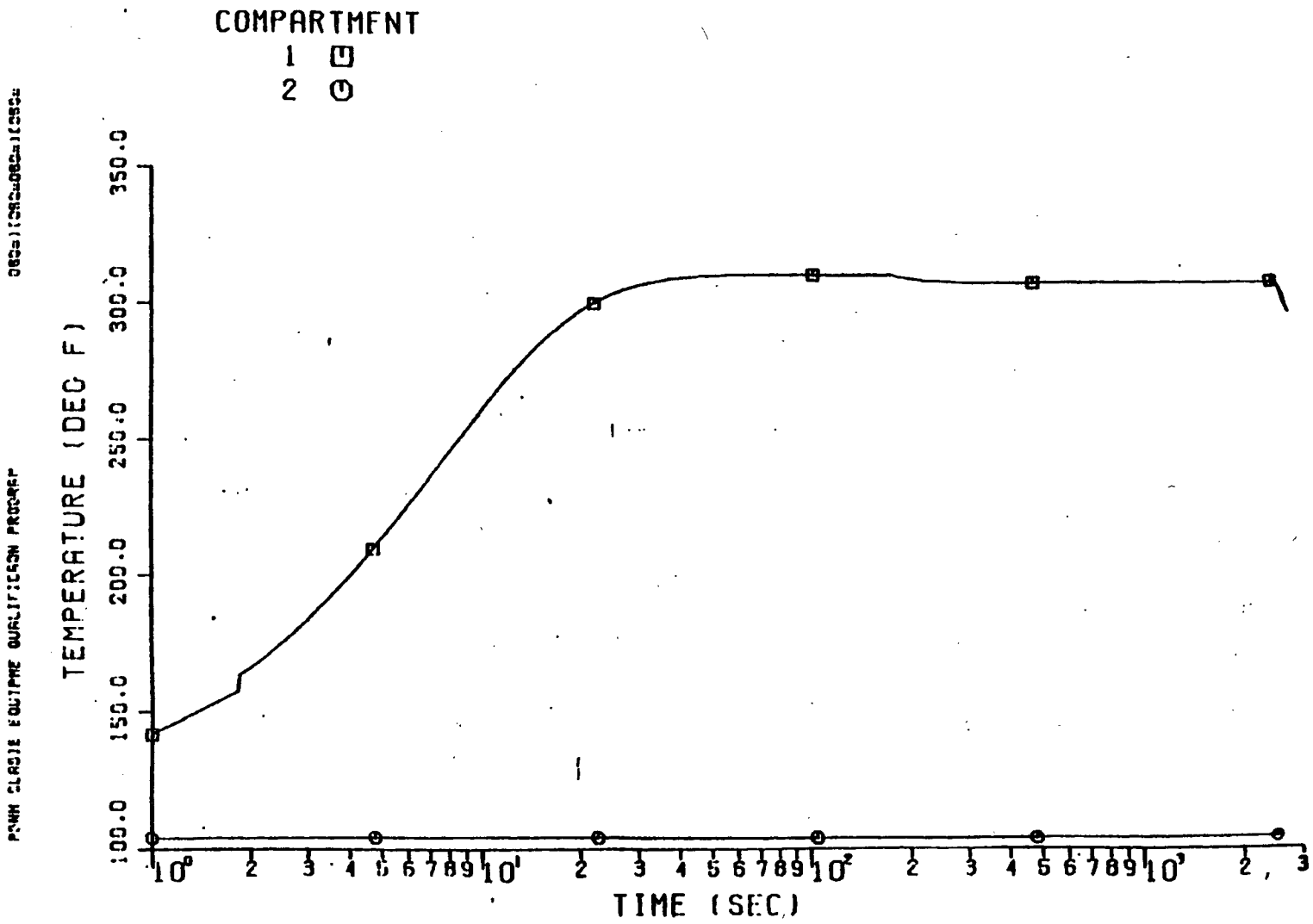
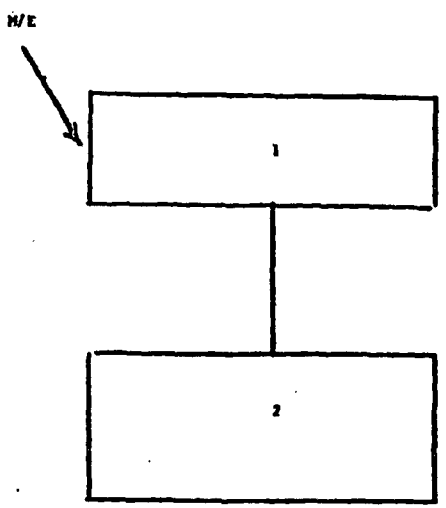


Figure 3.3-1C: Temperature Response of Main Steam/Feedwater Pipe Chase Following a Small (0.10 Square Feet) Rupture of Main Steam Line



NODE	VOLUME (ft^3)	HEAT SINK AREA(ft^2)
1	240,000	11,000
2	Atmosphere	

FLOW PATH CHARACTERISTICS

FROM NODE	TO NODE	AREA (ft^2)	INERTIA (ft^{-1})	K_c	LOSS FACTORS K_{exp}	K_{fric}	K_{total}
1	2	10.00	.006	.78	1.0	0.01	1.79

Figure 3.4-1A: Nodal Parameters of Tank Farm Area for Auxiliary Steam Line AS-2302-32-8" Break Analysis

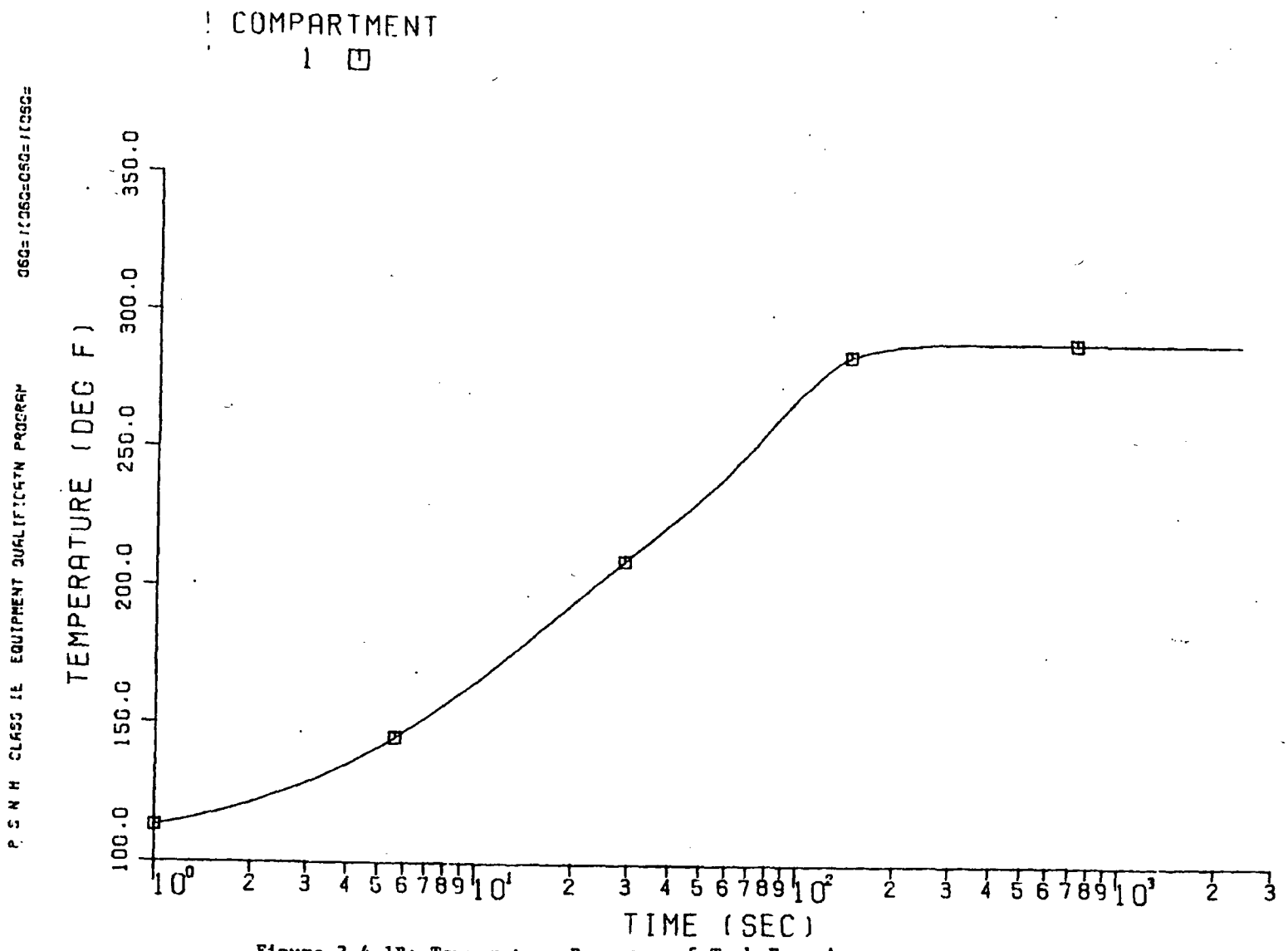


Figure 3.4-1B: Temperature Response of Tank Farm Area Following a Rupture of 8" Auxiliary Steam Line

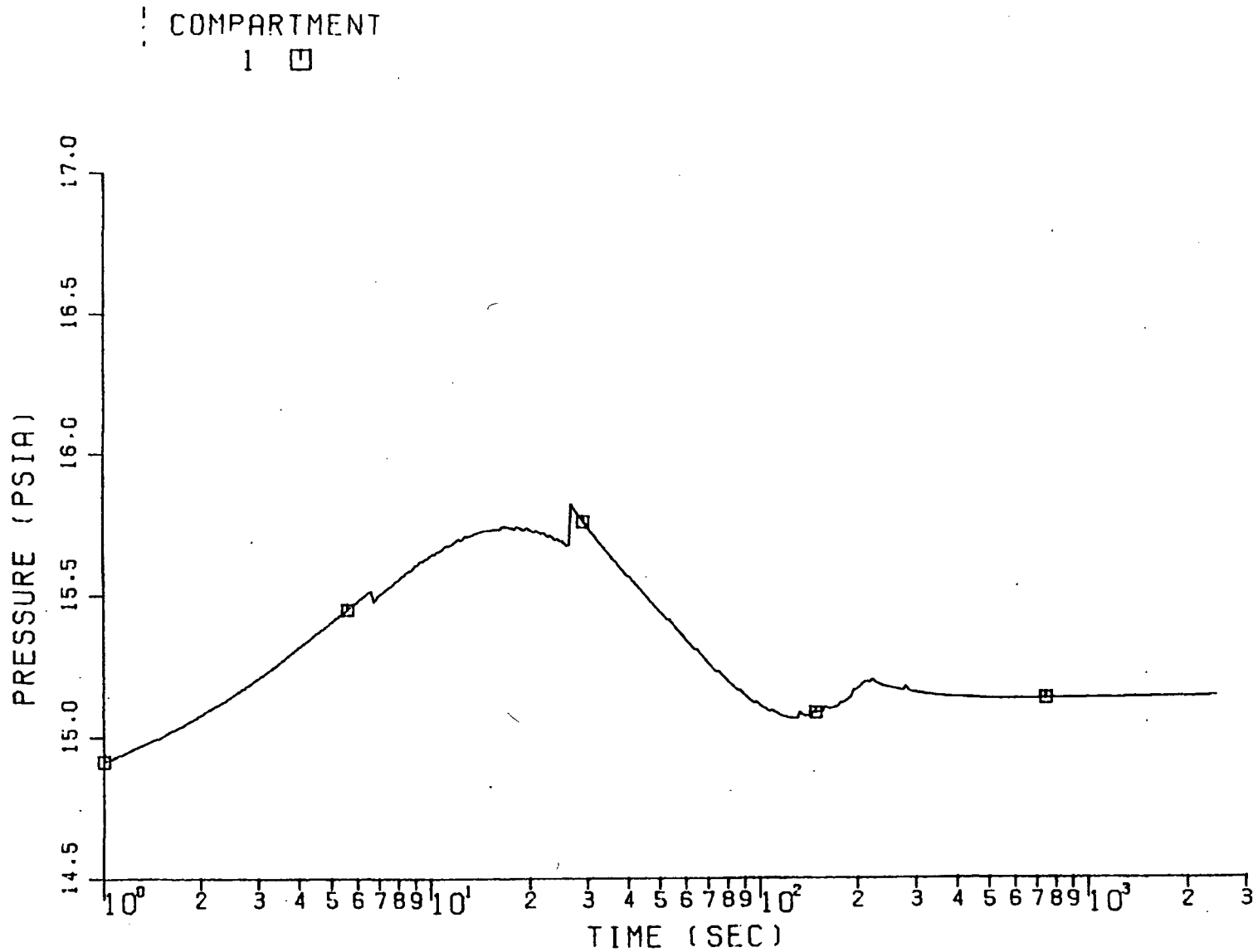
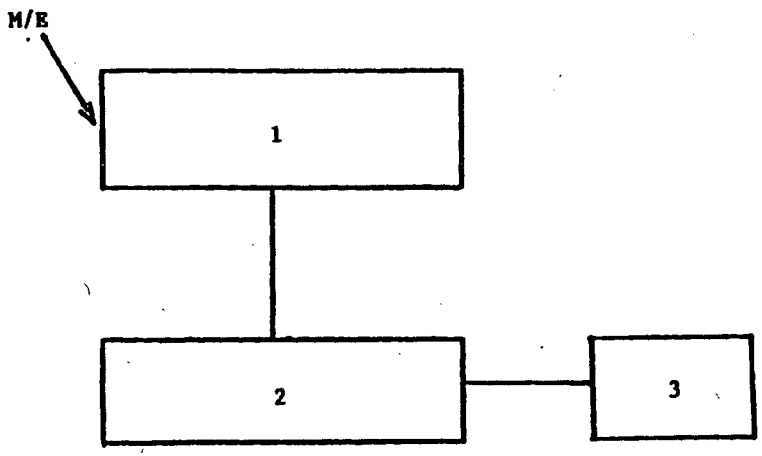


Figure 3.4-1C: Pressure Response of Tank Farm Area Following a Rupture of 8" Auxiliary Steam Line



NODE	VOLUME (ft^3)	HEAT SINK AREA(ft^2)
1	19,030	3,890
2	17,500	4,760
3	Atmosphere	

FLOW PATHS CHARACTERISTICS

FROM NODE	TO NODE	AREA (ft^2)	INERTIA (ft^{-1})	LOSS FACTORS K_c	K_{exp}	K_{fric}	K_{total}
1	2	8.40	0.24	.78	1.0	0.01	1.79
2	3	1.75	1.14	.78	1.0	0.06	1.84

Figure 3.5-1A: Nodal Parameters of Waste Processing Building/Primary Auxiliary Building Chase for Auxiliary Steam Line AS-2339-1-1 1/2" Break Analysis

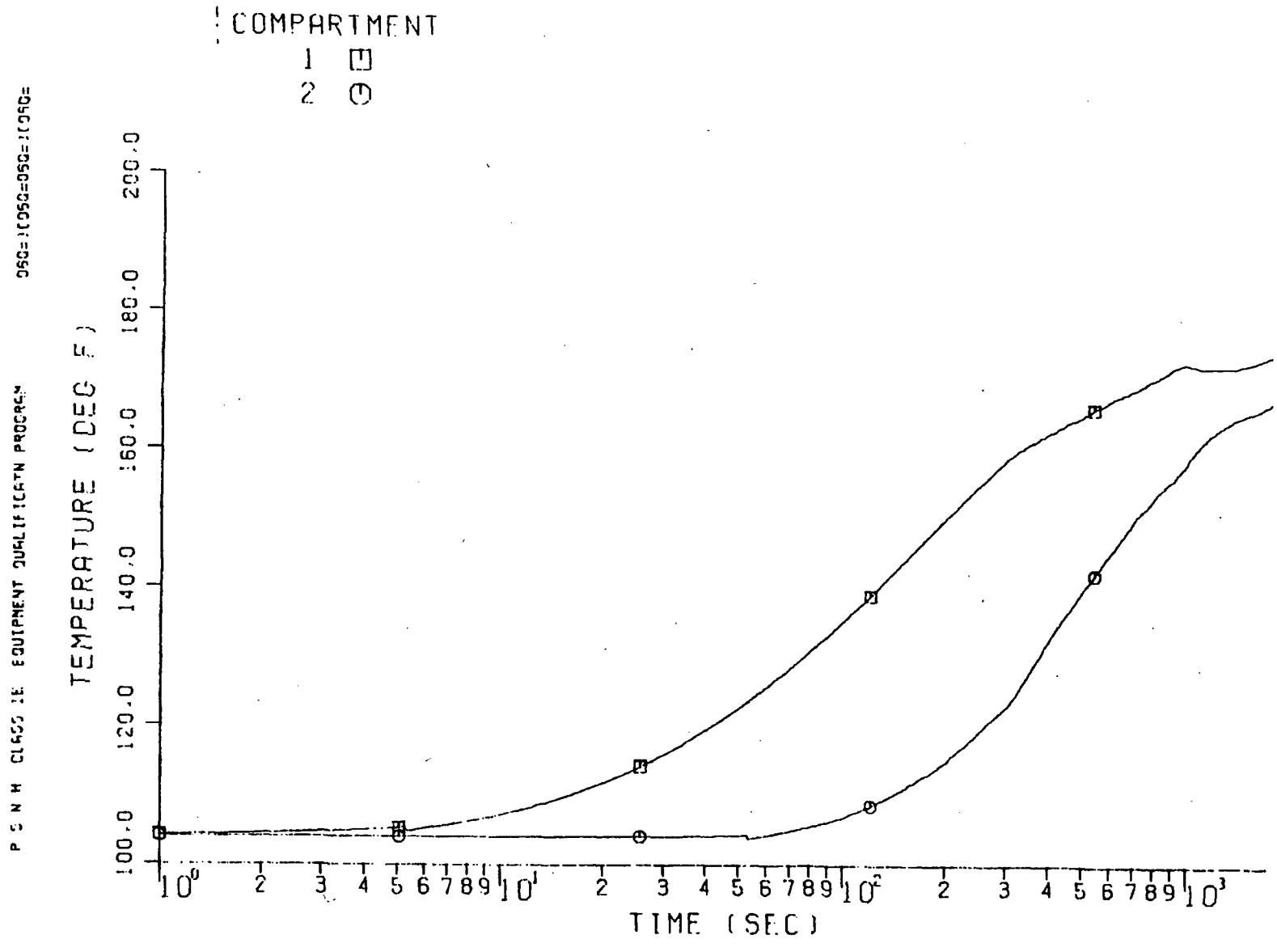


Figure 3.5-1B: Temperature Response of WPB/PAB Chase
Following a Rupture of 1 1/2" Auxiliary Steam Line

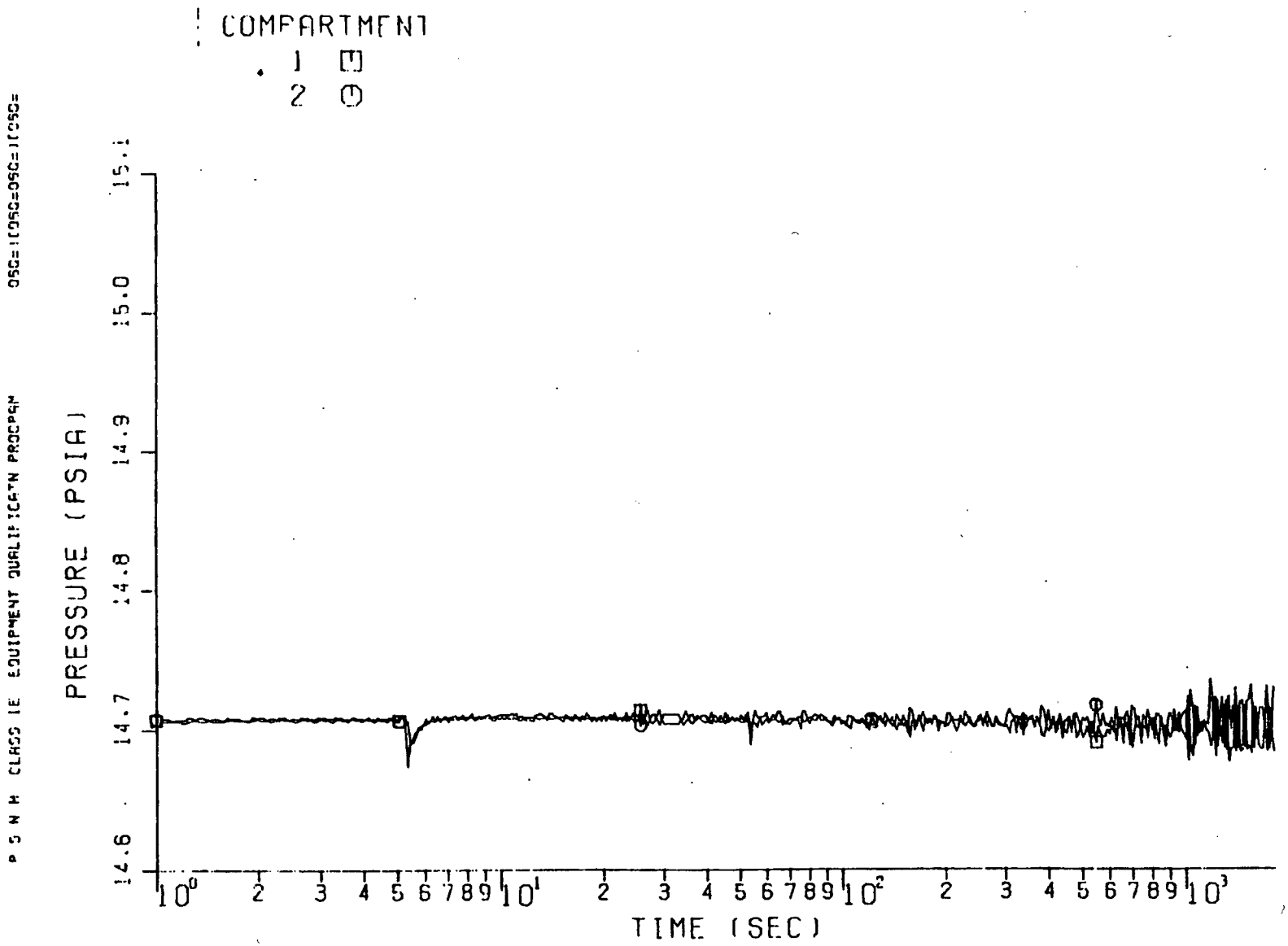


Figure 3.5-1C: Pressure Response of WPB/PAB Chase
Following a Rupture of 1 1/2" Auxiliary Steam Line

temperature detection system is use, the peak pressures correspond to approximately 95 seconds after the break. These peak pressures are listed in Table 3.2-2. The relative humidity throughout all CEA compartments would reach 100%.

3.3 Main Steam/Feedwater Pipe Chase

The breaks evaluated for the Main Steam/Feedwater Pipe Chase are listed in Table 3.3-1. It was concluded that the MS line breaks will result in more severe environmental conditions than the FW line breaks.

Figures 3.3-1A and 3.3-1B define the MS/FW Pipe Chase arrangement and nodalization. The MS/FW Pipe Chase reaches a maximum of 325°F for a spectrum of MS line break sizes from 0.10 ft^2 to 1.0 ft^2. The temperature transient resulting from a 0.10 ft^2 break is provided in Figure 3.3-1C and the results are summarized in Table 3.3-2.

3.4 Tank Farm Area

The break evaluated for the Tank Farm Area is listed in Table 3.4-1. Since no HELB temperature detectors are located in the Tank Farm Area, the Auxiliary Steam line break releases will continue until the operator detects the break and isolates the line.

Figure 3.4-1A defines the nodal parameters used for the Tank Farm Area HELB analysis. The resulting temperature and pressure transients are provided in Figures 3.4-1B and 3.4-1C, respectively, and the peak values summarized in Table 3.4-2.

3.5 Waste Processing Building/Primary Auxiliary Building Chase

The WPB/PAB Chase, which is located between the WPB and Column Line A of the PAB, contains both Class 1E equipment and several Auxiliary Steam and Condensate lines. The line ruptures which have been evaluated are listed in Table 3.5-1.

Figure 3.5-1A defines the nodal parameters used for evaluation of the WPB/PAB Chase response to postulated HELB. Figures 3.5-1B and 3.5-1C provide the temperature and pressure transients that result for the enveloping HELB. The peak values for pressure and temperature are summarized in Table 3.5-2.

4.0 HWHLB ANALYSES AND RESULTS

The environmental response following postulated HWHLB has been calculated for those plant buildings with Hot Water Heating (HWH) systems which operate in the high energy region, i.e. pressure greater than 275 psig or temperature greater than 200°F. The HWHLB postulated are listed in Table 4.0-1. The results of these HWHLB analyses are presented individually in the following sections and are summarized in Table 4.0-2.

4.1 Primary Auxiliary Building

The peak environmental conditions at the 53' elevation of the PAB due to postulated HWHLB were found to be 110°F with a relative humidity of 100%. These conditions are enveloped by the consequences resulting from other HELB postulated to occur in the PAB.

4.2 Containment Enclosure Area

The HWH system piping which serves the PAB and FSB passes through the CEA. A postulated rupture of one of these lines results in temperatures and relative humidities throughout the CEA of approximately 106°F and 100%, respectively. Due to the location of this piping, very localized conditions may be slightly more severe although the large recirculation air flows will tend to mitigate these effects to a certain extent. With the exception of these localized effects the environmental conditions that result from a CVCS letdown line break will envelope those resulting from a HWHLB.

4.3 Fuel Storage Building

Since the hot water heating piping are the only high energy lines present in the FSB, the environmental conditions that result from a postulated HWHLB will define the enveloping conditions for high

energy line ruptures. The resulting environmental conditions are 100°F with a 100% relative humidity.

4.4 Emergency Feedwater Pumphouse

Since the hot water heating piping are the only high energy lines present in the EFWPH, the environmental conditions that result from a postulated HWHLB will define the enveloping conditions for high energy line ruptures. The resulting environmental conditions are 88°F with a 100% relative humidity.

4.5 Service Water Pumphouse

Since the hot water heating piping are the only high energy lines present in the SWPH, the environmental conditions that result from a postulated HWHLB will define the enveloping conditions for high energy line ruptures. The maximum temperature that would be expected to result in the SWPH is 90°F. Due to the relatively large room volume and small volume of hot water heating piping for the SWPH, the maximum relative humidity that is expected to result following a HWHLB is 90%

5.0 CONCLUSIONS

The analysis of high energy line ruptures outside containment has yielded a realistic evaluation of the elevated temperatures, pressures, and humidities that can result in the various buildings of Units 1 and 2. These results provide the HELB environmental envelopes for evaluation of the Class 1E equipment. These envelopes should be evaluated along with the conditions that result following postulated moderate energy line breaks, loss of ventilation air flow, and any other events which may cause adverse environmental conditions to develop.

6.0 REFERENCES

1. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", August, 1979.

2. NUREG-0800, U.S. NRC Standard Review Plans 3.6.1 and 3.6.2, July, 1981.

3. Moody, F. J., "Maximum Two-Phase Vessel Blowdown from Pipes", Journal of Heat Transfer, August 1966.

4. UEC-TR-004-1, "COMPRESS - A Code for Calculating Subcompartment Pressure Responses", July, 1976.

5. Appendix E attached to ANSI Standard N176, "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture".

6. Calculation Set No. 4.3.35-F03

7. Calculation Set No. MSVCS-FAG-07

8. Calculation Set No. 4.3.35-F01

TABLE 2.1-1

DETERMINATION OF MASS/ENERGY RELEASE

Line	CVCS Letdown Line	Steam Generator Blowdown	Auxiliary Steam and Condensate Lines	Main Steam Line	Feedwater Line	Hot Water Heating Line
Plant Condition	Heatup Phase	Hot Standby	Full Power	Full Power	Full Power	Full Power
Line Conditions	P = 435 psia T = 380°F	P = 1100 psia T = 550°F	P = 165 psia T = 358°F	P = 1000 psia T = 545°F	P = 1100 psia T = 440°F	P = 157 psia T = 250°F
Break Flow	Limited by CVCS Letdown Line Control Valves	Moody critical flow with piping system frictional effects included (Methodology of App. E attached to ANSI Std. N176[5])	Limited by upstream pressure control valves	Releases calculated using Westinghouse information package methodology	Releases calculated using Westinghouse information package methodology	Moody Critical Flow Model
Isolation Mechanism	Dependent on Location: HELB Temperature Detection System or Operator Action at 30 min.	HELB Temperature Detection System	Dependent on Location: HELB Temperature Detection System or operator action at 30 minutes.	Reactor Protection System and Emergency Feedwater Discontinued at 30 min.	Reactor Protection System	No Isolation Occurs
Isolation Valve Closure Time	10 Seconds	5 Seconds	15 Seconds	Isolation Valve in Faulted Loop Fails.	Isolation Valve in Faulted Loop Fails	No Isolation Occurs

TABLE 2.2-1

Design Basis Information

A. Ambient Conditions	
1. HELB, other than HWHLB:	14.7 psia/88°F/100% RH
2. HWHLB:	14.7 psia/70°F/95% RH
B. Building Initial Conditions	
1. HELB, other than HWHLB:	14.7 psia/104°F/95% RH
2. HWHLB:	14.7 psia/86°F/56% RH
C. HELB Temperature Detection System	
1. Temperature at Isolation Signal Initiation: (Intended to cover setpoint plus instrument error margins of up to 10°F)	130°F
2. System Response Time-time delay: until signal at isolation valves	8.1 Seconds
D. Ventilation System Operation	
1. No credits are taken for energy removal or air exchange by non-Class 1E ventilation systems.	
2. Credits are taken for Class 1E ventilation systems according to their performance characteristics following postulated HELB.	
E. Unit Trip	
1. A concurrent loss of offsite power or unit trip has not been assumed.	

TABLE 3.1-1

Primary Auxiliary Building
High Energy Line Break Locations

1. Steam Generator Blowdown Line (Lines No. SG-1301-5-3", SG1304-5-3", SG-1307-5-3", or SG-1310-5-3")

   a. At 53' elevation of PAB in vicinity of blowdown flash tank.

2. Auxiliary Steam and Condensate Lines

   a. Line No. 2302-2-8" - At 53' elevation of PAB along Column Line 5 between Columns A & B.

   b. Line No. 2303-1-6" - At 7' elevation of PAB between Column Lines 5 & 6.

   c. Line No. 2404-2-3" - At (-) 6' elevation of PAB along Column Line C.

   d. Line No. 2406-1-4" - At (-) 6' elevation of PAB along Column Line 2.

3. Chemical and Volume Control System Letdown Line (Line No. CS-360-9-3")

   a. At 7' elevation of PAB in the CVCS equipment vault area.

TABLE 3.1-2

Primary Auxiliary Building
Summary of Results

ZONE DESIGNATION	SG-1310-5-3" Break @ Zone 32E		AS-2302-2-8" Break @ Zone 33E		AS-2303-1-6" Break @ Zone 33C		AS-2404-2-3" Break @ Zone 32B		AS-2406-1-4" Break @ Zone 32B		CS-360-9-3" Break @ Zone 47		Enveloping Conditions	
	Temp. °F	Press. psig	Temp. °F	Press. psig	Temp. °F	Press. psig	Temp. °F	Press. psig	Temp. °F	Press. psig	Temp. °F	Press. psig	Temp. °F	Press. psig
32A	108.	0.4	104	.04	104.	0.1	220	0.4	190.	0.3	114	.05	220.	0.4
32B	108.	0.4	104.	.04	104.	0.1	220	0.4	190.	0.3	114.	.05	220.	0.4
32C	108.	0.4	104.	.04	104.	0.1	132	0.1	136.	0.1	112.	.05	136.	0.4
33C	108.	0.4	104.	.04	163.	0.1	105	0.1	104.	0.1	107.	.05	163.	0.4
32D	111.	0.4	105.	.04	113.	0.1	105	0.1	104.	0.1	108.	.05	113.	0.4
33D	111.	0.4	105.	.04	113.	0.1	105	0.1	104.	0.1	108.	.05	113.	0.4
32E	165.	0.5	112.	.04	104.	0.1	105	0.1	104.	0.1	107.	.05	165.	0.5
33E	131.	0.5	158.	.06	104.	0.1	105	0.1	104.	0.1	107.	.05	158.	0.5
47	108.	0.4	104.	.04	134.	0.1	105	0.1	120.	0.1	185.	.15	185.	0.4
48	108.	0.4	104.	.04	134.	0.1	105	0.1	120.	0.1	185.	.15	185.	0.4

TABLE 3.2-1

Containment Enclosure Area
High Energy Line Break Locations

1. Chemical and Volume Control System Letdown Line (Line No. CS-360-9-3")

   a. In Mechanical Penetration Area (MPA) at (-) 34'-6" elevation near containment wall penetration.

TABLE 3.2-2

Containment Enclosure Area
Summary of Results

Compartment	CVCS Letdown Line Rupture (CS-360-9-3")	
	Peak Temperature (°F)	Peak Pressure (psig)
Mechanical Penetration Area	134	0.35
Remainder of Enclosure Volume (Including Charging Pump Cubicles & Ventilation Equipment Area)	108	0.35

TABLE 3.3-1

Main Steam/Feedwater Pipe Chase
High Energy Line Break Locations

1. Main Steam Line

   a. At 21' elevation of MS/FW Pipe Chase

2. Feedwater Line

   a. At 3' elevation of MS/FW Pipe Chase

TABLE 3.3-2

Main Steam/Feedwater Pipe Chase
Summary of Results

Main Steam Line Rupture	
Peak Temperature (°F)	Peak Pressure (psig)
325	Pressure Varies dependent upon location with respect to break location and has been studied in detail in a separate analysis. Maximum Pressure: 4.8

TABLE 3.4-1

Tank Farm Area
High Energy Line Break Locations

1. Auxiliary Steam and Condensate Lines

   a. Line No. AS-2302-32-8"

TABLE 3.4-2

Tank Farm Area
Summary of Results

Auxiliary Steam Line Rupture	
Peak Temperature (°F)	Peak Pressure (psig)
290	1.1

TABLE 3.5-1

Waste Processing Building/Primary Auxiliary Building Chase
High Energy Line Break Locations

1. Auxiliary Steam and Condensate Lines

   a. Line No. 2339-1-1 1/2" - At 53' elevation of WPB/PAB Chase

   B. Line No. 2341-1-1 1/2" - At 25' elevation of WPB/PAB Chase

TABLE 3.5-2

Waste Processing Building/Primary Auxiliary Building Chase
Summary of Results

Compartment	AS-2339-1-1 1/2" Break @ 53' elevation		AS-2341-1-1 1/2" Break @ 25' elevation	
	Temp. (°F)	Pressure (psig)	Temp. (°F)	Pressure (psig)
WPB/PAB Chase 53' elevation	175	0.05	168	0.05
WPB/PAB Chase 25' elevation and 15' 5" elevation	168	0.05	175	0.05

NOTE: Due to the general arrangement of the WPB/PAB Chase area, the results obtained for a break of Line No. AS-2339-1-1 1/2" have been extrapolated to be representative of the environmental conditions that would result from a break of Line No. AS-2341-1-1 1/2".

TABLE 4.0-1

Hot Water Heating Line Break Locations

1. Primary Auxiliary Building

   a. At 53' elevation of PAB

2. Containment Enclosure Area

   a. At 21'-6" elevation of CEA

3. Fuel Storage Building

   a. At 21'-6" elevation of FSB

4. Emergency Feedwater Pumphouse

   a. At 27' elevation of EFWPH

5. Service Water Pumphouse

   a. At 21' elevation of SWPH

TABLE 4.0-2

Hot Water Heating Line Breaks
Summary of Results

1. Primary Auxiliary Building: (53' elevation)	110°F/100% RH
2. Containment Enclosure Area:	106°F/100% RH
3. Fuel Storage Building:	100°F/100% RH
4. Emergency Feedwater Pumphouse:	88°F/100% RH
5. Service Water Pumphouse:	90°F/90% RH

PSNH SEABROOK STATION PRIMARY AUXILIARY BUILDING

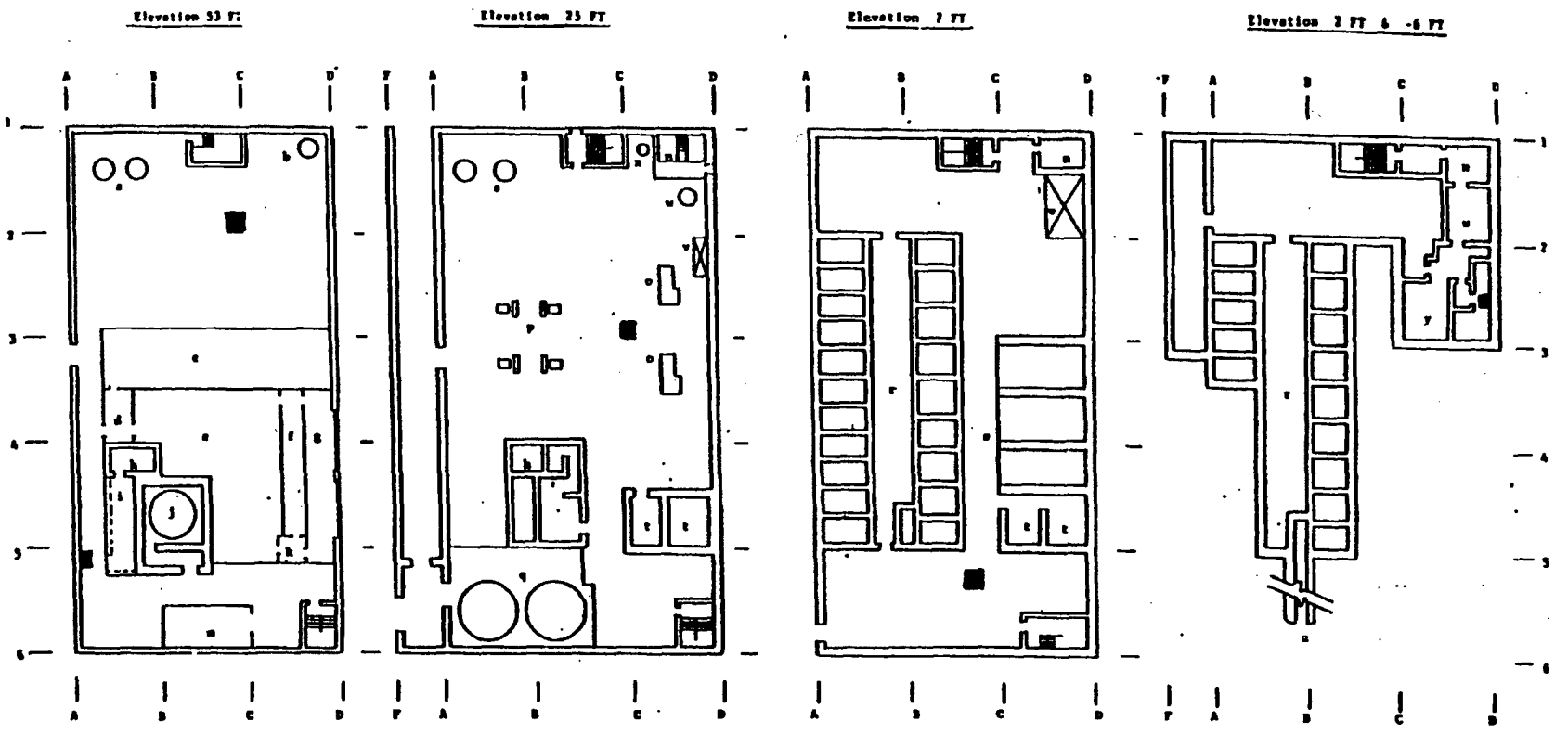


Figure 2.1-1: Primary Auxiliary Building Showing Locations of HELB Temperature Detection Thermocouples

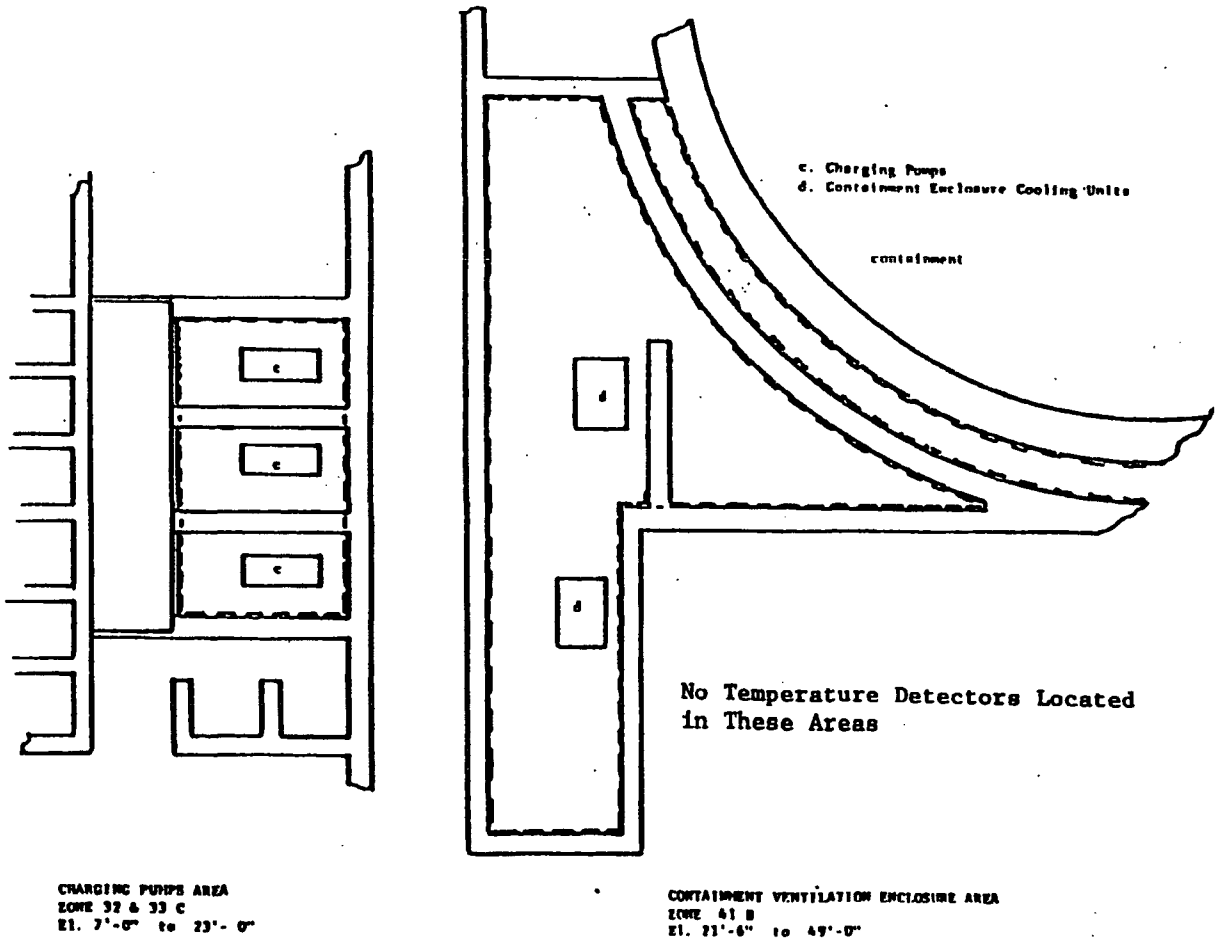
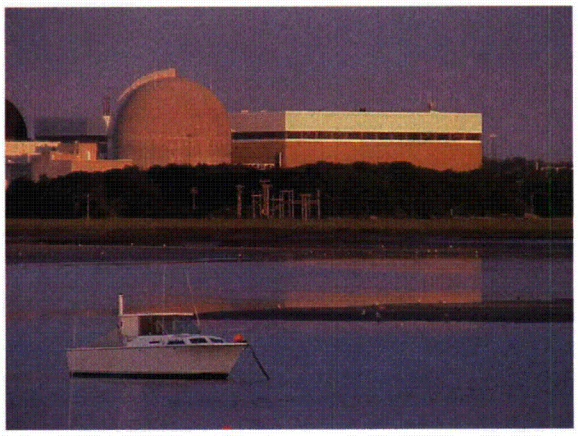


Figure 2.1-2: Containment Enclosure Area Showing Locations of HELB Temperature Detection Thermocouples

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 4
REACTOR



4.1 SUMMARY DESCRIPTION

This chapter describes: (1) the mechanical components of the reactor and reactor core including the fuel rods and fuel assemblies, (2) the nuclear design, and (3) the thermal-hydraulic design.

The reactor core comprises multiple regions of fuel assemblies which are similar in mechanical design, but different in fuel enrichment. Reload fuel is similar in mechanical design to the initial core; the differences are described in the following sections. The initial core design employed three enrichments in a three-region core, whereas more enrichments may be employed for a particular refueling scheme. Fuel cycle times of six months to over eighteen months are possible, and may be employed with the core described herein.

The core is cooled and moderated by light water at a pressure of 2250 pounds per square inch absolute (psia) in the Reactor Coolant System. The moderator coolant contains boron as a neutron poison. The concentration of boron in the coolant is varied as required to control relatively slow reactivity changes including the effects of fuel burnup. Additional boron, in the form of burnable poison rods, were employed in the initial core to establish the desired initial reactivity. Integral Fuel Burnable Absorbers (IFBA) are employed in reload fuel for this purpose. IFBAs are fuel rods in which a thin zirconium diboride coating is applied directly to the fuel pellets.

Two hundred and sixty four fuel rods are mechanically joined in a square, 17x17 array to form a fuel assembly. The fuel rods are supported at intervals along their length by grid assemblies and intermediate flow mixer (IFM) grids (for the RFA (w IFMs) design) which maintain the lateral spacing between the rods throughout the design life of the assembly. The grid assembly consists of an "egg-crate" arrangement of interlocked straps. The straps contain springs and dimples for fuel rod support as well as coolant mixing vanes. The fuel rods consist of enriched uranium dioxide ceramic cylindrical pellets contained in hermetically sealed zirconium alloy tubing. All fuel rods are pressurized with helium during fabrication to reduce stresses and strains and to increase fatigue life.

The center position in the assembly is reserved for use by the incore instrumentation, while the remaining 24 positions in the array are equipped with guide thimbles joined to the grids and the top and bottom nozzles. The guide thimbles may be used as core locations for Rod Cluster Control Assemblies (RCCAs), neutron source assemblies, or burnable poison rods. Otherwise, the guide thimbles can be fitted with plugging devices to limit bypass flow.

The bottom nozzle is a bottom structural element of the fuel assembly, and admits the coolant flow to the assembly.

The top nozzle assembly is a box-like structure which serves as the upper structural element of the fuel assembly, in addition to providing a partial protective housing for the RCCA or other components.

The RCCAs each consist of a group of individual absorber rods fastened at the top end to a common hub called a spider assembly. These assemblies contain absorber material to control the reactivity of the core, and to control axial power distribution.

The nuclear design analyses and evaluations established physical locations for control rods, burnable poison rods and physical parameters such as fuel enrichments and boron concentration in the coolant. The nuclear design evaluation established that the reactor core has inherent characteristics which, together with corrective actions of the reactor control and protective systems, provide adequate reactivity control even if the highest reactivity worth RCCA is stuck in the fully withdrawn position.

The design also provides for inherent stability against diametral and azimuthal power oscillations and for control of induced axial power oscillation through the use of control rods.

The thermal-hydraulic design analyses and evaluations establish coolant flow parameters which assure that adequate heat transfer is provided between the fuel cladding and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution and mixing. The mixing vanes incorporated in the RFA spacer grid design and the IFMs induce additional flow mixing between the various flow channels within a fuel assembly as well as between adjacent assemblies.

Instrumentation is provided in and out of the core to monitor the nuclear, thermal-hydraulic, and mechanical performance of the reactor and to provide inputs to automatic control functions.

Table 4.1-1 presents a comparison of the principal nuclear, thermal-hydraulic and mechanical design parameters between the Seabrook Station Unit 1 initial case and the W. B. McGuire Nuclear Station Units 1 and 2 (Docket Nos. 50-369 and 50-370).

The analytical techniques employed in the core design are tabulated in Table 4.1-2. The loading conditions considered in general for the core internals and components are tabulated in Table 4.1-3. Specific or limiting loads considered for design purposes of the various components are listed as follows: fuel assemblies in Subsection 4.2.1.5; neutron absorber rods, burnable poison rods, neutron source rods and thimble plug assemblies in Subsection 4.2.1.6. The dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9(N).

4.1.1 References

None

4.2 FUEL SYSTEM DESIGN

The plant design conditions are divided into four categories in accordance with their anticipated frequency of occurrence and risk to the public: Condition I - Normal Operation; Condition II - Incidents of Moderate Frequency; Condition III - Infrequent Incidents; and Condition IV - Limiting Faults. Chapter 15 describes bases and plant operation and events involving each condition.

The reactor is designed so that its components meet the following performance and safety criteria:

a. The mechanical design of the reactor core components and their physical arrangement, together with corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) ensure that:

   1. Fuel damage (defined as penetration of the fission product barrier i.e., the fuel rod clad) is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the plant cleanup system and are consistent with plant design bases.

   2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged (in any case, the fraction of fuel rods damaged must be limited to meet the dose guidelines of 10 CFR 100) although sufficient fuel damage might occur to preclude immediate resumption of operation.

   3. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from \Condition IV events.

b. The fuel assemblies are designed to withstand loads induced during shipping, handling, and core loading without exceeding the criteria of Subsection 4.2.1.5.

c. The fuel assemblies are designed to accept control rod insertions to provide the required reactivity control for power operations and reactivity shutdown conditions.

d. All fuel assemblies have provisions for the insertion of incore instrumentation necessary for plant operation.

e. The reactor internals, in conjunction with the fuel assemblies and incore control components, direct reactor coolant through the core. This achieves acceptable flow distribution and restricts bypass flow so that the heat transfer performance requirements can be met for all modes of operation.

4.2.1 Design Bases

The RFA fuel rod and fuel assembly design bases are established to satisfy the general performance and safety criteria presented in Section 4.2.

The fuel rods are designed for a peak rod burnup of approximately 60,000 megawatt days per metric ton of uranium (MWd/Mtu) in the fuel cycle equilibrium condition. Peak rod burnups as high as 62,000 MWd/Mtu can be licensed for Westinghouse fuel in individual fuel cycles using the Westinghouse Fuel Criteria Evaluation Process (References 20 and 22).

Design values for the properties of the materials which comprise the fuel rod, fuel assembly and incore control components are given in Reference 2 for Zircaloy clad in Reference 16 for ZIRLO™ clad fuel. The structural component hydrogen pickup limit has been replaced by structural component stress criterion in Reference 21. Other supplementary fuel design criteria/limits are given in Reference 20.

4.2.1.1 Cladding

a. Material and Mechanical Properties

Zircaloy-4 and ZIRLO™ combine neutron economy (low absorption cross section); high corrosion resistance to coolant, fuel, and fission products; and high strength and ductility at operating temperatures. Reference 1 documents the operating experience with Zircaloy-4 and ZIRLO™ as a clad material. Information on the material chemical and mechanical properties of the cladding is given in Reference 2 and Reference 16 with due consideration of temperature and irradiation effects.

b. Stress-Strain Limits

1. Clad Stress

The von Mises criterion is used to calculate the effective stresses. The cladding stresses under Condition I and II events are less than the Zircaloy 0.2% offset yield stress, with due consideration of temperature and irradiation effects. While the cladding has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design basis.

2. Clad Tensile Strain

The total tensile creep strain is less than 1 percent from the unirradiated condition. The elastic tensile strain during a transient is less than 1 percent from the pretransient value. These limits are consistent with proven practice.

c. Vibration and Fatigue

1. Strain Fatigue

The cumulative strain fatigue cycles are less than the design strain fatigue life. This basis is consistent with proven practice.

2. Vibration

Potential fretting wear due to vibration is prevented by design of the fuel assembly grid springs and dimples, assuring that the stress-strain limits are not exceeded during design life. Fretting of the clad surface can occur due to flow-induced vibration between the fuel rods and fuel assembly grid springs. Vibration and fretting forces vary during the fuel life due to clad diameter creepdown combined with grid spring relaxation.

d. Chemical Properties

Chemical properties of the cladding are discussed in Reference 2 for Zircaloy-4 and Reference 16 for ZIRLO™.

4.2.1.2 Fuel Material/Integral Fuel Burnable Absorber (IFBA)

a. Thermal-Physical Properties

The thermal-physical properties of UO_2 are described in Reference 2 with due consideration of temperature and irradiation effects.

Fuel pellet temperatures - The center temperature of the hottest pellet is to be below the melting temperature of the UO_2 (melting point of 5080 F (Reference 2) unirradiated and decreasing by 58°F per 10,000 MWd/Mtu). While a limited amount of center melting can be tolerated, the design conservatively precludes center melting. A calculated fuel centerline temperature of 4700 F has been selected as an overpower limit to assure no fuel melting. This provides sufficient margin for uncertainties as described in Subsection 4.4.2.9.

The normal design density of the fuel is approximately 95 percent of theoretical. Additional information on fuel properties is given in Reference 2.

b. Fuel Densification and Fission Product Swelling

The design bases and models used for fuel densification and swelling are provided in References 4 and 17.

c. Chemical Properties

References 2 and 16 provide the basis for justifying that no adverse chemical interactions occur between the fuel and adjacent cladding material.

4.2.1.3 Fuel Rod Performance

The detailed fuel rod design establishes such parameters as pellet size and density, cladding-pellet diameter gap, gas plenum size, and helium prepressurization level. The design also considers effects such as fuel density changes, fission gas release, cladding creep, and other physical properties which vary with burnup. The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive internal rod gas pressures due to fission gas releases, and excessive cladding stresses and strains. This is achieved by designing the fuel rods to satisfy the conservative design basis in the following subsections during Condition I and II events over the fuel lifetime. For each design basis, the performance of the limiting fuel rod must not exceed the limits specified.

a. Fuel Rod Models

The basic fuel rod models and the ability to predict operating characteristics are given in References 16, 17, and 23 and Subsection 4.2.3.

b. Mechanical Design Limits

Fuel rod design methodology described in Reference 18 demonstrates that clad flattening will not occur in Westinghouse fuel designs. The rod internal gas pressure will remain below the value which causes the fuel/clad diametral gap to increase due to outward cladding creep during steady state operation. The maximum rod pressure is also limited so that extensive Departure from Nucleate Boiling (DNB) propagation will not occur during normal operation or any accident event. Reference 7 shows that the DNB propagation criteria is satisfied.

4.2.1.4 Spacer Grids

a. Mechanical Limits and Materials Properties

The grid component strength criteria are based on experimental tests. The limit is established at 0.9 Pc, where Pc is the experimental collapse load. This limit is sufficient to assure that under worst-case combined seismic and blowdown loads the core will maintain a geometry amenable to cooling. As an integral part of the fuel assembly structure, the grids must satisfy the applicable fuel assembly design bases and limits defined in Subsection 4.2.1.5.

The grid material and chemical properties are given in References 2 and 16.

b. Vibration and Fatigue

The grids are designed to provide sufficient fuel rod support to limit fuel rod vibration and maintain clad fretting wear to within acceptable limits.

4.2.1.5 Fuel Assembly

a. Structural Design

As previously discussed in Subsection 4.2.1, the structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various non-operational, operational and accident loads. These limits are applied to the design and evaluation of the top and bottom nozzles, guide thimbles, grids, and the thimble joints.

The design bases for evaluating the structural integrity of the fuel assemblies are:

1. Non-operational 4g axial and 6g lateral loading with dimensional stability.
2. For the normal operating and upset conditions, the fuel assembly component structural design criteria are established for the two primary material categories, namely austenitic stainless steels and zirconium alloys. The stress categories and strength theory presented in the ASME Boiler and Pressure Vessel Code, Section III, are used as a general guide. The maximum shear-theory (Tresca criterion) for combined stresses is used to determine the stress intensities for the austenitic stainless steel components. The stress intensity is defined as the numerically largest difference between the various principal stresses in a three dimensional field. The design stress intensity, Sm, for austenitic stainless steels such as nickel-chromium-iron alloys, is given by the lowest of the following:
   (a) One-third of the specified minimum tensile strength or two-thirds of the specified minimum yield strength at room temperature
   (b) One-third of the tensile strength or 90 percent of the yield at temperature, but not to exceed two-thirds of the specified minimum yield strength at room temperature. The stress limits for the austenitic stainless steel components are given below. All stress nomenclature is per the ASME Code, Section III.

Stress Intensity Limits

Category	Limit
General Primary Membrane Stress Intensity	Sm
Local Primary Membrane Stress Intensity	1.5 Sm
Primary Membrane plus	1.5 Sm

Stress Intensity Limits

Category	Limit
Bending Stress Intensity	
Total Primary plus Secondary	3.0 Sm Stress Intensity

The zirconium alloy structural components, which consist of spacer grids, guide thimble and fuel tubes, are in turn subdivided into two categories because of material differences and functional requirements. The fuel tube design criteria are covered separately in Subsection 4.2.1.1. For the guide thimble design, the stress intensities, the design stress intensities and the stress intensity limits are calculated using the same methods as for the austenitic stainless steel structural components. For conservative purposes, the zirconium alloy unirradiated properties are used to define the stress limits.

(c) Abnormal loads during Conditions III or IV - worst cases represented by combined seismic and blowdown loads.

(1) Deflections or failures of components cannot interfere with the reactor shutdown or emergency cooling of the fuel rods.

(2) The fuel assembly structural component stresses under faulted conditions are evaluated using primarily the methods outlined in Appendix F of the ASME Code, Section III. Since the current analytical methods utilize elastic analysis, the stress allowables are defined as the smaller value of 2.4 Sm or 0.70 Su (ultimate strength per ASME nomenclature) for primary membrane and 3.6 Sm or 1.05 Su for primary membrane plus primary bending. For the austenitic steel fuel assembly components, the stress intensity is defined in accordance with the rules described in the previous section for normal operating conditions. For the zirconium alloy components the stress intensity limits are set at two-thirds of the material yield strength, Sy, at reactor operating temperature. This results in zirconium alloy stress limits being the smaller of 1.6 Sy (yield strength per ASME nomenclature) or 0.70 Su for primary membrane and 2.4 Sy or 1.05 Su for primary membrane plus bending. For conservative purposes the zirconium alloy unirradiated properties are used to define the stress limits.

The material and chemical properties of the fuel assembly components are given in References 2 and 16.

b. Thermal-Hydraulic Design

This topic is discussed in Section 4.4.

c. Reconstituted Fuel Assemblies

Those assemblies which contain zirconium alloy or stainless steel filler rods (as discussed in Subsection 4.2.2.1) will be incorporated into core loading plans as normal assemblies. These reconstituted assemblies will typically be grouped with other fuel assemblies with similar exposure histories, and the assemblies in these groups will then be placed in symmetric locations. A single reconstituted assembly may be placed in the center of the core. Appropriate core physics models will be applied to reflect the actual geometry of the reconstituted assemblies in each reload cycle. In the nuclear design analysis for each reload, reconstituted assemblies will be explicitly modeled on a pin-by-pin basis to ensure these assemblies are treated in a conservative manner.

4.2.1.6 Core Components

The core components are subdivided into permanent and temporary devices.

The permanent type components are the Rod Cluster Control Assemblies, secondary neutron source assemblies, and thimble plug assemblies. Thimble plugs may be installed if safety analysis shows the need for them. The temporary components are the burnable poison assemblies and the primary neutron source assemblies, which are normally used only in the initial core. Installation of the secondary sources is optional, provided a sufficient neutron source exists in their absence.

Materials are selected for compatibility in a pressurized water reactor environment, for adequate mechanical properties at room and operating temperature, for resistance to adverse property changes in a radioactive environment, and for compatibility with interfacing components. Materials properties are given in Reference 2.

For Conditions I and II, the stress categories and strength theory presented in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG-3000 are used as a general guide to establish core component rod cladding stress/strain limits. The code methodology is applied as with fuel assembly structure design, where possible. For Conditions III and IV, code stresses are not limiting.

Additional design bases for each of the mentioned components are given in the following subsections.

a. Control Rods

Design conditions which are considered under Article NB-3000 of the ASME Code, Section III are as follows:

1. External pressure equal to the reactor coolant system operating pressure with appropriate allowance for over-pressure transients
2. Wear allowance equivalent to 1000 reactor trips
3. Bending of the rod due to a misalignment in the guide tube
4. Forces imposed on the rods during rod drop
5. Loads imposed by the accelerations of the control rod drive mechanism
6. Radiation exposure during maximum core life

The stress intensity limit, Sm, for the control rod cladding material is defined at two-thirds of the 0.2 percent offset yield stress.

The absorber material temperature shall not exceed its melting temperature which is 1454°F for Ag-In-Cd absorber material, Reference 8. (The melting point basis is determined by the nominal material melting point minus uncertainty.)

b. Burnable Poison Rods

Failures of burnable poison rods during Conditions I through IV events will not interfere with reactor shutdown or cooling of the fuel rods.

The burnable poison absorber material is nonstructural. The structural elements of the burnable poison rod are designed to maintain the absorber geometry even if the absorber material is fractured. The rods are designed so that the absorber material is below its softening temperature which is 1492°F for Reference 12.5 weight percent boron rods. The absorber material used in burnable poison rods is Borosilicate glass. The softening temperature, as defined in ASTM C338-73, is 720°C.) In addition, the structural elements are designed to prevent excessive slumping.

c. Neutron Source Rods

The neutron source rods are designed to withstand the following:

1. The external pressure equal to the reactor coolant system operating pressure with appropriate allowance for over-pressure transients, and
2. An internal pressure equal to the pressure generated by released gases over the source rod life.

d. Thimble Plug Assembly

The thimble plug assembly may be used to restrict bypass flow through those thimbles not occupied by absorber, source or burnable poison rods.

The thimble plug assemblies satisfy the following criteria:

1. Accommodate the differential thermal expansion between the fuel assembly and the core internals
2. Limit the flow through each occupied thimble

4.2.1.7 Surveillance Program

Subsection 4.2.4.5 and Sections 8 and 23 of Reference 9 discuss the testing and fuel surveillance operational experience program that has, and is, being conducted to verify the adequacy of the fuel performance and design bases. An evaluation of the test program for the IFBA design features is given in Section 2.5 of Reference 14. Fuel surveillance and testing results, as they become available, are used to improve fuel rod design and manufacturing processes and assure that the design bases and safety criteria are satisfied.

4.2.2 Design Description

Each standard fuel assembly consists of 264 fuel rods, 24-guide thimble tubes and one instrumentation thimble tube arranged within a supporting structure.

The instrumentation thimble is located in the center position and provides a channel for insertion of an incore neutron detector, if the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of either a Rod Cluster Control Assembly, a neutron source assembly, a burnable poison assembly or a thimble plug assembly. Figure 4.2-1 shows a cross section of the fuel assembly array, and Figure 4.2-2A and Figure 4.2-2B show a fuel assembly full length view. The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles.

Fuel assemblies are installed vertically in the reactor vessel and stand upright on the lower core plate, which is fitted with alignment pins to locate and orient each assembly. After all fuel assemblies are set in place. The upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the top nozzle of each fuel assembly. The upper core plate then bears downward against the holddown springs on the top nozzle of each fuel assembly to hold the fuel assemblies in place.

Improper orientation of fuel assemblies within the core is prevented by the use of an indexing hole in one corner of the top nozzle top plate (see Figure 4.2-2A and Figure 4.2-2B). The assembly is oriented with respect to the handling tool and the core by means of a pin which is inserted into this indexing hole. Visual confirmation of proper orientation is also provided by an identification number on the opposite corner clamp.

4.2.2.1 Fuel Rods

The fuel rods consist of fuel pellets contained in hermetically sealed zirconium alloy tubing. The fuel pellets are right circular cylinders consisting of slightly enriched ceramic uranium dioxide which has been sintered to approximately 95% of theoretical density. Some of the pellets may be coated with a thin layer of zirconium di-boride for local reactivity control. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used.

Void volume and clearances are provided within the rods to accommodate gases which are released from the fuel pellets during irradiation, differential thermal expansion between the clad and the fuel, and fuel density changes during irradiation. The ends of the fuel pellets may be dished to allow for greater axial expansion at the pellet centerline, and contribute to the void volume available for accommodation of gases. Shifting of the fuel within the clad during handling or shipping prior to core loading is prevented by a spring which bears on top of the fuel.

Some fuel rods may contain annular axial blankets at the top and bottom of the fuel stack. The blankets contain mid-enriched fuel pellets with an annulus through the center and no dish on the ends of the pellet. Mid-enriched annular axial blanket pellets reduce neutron leakage, improve fuel utilization and provide additional void volume to accommodate fission gas release.

With respect to prepressurization, the rods are designed so that (1) the internal gas pressure mechanical design limits given in Subsection 4.2.1.3 are not exceeded, (2) the cladding stress-strain limits (see Subsection 4.2.1.1) are not exceeded for Conditions I and II events, and (3) clad flattening will not occur during the fuel core life.

4.2.2.2 Fuel Assembly Structure

The fuel assembly structure consists of a bottom nozzle, top nozzle, guide thimbles and grids, as shown in Figure 4.2-2A and Figure 4.2-2B.

a. Bottom Nozzle

The bottom nozzle serves as the bottom structural element of the fuel assembly and admits the coolant flow to the assembly. It is fabricated from austenitic stainless steel, and consists of a perforated plate and four angle legs with bearing plates as shown in Figure 4.2-2A and Figure 4.2-2B. The legs form a plenum for the inlet coolant flow to the fuel assembly. The plate also prevents downward ejection of the fuel rods from the fuel assembly. The bottom nozzle is fastened to the fuel assembly guide tubes by screws which penetrate the nozzle and engage threaded plugs in the guide thimbles.

Axial loads (holddown) imposed on the fuel assembly and the weight of the fuel assembly are transmitted through the bottom nozzle to the lower core plate. Indexing and positioning of the fuel assembly are provided by alignment holes in two diagonally opposite bearing plates which mate with locating pins in the lower core plate. Lateral loads on the fuel assembly are transmitted to the lower core plate through the locating pins.

b. Top Nozzle

The top nozzle assembly functions as the upper structural element of the fuel assembly. The top nozzle assembly consists of a box-like structure with holddown springs mounted as shown in Figure 4.2-2A and Figure 4.2-2B. The springs and bolts are made of Inconel, whereas other components are made of austenitic stainless steel.

The square adapter plate is provided with openings to permit the flow of coolant upward through the top nozzle. Other holes are provided to accept the thimble tubes. The ligaments in the plate cover the tops of the fuel rods and prevent their upward ejection from the fuel assembly. The enclosure is a box-like structure which sets the distance between the adapter plate and the top plate. The top plate has a large square hole in the center to permit access for the control rod assembly or other components. Holddown springs are mounted on the top plate and are fastened in place by bolts and clamps located at two diagonally opposite corners. On the other two corners, pads are positioned which contain alignment holes for locating the upper end of the fuel assembly.

c. Guide and Instrument Thimbles

The guide thimbles are structural members which also provide channels for the neutron absorber rods, burnable poison rods, neutron source or thimble plug assemblies. Each thimble is fabricated from zirconium alloy tubing having two different diameters. The tube diameter at the top section provides the annular area necessary to permit rapid control rod insertion during a reactor trip. The lower portion of the guide thimble is of a smaller diameter to reduce diametral clearances and produce a dashpot action near the end of the control rod travel. Holes are provided in the thimble tube above the dashpot to reduce the rod drop time. The dashpot is closed at the bottom by means of an end plug which is provided with a small flow port to avoid fluid stagnation. The top end of the guide thimble is fastened to a tubular sleeve by expansion swages. The sleeve fits into and is fastened to the top nozzle adapter plate. The lower end of the guide thimble is fitted with an end plug which is then fastened to the bottom nozzle by a screw.

Fuel rod support grids are fastened to the guide thimble assemblies to create an integrated structure. A mechanical fastening technique depicted in Figure 4.2-4 and Figure 4.2-5 is used for all but the bottom grids in a fuel assembly.

An expanding tool is inserted into the thimble tube at the elevation of the sleeves that have been attached to the grid assemblies. The four-lobed tool forces the thimble and sleeve outward to a predetermined diameter, thus joining the two components.

The top grid to thimble attachment for the initial core is shown in Figure 4.2-6A. The stainless steel sleeves are brazed into the Inconel grid assembly. The zirconium alloy guide thimbles are fastened to the sleeves by expanding the two members as shown in Figure 4.2-4 and Figure 4.2-5. Finally, the top ends of the sleeves are attached to the top nozzle adapter plate as shown in Figure 4.2-6A and Figure 4.2-6B.

The bottom grid assembly is joined to the skeleton assembly as shown in Figure 4.2-7. The stainless steel insert is attached to the bottom grid and later captured between the guide thimble end plug and the bottom nozzle by a screw fastener.

The described methods of grid fastening are standard and have been used successfully since the introduction of Zircaloy guide thimbles in 1969.

The central instrumentation thimble of each fuel assembly is constrained by seating in counterbores in each nozzle. This tube is a constant diameter and guides the incore detector assembly. It is expanded at the top and mid-grids in the same manner as the previously discussed expansion of the guide thimbles to the grids.

d. Grid Assemblies

The fuel rods, as shown in Figure 4.2-2A and Figure 4.2-2B, are supported at intervals along their length by grid assemblies which maintain the lateral spacing between the rods. Each fuel rod is supported within each grid by the combination of support dimples and springs.

The grid assembly consists of individual slotted straps interlocked and brazed or welded in an "egg-crate" arrangement to join the straps permanently at their points of intersection. The straps contain springs, support dimples and mixing vanes.

The grid material is Inconel or zirconium alloy, chosen because of its corrosion resistance and strength. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

Nine grids, with mixing vanes projecting from the edges of the straps into the coolant stream, are used in the high heat flux region of the fuel assemblies to promote mixing of the coolant. The top, bottom, and protective grids do not contain mixing vanes on the internal straps. The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core.

4.2.2.3 Core Components

Reactivity control is provided by neutron absorbing rods and a soluble chemical neutron absorber (boric acid). The boric acid concentration is varied to control long-term reactivity changes such as:

a. Fuel depletion and fission product buildup

b. Cold to hot, zero power reactivity change

c. Reactivity change produced by intermediate term fission products such as xenon and samarium

d. Burnable poison depletion.

The Chemical and Volume Control System is discussed in Chapter 9.

The Rod Cluster Control Assemblies (RCCAs) provide reactivity control for:

a. Shutdown

b. Reactivity changes resulting from coolant temperature changes in the power range

c. Reactivity changes associated with the power coefficient of reactivity

d. Reactivity changes resulting from void formation.

Figure 4.2-8 illustrates the RCCA and control rod drive mechanism assembly, in addition to the arrangement of these components in the reactor relative to the interfacing fuel assembly and guide tubes. In the following paragraphs, each reactivity control component is described in detail. The control rod drive mechanism assembly is described in Subsection 3.9(N).4.

The neutron source assemblies provide a means of monitoring the core during periods of low neutron level. The thimble plug assemblies may be used to limit bypass flow through those fuel assembly thimbles which do not contain control rods, burnable poison rods, or neutron source rods.

a. (RCCA)

The RCCAs are divided into two categories: control and shutdown. The control groups compensate for reactivity changes associated with variations in operating conditions of the reactor, i.e., power and temperature variations. Two nuclear design criteria have been employed for selection of the control group. First the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to ensure that the power capability is met. Additional shutdown banks are provided which, together with the control banks A, B, C and D, supply reactivity insertion to cover the power defect, plus (a) transient cooldowns below the hot zero power critical state, (b) an NRC requirement for a minimum of 1 percent hot standby shutdown reactivity, (c) the worth of any full length control rod stuck out of the core, and (d) a margin for uncertainty in rod worth and reactivity change calculations. The control and shutdown groups together provide adequate shutdown margin.

The Ag-In-Cd Rod Cluster Control Assembly comprises 24 neutron absorber rods fastened at the top end to a common spider assembly, as illustrated in Figure 4.2-9A and Figure 4.2-9B.

The absorber material used in the control rods is a silver-indium-cadmium alloy which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The Ag-In-Cd absorber rod is illustrated in Figure 4.2-10.

The bottom plugs are tapered to reduce the hydraulic drag during reactor trip and to guide the absorber rods smoothly into the dashpot section of the fuel assembly guide thimbles.

The allowable stresses used as a function of temperature are listed in Table 1.1-2 of Section III of the ASME Code. The fatigue strength is based on the S-N curve for austenitic stainless steels in Figure 1.9-2 of Section III.

The spider assembly is in the form of a central hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive rod assembly are machined into the upper end of the hub. A coil spring inside the spider body absorbs the impact energy at the end of a trip insertion. A center-post which holds the spring and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from austenitic stainless steel or other corrosion-resistant material such as Inconel.

The absorber rods are fastened securely to the spider. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small misalignments.

The overall length is such that when the assembly is withdrawn through its full travel, the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble.

b. Burnable Poison Assembly

Each burnable poison assembly consists of burnable poison rods attached to a holddown assembly. A burnable poison assembly is shown in Figure 4.2-11. When needed for nuclear considerations, burnable poison assemblies are inserted into selected thimbles within fuel assemblies.

The poison rods consist of borosilicate glass tubes contained within austenitic stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall tubular inner liner. The top end of the liner is open to permit the diffused helium to pass into the void volume, and the liner overhangs the glass. The liner has an outward flange at the bottom end to maintain the position of the liner with the glass. A typical burnable poison rod is shown in longitudinal and transverse cross sections in Figure 4.2-12.

The poison rods in each fuel assembly are grouped and attached together at the top end of the rods to a holddown assembly by a flat perforated retaining plate which fits within the fuel assembly top nozzle and rests on the adaptor plate. The retaining plate and poison rods are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals assembly is lowered into the reactor. This arrangement ensures that the poison rods cannot be ejected from the core by flow forces. Each rod is permanently attached to the base plate by a nut which is lock-welded into place.

The cladding of the burnable poison rods and all other structural materials are austenitic stainless steel except for the springs which are Inconel. The borosilicate glass tube provides sufficient boron content to meet the criteria discussed in Subsection 4.3.1.

c. Neutron-Source Assembly

The purpose of the neutron source assembly is to provide base neutron level to ensure that the neutron detectors are operational and responding to core multiplication neutrons. A neutron source is placed in the reactor to provide a positive neutron count on the source range detectors attributable to core neutrons. The detectors, called source range detectors, are used primarily when the core is subcritical and during special subcritical modes of operations.

The source assembly permits detection of changes in the core multiplication factor during core loading and approach to criticality.

This can be done since the multiplication factor is related to an inverse function of the detector count rate. Changes in the multiplication factor can be detected during addition of fuel assemblies while loading the core, changes in control rod positions, and changes in boron concentration.

Both primary and secondary neutron source rods are used in the initial core. Subsequent cycles do not require a primary source. The primary source rod, containing a radioactive material, spontaneously emits neutrons during initial core loading, reactor startup and initial operation of the first core. After the primary source rod decays beyond the desired neutron flux level, neutrons are then supplied by the secondary source rod. The secondary source rod contains a stable material which is activated during reactor operation. The activation results in the subsequent release of neutrons.

Four source assemblies were installed in the initial reactor core: two primary source assemblies and two secondary source assemblies. Each primary source assembly contains one primary source rod and a number of burnable poison rods. Each secondary source assembly contains a symmetrical grouping of four secondary source rods. Locations not filled with a source rod or burnable poison rod contain a thimble plug rodlet. Two additional secondary source assemblies may be incorporated for future cycles. The source assemblies are shown in Figure 4.2-13, Figure 4.2-14A and Figure 4.2-14B.

The source assemblies are inserted into the rod cluster control guide thimbles in fuel assemblies at selected unrodded locations.

As shown in Figure 4.2-13 and Figure 4.2-14, the source assemblies contain a holddown assembly identical to that of the burnable poison assembly. The additional secondary sources contain a single holddown spring with similar holddown characteristics to that of the original sources.

The primary and secondary source rods have the same cladding as the absorber rods. The secondary source rods contain Sb-Be pellets stacked to a height of approximately 88 inches. The primary source rods contain capsules of californium source material and alumina spacer to position the source material within the cladding. The rods in each assembly are permanently fastened at the top end to a holddown assembly.

The other structural members such as the spider head and vanes are constructed of austenitic stainless steel or Inconel.

d. Thimble Plug Assembly

Thimble plug assemblies may be used to limit bypass flow through the rod cluster control guide thimbles in fuel assemblies which do not contain control rods, source rods, or burnable poison rods.

The thimble plug assemblies consist of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly as shown in Figure 4.2-15. The 24 short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow. Each thimble plug is permanently attached to the base plate. Similar short rods are also used on the source assemblies and burnable poison assemblies to plug the ends of all vacant fuel assembly guide thimbles. When in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly guide thimbles tubes. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place.

All components in the thimble plug assembly, except for the springs, are constructed from austenitic stainless steel or Inconel.

4.2.3 Design Evaluation

The fuel assemblies, fuel rods and incore control components are designed to satisfy the performance and safety criteria of the introduction to Section 4.2, the mechanical design bases of Subsection 4.2.1, and other interfacing nuclear and thermal hydraulic design bases specified in Sections 4.3 and 4.4. Effects of Conditions II, III, IV or Anticipated Transients without Trip on fuel integrity are presented in Chapter 15 or supporting topical reports.

The initial step in fuel rod design evaluation for a region of fuel is to determine the limiting rod(s). Limiting rods are defined as those rod(s) whose predicted performance provides the minimum margin to each of the design criteria. For a number of design criteria, the limiting rod is the lead burnup rod of a fuel region. In other instances it may be the maximum power or the minimum burnup rod. For the most part, no single rod will be limiting with respect to all design criteria.

After identifying the limiting rod(s), a worst-case performance evaluation is made which uses the limiting rod design basis power history and considers the effects of model uncertainties and dimensional variations. Furthermore, to verify adherence to the design criteria, the conservative case evaluation also considers the effects of postulated transient power increases which are achievable during operation consistent with Conditions I and II events. These transient power increases can affect both rod and local power levels. The analytical methods used in the evaluation result in performance parameters which demonstrate the fuel rod behavior. Examples of parameters considered include rod internal pressure, fuel temperature, clad stress, and clad strain. In fuel rod design analyses, these performance parameters provide the basis for comparison between expected fuel rod behavior and the corresponding design criteria limits.

Fuel rod and fuel assembly models used for the various evaluations are documented and maintained under an appropriate control system. Properties of materials used in the design evaluations are given in References 2 and 16.

4.2.3.1 Cladding

a. Vibration and Wear

Fuel rod vibrations are flow induced. The effect of the vibration on the fuel assembly and individual fuel rods is minimal. The cyclic stress range associated with deflections of such small magnitude is insignificant and has no effect on the structural integrity of the fuel rod.

The reaction force on the grid supports due to rod vibration motions is also small and is much less than the spring preload. Firm fuel clad spring contact is maintained. No significant wear of the clad or grid supports is expected during the life of the fuel assembly, based on out-of-pile flow tests performance of similarly designed fuel in operating reactors, and design analysis.

Clad fretting and fuel vibration has been experimentally investigated as shown in Reference 10.

b. Fuel Rod Internal Pressure and Cladding Stresses

The burnup dependent fission gas release model (References 17 and 23) is used in determining the internal gas pressures as a function of irradiation time. The plenum volume of the fuel rod has been established to ensure that the maximum internal pressure of the fuel rod will not exceed the value which would cause (1) the fuel/clad diametral gap to increase during steady state operation and (2) extensive DNB propagation to occur (see Subsection 4.2.1.3b). The clad stresses at a constant local fuel rod power are low. Compressive stresses are created by the pressure differential between the coolant pressure and the rod internal gas pressure.

Stresses due to the temperature gradient are not included in the average effective stress because thermal stresses are, in general, negative at the clad inside diameter and positive at the clad outside diameter and their contribution to the clad volume average stress is small. Furthermore, the thermal stress decreases with time during steady state operation due to stress relaxation. The stress due to pressure differential is highest in the minimum power rod at the beginning-of-life due to low internal gas pressure. The thermal stress is highest in the maximum power rod due to steep temperature gradient.

Tensile stresses could be created once the clad has come in contact with the pellet. These stresses would be induced by the fuel pellet swelling during irradiation. Fuel swelling can result in small clad strains (< 1 percent) for expected discharge burnups but the associated clad stresses are very low because of clad creep (thermal and irradiation-induced creep). The 1 percent strain criterion is extremely conservative for fuel-swelling driven clad strain because the strain rate associated with solid fission products swelling is very slow. A detailed discussion on fuel rod performance is given in Subsection 4.2.3.3.

c. Materials and Chemical Evaluation

Zircaloy-4 and ZIRLO™ clad has a high corrosion resistance to the coolant, fuel and fission products. As shown in Reference 1, there is pressurized water reactor operating experience on the capability of Zircaloy-4 and ZIRLO™ as a clad material. Controls on fuel fabrication specify maximum moisture levels to preclude clad hydriding.

Metallographic examination of irradiated commercial fuel rods have shown occurrences of fuel/clad chemical interaction. Reaction layers of 1 mil in thickness have been observed between fuel and clad at limited points around the circumference. Metallographic data indicate that this interface layer remains very thin even at high burnup. Thus, there is no indication of propagation of the later and eventual clad penetration.

Stress corrosion cracking is another postulated phenomenon related to fuel/clad chemical interaction. Out-of-pile tests have shown that in the presence of high clad tensile stresses, large concentrations of selected fission products (such as iodine) can chemically attack the tubing and can lead to eventual clad cracking. Extensive post-irradiation examination has produced no inpile evidence that this mechanism is operative in Westinghouse produced commercial fuel.

d. Rod Bowing

Reference 11 presents the model used for evaluation of fuel rod bowing. The effects of rod bowing or DNBR are described in Subsection 4.4.2.2e. Also refer to item e in Section 4.2.

e. Consequences of Power-Coolant Mismatch

This subject is discussed in Chapter 15.

f. Creep Collapse and Creepdown

This subject and the associated irradiation stability of cladding have been evaluated using the models described in References 6 and 18. It has been established that the design basis of no clad collapse during planned core life can be satisfied by limiting fuel densification and by having a sufficiently high initial internal rod pressure.

g. Irradiation Stability of the Cladding

As shown in Reference 1, there is PWR operating experience on the capability of Zircaloy and ZIRLO™ as a cladding material. Extensive experience with irradiated Zircaloy-4 is summarized in Reference 2 and Reference 16 for ZIRLO™.

h. Cycling and Fatigue

A comprehensive review of the available strain fatigue models was conducted by Westinghouse as early as 1968. This review included the Langer-O'Donnell model (Reference 12), the Yao-Munse model and the Manson-Halford model. Upon completion of this review and using the results of the Westinghouse experimental programs discussed below, it was concluded that the approach defined by Langer-O'Donnell would be retained and the empirical factors of their correlation modified in order to conservatively bound the results of the Westinghouse testing program.

The Westinghouse testing program was subdivided into the following subprograms:

1. A rotating bend fatigue experiment on unirradiated Zircaloy-4 specimens at room temperature and at 725F. Both hydrided and non-hydrided Zircaloy-4 cladding were tested.
2. A biaxial fatigue experiment in gas autoclave on unirradiated Zircaloy-4 cladding, both hydrided and non-hydrided.
3. A fatigue test program on irradiated cladding from the CVS and Yankee Core V conducted at Battelle Memorial institute.

The results of these test programs provided information on different cladding conditions including the effects of irradiation, of hydrogen levels and of temperature.

The design equations followed the concept for the fatigue design criterion according to the ASME Boiler and Pressure Vessel Code, Section III.

it is recognized that a possible limitation to the satisfactory behavior of the fuel rods in a reactor which is subjected to daily load follow is the failure of the cladding by low cycle strain fatigue. During their normal residence time in reactor, the fuel rods may be subjected to ~1000 cycles with typical changes in power level from 50% to 100% of their steady-state values.

The assessment of the fatigue life of the fuel rod cladding is subject to a considerable uncertainty due to the difficulty of evaluating the strain range which results from the cyclic interaction of the fuel pellets and cladding. This difficulty arises, for example, from such high unpredictable phenomena as pellet cracking, fragmentation, and relocation. Since early 1968, this particular phenomenon has been investigated analytically and experimentally.

Strain fatigue tests on irradiated and non-irradiated hydrided Zr-4 claddings were performed, which permitted a definition of a conservative fatigue life limit and recommendation on a methodology to treat the strain fatigue evaluation of Westinghouse reference fuel rod designs.

It is believed that the final proof of the adequacy of a given fuel rod design to meet the load follow requirements can only come from incore experiments performed on actual reactors. Experience in load follow operation dates back to early 1970 with the load follow operation of the Saxton reactor. Successful load follow operation has been performed on reactor A (>400 load follow cycles) and reactor B (>500 load follow cycles). In both cases, there was no significant coolant activity increase that could be associated with the load follow mode of operation.

4.2.3.2 Fuel Materials Considerations

Sintered, high density uranium dioxide fuel reacts only slightly with the clad at core operating temperatures and pressures. In the event of clad defects, the high resistance of uranium dioxide to attack by water protects against fuel deterioration although limited fuel erosion can occur. As has been shown by operating experience and extensive experimental work, the thermal design parameters conservatively account for changes in the thermal performance of the fuel elements due to pellet fracture which may occur during power operation. The consequences of defects in the clad are greatly reduced by the ability of uranium dioxide to retain fission products, including those which are gaseous or highly volatile. Observations from several operating Westinghouse-supplied pressurized water reactors (Reference 9) have shown that fuel pellets can densify under irradiation to a density higher than the manufactured values. Fuel densification and subsequent settling of the fuel pellets can result in local and distributed gaps in the fuel rods. Fuel densification has been minimized by improvements in the fuel manufacturing process and by specifying a nominal 95 percent initial fuel density.

The evaluation of fuel densification effects and its consideration in fuel design are described in References 17 and 23. The treatment of fuel swelling and fission gas release are described in References 17 and 23.

The effects of waterlogging on fuel behavior are discussed in Subsection 4.2.3.3.

4.2.3.3 Fuel Rod Performance

In the calculation of the steady state performance of a nuclear fuel rod, the following interacting factors must be considered:

a. Clad creep and elastic deflection

b. Pellet density changes, thermal expansion, gas release, and thermal properties as a function of temperature and fuel burnup

c. Internal pressure as a function of fission gas release, rod geometry, and temperature distribution.

These effects are evaluated using a fuel rod design model (References 17 and 23). The model modifications for time dependent fuel densification are given in References 17 and 23. With the above interacting factors considered, the model determines the fuel rod performance characteristics for a given rod geometry, power history, and axial power shape. In particular, internal gas pressure, fuel and clad temperatures, and clad deflections are calculated. The fuel rod is divided into several axial sections and radially into a number of annular zones. Fuel density changes are calculated separately for each segment. The effects are integrated to obtain the internal rod pressure.

The initial rod internal pressure is selected to delay fuel/clad mechanical interaction and to avoid the potential for flattened rod formation. It is limited, however, by the design criteria for the rod internal pressure (see Subsection 4.2.1.3).

The gap conductance between the pellet surface and the clad inner diameter is calculated as a function of the composition, temperature, and pressure of the gas mixture, and the gap size or contact pressure between clad and pellet. After computing the fuel temperature for each pellet annular zone, the fractional fission gas release is assessed using an empirical model derived from experimental data (References 17 and 23). The total amount of gas released is based on the average fractional release within each axial and radial zone and the gas generation rate which in turn is a function of burnup. Finally, the gas released is summed over all zones and the pressure is calculated.

The code shows good agreement with a variety of published and proprietary data on fission gas release, fuel temperatures and clad deflections (References 17 and 23). These data include variations in power, time, fuel density, and geometry.

a. Fuel/Cladding Mechanical Interaction

One factor in fuel element duty is potential mechanical interaction of fuel and clad. This fuel/clad interaction produces cyclic stresses and strains in the clad, and these in turn consume clad fatigue life. The reduction of fuel/clad interaction is therefore a goal of design. The technology of using prepressurized fuel rods has been developed to further this objective.

The gap between the fuel and clad is sufficient to prevent hard contact between the two. However, during power operation, a gradual compressive creep of the clad onto the fuel pellet occurs due to the external pressure exerted on the rod by the coolant. Clad compressive creep eventually results in the fuel/clad contact. Once fuel/clad contact occurs, changes in power level result in changes in clad stresses and strains. By using prepressurized fuel rods to partially offset the effect of the coolant external pressure, the rate of clad creep toward the surface of the fuel is reduced. Fuel rod prepressurization delays the time at which fuel/clad contact occurs and hence significantly reduces the extent of cyclic stresses and strains experienced by the clad both before and after fuel/clad contact. These factors result in an increase in the fatigue life margin of the clad and lead to greater clad reliability. If gaps should form in the fuel stacks, clad flattening will be prevented by the rod prepressurization so that the flattening time will be greater than the fuel life time.

A two-dimensional (r,θ) finite element model has been developed to investigate the effects of radial pellet cracks on stress concentrations in the clad. Stress concentration is defined here as the difference between the maximum clad stress in the θ direction and the mean clad stress. The first case has the fuel and clad in mechanical equilibrium, and as a result the stress in the clad is close to zero. In subsequent cases, the pellet power is increased in steps and the resultant fuel thermal expansion imposes tensile stress in the clad. In addition to uniform clad stresses, stress concentrations develop in the clad adjacent to radial cracks in the pellet. These radial cracks have a tendency to open during a power increase but the frictional forces between fuel and clad oppose the opening of these cracks and result in localized increases in clad stress. As the power is further increased, large tensile stresses exceed the ultimate tensile strength of UO2, and additional cracks develop in the fuel thus limiting the magnitude of the stress concentration in the clad.

As part of the standard fuel rod design analysis, the maximum stress concentration evaluated from finite element calculations is added to the volume averaged effective stress in the clad as determined from one-dimensional stress/strain calculations. The resultant clad stress is then compared to the temperature-dependent yield strength to assure that the stress/strain criteria are satisfied.

1. Transient Evaluation Method

Pellet thermal expansion due to power increases is considered the only mechanism by which significant stresses and strains can be imposed on the clad. Such increases are a consequence of fuel shuffling (e.g., Region 3 positioned near the center of the core for Cycle 2 operation after operating near the periphery during Cycle 1), reactor power escalation following extended reduced power operation, and full length control rod movement. In the mechanical design model, lead rod burnup values are obtained using best estimate power histories, as determined by core physics calculations. During burnup, the amount of diametral gap closure is evaluated based upon the pellet expansion cracking model, clad creep model, and fuel swelling model. At various times during the depletion, the power is increased locally on the rod to the burnup dependent attainable power density as determined by core physics calculations. The radial, tangential and axial clad stresses -resulting from the power increase are combined into a volume average effective clad stress.

The Von Mises criterion is used to determine if the clad yield-strength has been exceeded. This criterion states that an isotropic material in multi-axial stress will begin to yield plastically when the effective stress exceeds the yield strength as determined by an axial tensile test. The yield strength correlation is for irradiated cladding since fuel/clad interaction occurs at high burnup. Furthermore, the effective stress is increased by an allowance, which accounts for stress concentrations in the clad adjacent to radial cracks in the pellet, prior to the comparison with the yield stress. This allowance was evaluated using a two-dimensional (r,θ) finite element model.

Slow transient power increases can result in large clad strains without exceeding the clad yield strength because of clad creep and stress relaxation. Therefore, in addition to the yield strength criterion, a criterion on allowable clad strain is necessary. Based upon high strain rate burst and tensile test data on irradiated tubing, 1 percent strain was determined to be a conservative lower limit on irradiated clad deformation and was thus adopted as a design criterion.

A comprehensive review of the available strain fatigue models was conducted by Westinghouse as early as 1968. This included the Langer-O'Donnell model (Reference 12), the Yao-Munse model, and the Manson Halford model. Upon completion of this review and using the results of the Westinghouse experimental programs discussed below, it was concluded that the approach defined by Langer-O'Donnell would be retained and the empirical factors of their correlation modified in order to conservatively bound the results of the Westinghouse testing program.

The Langer-O'Donnell empirical correlation has the following form:

$$S_a = \frac{E}{4\sqrt{N_f}} \ln\left(\frac{100}{100 - RA}\right) + S_e$$

where:

S_a = 1/2 E $\Delta\varepsilon_t$ = pseudo-stress amplitude which causes failure in Nf cycles (lb./in.2)

$\Delta\varepsilon_t$ = total strain range (in./in.)

E = Young's Modulus (lb./in.2)

N_f = number of cycles to failure

RA = reduction in area at fracture in a uniaxial tensile test (percent)

S_e = endurance limit (lb/in^2)

Both RA and S_e are empirical constants which depend on the type of material, the temperature and irradiation. The Westinghouse testing program was subdivided into the following subprograms:

(a) A rotating bend fatigue experiment on unirradiated Zircaloy-4 specimens at room temperature and at 725°F. Both hydrided and nonhydrided Zircaloy-4 cladding were tested.

(b) A biaxial fatigue experiment in gas autoclave on unirradiated Zircaloy-4 cladding, both hydrided and nonhydrided

(c) A fatigue test program on irradiated cladding from the Carolina-Virginia Tube Reactor and Yankee Core V conducted at Battelle Memorial Institute.

The results of these test programs provided information on different cladding conditions including the effect of irradiation, of hydrogen level, and temperature.

The design equations followed the concept for the fatigue design criterion according to the ASME Code, Section III, namely:

(a) The calculated pseudo-stress amplitude (S_a) has to be multiplied by a factor of 2 in order to obtain the allowable number of cycles (Nf).

(b) The allowable number of cycles for a given S_a is 5 percent of N, maintaining a safety factor of 20 on cycles.

The lesser of the two allowable number of cycles is selected. The cumulative fatigue life fraction is then computed as:

$$\sum_{1}^{k} \frac{n_k}{N_{fk}} \leq 1$$

where:

n_k = number of diurnal cycles of mode k

N_{fk} = number of allowable cycles

It is recognized that a possible limitation to the satisfactory behavior of the fuel rods in a reactor which is subjected to daily load follow is the failure of the clad by low cycle strain fatigue. During their normal residence time in reactor, the fuel rods may be subjected to approximately 1000 cycles with typical changes in power level from 50 to 100 percent of their steady state values.

The assessment of the fatigue life of the fuel rod clad is subject to a considerable uncertainty due to the difficulty of evaluating the strain range which results from the cyclic interaction of the fuel pellets and clad. This difficulty arises, for example, from such highly unpredictable phenomena as pellet cracking, fragmentation, and relocation. Nevertheless, since early 1968, this particular phenomenon has been investigated analytically and experimentally (Reference 12). Strain fatigue tests on irradiated and nonirradiated hydrided Zircaloy-4 claddings were performed which permitted a definition of a conservative fatigue life limit and recommendation on a methodology to treat the strain fatigue evaluation of the Westinghouse reference fuel rod designs.

It is believed that the final proof of the adequacy of a given fuel rod design to meet the load follow requirements can only come from incore experiments performed on actual reactors. Experience in load follow operation dates back to early 1970 with the load follow operation of the Saxton reactor. Successful load follow operation has been performed on reactor A (approximately 400 load follow cycles) and reactor B (approximately 500 load follow cycles). In both cases, there was no significant coolant activity increase that could be associated with the load follow mode of operation.

b. Irradiation Experience

Westinghouse fuel operational experience is presented in Reference 1. Additional test assembly and test rod experience are given in Sections 8 and 23 of Reference 9.

c. Fuel and Cladding Temperature

The methods used for evaluation of fuel rod temperatures are presented in Subsection 4.4.2.11.

d. Waterlogging

Local cladding deformations typical for waterlogging bursts have never been observed in commercial Westinghouse-supplied fuel. (Waterlogging damage of a previously defected fuel rod has occasionally been postulated as a mechanism for subsequent rupture of the cladding. Such damage has been postulated as a consequence of a power increase on a rod after water has entered such a rod through a clad defect of appropriate size. Rupture is postulated upon power increase if the rod internal pressure increase is excessive due to insufficient venting of water to the reactor coolant.) Experience has shown that the small number of rods which have acquired clad defects, regardless of primary mechanism, remain intact and do not progressively distort or restrict coolant flow. In fact such small defects are normally observed, through reductions in coolant activity, to be progressively closed upon further operation due to the buildup of zirconium oxide and other substances. Secondary failures which have been observed in defective rods are attributed to hydrogen embrittlement of the cladding. Post-irradiation examinations point to the hydriding failure mechanism rather than a waterlogging mechanism; the secondary failures occur as axial cracks or blisters in the cladding and are similar regardless of the primary failure mechanism. Such cracks do not result in flow blockage, or increase the effects of any postulated transients. More information is provided in Reference 19.

e. Potentially Damaging Temperature Effects During Transients

The fuel rod experiences many operational transients (intentional maneuvers) during its residence in the core. A number of thermal effects must be considered when analyzing the fuel rod performance.

The clad can be in contact with the fuel pellet at some time in the fuel lifetime. Clad/pellet interaction occurs if the fuel pellet temperature is increased after the clad is in contact with the pellet. Clad/pellet interaction is discussed in Subsection 4.2.3.3a.

The potential effects of operation with waterlogged fuel are discussed in Subsection 4.2.3.3d, which concluded that waterlogging is not a concern during operational transients.

Clad flattening, as shown in Reference 6, has been observed in some operating Westinghouse supplied power reactors. Thermal expansion (axial) of the fuel rod stack against a flattened section of clad could cause failure of the clad. This is no longer a concern because clad flattening is precluded by design during the fuel residence in the core (see Subsection 4.2.3.1).

Potential differential thermal expansion between the fuel rods and the guide thimbles during a transient is considered in the design. Excessive bowing of the fuel rods is precluded because the grid assemblies allow axial movement of the fuel rods relative to the grids. Specifically, thermal expansion of the fuel rods is considered in the grid design so that axial loads imposed on the fuel rods during a thermal transient will not result in excessively bowed fuel rods.

f. Fuel Element Burnout and Potential Energy Release

As discussed in Subsection 4.4.2.2, the core is protected from DNB over the full range of possible operating conditions. In the extremely unlikely event that DNB should occur, the clad temperature will rise due to degradation in heat transfer caused by steam blanketing at the rod surface. During this time, some chemical reaction between the cladding and the coolant will occur. However, because of the relatively good film boiling heat transfer following DNB, and the short time of the transient, the energy release resulting from this reaction is insignificant compared to the power produced by the fuel.

g. Coolant Flow Blockage Effects on Fuel Rods

This evaluation is presented in Subsection 4.4.4.7.

4.2.3.4 Spacer Grids

The coolant flow channels are established and maintained by the structure composed of grids and guide thimbles. The lateral spacing between fuel rods is provided and controlled by the support dimples of adjacent grid cells. Contact of the fuel rods on the dimples is maintained by the clamping force of the grid springs. Lateral motion of the fuel rods is opposed by the spring force and the internal moments generated between the spring and the support dimples. Grid testing is discussed in Reference 13.

As shown in Reference 13, grid crushing tests and seismic and loss-of-coolant accident evaluations show that the grids will maintain a geometry that is capable of being cooled under the worst-case accident Condition IV event.

4.2.3.5 Fuel Assembly

a. Stresses and Deflections

The fuel assembly component stress levels are limited by the design. For example, stresses in the fuel rod due to axial thermal expansion and zirconium alloy irradiation growth are limited by the relative motion of the rod as it slips over the grid spring and dimple surfaces. Clearances between the fuel rod ends and nozzles are provided so that zircaloy irradiation growth does not result in rod end interferences. Stresses in the fuel assembly caused by tripping of the Rod Cluster Control Assembly have little influence on fatigue because of the small number of events during the life of an assembly. Assembly components and prototype fuel assemblies made from production parts have been subjected to structural tests to verify that the design bases requirements are met.

The fuel assembly design loads for shipping and handling have been established at 4g axial and 6g lateral. Accelerometers are permanently placed into the shipping cask to monitor and detect fuel assembly accelerations that would exceed the criteria. Past history and experience have indicated that loads which exceed the allowable limits rarely occur. Exceeding the limits requires reinspection of the fuel assembly for damage. Tests on various fuel assembly components such as the grid assembly, sleeves, inserts and structure joints have been performed to assure that the shipping design limits do not result in impairment of fuel assembly function. Seismic analysis of the fuel assembly is presented in Reference 13.

b. Dimensional Stability

A prototype fuel assembly has been subjected to column loads in excess of those expected in normal service and faulted conditions (see Reference 13).

No interference between control rod and thimble tubes will occur during insertion of the rods following a postulated loss-of-coolant accident transient due to fuel rod swelling, thermal expansion, or bowing. In the early phase of the transient following the coolant break, the high axial loads, which could be generated by the difference in thermal expansion between fuel clad and thimbles, are relieved by slippage of the fuel rods through the grids. The relatively low drag force restraint on the fuel rods will induce only minor thermal bowing, which is insufficient to lose the gap between the fuel rod and thimble tube.

Reference 13 shows that the fuel assemblies will maintain a geometry that is capable of being cooled during a combined seismic and double-ended loss-of-coolant accident.

4.2.3.6 Reactivity Control Assembly and Burnable Poison Rods

a. Internal Pressure and Cladding Stresses During Normal, Transient and Accident Conditions

The designs of the burnable poison and source rods provide a sufficient void volume to accommodate the internal pressure increase during operation caused by release of helium generated by neutron absorption. This is not a concern for the Ag-In-Cd control rods, because no gas is generated in or released by the absorber material. For the burnable poison rod, the use of glass in tubular form provides a central void volume along the length of the rods.

The stress analysis of the burnable poison end source rods assumes 100 percent gas release to the rod void volume in addition to the initial pressure within the rod.

During normal transient and accident conditions the void volume limits the internal pressures to values which satisfy the criteria in Subsection 4.2.1.6.

These limits are established not only to assure that peak stresses do not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of the fatigue characteristics of the materials.

Rod, guide thimble, and dashpot flow analyses indicate that the flow is sufficient to prevent coolant boiling. Therefore, clad temperatures at which the clad material has adequate strength to resist coolant operating pressures and rod internal pressures are maintained.

b. Thermal Stability of the Absorber Material, Including Phase Changes and Thermal Expansion

The radial and axial temperature profiles have been determined by considering gap conductance, thermal expansion, and neutron or gamma heating of the contained material as well as gamma heating of the clad.

The maximum temperature of the absorber material was calculated to be less than 1010°F for Ag-In-Cd and occurs axially at only the highest flux region. This temperature is well below the absorber melting temperature stated in Subsection 4.2.1.6. The thermal expansion properties of the absorber material and the phase changes are discussed in Reference 2.

The maximum temperature of the borosilicate glass was calculated to be about 1300°F and takes place following the initial rise to power.

As the operating cycle proceeds, the glass temperature decreases for the following reasons: (1) reduction in power generation due to boron 10 depletion, (2) better gap conductance as the helium produced diffuses to the gap, and (3) external gap reduction due to borosilicate glass swelling.

Sufficient diametral and end clearances have been provided in the neutron absorber, burnable poison, and source rods to accommodate the relative thermal expansions between the enclosed material and the surrounding clad and end plug.

c. Irradiation Stability of the Absorber Material, Taking into Consideration Gas Release and Swelling

The irradiation stability of the absorber material is discussed in Reference 2. Irradiation produces no deleterious effects in the absorber material.

Sufficient diametral and end clearances are provided to accommodate swelling of the absorber material.

Based on experience with borosilicate glass, and on nuclear and thermal calculations, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube could occur, but would continue only until the glass came in contact with the inner liner. The wall thickness of the inner liner is sized to provide adequate support in the event of slumping, and to collapse locally before rupture of the exterior cladding if unexpected large volume changes, due to swelling or cracking, should occur. The ends of the inner liner are open to allow helium, which diffuses out of the glass, to occupy the central void.

d. Potential for Chemical Interaction, Including Possible Waterlogging Rupture

The structural materials selected have good resistance to irradiation damage and are compatible with the reactor environment.

Corrosion of the materials exposed to the coolant is quite low, and proper control of chloride and oxygen in the coolant will prevent the occurrence of stress corrosion. The potential for the interference with rod cluster control movement due to possible corrosion phenomena is very low.

Waterlogging rupture is not a failure mechanism associated with Westinghouse-designed control rods. However, a breach of the cladding for any postulated reason does not result in serious consequences. The silver-indium-cadmium absorber material is relatively inert and would still remain remote from high coolant velocity regions. Rapid loss of material resulting in significant loss of reactivity control material would not occur. Bettis test results (Reference 8) concluded that additions of indium and cadmium to silver, in the amounts to form the Westinghouse absorber material composition, result in small corrosion rates.

4.2.4 Testing and Inspection Plan

4.2.4.1 Quality Assurance Program

The quality assurance program plan of the Westinghouse Nuclear Fuel Division is discussed in Section 17.1.

The program provides for control over all activities affecting product quality, commencing with design and development and continuing through procurement, materials handling, fabrication, testing and inspection, storage, and transportation. The program also provides for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

Westinghouse drawings and product, process, and material specifications identify the inspection to be performed.

4.2.4.2 Quality Control

Quality control philosophy is generally based on the following inspections being performed to a 95 percent confidence that at least 95 percent of the product meets specification, unless otherwise noted.

a. Fuel System Components and Parts

The characteristics inspected depends upon the component parts and includes dimensional, visual, check audits of test reports, material certification and nondestructive examination such as X-ray and ultrasonic.

All material used in this core is accepted and released by Quality Control.

b. Pellets

Inspection is performed for dimensional characteristics such as diameter, density, length and squareness of ends. Additional visual inspections are performed for cracks, chips and surface conditions according to approved standards.

Density is determined in terms of weight per unit length. Chemical analyses are taken on a specified sample basis throughout pellet production.

c. Rod Inspection

Fuel rod, control rod, burnable poison and source rod inspection consists of the following nondestructive examination techniques and methods, as applicable.

1. Leak Testing

   Each rod is tested using a calibrated mass spectrometer with helium as the detectable gas.

2. Closure Welds

   Rod closure welds are inspected by ultrasonic test or X-ray in accordance with a qualified technique and Westinghouse specifications.

3. Dimensional

   All rods are dimensionally inspected prior to final release. The requirements include such items as length, camber, and visual appearance.

4. Plenum Dimensions

   All fuel rods are inspected by gamma scanning or other approved methods as discussed in Subsection 4.2.4.4 to ensure proper plenum dimensions.

5. Pellet-to-Pellet Gaps

   All fuel rods are inspected by gamma scanning or other approved methods as discussed in Subsection 4.2.4.4 to ensure that no significant gaps exist between pellets.

6. Enrichment Control

   All fuel rods are gamma scanned to verify enrichment control prior to acceptance for assembly loading.

7. Traceability

   Traceability of rods and associated rod components is established by Quality Control.

d. Assemblies

Each fuel, control, burnable poison and source rod assembly is inspected for compliance with drawing and/or specification requirements. Other incore control component inspection and specification requirements are given in Subsection 4.2.4.3.

e. Other Inspections

The following inspections are performed as part of the routine inspection operation:

1. Tool and gage inspection and control including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and conditions of tools.

2. Audits are performed of inspection activities and records to assure that prescribed methods are followed and that records are correct and properly maintained.

3. Surveillance inspection, where appropriate, and audits of outside contractors are performed to ensure conformance with specified requirements.

f. Process Control

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

The UO2 powder is kept in sealed containers or is processed in a closed system. The containers are either fully identified both by descriptive tagging and preselected color coding or, for the closed system, the material is monitored by a computer data management information system. For the sealed container system, a Westinghouse identification tag completely describing the contents is affixed to the containers before transfer to powder storage. Isotopic content is confirmed by analysis.

Powder withdrawal from storage can only be made by an authorized group, which directs the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single nominal enrichment and density are produced in a given production line at any given time.

Finished pellets are transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for physical and chemical properties including isotopic content and impurity levels prior to acceptance by Quality Control. Physical barriers prevent mixing of pellets of different nominal designs and enrichment in this storage area. Unused powder and substandard pellets are returned to storage for disposition.

Pellets are loaded into fuel cladding tubes on isolated production lines. Each production line contains only rods of one fuel type at any one time.

A unique code is placed on each fuel tube for traceability purposes. The end plugs are inserted and welded to seal the tube. The fuel tube remains identifiable by this code throughout the fabrication process.

At the time of installation into an assembly, a matrix is generated to identify each rod in its position within a given assembly. After the fuel rods are installed, an inspector verifies that all fuel rods in an assembly carry the correct identification character describing the fuel enrichment and density for the core region being fabricated. The top nozzle is inscribed with a permanent identification number providing traceability of the assembly and the fuel rods contained in the assembly.

Similar traceability is provided for burnable poison, source rods and control rodlets as required.

4.2.4.3 Core Component Testing and Inspection

Tests and inspections are performed on each reactivity control component to verify the mechanical characteristics. In the case of the Rod Cluster Control Assembly, prototype testing has been conducted and both manufacturing test/inspections and functional testing at the plant site are performed.

During the component manufacturing phase, the following requirements apply to the reactivity control components to assure the proper functioning during reactor operation:

a. All materials are procured to specifications to attain the desired standard of quality.

b. A spider from each braze lot is proof tested by applying a 5000 pound load to the spider body, so that approximately 310 pounds is applied to each vane. This proof load provides a bending moment at the spider body approximately equivalent to 1.4 times the load caused by the acceleration imposed by the control rod drive mechanism.

c. All rods are checked for integrity by methods described in Subsection 4.2.4.2, item c.

d. To assure proper fitup with the fuel assembly, the rod cluster control, burnable poison and source assemblies are installed in the fuel assembly without restriction or binding in the dry condition. Also a straightness of 0.01 in./ft is required on the entire inserted length of each rod assembly.

The Rod Cluster Control Assemblies are functionally tested following core loading but prior to criticality to demonstrate reliable operation of the assemblies. The testing performed during the initial plant startup is described in Chapter 14. Following each refueling, each assembly is fully withdrawn and dropped at full flow/operating temperature conditions specified by Technical Specifications.

In order to demonstrate continuous free movement of the Rod Cluster Control Assemblies, and to ensure acceptable core power distributions during operations, partial movement checks are performed on every Rod Cluster Control Assembly as required by the Technical Specifications.

If a Rod Cluster Control Assembly cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus inability to move one Rod Cluster Control Assembly can be tolerated. More than one inoperable Rod Cluster Control Assembly could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable Rod Cluster Control Assemblies has been limited to one.

4.2.4.4 Tests and Inspections by Others

If any tests and inspections are to be performed on behalf of Westinghouse, Westinghouse will review and approve the quality control procedures, inspection plans, etc., to be utilized to ensure that they are equivalent to the description provided in Subsections 4.2.4.1 and 4.2.4.3 and are performed properly to meet all Westinghouse requirements.

4.2.4.5 In-Service Surveillance

Westinghouse has extensive experience with the use of 17x17 standard fuel assemblies in other operating plants. This experience is summarized in WCAP-8183, Reference 1, which is periodically updated to provide the most recent information operating plants. Additional test assembly and test rod experience is given in Sections 8 and 23 of Reference 9.

4.2.4.6 Onsite Inspection

Detailed written procedures are used by the station staff and nuclear fuel quality assurance personnel for the receipt inspection of all new fuel and associated components such as control rods and plugs. The procedures are specific and are written to take into account the manufacturer's procedures and processes. The specific procedures incorporate the following minimum requirements:

a. Survey of the new fuel shipping containers for radiation and contamination levels

b. External inspection of shipping container for visible signs of damage, including integrity of seals

c. Check of condition of new fuel shipping container accelerometers

d. Inspection of physical condition of inside of shipping containers including hardware utilized to secure the component and protective covers if used

e. Verification of serial numbers if serial numbers are required

f. Visual inspection of component for dirt, debris, water, deep scars, abrasions and other irregularities or evidence of damage

g. Survey of radiation and contamination levels of new fuel assembly.

Surveillance of fuel and reactor performance is routinely conducted. Power distribution is monitored using excore fixed and incore detectors. Coolant activity and chemistry are followed to permit early detection of any fuel clad defects.

Visual irradiated fuel inspections will be conducted as necessary during each refueling. Selected fuel assemblies may be inspected for fuel rod failure, structural integrity, crud deposition, rod bow and other irregularities. Fuel assemblies will be selected for inspection based upon performance history and recommendations made by the fuel supplier.

The fuel inspection program will be expanded to include more fuel assemblies or greater detail of examination if high coolant activity is experienced during operation, irregularities are noted in fuel performance, irregularities are noted during routine inspections, or if a new fuel design is incorporated.

4.2.5 References

1. Slagle, W. H., "Operational Experience with Westinghouse Cores," WCAP-8183, (latest revision).
2. Beaumont, M. D., and Iorii, J .A., "Properties of Fuel and Core Component Materials," WCAP-9179, Revision 1 (Proprietary), July 1978.
3. Not used
4. Hellman, J. M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A (Proprietary) and WCAP-8219-A (Nonproprietary), March 1975.
5. Not used
6. George, R. A., Lee, Y. C. and Eng. G. H., "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Nonproprietary), July 1974.
7. Risher, D. H., et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963 (Proprietary), November 1976 and WCAP-8964 (Nonproprietary), August 1977.
8. Cohen, J., "Development and Properties of Silver Base Alloys as Control Rod Materials for Pressurized Water Reactors," WAPD-214, December 1959.
9. Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries-Winter 1977-Summer 1978," WCAP-8768, Revision 2, October 1978.
10. Demario, E. E., "Hydraulic Flow Test of the 17x17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Nonproprietary), February 1974.
11. Skaritka, J., et al., "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (Proprietary) and WCAP-8692, Revision 1 (Nonproprietary), July 1979.

12. O'Donnell, W. J. and Langer, B. F., "Fatigue Design Basis for Zircaloy Components," Nuclear Science and Engineering 20 1-12, 1964.

13. Gesinski, L., and Chiang, D., "Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," WCAP-8236 (Proprietary) and WCAP-8288 (Nonproprietary) December 1973.

14. Davidson. S. L. (Ed.), et. al., "VANTAGE-5 Fuel Assembly, Reference Core Report," WCAP-10444-P-A, Westinghouse Electric Corporation, September 1985.

15. Davidson, S. L. (Ed.), et. al., "VANTAGE-5H Fuel Assembly," WCAP-10444-P-A Addendum 2-A, Westinghouse Electric Corporation, April 1988.

16. Davidson, S. L. and Ryan, T.L., "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A and Appendices A through D, Westinghouse Electric Corporation, April 1995.

17. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), August 1988.

18. Kersting, P. J., et al., "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," WCAP-13589-A, March 1995.

19. Stephan, L. A., "The Effects of Cladding Material and Heat Treatment on the Response of Water Logged UO_2 Fuel Rods to Power Bursts," IN-ITR 1970.

20. Davidson, S. L., "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A, October 1994.

21. "Addendum 1 to WCAP-12488-A, Revision To Design Criteria," WCAP-12488-A, Addendum 1-A, Revision 1, January 2002. (Westinghouse Proprietary)

22. Letter from N. J. Liparolu (Westinghouse) to R. S. Jones (NRC), "Westinghouse Interpretation of Staff's Position on Extended Burnup (Proprietary)," NTD-NRC-94-4275, August 29, 1994.

23. Slagle, W. G., (Ed.), "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Rev. 1 w/ errata July 2000.

4.3 NUCLEAR DESIGN

4.3.1 Design Bases

This section describes the design bases and functional requirements used in the nuclear design of the Fuel and Reactivity Control System, and relates these design bases to the General Design Criteria (GDC) in 10 CFR 50, Appendix A. Where appropriate, supplemental criteria such as 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors are addressed. Before discussing the nuclear design bases, it is appropriate to briefly review the four major categories ascribed to conditions of plant operation.

The full spectrum of plant conditions is divided into four categories, in accordance with the anticipated frequency of occurrence and risk to the public (as defined in ANSI Standard N18.2):

1. Condition I - Normal Operation
2. Condition II - Incidents of Moderate Frequency
3. Condition III - Infrequent Faults
4. Condition IV - Limiting Faults

In general, the Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Condition II incidents are accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action. Fuel damage (fuel damage as used here is defined as penetration of the fission product barrier; i.e., the fuel rod clad) is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the plant cleanup system and are consistent with the plant design basis.

Condition III incidents shall not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude immediate resumption of operation. The release of radioactive material due to Condition III incidents should not be sufficient to interrupt or restrict public use of these areas beyond the exclusion radius. Furthermore, a Condition III incident shall not, by itself generate a Condition IV fault or result in a consequential loss of function of the reactor coolant or reactor containment barriers.

Condition IV occurrences are faults that are not expected to occur, but are defined as limiting faults which must be designed against. Condition IV faults shall not cause a release of radioactive material that results in an undue risk to public health and safety.

The core design power distribution limits related to fuel integrity are met for Condition I occurrences through conservative design, and maintained by the action of the control system. The requirements for Condition II occurrences are met by providing an adequate protection system which monitors reactor parameters. The control and protection systems are described in Chapter 7, and the consequences of Condition II, III and IV occurrences are given in Chapter 15.

4.3.1.1 Fuel Burnup

a. Basis

The fuel rod design basis is described in Section 4.2. The nuclear design basis is to install sufficient reactivity in the fuel to attain a region average discharge burnup of between 45,000 and 50,000 MWd/Mtu. The above, along with the design basis in Subsection 4.3.1.3, satisfies GDC-10.

b. Discussion

Fuel burnup is a measure of fuel depletion, which represents the integrated energy output of the fuel (MWd/Mtu), and is a convenient means for quantifying fuel exposure criteria.

The core design lifetime or design discharge burnup is achieved by installing sufficient initial excess reactivity in each fuel region and by following a fuel replacement program (such as that described in Subsection 4.3.2) that meets all safety-related criteria in each cycle of operation.

Initial excess reactivity installed in the fuel, although not a design basis, must be sufficient to maintain core criticality at full power operating conditions throughout cycle life with equilibrium xenon, samarium, and other fission products present. The end of design cycle life is defined to occur when the chemical shim concentration is essentially zero with control rods present to the degree necessary for operational requirements (e.g., the controlling bank at the "bite" position). In terms of chemical shim boron concentration this represents approximately 10 parts per million (ppm) with no control rod insertion.

A limitation on initial installed excess reactivity is not required other than as is quantified in terms of other design bases such as core reactivity feedback and shutdown margin discussed below.

4.3.1.2 Reactivity Feedbacks (Reactivity Coefficient)

a. Basis

The fuel temperature coefficient will be negative and the moderator temperature coefficient of reactivity will be nonpositive for power operating conditions, above 20% power, thereby providing negative reactivity feedback characteristics. The design basis conservatively includes analysis for positive moderator temperature coefficients; however, actual core loading designs meet the above restrictions and thus GDC 11.

b. Discussion

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption effects (Doppler) associated with changing fuel temperature, and the spectrum effect resulting from changing moderator density. These basic physics characteristics are often identified by reactivity coefficients. The use of slightly enriched uranium ensures that the Doppler coefficient of reactivity is negative. This coefficient provides the most rapid reactivity compensation. The core is also designed to have an overall negative moderator temperature coefficient of reactivity so that average coolant temperature or void content provides another, slower compensatory effect. Nominal power operation is permitted only in a range of overall negative moderator temperature coefficient. The negative moderator temperature coefficient can be achieved through use of fixed burnable poison and/or control rods by limiting the reactivity held down by soluble boron.

Burnable poison content (quantity and distribution) is not stated as a design basis other than as it relates to accomplishment of a nonpositive moderator temperature coefficient at power operating conditions discussed above.

4.3.1.3 Control of Power Distribution

a. Basis

The nuclear design basis is that, with at least a 95 percent confidence level:

1. The fuel will not be operated at greater than 14.6 kW/ft* under normal operating conditions.
2. Under abnormal conditions including the maximum overpower condition, the fuel peak power will not cause melting as defined in Subsection 4.4.1.2.
3. The fuel will not operate with a power distribution that violates the Departure from Nucleate Boiling (DNB) design basis (i.e., the DNBR shall not be less than the safety analysis limit value, as discussed in Subsection 4.4.1.1) under Condition I and II events including the maximum overpower condition.
4. Fuel management will be such to produce rod powers and burnups consistent with the assumptions in the fuel rod mechanical integrity analysis of Section 4.2.

The above basis meets GDC-10.

b. Discussion

Calculation of extreme power shapes which affect fuel design limits is performed with proven methods and verified frequently with measurements from operating reactors. The conditions under which limiting power shapes are assumed to occur are chosen conservatively with regard to any permissible operating state.

Even though there is good agreement between peak power calculations and measurements, a nuclear uncertainly margin (see Subsection 4.3.2.2g) is applied to calculated peak local power. Such a margin is provided both for the analysis for normal operating states and for anticipated transients.

---

* Due to LOCA analysis. Average kW/ft (5.84), (assuming maximum reactor rated thermal power of $\leq$ 3659 MWt) x F_Q (2.50) = 14.6 kW/ft

4.3.1.4 Maximum Controlled Reactivity Insertion Rate

a. Basis

The maximum reactivity insertion rate due to withdrawal of Rod Cluster Control Assemblies at power or by boron dilution is limited. During normal at power operation with normal control rod overlap, the maximum controlled reactivity rate change is limited to less than 110 pcm/sec. (1 pcm = 10^{-5} $\Delta\rho$, see footnote to Table 4.3-2.) At zero power conditions, a maximum reactivity change rate of 75 pcm/sec for accidental simultaneous withdrawal of two control banks is set so that peak heat generation rate and DNBR do not exceed the maximum allowable at overpower conditions. This satisfies GDC-25.

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to preclude rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity due to a rod withdrawal or ejection accident (see Chapter 15).

Following any Condition IV event (rod ejection, steam line break, etc.), the reactor can be brought to the shutdown condition and the core will maintain acceptable heat transfer geometry. This satisfies GDC-28.

b. Discussion

Reactivity addition associated with an accidental withdrawal of a control bank (or banks) is limited by the maximum rod speed (or travel rate) and by the worth of the bank(s). The maximum control rod speed is 45 inches per minute and the maximum rate of reactivity change considering two control banks moving is less than 75 pcm/sec. During normal operation at power and with normal control rod overlap, the maximum reactivity change rate is limited to less than 110 pcm/sec.

The reactivity change rates are conservatively calculated assuming unfavorable axial power and xenon distributions. The peak xenon burnout rate is 25 pcm/min, significantly lower than the maximum reactivity addition rate of 110 pcm/sec for normal operation and 75 pcm/sec for accidental withdrawal of two banks.

4.3.1.5 Shutdown Margins

a. Basis

Minimum shutdown margin as specified in Technical Specifications and the Core Operating Limits Report is required at any power operating condition, in the hot standby shutdown condition and in the cold shutdown condition.

In all analyses involving reactor trip, the single, highest worth Rod Cluster Control Assembly is postulated to remain untripped in its full out position (stuck rod criterion). This satisfies GDC-26.

b. Discussion

Two independent reactivity control systems are provided, namely control rods and soluble boron in the coolant. The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full-load to no-load. In addition, the Control Rod System provides the minimum shutdown margin under Condition I events and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits assuming that the highest worth control rod is stuck out upon trip.

The boron system can compensate for all xenon burnout reactivity changes and will maintain the reactor in the cold shutdown condition. Thus, backup and emergency shutdown provisions are provided by a mechanical and a chemical shim control system, which satisfies GDC-26.

c. Basis

When fuel assemblies are in the pressure vessel and the vessel head is not in place, k_{eff} will be maintained at or below 0.95 with control rods and soluble boron. Further, the fuel will be maintained sufficiently subcritical that removal of all Rod Cluster Control Assemblies will not result in criticality.

d. Discussion

ANSI Standard N18.2 specifies a k_{eff} not to exceed 0.95 in spent fuel storage racks and transfer equipment flooded with pure water and a k_{eff} not to exceed 0.98 in normally dry new fuel storage racks assuming optimum moderation. No criterion is given for the refueling operation; however, a 5 percent margin, which is consistent with spent fuel storage and transfer, is adequate for the controlled and continuously monitored operations involved.

The boron concentration required to meet the refueling shutdown criteria is specified in the Technical Specifications. Verification that this shutdown criteria is met, including uncertainties, is achieved using qualified nuclear design methods such as the CASMO Code (Reference 1) and SIMULATE Code (Reference 2), per the Phoenix-P/ANC Code System (Reference 11). The subcriticality of the core is continuously monitored as described in the Technical Specifications.

4.3.1.6 Stability

a. Basis

The core will be inherently stable to power oscillations at the fundamental mode. This satisfies GDC-12. Spatial power oscillations within the core with a constant core power output, should they occur, can be reliably and readily detected and suppressed.

b. Discussion

Oscillations of the total power output of the core, from whatever cause, are readily detected by the loop temperature sensors and by the nuclear instrumentation. The core is protected by these systems and a reactor trip would occur if power increased unacceptably, preserving the design margins to fuel design limits. The stability of the Turbine/Steam Generator/Core Systems and the Reactor Control System is such that total core power oscillations are not normally possible. The redundancy of the protection circuits ensures an extremely low probability of exceeding design power levels.

The core is designed so that diametral and azimuthal oscillations due to spatial xenon effects are self-damping and no operator action or control action is required to suppress them. The stability to diametral oscillations is so great that this excitation is highly improbable. Convergent azimuthal oscillations can be excited by prohibited motion of individual control rods. Such oscillations are readily observable and alarmed using the excore long ion chambers. Indications are also continuously available from incore thermocouples and loop temperature measurements. Moveable and fixed incore detectors can be activated to provide more detailed information.

In all presently proposed cores, these horizontal plane oscillations are self-damping by virtue of reactivity feedback effects designed into the core.

However, axial xenon spatial power oscillations may occur late in core life. The control banks and excore detectors are provided for control and monitoring of axial power distributions. Assurance that fuel design limits are not exceeded is provided by reactor Overpower ΔT and Overtemperature ΔT trip functions which use the measured axial power imbalance as input.

4.3.1.7 Anticipated Transients without Trip

The effects of anticipated transients with failure to trip are not considered in the design bases for transients analyzed in Chapter 15. Analysis has shown that the likelihood of such a hypothetical event is negligibly small. Furthermore, analysis of the consequences of a hypothetical failure to trip following anticipated transients has shown that no significant core damage would result, system peak pressures would be limited to acceptable values and no failure of the Reactor Coolant System would result (see Reference 3). The final NRC ATWTS Rule (Reference 4) requires that Westinghouse-designed plants install ATWTS mitigation systems to initiate a turbine trip and actuate emergency feedwater flow independent of the Reactor Protection System. The Seabrook ATWTS mitigation system is described in Subsection 7.6.12.

4.3.2 Description

4.3.2.1 Nuclear Design Description

The reactor core consists of a specified number of fuel rods which are held in bundles by spacer grids and top and bottom fittings. The fuel rods are constructed of zirconium alloy cylindrical tubes containing UO_2 fuel pellets. The bundles, known as fuel assemblies, are arranged in a pattern which approximates a right circular cylinder.

Each fuel assembly contains a 17x17 rod array composed of 264 fuel rods, 24 rod cluster control thimbles and an incore instrumentation thimble. Figure 4.2-1 shows a cross-sectional view of a 17x17 fuel assembly and the related rod cluster control locations. Further details of the fuel assembly are given in Section 4.2.

The fuel rods within a given assembly have the same uranium enrichment in both the radial and axial planes. Fresh fuel assemblies of different enrichments are used in the reload core to establish a favorable radial power distribution. Figure 4.3-1 shows a sample fuel loading pattern to be used in the reload cores. The premise for reload designs is for low radial leakage, achieved by placing low reactivity assemblies around the perimeter of the core. Fresh assemblies are then distributed within the core interior to generate a favorable radial power distribution. The enrichments for these cores vary with the expected cycle length; typical values are shown in Table 4.3-1. Axial fuel blankets composed by mid-enriched annular fuel pellets may be used to reduce axial neutron leakage and improve fuel utilization.

The reference reloading pattern is typically similar to Figure 4.3-1, with depleted fuel on the periphery and fresh fuel interspersed in the center with depleted fuel. The core will operate between eighteen and twenty-four months between refueling, accumulating between 16,000 MWd/Mtu and 24,000 MWd/Mtu per cycle. The exact reloading pattern, initial and final positions of assemblies, number of fresh assemblies and their placement are dependent on the energy requirement for the next cycle, and burnup and power histories of the previous cycles.

The core average enrichment is determined by the amount of fissile material required to provide the desired core lifetime and energy requirements, namely a region average discharge burnup of between 45,000 and 50,000 MWd/Mtu. The physics of the burnout process is such that operation of the reactor depletes the amount of fuel available due to absorption of neutrons by the U-235 atoms and their subsequent fission. The rate of U-235 depletion is directly proportional to the power level at which the reactor is operated. In addition, the fission process results in the formation of fission products, some of which readily absorb neutrons. These effects, depletion and the buildup of fission products, are partially offset by the buildup of plutonium shown in Figure 4.3-2 for the 17x17 fuel assembly, which occurs due to the nonfission absorption of neutrons in U-238. Therefore, at the beginning of any cycle a reactivity reserve equal to the depletion of the fissionable fuel and the buildup of fission product poisons over the specified cycle life must be "built" into the reactor. This excess reactivity is controlled by removable neutron absorbing material in the form of boron dissolved in the primary coolant and burnable poison rods.

The concentration of boric acid in the primary coolant is varied to provide control and to compensate for long-term reactivity requirements. The concentration of the soluble neutron absorber is varied to compensate for reactivity changes due to fuel burnup, fission product poisoning including xenon and samarium, burnable poison depletion, and the cold-to-operating moderator temperature change. Using its normal makeup path, the Chemical and Volume Control System (CVCS) is capable of inserting negative reactivity at a rate of approximately 30 pcm/min when the reactor coolant boron concentration is 1000 ppm and approximately 35 pcm/min when the reactor coolant boron concentration is 100 ppm. If the emergency boration path is used, the CVCS is capable of inserting negative reactivity at a rate of approximately 65 pcm/min when the reactor coolant concentration is 1000 ppm and approximately 75 pcm/min when the reactor coolant boron concentration is 100 ppm. The peak burnout rate for xenon is 25 pcm/min (Subsection 9.3.4 discusses the capability of the CVCS to counteract xenon decay). Rapid transient reactivity requirements and safety shutdown requirements are met with control rods.

As the boron concentration is increased, the moderator temperature coefficient becomes less negative. The use of a soluble poison alone would result in a positive moderator coefficient at beginning-of-life for the cycle. Therefore, burnable absorber fuel rods are used to reduce the soluble boron concentration sufficiently to ensure that the moderator temperature coefficient is negative for power operating conditions above 20% power*. During operation the poison content in these rods is depleted thus adding positive reactivity to offset some of the negative reactivity from fuel depletion and fission product buildup. The depletion rate of the burnable absorber fuel rods is not critical since chemical shim is always available and flexible enough to cover any possible deviations in the expected burnable absorber depletion rate. Figure 4.3-3 is a graph of a typical core depletion.

In addition to reactivity control, the burnable absorber fuel rods are strategically located to provide a favorable radial power distribution. Figure 4.3-4 shows the integral burnable absorber fuel rod distribution within a fuel assembly for the several fuel rod patterns used in a 17x17 array. A typical integral burnable absorber fuel rod loading pattern is shown in Figure 4.3-5.

Control rods are located for use in the core to provide control for rapid changes in reactivity. The reactivity worth of the control rods is dependent on the particular absorber material used, but the power distribution effects and reactivity worth depend primarily on the number and location of the inserted control rods.

Table 4.3-1, Table 4.3-2 and Table 4.3-3 contain a summary of the reactor core design parameters for a typical reload fuel cycle, including reactivity coefficients, delayed neutron fraction and neutron lifetimes. Sufficient information is included to permit an independent calculation of the nuclear performance characteristics of the core.

4.3.2.2 Power Distributions

The accuracy of power distribution calculations has been confirmed through analytic benchmarks and experience of operation under conditions very similar to those expected. Details of this confirmation are given in Reference 2 and in Subsection 4.3.2.2f.

a. Definitions

Power distributions are quantified in terms of hot channel factors. These factors are a measure of the peak pellet power within the reactor core and the total energy produced in a coolant channel and are expressed in terms of quantities related to the nuclear or thermal design, namely:

---

* Note: A non-negative moderator temperature coefficient is allowed by Technical Specifications for all power levels, provided that compliance with the ATWS Rule and its basis are maintained, as described in the Bases for Technical Specification 3/4.1.1.3. The Seabrook core design philosophy meets this requirement by ensuring that a non-positive MTC exists for operating conditions above 20% power.

1. Power density, is the thermal power produced per unit volume of the core (kW/liter).

2. Linear power density, is the thermal power produced per unit length of active fuel (kW/ft). Since fuel assembly geometry is standardized, this is the unit of power density most commonly used. For all practical purposes, it differs from kW/liter by a constant factor which includes geometry and the fraction of the total thermal power which is generated in the fuel rod.

3. Average linear power density, is the total thermal power produced in the fuel rods divided by the total active fuel length of all rods in the core.

4. Local heat flux, is the heat flux at the surface of the cladding (Btu/ft^2-hr). For nominal rod parameters, this differs from linear power density by a constant factor.

5. Rod power or rod integral power, is the length integrated linear power density in one rod (kW).

6. Average rod power, is the total thermal power produced in the fuel rods divided by the number of fuel rods (assuming all rods have equal length).

7. The hot channel factors used in the discussion of power distribution in this section are defined as follows:

   (a) F_Q, heat flux hot channel factor is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

   (b) F^N_Q, nuclear heat flux hot channel factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod parameters.

   (c) F^E_Q, engineering heat flux hot channel factor is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically, the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

   (d) $F_{\Delta H}{}^N$ nuclear enthalpy rise hot channel factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Manufacturing tolerances, hot channel power distribution and surrounding channel power distributions are treated in the calculation of the DNBR as described in Section 4.4.

It is convenient for the purposes of discussion to define subfactors of F_Q; however, design limits are set in terms of the total peaking factor.

$$F_Q = \text{Total peaking factor or heat flux hot-channel factor}$$

$$= \frac{\text{Maximum kW/ft}}{\text{Average kW/ft}}$$

Without densification effects,

$$F_Q = F^N_Q \times F^E_Q \times F^N_U$$

where

F^N_Q and F^E_Q are defined above.

F^N_U = uncertainty associated with the incore detector system, given in the COLR.

To include the allowances made for densification effect, which are height dependent, the following quantities are defined.

$S(Z)$ = the allowance made for densification effects at height Z in the core. See Subsection 4.3.2.2e.

Then

$$F_Q = \text{Total peaking factor}$$

$$= \frac{\text{Maximum kW/ft}}{\text{Average kW/ft}}$$

Including densification allowance

$$F_Q = \max\left(S(Z) \times F^N_Q \times F^E_Q \times F^N_U \right.$$

b. Radial Power Distributions

While radial power distributions in various axial planes of the core contribute to the axial F_Q, the core radial enthalpy rise distribution as determined by the integral of power up each channel is of greater interest. The power shape is axially integrated to yield a two dimensional representation of assembly and pin powers ($F_{\Delta H}^N$). Figure 4.3-6, Figure 4.3-7, Figure 4.3-8, Figure 4.3-9, Figure 4.3-10 and Figure 4.3-11 show typical representative operating conditions. These conditions are: (1) hot full power (HFP) near beginning-of-life (BOL) - unrodded - no xenon, (2) HFP near BOL - unrodded - equilibrium xenon, (3) HFP near BOL - bank D in - equilibrium xenon (4) HFP near middle-of-life (MOL) - unrodded - equilibrium xenon, (5) HFP near end-of-life (EOL) - unrodded - equilibrium xenon, and (6) HFP near EOL - bank D in - equilibrium xenon.

c. Assembly Power Distributions

For the purpose of illustration, assembly power distributions from the BOL and EOL conditions corresponding to Figure 4.3-7 and Figure 4.3-10, respectively, are given for the same assembly in Figure 4.3-12 and Figure 4.3-13, respectively.

d. Axial Power Distributions

The shape of the power profile in the axial, or vertical, direction is largely under the control of the operator either through the manual operation of the full length control rods or automatic motion of full length rods responding to manual operation of the CVCS. Nuclear effects which cause variations in the axial power shape include moderator density, Doppler effect on resonance absorption, spatial xenon and burnup.

Automatically controlled variations in total power output and full length rod motion are also important in determining the axial power shape at any time. Signals are available to the operator from the excore ion chambers, which are long ion chambers outside the reactor vessel running parallel to the axis of the core. Separate signals are taken from the top and bottom halves of the chambers. The difference between top and bottom signals from each of four pairs of detectors is displayed on the control panel and called the flux difference, ΔI. Calculations of core average peaking factor for many plants and measurements from operating plants under many operating situations are associated with either ΔI or axial offset in such a way that an upper bound can be placed on the peaking factor. For these correlations, axial offset is defined as:

$$\text{axial offset} = \frac{\varphi_t - \varphi_b}{\varphi_t + \varphi_b}$$

where φ_t and φ_b are the top and bottom detector readings, respectively.

Representative axial power shapes for typical BOL and EOL unrodded conditions are shown in Figure 4.3-14 and Figure 4.3-15. Comparative partially rodded axial power shapes are shown in Figure 4.3-16. These figures cover a wide range of axial offset including values not permitted at full power.

The radial power distributions shown in Figure 4.3-8 and Figure 4.3-11 involving the partial insertion of control rods represent a synthesis of power shapes from the rodded and unrodded planes. The applicability of the separability assumption upon which this procedure is based is assured through extensive three-dimensional calculations of possible rodded conditions. As an example, Figure 4.3-17 compares the axial power distribution for several assemblies at different distances from inserted control rods with the core average distribution.

The only significant difference from the average occurs in the low power peripheral assemblies, thus confirming the validity of the separability assumption.

e. Local Power Peaking

Fuel densification, which has been observed to occur under irradiation in several operating reactors, causes the fuel pellets to shrink both axially and radially. The pellet shrinkage combined with random hang-up of fuel pellets results in gaps in the fuel column when the pellets below the hung-up pellet settle in the fuel rod. The gaps vary in length and location in the fuel rod. Because of decreased neutron absorption in the vicinity of the gap, power peaking occurs in the adjacent fuel rods resulting in an increased power peaking factor. A quantitative measure of this local peaking is given by the power spike factor S(Z), where Z is the axial location in the core.

Results reported in Reference 5 show that fuel manufactured by Westinghouse will not densify to the point that significant axial gaps will occur in the fuel stack, and that no power spike penalty should be included in the safety analysis.

f. Limiting Power Distributions

According to the ANSI classification of plant conditions (see Chapter 15), Condition I occurrences are those which are expected frequently or regularly in the course of power operation, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during Condition I operation.

The list of steady state and shutdown conditions, permissible deviations, and operational transients is given in Chapter 15. Implicit in the definition of normal operation is proper and timely action by the reactor operator. That is, the operator follows recommended operating procedures for maintaining appropriate power distributions and takes any necessary remedial actions when alerted to do so by the plant instrumentation. Thus, as stated above, the worst or limiting power distribution which can occur during normal operation is to be considered as the starting point for analysis of ANSI Condition II, III and IV events.

Improper procedural actions or errors by the operator are assumed in the design as occurrences of moderate frequency (ANSI Condition II). Some of the consequences which might result are listed in Chapter 15. Therefore, the limiting power shapes which result from such Condition II events are those power shapes which deviate from the normal operating condition at the recommended axial offset band; e.g., due to lack of proper action by the operator during a xenon transient following a change in power level brought about by control rod motion. Power shapes which fall in this category are used for determination of the reactor protection system setpoints so as to maintain margin to overpower or DNB limits.

The means for maintaining power distributions within the required hot channel factor limits are described in the Surveillance and Action requirements of Technical Specifications.

The calculations used to establish the limits on core power distribution are described in Reference 15. All of the nuclear effects which influence the radial and/or axial power distributions throughout core life for various modes of operation, including load follow, reduced power operation, and axial xenon transients are considered.

Power distributions are calculated for the full power condition and reduced power operation with fuel and moderator temperature feedback effects included. The steady state nuclear design calculations are done for normal flow with the same mass flow in each channel. Flow redistribution is calculated explicitly where it is important in the DNB analysis of accidents. The effect of xenon on radial power distribution is small (compare Figure 4.3-6 and Figure 4.3-7) but is included as part of the normal design process.

The core average axial profile can experience significant changes which can occur rapidly as a result of rod motion and load changes and more slowly due to xenon distribution. For the study of points of closest approach to axial power distribution limits, several thousand cases are examined. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries on the limits of interest can be set in terms of the parameters which are readily observed on the plant. Specifically, the nuclear design parameters which are significant to the axial power distribution analysis are:

1. Core power level
2. Core height
3. Coolant temperature and flow
4. Coolant temperature program as a function of reactor power
5. Fuel cycle lifetimes
6. Rod bank worths
7. Rod bank overlaps

Normal operation of the plant assumes compliance with the following conditions:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 12 steps (indicated) from the bank demand position;
2. Control banks are sequenced with overlapping banks;
3. The control bank insertion limits are not violated; and
4. Axial power distribution procedures, which are given in terms of flux difference control and control bank position, are observed.

The axial power distribution procedures referred to above are part of the required operating procedures which are followed in normal operation. Briefly, they require control of the axial offset (flux difference divided by fractional power) at all power levels within a permissible operating band.

Calculations are performed for normal operation of the reactor, including axial xenon transients. Beginning, middle and end-of-cycle conditions are included in the calculations. These cases represent many possible reactor states in the life of one fuel cycle, and they have been chosen as sufficiently definitive of the cycle. It is not possible to single out any transient or steady-state condition which defines the most limiting case. The process of generating a myriad of power distributions is essential to the philosophy that leads to the required level of confidence for the level of protection provided by the core thermal limit protection function setpoints and core power distribution limits.

The calculated power distributions are the result of power level and control rod configurations run with reconstructed axial xenon distributions. The specific xenon distributions are preconditioned by the presence of control rods and then allowed to redistribute for several hours. A detailed discussion of the method used to generate allowable xenon conditions may be found in Reference 15.

The envelope drawn over the calculated max (F_Q x Power) points in Figure 4.3-21 represents an upper bound envelope on local power density versus elevation in the core. The calculated values have been increased by the nuclear uncertainty factor F^N_U for conservatism and a factor of 1.03 for the engineering factor F^E_Q. It should be emphasized that this envelope is a conservative representation of the bounding values of local power density. Expected values are considerably smaller.

Allowing for fuel densification effects, the average linear power at a maximum analyzed power level of 3659 MWt is 5.84 kW/ft. From Figure 4.3-21, the conservative upper bound value of normalized local power density, including uncertainly allowances, is 2.50 corresponding to a peak linear power of 14.6 kW/ft at full power.

The confirmation of protection system setpoints with respect to power distributions is described in Reference 15. In evaluating the required setpoints the core is assumed to be operating within the four constraints described above.

The required Overpower ΔT and Overtemperature ΔT reactor trip setpoints as a function of power and flux difference are cycle dependent. Setpoints for a typical reload core are shown in Figure 4.3-22 and Figure 4.3-23. The peak power density which can occur in the core assuming reactor trip at the Overpower ΔT reactor trip setpoint is less than that required for center-line melt including uncertainties. Similarly, assuming the reactor is tripped at the Overtemperature ΔT setpoint, the minimum DNBR during events for which the Overtemperature ΔT provides protection will be greater than the safety analysis limit value.

It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error. Additional detailed discussion of these analyses is presented in Reference 15.

F_Q can be increased with decreasing power as shown in the Technical Specifications. Increasing $F_{\Delta H}$ with decreasing power is also permitted. The allowance for increased $F_{\Delta H}$ permitted is cycle-dependent and shown in the Core Operating Limits Report. The allowed increase for a typical reload core is shown in Figure 4.3-26.

Typical radial factors and radial power distributions are shown in Figure 4.3-6, Figure 4.3-7, Figure 4.3-8, Figure 4.3-9, Figure 4.3-10 and Figure 4.3-11. The worst values generally occur when the rods are assumed to be at their insertion limits.

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the precondition for a subsequent hypothetical accident, but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition. These alarms are described in detail in Chapters 7 and 16.

g. Experimental Verification of Power Distribution Analysis

This subject is discussed in depth in Reference 2. A summary of this report is given below. It should be noted that power distribution-related measurements are incorporated into the evaluation of calculated power distribution information using the FINC code described in Reference 8. The measured versus calculational comparison is normally performed periodically throughout the cycle lifetime of the reactor as required by Technical Specifications.

In a measurement of the heat flux hot channel factor, F_Q, with the incore detector system described in Subsections 7.7.1 and 4.4.6, the following uncertainties have to be considered:

1. Reproducibility of the measured signal
2. Errors in the physics analytical methods employed in inferring the power distribution.
3. Errors in the calculated relationship between detector flux and peak rod power some distance from the measurement thimble.
4. Errors in constructing an axial power profile from five fixed points.

The appropriate allowance for category 1 above has been quantified by repetitive measurements made with the Incore Detector System. This system stores data every minute, thus the reproducibility of the detector's signal can be determined by monitoring the signals over time with the core in steady state. Local power distribution predictions are verified in critical experiments on arrays of rods with simulated guide thimbles, control rods, burnable poisons, etc. These critical experiments provide quantification of errors of types 2 and 3 above. Errors in category 3 above are quantified to the extent possible, by using the comparisons of data measured and predicted over 22 full core measurements. Axial power construction was verified by direct measurements of the incore axial neutron flux profile to the predictions of the analytical prediction of that profile. As well as comparisons of axial offset determined from both the fixed incore detector system and other means.

Reference 8 describes the foundations and results of the uncertainty analysis, along with comparisons to data collected with the movable detector system. The report concludes that the uncertainty associated with F_Q (heat flux) is 5.21 percent at the 95 percent confidence level with only 5 percent of the measurements greater than the inferred value. This is the equivalent of a 1.645σ limit on a normal distribution and is the uncertainty to be associated with a full core flux map with fixed detectors reduced with a reasonable set of input data incorporating the influence of burnup on the radial power distribution.

In comparing measured power distributions (or detector currents) against the calculations for the same situation, it is not possible to subtract out the detector reproducibility. Thus a comparison between measured and predicted power distributions has to include some measurement error. Such a comparison is given in Figure 4.3-24 for one of the maps used in Reference 8. The report results confirm the adequacy of the 5.21 percent uncertainty allowance on the calculated F_Q.

A similar analysis for the uncertainty in $F^N_{\Delta H}$ (rod integral power) measurements results in an allowance of 4.12 percent at the equivalent of a 1.645σ confidence level. For historical reasons, an 8 percent uncertainty factor is allowed in the nuclear design calculational basis; that is, the predicted rod integrals at full power must not exceed the design $F^N_{\Delta H}$ less 8 percent.

The accumulated data on power distributions in actual operation is basically of three types:

1. Much of the data is obtained in steady state operation at constant power in the normal operating configuration;
2. Data with unusual values of axial offset are obtained as part of the excore detector calibration exercise which is performed quarterly;
3. Special tests have been performed in load-follow and other transient xenon conditions which have yielded useful information on power distributions.

These data are presented in detail in Reference 2.

h. Testing

A very extensive series of physics tests is performed on the first cores. These tests and the criteria for satisfactory results are described in detail in Chapter 14. Since not all limiting situations can be created at BOL, the main purpose of the tests is to provide a check on the calculational methods used in the predictions for the conditions of the tests. Tests performed at the beginning of each reload cycle are limited to verification of steady state power distributions, on the assumption that the reload fuel is supplied by the first core designer.

i. Monitoring Instrumentation

The adequacy of instrument numbers, spatial deployment, required correlations between readings and peaking factors, calibration and errors are described in Reference 8. The relevant conclusions are summarized here in Subsections 4.3.2.2g and 4.4.6.1.

Provided the limitations given in Subsection 4.3.2.2f on rod insertion and flux difference are observed, the Excore Detector System provides adequate online monitoring of power distributions. Further details of specific limits on the observed rod positions and power distributions are given in the Technical Specifications together with a discussion of their bases.

Limits for alarms, reactor trip, etc., are given in the Technical Specifications. Descriptions of the systems provided are given in Section 7.7.

4.3.2.3 Reactivity Coefficients

The kinetic characteristics of the reactor core determine the response of the core to changing plant conditions or to operator adjustments made during normal operation, as well as the core response during abnormal or accidental transients. These kinetic characteristics are quantified in reactivity coefficients. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions such as power, moderator or fuel temperatures, or less significantly due to a change in pressure or void conditions. Since reactivity coefficients change during the life of the core, ranges of coefficients are employed in transient analysis to determine the response of the plant throughout life. The results of such simulations and the reactivity coefficients used are presented in Chapter 15. The reactivity coefficients are calculated on a corewise basis by advanced nodal analysis methods. The effects of radial and axial power distribution on core average reactivity coefficients is implicit in those calculations and is not significant under normal operating conditions. For example, a skewed xenon distribution which results in changing axial offset by 5 percent changes the moderator and Doppler temperature coefficients by less than 0.01 pcm/°F and 0.03 pcm/°F, respectively. An artificially skewed xenon distribution which results in changing the radial $F^N_{\Delta H}$ by 3 percent changes the moderator and Doppler temperature coefficients by less than 0.03 pcm/°F and 0.001 pcm/°F, respectively. The spatial effects are accentuated in some transient conditions; for example, in postulated rupture of a main steam line and rupture of rod cluster control assembly mechanism housing described in Subsections 15.1.5 and 15.4.8, and are included in these analyses.

The analytical methods and calculational models used in calculating the reactivity coefficients are given in Subsection 4.3.3. These models have been confirmed through extensive testing of more than thirty cores similar to the plant described herein; results of these tests are discussed in Subsection 4.3.3.

Quantitative information for calculated reactivity coefficients, including fuel-Doppler coefficient, moderator coefficients (density, temperature, pressure, void) and power coefficient is given in the following sections.

a. Fuel Temperature (Doppler) Coefficient

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature and is primarily a measure of the Doppler broadening of U-238 and Pu-240 resonance absorption peaks. Doppler broadening of other isotopes such as U-236, Np-237 etc., are also considered but their contributions to the Doppler effect is small. An increase in fuel temperature increases the effective resonance absorption cross sections of the fuel and produces a corresponding reduction in reactivity.

The fuel temperature coefficient is calculated by performing calculations using the SIMULATE-3 code (Reference 2) or the ANC code (Reference 12). Moderator temperature reactivity changes are removed as the power level is varied. Spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of power density as discussed in Subsection 4.3.3.1.

The Doppler temperature coefficient is shown in Figure 4.3-27 as a function of the effective fuel temperature (at BOL and EOL conditions). The effective fuel temperature is lower than the volume averaged fuel temperature since the neutron flux distribution is nonuniform through the pellet and gives preferential weight to the surface temperature. The Doppler-only contribution to the power coefficient, defined later, is shown in Figure 4.3-28 as a function of relative core power. The integral of the differential curve on Figure 4.3-28 is the Doppler contribution to the power defect and is shown in Figure 4.3-29 as a function of relative power. The Doppler coefficient becomes more negative as a function of life as the Pu-240 content increases, thus increasing the Pu-240 resonance absorption, but overall becomes less negative since the fuel temperature changes with burnup as described in Subsection 4.3.3.1. The upper and lower limits of Doppler coefficient used in accident analyses are given in Chapter 15.

b. Moderator Coefficients

The moderator coefficient is a measure of the change in reactivity due to a change in specific coolant parameters such as density, temperature, pressure or void. The coefficients so obtained are moderator density, temperature, pressure and void coefficients.

1. Moderator Density and Temperature Coefficients

The moderator temperature (density) coefficient is defined as the change in reactivity per unit change in the moderator temperature. Generally, the effect of the changes in moderator density as well as the temperature are considered together. A decrease means less moderation which results in a negative moderation coefficient. An increase in coolant temperature, keeping the density constant, leads to a hardened neutron spectrum and results in an increase in resonance absorption in U-238, Pu-240 and other isotopes. The hardened spectrum also causes a decrease in the fission to capture ratio in U-235 and Pu-239. Both of these effects make the moderator coefficient more negative. Since water density changes more rapidly with temperature as temperature increases, the moderator temperature coefficient becomes more negative with increasing temperature.

The soluble boron used in the reactor as a means of reactivity control also has an effect on moderator temperature coefficient since the soluble boron poison density as well as the water density is decreased when the coolant temperature rises. A decrease in the soluble poison concentration introduces a positive component in the moderator temperature coefficient.

Thus, if the concentration of soluble poison is large enough, the net value of the coefficient may be positive. With the burnable poison rods present, however, the initial hot boron concentration is sufficiently low that the moderator temperature coefficient is negative at power operating conditions above 20% power*. The effect of control rods is to make the moderator coefficient more negative by reducing the required soluble boron concentration and by increasing the "leakage" of the core.

With burnup, the moderator temperature coefficient becomes more negative primarily as a result of boric acid dilution but also to an extent from the effects of the buildup of plutonium and fission products.

The moderator coefficient is calculated for the various plant conditions discussed above by performing two-group nodal calculations, varying the moderator temperature (and density) by about ±5°F about each of the mean temperatures. The moderator coefficient is shown as a function of core temperature and boron concentration for the unrodded and rodded core in Figure 4.3-30, Figure 4.3-31 and Figure 4.3-32. The temperature range covered is from cold (68°F) to about 600°F. The contribution due to Doppler coefficient (because of change in moderator temperature) has been subtracted from these results. Figure 4.3-33 shows the hot, full power moderator temperature coefficient plotted as a function of cycle lifetime for the just critical boron concentration condition based on the design boron letdown condition.

The moderator coefficients presented here are calculated on a corewide basis, since they are used to describe the core behavior in normal and accident situations when the moderator temperature changes can be considered to affect the entire core.

---

* Note: A non-negative moderator temperature coefficient is allowed by Technical Specifications for all power levels, provided that compliance with the ATWS Rule and its basis are maintained, as described in the Bases for Technical Specification 3/4.1.1.3. The Seabrook core design philosophy meets this requirement by ensuring that a non-positive MTC exists for operating conditions above 20% power.

2. Moderator Pressure Coefficient

The moderator pressure coefficient relates the change in moderator density, resulting from a reactor coolant pressure change, to the corresponding effect on neutron production. This coefficient is of much less significance in comparison with the moderator temperature coefficient. A change of 50 psi in pressure has approximately the same effect on reactivity as a half degree change in moderator temperature. This coefficient can be determined from the moderator temperature coefficient by relating change in pressure to the corresponding change in density.

3. Moderator Void Coefficient

The moderator void coefficient relates the change in neutron multiplication to the presence of voids in the moderator. In a PWR, this coefficient is not very significant because of the low void content in the coolant. The core void content is less than one-half of one percent and is due to local or statistical boiling. The void coefficient can be determined from the moderator temperature coefficient by relating change in void to corresponding change in density.

c. Power Coefficient

The combined effect of moderator and fuel temperature change as the core power level changes is called the total power coefficient and is expressed in terms of reactivity change per percent power change. The power coefficient at BOL and EOL conditions is given in Figure 4.3-34.

It becomes more negative with burnup reflecting the combined effect of moderator and fuel temperature coefficients with burnup. The power defect (integral reactivity effect) at BOL and EOL is given in Figure 4.3-35.

d. Comparison of Calculated and Experimental Reactivity Coefficients

Subsection 4.3.3 describes the comparison of calculated and experimental reactivity coefficients in detail. Based on the data presented there, the accuracy of the current analytical model is:

1. ±0.2 percent $\Delta\rho$ for Doppler defect
2. ±2 pcm/°F for the moderator coefficient

Experimental evaluation of the calculated coefficients will be completed during the physics tests described in Chapter 14.

e. Reactivity Coefficients Used in Transient Analysis

Table 4.3-2 gives the limiting values as well as the best estimate values for the reactivity coefficients. The limiting values are used as design limits in the transient analysis. The exact values of the coefficient used in the analysis depend on whether the transient of interest is examined at the BOL or EOL, whether the most negative or the most positive (least negative) coefficients are appropriate, and whether spatial nonuniformity must be considered in the analysis. Conservative values of coefficients, considering various aspects of analysis are used in the transient analysis. This is completely described in Chapter 15.

The reactivity coefficients shown in Figure 4.3-27, Figure 4.3-28, Figure 4.3-29, Figure 4.3-30, Figure 4.3-31, Figure 4.3-32, Figure 4.3-33, Figure 4.3-34 and Figure 4.3-35 are best estimate values calculated for this cycle and apply to the core described in Table 4.3-1. The limiting values shown in Table 4.3-2 are chosen to encompass the best estimate reactivity coefficients, including the uncertainties given in Subsection 4.3.3.3 over appropriate operating conditions calculated for this cycle and the expected values for the subsequent cycles. The most positive as well as the most negative values are selected to form the design basis range used in the transient analysis. A direct comparison of the best estimate and design limit values shown in Table 4.3-2 can be misleading since in many instances, the most conservative combination of reactivity coefficients is used in the transient analysis even though the extreme coefficients assumed may not simultaneously occur at the condition of lifetime, power level, temperature and boron concentration assumed in the analysis. The need for re-evaluation of any accident in a subsequent cycle is contingent upon whether or not the coefficients for that cycle fall within the identified range used in the analysis presented in Chapter 15 with due allowance for the calculational uncertainties given in Subsection 4.3.3.3. Control rod requirements are given in Table 4.3-3 for the core described and for a hypothetical equilibrium cycle since these are markedly different. These latter numbers are provided for information only and their validity in a particular cycle would be an unexpected coincidence.

4.3.2.4 Control Requirements

To ensure the shutdown margin stated in the Technical Specifications and the Core Operating Limits Report under conditions where a cooldown to ambient temperature is required, concentrated soluble boron is added to the coolant. Boron concentrations for several core conditions are listed in Table 4.3-2. For all core conditions including refueling, the boron concentration is well below the solubility limit. The Rod Cluster Control Assemblies are employed to bring the reactor to the hot shutdown condition. The minimum required shutdown margin is given in the Technical Specifications.

The ability to accomplish the shutdown for hot conditions is demonstrated in Table 4.3-3 by comparing the difference between the Rod Cluster Control Assembly reactivity available with an allowance for the worst stuck rod with that required for control and protection purposes. The shutdown margin includes an allowance of 10 percent for analytic uncertainties (see Subsection 4.3.2.4i). The largest reactivity control requirement appears at the EOL when the moderator temperature coefficient reaches its peak negative value as reflected in the larger power defect.

The control rods are required to provide sufficient reactivity to account for the power defect from full power to zero power and to provide the required shutdown margin. The reactivity addition resulting from power reduction consists of contributions from Doppler, variable average moderator temperature, flux redistribution, and reduction in void content as discussed below.

a. Doppler

The Doppler effect arises from the broadening of U-238 and Pu-240 resonance peaks with an increase in effective pellet temperature. This effect is most noticeable over the range of zero power to full power due to the large pellet temperature increase with power generation.

b. Variable Average Moderator Temperature

When the core is shutdown to the hot, zero power condition, the average moderator temperature changes from the equilibrium full load value determined by the steam generator and turbine characteristics (steam pressure, heat transfer, tube fouling, etc.) to the equilibrium no load value, which is based on the steam generator shell side design pressure. The design change in temperature is conservatively increased by 6°F to account for the control dead band and measurement errors.

c. Redistribution

During full power operation, the coolant density decreases with core height, and this, together with partial insertion of control rods, results in less fuel depletion near the top of the core. Under steady state conditions, the relative power distribution will be slightly asymmetric towards the bottom of the core. On the other hand, at Hot Zero Power conditions, the coolant density is uniform up the core, and there is no flattening due to the Doppler. The result will be a flux distribution which at zero power can be skewed toward the top of the core. The reactivity insertion due to the skewed distribution is calculated with an allowance for effects of xenon distribution.

d. Void Content

A small void content in the core is due to nucleate boiling at full power. The void collapse coincident with power reduction makes a small reactivity contribution.

e. Rod Insertion Allowance

At full power, the control bank is operated within a prescribed band of travel to compensate for small periodic changes in boron concentration, changes in temperature and very small changes in the xenon concentration not compensated for by a change in boron concentration. When the control bank reaches either limit of this band, a change in boron concentration is required to compensate for additional reactivity changes. Since the insertion limit is set by a rod travel limit, a conservatively high calculation of the inserted worth exceeds the normally inserted reactivity.

f. Burnup

Excess reactivity of 10 percent $\Delta\rho$ (hot) is installed at the beginning of each cycle to provide sufficient reactivity to compensate for fuel depletion and fission products throughout the cycle. This reactivity is controlled by the addition of soluble boron to the coolant and by burnable poison. The soluble boron concentration for several core configurations, the unit boron worth, and burnable poison worth are given in Table 4.3-1 and Table 4.3-2. Since the excess reactivity for burnup is controlled by soluble boron and/or burnable poison, it is not included in control rod requirements.

g. Xenon and Samarium Poisoning

Changes in xenon and samarium concentrations in the core occur at a sufficiently slow rate, even following rapid power level changes, that the resulting reactivity change is controlled by changing the soluble boron concentration.

h. pH Effects

Changes in reactivity due to a change in coolant pH, if any, are sufficiently small in magnitude and occur slowly enough to be controlled by the boron system. Further details are available in Reference 9.

i. Experimental Confirmation

Following a normal shutdown, the total core reactivity change during cooldown with a stuck rod has been measured on a 121 assembly, 10-foot high core and 121 assembly, 12-foot high core. In each case, the core was allowed to cool down until it reaches criticality simulating the steamline break accident. For the 10-foot core, the total reactivity change associated with the cooldown is over-predicted by about 0.3 percent $\Delta\rho$ with respect to the measured result. This represents an error of about 5 percent in the total reactivity change and is about half the uncertainty allowance for this quantity. For the 12-foot core, the difference between the measured and predicted reactivity change was an even smaller 0.2 percent $\Delta\rho$. These measurements and others demonstrate the ability of the methods described in Subsection 4.3.3.

j. Control

Core reactivity is controlled by means of a chemical poison dissolved in the coolant, Rod Cluster Control Assemblies, and burnable absorber fuel rods as described below.

k. Chemical Poison

Boron in solution as boric acid is used to control relatively slow reactivity changes associated with:

1. The moderator temperature defect in going from cold shutdown at ambient temperature to the hot operating temperature at zero power,
2. The transient xenon and samarium poisoning, such as that following power changes or changes in rod cluster control position,
3. The excess reactivity required to compensate for the effects of fissile inventory depletion and buildup of long-life fission products,
4. The burnable absorber fuel rod depletion.

The boron concentration for various core conditions is presented in Table 4.3-2.

l. Rod Cluster Control Assemblies

Full length Rod Cluster Control Assemblies exclusively are employed in this reactor.

The number of respective full length assemblies is shown in Table 4.3-1. The full length Rod Cluster Control Assemblies are used for shutdown and control purposes to offset fast reactivity changes associated with:

1. The required shutdown margin in the Hot Zero Power, stuck rod condition,
2. The reactivity compensation as a result of an increase in power above Hot Zero Power (power defect including Doppler, and moderator reactivity changes),
3. Unprogrammed fluctuations in boron concentration, coolant temperature or xenon concentration (with rods not exceeding the allowable rod insertion limits),
4. Reactivity ramp rates resulting from load changes.

The allowed full length control bank insertion is limited at full power to maintain shutdown capability. As the power level is reduced, control rod reactivity requirements are also reduced and more rod insertion is allowed. The control bank position is monitored and the operator is notified by an alarm if the limit is approached. The determination of the insertion limit uses conservative xenon distributions and axial power shapes. In addition, the rod cluster control assembly withdrawal pattern determined from these analyses is used in determining power distribution factors and in determining the maximum worth of an inserted rod cluster control assembly ejection accident. For further discussion, refer to the Technical Specifications on rod insertion limits.

Power distribution, rod ejection and rod misalignment analyses are based on the arrangement of the shutdown and control groups of the Rod Cluster Control Assemblies shown in Figure 4.3-36. All shutdown Rod Cluster Control Assemblies are withdrawn before withdrawal of the control banks is initiated. In going from zero to 100 percent power, control banks A, B, C and D are withdrawn sequentially. The limits of rod positions and further discussion on the basis for rod insertion limits are provided in the Technical Specifications and the Core Operating Limits Report.

m. Reactor Coolant Temperature

Reactor coolant (or moderator) temperature control has added flexibility in reactivity control of the Westinghouse PWR. This feature takes advantage of the negative moderator temperature coefficient inherent in a PWR to:

1. Maximize return to power capabilities

2. Provide ±5 percent power load regulation capabilities without requiring control rod compensation

3. Extend the time in cycle life to which daily load follow operation can be accomplished.

Reactor coolant temperature control supplements the dilution capability of the plant by lowering the reactor coolant temperature to supply positive reactivity through the negative moderator coefficient of the reactor. After the transient is over, the system automatically recovers the reactor coolant temperature to the programmed value.

Moderator temperature control of reactivity, like soluble boron control, has the advantage of not significantly affecting the core power distribution. However, unlike boron control, temperature control can be rapid enough to achieve reactor power change rates of 5 percent/minute.

n. Integral Fuel Burnable Absorber Rods

Integral Fuel Burnable Absorber (IFBA) rods provide partial control of the excess reactivity available during the beginning of the fuel cycle. In doing so, these rods prevent the moderator temperature coefficient from being positive at normal operating conditions above 20% power. They perform this function by reducing the requirement of soluble poison in the moderator at the beginning of the fuel cycle as described previously. For purposes of illustration, a typical IFBA rod pattern in the core together with the number of rods per assembly are shown in Figure 4.3-5, while the arrangements within an assembly are displayed in Figure 4.3-4. The reactivity worth of these rods is shown in Table 4.3-1. The boron in the rods is depleted with burnup but at a sufficiently slow rate so that the resulting critical concentration of soluble boron is such that the moderator temperature coefficient remains negative at all times for power operating conditions above 20% power*.

* Note: A non-negative moderator temperature coefficient is allowed by Technical Specifications for all power levels, provided that compliance with the ATWS Rule and its basis are maintained, as described in the Bases for Technical Specification 3/4.1.1.3. The Seabrook core design philosophy meets this requirement by ensuring that a non-positive MTC exists for operating conditions above 20% power.

o. Peak Xenon Startup

Compensation for the peak xenon buildup is accomplished using the Boron Control System. Startup from the peak xenon condition is accomplished with a combination of rod motion and boron dilution. The boron dilution may be made at any time, including during the shutdown period, provided the shutdown margin is maintained.

p. Load Follow Control and Xenon Control

During load follow maneuvers, power changes are accomplished using control rod motion and dilution or boration by the boron system as required. Control rod motion is limited by the control rod insertion limits on full length rods as provided in the Technical Specifications and discussed in Subsections 4.3.2.4l and 4.3.2.4m. The power distribution is maintained within acceptable limits through the location of the full length rod bank. Reactivity changes due to the changing xenon concentration can be controlled by rod motion and/or changes in the soluble boron concentration. Late in cycle life, extended load follow capability is obtained by augmented the limited boron dilution capability at low soluble boron concentration by temporary moderator temperature reductions.

Rapid power increases (5 percent/min) from part power load follow operation are accomplished with a combination of rod motion, moderator temperature reduction, and boron dilution. Compensation for the rapid power increase is accomplished initially by a combination of rod withdrawal and moderator temperature reduction. As the slower boron dilution takes affect after the initial rapid power increase, the moderator temperature returns to the programmed value.

q. Burnup

Control of the excess reactivity for burnup is accomplished using soluble boron and/or burnable poison. The boron concentration must be limited during operating conditions to ensure the moderator temperature coefficient is negative. Sufficient burnable poison is installed at the beginning of a cycle to give the desired cycle lifetime without exceeding the boron concentration limit. The practical minimum boron concentration is 10 ppm.

4.3.2.5 Control Rod Patterns and Reactivity Worth

The full length Rod Cluster Control Assemblies are designated by function as the control groups and the shutdown groups. The terms "group" and "bank" are used synonymously throughout this report to describe a particular grouping of control assemblies. The rod cluster assembly pattern is displayed in Figure 4.3-36, which is not expected to change during the life of the plant. The control banks are labeled A, B, C, and D and the shutdown banks are labeled SA, SB, etc., as applicable. Each bank, although operated and controlled as a unit, is comprised of two subgroups. The axial position of full length Rod Cluster Control Assemblies may be controlled manually or automatically. The Rod Cluster Control Assemblies are all dropped into the core following actuation of reactor trip signals.

Two criteria have been employed for selection of the control groups. First, the total reactivity worth must be adequate to meet the requirements specified in Table 4.3-3. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to ensure that the power capability requirements are met. Analyses indicate that the first requirement can be met either by a single group or by two or more banks whose total worth equals at least the required amount. The axial power shape would be more peaked following movement of a single group of rods worth three to four percent $\Delta\rho$; therefore, four banks (described as A, B, C, and D in Figure 4.3-36) each worth approximately one percent $\Delta\rho$ have been selected.

The position of control banks for criticality under any reactor condition is determined by the concentration of boron in the coolant. On an approach to criticality, boron is adjusted to ensure that criticality will be achieved with control rods above the insertion limit set by shutdown and other considerations (see the Technical Specifications and the Core Operating Limits Report).

Ejected rod worths are given in Subsection 15.4.8 for several different conditions.

Allowable deviations due to misaligned control rods are noted in the Technical Specifications.

A representative calculation for two banks of control rods withdrawn simultaneously (rod withdrawal accident) is given in Figure 4.3-37.

Calculation of control rod reactivity worth versus time following reactor trip involves both control rod velocity and differential reactivity worth. The rod position versus time of travel after rod release, assumed is given in Figure 4.3-38. For nuclear design purposes, the reactivity worth versus rod position is calculated by a series of steady state calculations at various control rod positions assuming all rods out of the core as the initial position in order to minimize the initial reactivity insertion rate. Also to be conservative, the rod of highest worth is assumed stuck out of the core and the flux distribution (and thus reactivity importance) is assumed to be skewed to the bottom of the core. The result of these calculations is shown in Figure 4.3-39.

The shutdown groups provide additional negative reactivity to assure an adequate shutdown margin. Shutdown margin is defined as the amount by which the core would be subcritical at hot shutdown if all Rod Cluster Control Assemblies are tripped, but assuming that the highest worth assembly remains fully withdrawn and no changes in xenon or boron take place. The loss of control rod worth due to the material irradiation is negligible since only bank D and bank C may be in the core under normal operating conditions (near full power).

The values given in Table 4.3-3 show that the available reactivity in withdrawn rod cluster control assemblies provides the design bases minimum shutdown margin allowing for the highest worth cluster to be at its fully withdrawn position. An allowance for the uncertainty in the calculated worth of N-1 rods is made before determination of the shutdown margin.

4.3.2.6 Criticality of the Reactor During Refueling and Criticality of Fuel Assemblies

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping and storage facilities and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly and/or inserting neutron poisons between assemblies.

The design basis for preventing criticality outside the reactor is that, considering possible variation, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor, K_{eff}, of the fuel assembly array will be less than 0.95. In areas like the new fuel vault where sources of moderation such as those that could arise during fire fighting operations are included, the maximum design basis K_{eff} is 0.98 under conditions of low density, "optimum moderation." For further description of the criticality safety limits in the new fuel vault and spent fuel pool, see Subsections 9.1.1.3 and 9.1.2.3, respectively.

4.3.2.7 Stability

a. Introduction

The stability of the PWR cores against xenon-induced spatial oscillations and the control of such transients are discussed extensively in References 10, 13, and 14. A summary of these reports is given in the following discussion and the design bases are given in Subsection 4.3.1.6.

In a large reactor core, xenon-induced oscillations can take place with no corresponding change in the total power of the core. The oscillation may be caused by a power shift in the core which occurs rapidly by comparison with the xenon-iodine time constants. Such a power shift occurs in the axial direction when a plant load change is made by control rod motion and results in a change in the moderator density and fuel temperature distributions. Such a power shift could occur in the diametral plane of the core as a result of abnormal control action.

Due to the negative power coefficient of reactivity, PWR cores are inherently stable to oscillations in total power. Protection against total power instabilities is provided by the Control and Protection System as described in Section 7.7. Hence, the discussion on the core stability will be limited here to xenon-induced spatial oscillations.

b. Stability Index

Power distributions, either in the axial direction or in the X-Y plane, can undergo oscillations due to perturbations introduced in the equilibrium distributions without changing the total core power. The overtones in the current PWRs, and the stability of the core against xenon-induced oscillations can be determined in terms of the eigenvalues of the first flux overtones. Writing, either in the axial direction or in the X-Y plane, the eigenvalue of the first harmonic as:

$$\xi = b + ic, \tag{4.3-1}$$

then b is defined as the stability index and $T = 2\pi/c$ as the oscillation period of the first harmonic. The time-dependence of the first harmonic in the power distribution can now be represented as:

$$\delta\varphi(t) = A\, e^{\xi t} = ae^{bt} \cos ct, \tag{4.3-2}$$

where A and a are constants. The stability index can also be obtained approximately by:

$$b = \frac{1}{T} \ln \frac{A_{n+1}}{A_n} \tag{4.3-3}$$

where A_n, A_{n+1} are the successive peak amplitudes of the oscillation and T is the time period between the successive peaks.

c. Prediction of the Core Stability

The stability of the core described herein (i.e., with 17x17 fuel assemblies) against xenon-induced spatial oscillations is expected to be equal to or better than that of earlier designs. The prediction is based on a comparison of the parameters which are significant in determining the stability of the core against the xenon-induced oscillations, namely: (1) the overall core size is unchanged and spatial power distributions will be similar, (2) the moderator temperature coefficient is expected to be similar to or slightly more negative, and (3) the Doppler coefficient of reactivity is expected to be equal to or slightly more negative at full power.

Analysis of both the axial and X-Y xenon transient tests, discussed in Subsection 4.3.2.7e, shows that the calculational model is adequate for the prediction of core stability.

d. Stability Measurements

1. Axial Measurements

Two axial xenon transient tests conducted in a PWR with a core height of 12 feet and 121 fuel assemblies are reported in Reference 14, and will be briefly discussed here. The tests were performed at approximately 50 percent and 100 percent of cycle life.

Both transients lasted about 40 hours with the regulating control bank ranging in insertion from 214 steps to 179 steps withdrawn. These maneuvers produced measured axial offsets that ranged from 3.5% to -11.6%. Figure 4.3-40 shows the axial offset as a function of time through these measurements.

The total core power was maintained constant during these spatial xenon tests, and the stability index and the oscillation period were obtained from a least-square fit of the axial offset data in the form of Equation (4.3-2). The axial offset of power is the quantity that properly represents the axial stability in the sense that it essentially eliminates any contribution from even order harmonics including the fundamental mode. The conclusions of the tests are:

(a) The core was stable against induced axial xenon transients both at the core average burnups of 1550 MWd/Mtu and 7700 MWd/Mtu. The measured stability indices are -0.041 hr^{-1} for the first test (Curve 1 of Figure 4.3-40) and -0.014 hr^{-1} for the second test (Curve 2 of Figure 4.3-40). The corresponding oscillation periods are 32.4 hrs and 27.2 hrs, respectively.

(b) The reactor core becomes less stable as fuel burnup progresses and the axial stability index was essentially zero at 12,000 MWd/Mtu.

Additional tests conducted on a PWR 12 foot core with 193 assemblies indicate that full length control rods can be used to dampen axial xenon oscillations effectively.

2. Measurements in the X-Y Plane

Two X-Y xenon oscillation tests were performed at a PWR plant with a core height of 12 feet and 157 fuel assemblies. The first test was conducted at a core average burnup of 1540 MWd/Mtu and the second at a core average burnup of 12,900 MWd/Mtu. Both of the X-Y xenon tests show that the core was stable in the X-Y plane at both burnups. The second test shows that the core became more stable as the fuel burnup increased and all Westinghouse PWRs with 121 and 157 assemblies are expected to be stable throughout their burnup cycles.

In each of the two X-Y tests, a perturbation was introduced to the equilibrium power distribution through an impulse motion of one rod cluster control unit located along the diagonal axis. Following the perturbation, the uncontrolled oscillation was monitored using the moveable detector and thermocouple system and the excore power range detectors. The quadrant tilt difference (QTD) is the quantity that properly represents the diametral oscillation in the X-Y plane of the reactor core in that the differences of the quadrant average powers over two symmetrically opposite quadrants essentially eliminate the contributions to the oscillation from the azimuthal mode. The QTD data were fitted in the form of Equation (4.3-2) through a least-square method. A stability index of -0.076 hr^{-1} with a period of 29.6 hours was obtained from the thermocouple data shown in Figure 4.3-41.

It was observed in the second X-Y xenon test that the PWR core with 157 fuel assemblies had become more stable due to an increased fuel depletion and the stability index was not determined.

e. Comparison of Calculations with Measurements

Analysis of the axial xenon transients above was performed by Westinghouse using its neutronics methods. The results of the stability calculation for the axial tests are compared with the experimental data in Table 4.3-4. The calculations show conservative results for both of the axial tests with a margin of approximately -0.01 hr^{-1} in the stability index.

An analytical simulation of the first X-Y xenon oscillation test shows a calculated stability index of -0.081 hr^{-1}, in good agreement with the measured value of -0.076 hr^{-1}. As indicated earlier, the second X-Y xenon test showed that the core had become more stable compared to the first test and no evaluation of the stability index was attempted. This increase in the core stability in the X-Y plane due to increased fuel burnup is due mainly to the increased magnitude of the negative moderator temperature coefficient.

f. Stability Control and Protection

The Excore Detector System is utilized to provide indications of xenon-induced spatial oscillations. The readings from the multi-section excore detectors are available to the operator and also form part of the protection system.

1. Axial Power Distribution

For maintenance of proper axial power distributions, the operator is instructed to maintain axial power distribution within axial flux difference operating limit band specified in the Core Operating Limits Report, based on the excore detector readings. Should the axial flux difference move outside this band, power level will be reduced by the operators per Technical Specification requirements. If the operators do not reduce power, the protection limit will be reached and the reactor will be tripped.

Twelve foot PWR cores become less stable to axial xenon oscillations as fuel burnup progresses. However, free xenon oscillations are not allowed to occur except for special tests. The full-length control rod banks are sufficient to dampen and control any axial xenon oscillations present. Should the axial flux difference move outside the specified operating limit band due to a axial xenon oscillation, or any other reason, the core power level will be reduced by the operators per Technical Specification requirements, or the protection limit on axial flux difference will be reached and the reactor will be tripped.

2. Radial Power Distribution

The core described herein is calculated to be stable against X-Y xenon induced oscillations at all times in life.

The X-Y stability of large PWRs has been further verified as part of the startup physics test program for cores with 193 fuel assemblies. The measured X-Y stability of the cores with 157 and 193 assemblies was in good agreement with the calculated stability as discussed in Subsections 4.3.2.7d and 4.3.2.7e. In the unlikely event that X-Y oscillations occur, backup actions are possible and would be implemented if necessary, to increase the natural stability of the core as discussed in the Technical Specifications. This is based on the fact that several actions could be taken to make the moderator temperature coefficient more negative, which will increase the stability of the core in the X-Y plane.

Provisions for protection against nonsymmetric perturbations in the X-Y power distribution that could result from equipment malfunctions are made in the protection system design. This includes control rod drop, rod misalignment and asymmetric loss of coolant flow.

4.3.2.8 Vessel Irradiation

A brief review of the methods and analyses used in the determination of neutron and gamma ray flux attenuation between the core and the pressure vessel is given below. A more complete discussion on the pressure vessel irradiation and surveillance program is given in Section 5.3.

The materials that serve to attenuate neutrons originating in the core and gamma rays from both the core and structural components consist of the core baffle, core barrel, neutron pads and associated water annuli, all of which are within the region between the core and the pressure vessel.

In general, few group neutron diffusion theory codes are used to determine fission power density distributions within the active core, and the accuracy of these analyses is verified by incore measurements on operating reactors. Region and rodwise power sharing information from the core calculations is then used as source information in two-dimensional S_n transport calculations which compute the flux distributions throughout the reactor.

The neutron flux distribution and spectrum in the various structural components varies significantly from the core to the pressure vessel. Representative values of the neutron flux distribution and spectrum are presented in Table 4.3-5. The values listed are based on time averaged equilibrium cycle reactor core parameters and power distributions; and, thus, are suitable for long-term nvt projections and for correlation with radiation damage estimates.

As discussed in Section 5.3, the irradiation surveillance program utilizes actual test samples to verify the accuracy of the calculated fluxes at the vessel.

4.3.3 Analytical Methods

Calculations required in nuclear design consist of three distinct types, which are performed in sequence:

1. Determination of effective fuel temperatures
2. Generation of macroscopic few-group parameters
3. Space-dependent, few-group diffusion calculations.

These calculations are carried out by computer codes which can be executed individually; however, most of the codes required have been linked to form an automated design sequence which minimizes design time, avoids errors in transcription of data, and standardizes the design methods.

4.3.3.1 Fuel Temperature (Doppler) Calculations

Temperatures vary radially within the fuel rod, depending on the heat generation rate in the pellet, the conductivity of the materials in the pellet, gap, and clad, and the temperature of the coolant.

The fuel temperatures for use in nuclear design Doppler calculations are obtained from the fuel rod design model described in Subsection 4.2.1.3 which considers the effect of radial variation of pellet conductivity, expansion-coefficient and heat generation rate, elastic deflection of the clad, and a gap conductance which depends on the initial fill gap, the hot open gap dimension, fuel swelling, fission gas release, and plastic clad deformation. Further gap closure occurs with burnup and accounts for the decrease in Doppler defect with burnup.

4.3.3.2 Macroscopic Group Constants

Macroscopic few-group constants and analogous microscopic cross sections (needed for feedback and microscopic depletion calculations) are generated for fuel cells by a recent version of the CASMO or PHOENIX-P (References 1 and 11) code, which provide burnup dependent cross sections. Fast and thermal cross section library tapes contain microscopic cross sections taken from the ENDF/B-VI library, with a few exceptions where other data provided good agreement with critical experiments, isotopic measurements, and plant critical boron values. The effect on the unit fuel cell of nonlattice components in the fuel assembly is obtained by supplying an appropriate volume fraction of these materials in an extra region which is homogenized with the unit cell in the fast and thermal flux calculations. In the thermal calculation, the fuel rod, clad, and moderator are homogenized by energy-dependent disadvantage factors derived from an analytical fit to integral transport theory results.

Group constants for control rods, IFBA rods, guide thimbles, instrument thimbles and interassembly gaps are generated in a manner analogous to the fuel cell calculation. Baffle and reflector group constants are taken from two dimensional PHOENIX-P models of the core and baffle/reflector interface.

Nodal group constants are obtained by a flux-volume homogenization of the fuel cells, burnable poison cells, guide thimbles, instrumentation thimbles, interassembly gaps, and control rod cells from one mesh internal per cell X-Y unit fuel assembly diffusion calculations.

Validation of the cross section method is based on analysis of isotopic data, plant critical boron (C_B) values at HZP, BOL and at HFP as a function of burnup as shown in Reference 11. Control rod worth measurements are also shown in Reference 11.

Confirmatory critical experiments on burnable poisons are described in Reference 1.

4.3.3.3 Spatial Three Dimensional Calculations

Spatial three dimensional calculations consist primarily of two-group advanced nodal calculations using a version of ANC (Reference 12) or SIMULATE (Reference 2). Full three dimensional calculations are performed using four radial nodes per assembly and at least twenty four axial nodes. Pin power reconstruction is performed within the code to determine discrete pin powers and detailed detector reaction rates. The code also contains means to follow the core spectral history to compensate for depletion of nodes not at the general conditions used in generating the cross sections.

Validation of ANC and SIMULATE calculations is associated with the validation of the group constants themselves, as discussed in Subsection 4.3.3.2. Validation of the Doppler calculations is associated with the fuel temperature validation discussed in Subsection 4.3.3.1. Validation of the moderator coefficient calculations is obtained by comparison with plant measurements at Hot Zero Power conditions.

Validation of the spatial codes for calculating power distributions involves the use of incore and excore detectors and is discussed in Subsection 4.3.2.2g.

4.3.4 References

1. A.S. DiGiovine, K.B. Spinney, D.G. Napolitano, J. Pappas, "CASMO-3G Validation", YAEC-1363, April, 1988.
2. A.S. DiGiovine, J.P. Gorski, M.A. Tremblay, "SIMULATE-3 Validation and Verification", YAEC-1659, September, 1998.
3. "Westinghouse Anticipated Transients Without Reactor Trip Analysis", WCAP-8330, August 1974.
4. ATWS Final Rule - Code of Federal Regulations 10 CFR 50.62 and Supplementary Information Package, "Reduction of Risk from Anticipated Transient without Scram (ATWS) Events for Light Water-Cooled Nuclear Power Plants."

5. Oelrich, R. L. and Kersting, P. J., "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," WCAP-13589-A, March 1995.

6. Meyer, R.O., "The Analysis of Fuel Densification," Division of Systems Safety, USNRC, NUREG-0085, July 1976.

7. Guimond, P. J., et al., "Core Thermal Limit Protection Function Setpoint Methodology for Seabrook Station", YAEC-1854-P-A, October 1992.

8. J.P. Gorski, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis", YAEC-1855-P-A, October 1992.

9. Cermek, J. O., et al., "Pressurized Water Reactor pH - Reactivity Effect Final Report," WCAP-3696-8 (EURAEC-2074), October 1968.

10. Moore, J. S., Power Distribution Control of Westinghouse Pressurized Water Reactors," WCAP-7811, December 1971.

11. Nguyen, T. Q., et. al. "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores", WCAP-11596-P-A, June 1998 (Westinghouse Proprietary)

12. Liu, Y. S, Meliksetion, A, Rathkopt, J. A., Little D. C., Nakano, F., Poploski, M. J., "ANC-A Westinghouse Advanced Nodal Computer Code", WCAP-10965-P-A, September, 1986 (Westinghouse Proprietary)

13. Skogen, F. B. and McFarlane, A. F., "Xenon-Induced Spatial Instabilities in Three-Dimensions," WCAP-3680-22 (EURAEC-2116), September 1969.

14. Lee, J. C., et al., "Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor," WCAP-7964, June 1971.

15. Miller, R. W., et al., "Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification," WCAP-10216-P-A, Revision 1A, February 1994.

4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 Design Bases

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer which is compatible with the heat generation distribution in the core such that heat removal by the Reactor Coolant System or the Emergency Core Cooling System (when applicable) assures that the following performances and safety criteria requirements are met:

a. Fuel damage (defined as penetration of the fission product barrier, i.e., the fuel rod clad) is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These will be within the capability of the plant cleanup system, and are consistent with the plant design bases.

b. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged (see above definition) although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

c. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

In order to satisfy the above criteria, the following design bases have been established for the thermal and hydraulic design of the reactor core.

4.4.1.1 Departure From Nucleate Boiling Design Basis

a. Basis

There will be at least 95 percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Conditions I and II events), at 95 percent confidence level.

The design limit DNBR value for RFA is 1.22 for typical cells and 1.22 for thimble cells. For use in the DNB safety analyses, the limit DNBR is conservatively increased to provide DNB margin to offset the effect of rod bow, RCS flow anomaly and any other DNB penalties that may occur, and to provide flexibility in design and operation of the plant. For RFA fuel, Safety Analysis Limit DNBR value of 1.47 for both typical and thimble cells is employed in the analysis.

b. Discussion

By preventing DNB, adequate heat transfer is assured between the fuel clad and the reactor coolant, thereby preventing clad damage as a result of inadequate cooling. Maximum fuel rod surface temperature is not a design basis, as it will be within a few degrees of coolant temperature during operation in the nucleate boiling region.

Limits provided by the nuclear control and protection systems are such that this design basis will be met for transients associated with Condition II events including overpower transients. There is an additional large DNBR margin at rated power operation and during normal operating transients.

The thermal-hydraulic analysis of the RFA (w/IFMs) fuel used in Seabrook Station incorporates the use of the VIPRE-01 computer code and the Revised Thermal Design Procedure (RTDP). The WRB-2M DNB correlation is used for the RFA. The W-3 correlations are still used when conditions are outside the range of theWRB-2 or WRB-2M correlation and applicability of the RTDP.

The WRB-2M DNB correlation is based on rod bundle data and takes advantage of the DNB benefit of reduced grid spacings associated with IFMs. The approval of the NRC that a 95/95 limit DNBR of 1.14 is appropriate for RFA and has been documented.

The W-3 correlation with a 95/95 limit DNBR of 1.30 is used below the fuel assembly first mixing vane grid. The W-3 correlation with a 95/95 limit DNBR of 1.45 is used in the pressure range of 500 to 1000 psia.

With RTDP methodology, variations in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and DNB correlation predictions are considered statistically to obtain the overall DNBR uncertainty factor which is used to define the design limit DNBR that satisfies the DNB design criterion. The criterion is that the probability that DNB will not occur on the most limiting fuel rod is at least 95 percent (at 95 percent confidence level) for any Condition I or II event. Conservative uncertainty values are used to calculate the design limit DNBR. Since the uncertainties are all included in the uncertainty factor, the accident analysis is done with input parameters at their nominal or best-estimate values. RTDP analyses use the minimum measured flow (MMF), equal to thermal design flow (TDF) plus a flow uncertainty. Analyses by standard methods continue to use TDF.

The Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. In the STDP method, the parameters used in analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety analyses input values to give the lowest minimum DNBR. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

4.4.1.2 Fuel Temperature Design Basis

a. Basis

During modes of operation associated with Condition I and Condition II events, there is at least a 95 percent probability that the peak kW/ft fuel rods will not exceed the $U0_2$ melting temperature at the 95 percent confidence level. The melting temperature of $U0_2$ is taken as 5080°F (Reference 1), unirradiated, decreasing 58°F per 10,000 MWd/Mtu exposure.

Design evaluations for Condition I and II events have shown that fuel melting will not occur for achievable local burnups to 75,000 MWd/Mtu, Reference 5. The NRC approved design evaluations up to 60,000 MWd/Mtu in Reference 5 and up to 62,000 MWd/Mtu in Reference 6.

b. Discussion

By precluding $U0_2$ melting, the fuel geometry is preserved and possible adverse effects of molten $U0_2$ on the cladding are eliminated. Cycle-specific values for the peak linear heat generation rate precluding centerline melt are determined as a function of fuel rod average exposure. The determination of these values includes allowance of sufficient margin to accommodate the uncertainties in the thermal evaluations described in Subsection 4.4.2.9a. To preclude fuel centerline melting, these values are observed as an overpower limit for Condition I and II events, and employed as a basis for overpower protection system setpoints.

Fuel rod thermal evaluations are performed at various burnups to assure that this design basis as well as the fuel integrity design bases given in Section 4.2 is met.

4.4.1.3 Core Flow Design Basis

a. Basis

A minimum of 91.7 percent of the thermal flow rate will pass through the fuel rod region of the core and be effective for fuel rod cooling. Coolant flow through the thimble tubes, as well as the leakage from flow paths outside the core including the core barrel-baffle region, are not considered effective for heat removal.

b. Discussion

As noted in section 4.4.1.1, in core cooling evaluations the flow rate entering the reactor vessel is assumed to be the minimum measured flow rate (MMF), when the WRB-2M correlation and RTDP are applicable, and the thermal design flow rate (TDF) otherwise. A maximum of 6.8 percent of the MMF value is allotted as bypass flow. Similarly, a maximum of 8.3 percent of the TDF value is allotted as bypass flow. These values include rod cluster control guide thimble cooling flow for the case of all thimble plug assemblies removed, head cooling flow, baffle leakage, leakage to the vessel outlet nozzle, and the effect of IFM grids.

4.4.1.4 Hydrodynamic Stability Design Basis

Modes of operation associated with Conditions I and II events shall not lead to hydrodynamic instability.

4.4.1.5 Other Considerations

The above design bases, together with the fuel clad and fuel assembly design bases given in Subsection 4.2.1, are sufficiently comprehensive so additional limits are not required.

Fuel rod diametral gap characteristics, moderator-coolant flow velocity and distribution, and moderator void are not inherently limiting. Each of these parameters is incorporated into the thermal and hydraulic models used to ensure the above-mentioned design criteria are met. For instance, the fuel rod diametral gap characteristics change with time (see Subsection 4.2.3.3) and the fuel rod integrity is evaluated on that basis. The effect of the moderator flow velocity and distribution (see Subsection 4.4.2.2) and moderator void distribution (see Subsection 4.4.2.4) are included in the core thermal (VIPRE-01) evaluation and thus affect the design bases.

Meeting the fuel clad integrity criteria covers possible effects of clad temperature limitations. As noted in Subsection 4.2.3.3, the fuel rod conditions change with time. A single clad temperature limit for Condition I or Condition II events is not appropriate since, of necessity, it would be overly conservative. A clad temperature limit is applied to the loss-of-coolant accident (Subsection 15.6.5), control rod ejection accident, and locked rotor accident.

4.4.2 Description

4.4.2.1 Summary Comparison

Values of pertinent parameters, along with critical heat flux ratios, fuel temperatures and linear heat generation rates, are presented in Table 4.4-1 for both the Seabrook Station cycle 10 and the uprate cycles for all coolant loops in service. The thermal and hydraulic analyses cover both an uprate to 3659 MWt and any intermediate uprates between 3411 and 3659 MWt. It is also noted, that in this power capability evaluation there has not been any change in the design criteria. The reactor is still designed to meet the DNB design criterion of Section 4.4.1.1, as well as no fuel centerline melting during normal operation, operational transients and faults of moderate frequency.

All DNB analyses were performed such that the DNBR margins are available for offsetting rod bow penalties, RCS flow anomaly and any other DNB penalties that may occur and for flexibility in design.

Fuel densification has been considered in the DNB and fuel temperature evaluations.

4.4.2.2 Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology

The minimum DNBRs for the rated power, design overpower and anticipated transient conditions are given in Table 4.4-1. The minimum DNBR in the limiting flow channel is usually downstream of the peak heat flux location (hot spot) due to the increased downstream enthalpy rise.

DNBRs are calculated by using the correlation and definitions described in the following Subsections 4.4.2.2a and 4.4.2.2b. The VIPRE-01 computer code (discussed in Subsection 4.4.4.5a) is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is discussed in Subsection 4.4.4.3a (nuclear hot channel factors) and in Subsection 4.4.2.2d (hot channel factors).

a. Departure from Nucleate Boiling Technology

The WRB-2M DNB correlation is used to evaluate critical heat flux in the fuel assemblies. The W-3 or the WRB-2 correlation is used where the WRB-2M correlation is not applicable. These correlations are tested against DNB test data in order to establish correlation limits which satisfy the DNB design basis stated in Section 4.4.1.1.

b. Definition of Departure from Nucleate Boiling Ratio (DNBR)

The DNB heat flux ratio (DNBR) as applied to this design when all flow cell walls are heated, is:

$$DNBR = \frac{q''DNB,N}{q''_{loc}} \qquad (4.4\text{-}1)$$

where:

$$q''DNB,N = \frac{q''DNB,EU}{F} \tag{4.4-2}$$

$q''_{DNB,EU}$ is the uniform DNB heat flux as predicted by the WRB-2M (Reference 9), WRB-2 (References 79 and 80) or W-3 (Reference 8) DNB correlation.

F is the flux shape factor to account for axial heat flux distributions (References 7, 8, and 10) with the "C" term modified as in Reference 3.

q''_{loc} is the actual local heat flux.

A multiplier of 0.88 is applied for all DNB analyses using the W-3 correlation.

The DNBR when a cold wall is present is the same as equation 4.4-1 above when the WRB-2M correlation is applied. When the W-3 correlation is applied, the DNBR is:

$$DNBR = \frac{q''_{DNB,N,CW}}{q''_{loc}} \tag{4.4-4}$$

where:

$$CWF = 1.0 - Ru\left[13.76 - 1.37e^{1.78x} - 4.732\left\{\frac{G}{10^6}\right\}^{-0.0535} - 0.0619\left\{\frac{P}{1000}\right\}^{0.14} - 8.50Dh^{0.107}\right] \tag{4.4-6}$$

and Ru = 1 - De/Dh.

Values of minimum DNBR provided in Table 4.4-1 and Table 4.4-2 are the limiting values obtained by applying the above two definitions of DNBR to the appropriate cell (typical cell with all walls heated, or a thimble cold wall cell with a partial heated wall condition).

c. Mixing Technology

1. Flow Mixing

The subchannel mixing model incorporated in the VIPRE-01 Code and used in reactor design is based on experimental data (Reference 17). The mixing vanes incorporated in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances.

2. Thermal Diffusion

The rate of heat exchange by mixing between flow channels is proportional to the difference in the local mean fluid enthalpy of the respective channels of the local fluid density and flow velocity. The proportionality is expressed by the dimensionless thermal diffusion coefficient (TDC) which is defined as:

$$TDC = \frac{w'}{\rho V a} \qquad (4.4\text{-}7)$$

where:

w' = flow exchange rate per unit length, (lb_m/ft-sec)

ρ = fluid density, lb_m/ft^3

V = fluid velocity, ft/sec

a = lateral flow area between channels per unit length, ft^2/ft

The application of the TDC in the VIPRE-01 analysis for determining the overall mixing effect or heat exchange rate is presented in Reference 81.

As a part of an ongoing research and development program, Westinghouse has sponsored and directed mixing tests at Columbia University (Reference 12). These series of tests, using the "R" mixing vane grid design on 13, 26 and 32 inch grid spacing, were conducted in pressurized water loops at Reynolds numbers similar to that of PWR core under the following single and two phase (subcooled boiling) flow conditions:

Pressure	1500 to 2400 psia
Inlet temperature	332 to 642°F

Mass velocity	1.0 to 3.5×10^6 $lb_m/hr\text{-}ft^2$
Reynolds number	1.34 to 7.45×10^5
Bulk outlet quality	-52.1 to 13.5%

TDC is determined by comparing code predictions with the measured subchannel exit temperatures. Data for 26 inch axial grid spacing are presented in Figure 4.4-1 where the thermal diffusion coefficient is plotted versus the Reynolds number. TDC is found to be independent of Reynolds number, mass velocity, pressure and quality over the ranges tested. The two-phase data (local, subcooled boiling) fell within the scatter of the single-phase data. The effect of two-phase flow on the value of TDC has been demonstrated by Cadek (Reference 12), Rowe and Angle (References 13 and 14), and Gonzalez-Santalo and Griffith (Reference 15). In the subcooled boiling region, the values of TDC were indistinguishable from the single-phase values. In the quality region, Rowe and Angle show that in the case with rod spacing similar to that in PWR reactor core geometry, the value of TDC increased with quality to a point and then decreased, but never below the single-phase value. Gonzalez-Santalo and Griffith showed that the mixing coefficient increased as the void fraction increased.

The data from these tests on the "R" grid showed that a design TDC value of 0.038 (for 26 inch grid spacing) can be used in determining the effect of coolant mixing.

A mixing test program similar to the one described above was conducted at Columbia University for the 17x17 geometry and mixing vane grids on 26 inch spacing (Reference 16). The mean value of TDC obtained from these tests was 0.059, and all data was well above the current design value of 0.038.

Since the actual reactor grid spacing is approximately 20 inches, additional margin is available for this design, as the value of TDC increases as grid spacing decreases (Reference 12).

The inclusion of three intermediate flow mixer grids in the upper span of the RFA (w IFMs) results in a grid spacing of approximately 10 inches. Per Reference 80, a TDC value of 0.038 was chosen as a conservatively low value for use in RFA (w IFMs) to determine the effect of constant mixing in the core thermal performance analysis.

3. Inlet Flow Maldistribution

A conservatively low total core inlet flow is used in VIPRE-01 subchannel analysis. The applicable core inlet flow is reduced by a cycle-specific factor accounting for the effect of inlet flow maldistribution on core thermal performance. Determination of the flow reduction factor is discussed in Subsection 4.4.4.2b.

4. Flow Redistribution

Redistribution of flow in the hot channel resulting from the high flow resistance in the channel due to local or bulk boiling and the effect of the nonuniform power distribution is inherently considered in the VIPRE-01 analysis for every operating condition which is evaluated.

d. Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core average ratios of these quantities. The heat flux hot channel factor considers the local maximum linear heat generation rate at a point (the hot spot), and the enthalpy rise hot channel factor involves the maximum integrated value along a channel (the hot channel).

Each of the total hot channel factors considers a nuclear hot channel factor (see Subsection 4.4.4.3) describing the neutron power distribution and an engineering hot channel factor which allows for fabrication tolerances.

1. Heat Flux Engineering Hot Channel Factor, F_Q^E

The heat flux engineering hot channel factor is used to evaluate the maximum heat flux. This subfactor has a value of 1.03 and is determined by statistically combining the tolerances for the fuel pellet diameter, density, enrichment, and burnable absorber. Measured manufacturing data on Westinghouse 17x17 fuel were used to verify that this value was not exceeded for 95 percent of the limiting fuel rods at a 95 percent confidence level. Thus, it is expected that a statistical sampling of the fuel assemblies of this plant will yield a value no larger than 1.03. As shown in Reference 30, no DNB penalty needs to be taken for the relatively low intensity heat flux spikes caused by variations in the above parameter as well as fuel pellet eccentricity and fuel rod diameter variations.

2. Enthalpy Rise Engineering Hot Channel Factor, $F^{E}_{\Delta H}$

The effect of fabrication tolerances on the hot channel enthalpy rise is also considered in the core thermal subchannel analysis. The development of the WRB-2M DNBR design limit used with the RTDP included consideration of the fabrication tolerances for density, enrichment, and burnable absorber.

Values employed in the analysis related to the above fabrication variations are based on applicable limiting tolerances, such that design values are met for 95 percent of the limiting channels at a 95 percent confidence level. Measured manufacturing data on Westinghouse 17x17 fuel show the tolerances used are conservative. In addition, each fuel assembly is checked to assure the channel spacing design criteria are met.

When the W-3 or WRB-2 correlations are employed the effect of fabrication variations is applied in the VIPRE-01 analysis as a direct multiplier on the hot channel enthalpy rise.

e. Effects of Rod Bow on DNBR

The phenomenon of fuel rod bowing, as described in Reference 83, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR are used to offset the effect of rod bow.

For the safety analysis of Seabrook Unit 1, sufficient DNBR margin was maintained to accommodate the maximum full and low flow rod bow DNBR penalties which are based on the methodology in Reference 4.

The maximum rod bow penalty accounted for in the design safety analysis is based on an assembly average burnup of 24,000 MWd/Mtu. At burnups greater than 24,000 MWd/Mtu, credit is taken for the effect of F_H burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required (Reference 85).

In the upper spans of the RFA (w IFMs) fuel assembly, additional restraint is provided with the intermediate flow mixer grids such that the grid-to-grid spacing in those spans with IFM grids is approximately 10 inches compared to approximately 20 inches in the other spans. Using the NRC approved scaling factor results in predicted channel closure in the limiting 10-inch spans of less than 50-percent closure. Therefore, no rod bow DNBR penalty is required in the 10-inch spans in RFA (w IFMs) safety analyses.

4.4.2.3 Linear Heat Generation Rate

The core average and maximum LHGRs are given in Table 4.4-1. The method of determining the maximum LHGR is given in Subsection 4.3.2.2.

4.4.2.4 Void Fraction Distribution

The calculated core average and the hot subchannel maximum and average void fractions are presented in Table 4.4-2 for operation at full power with the original design hot channel factors. The void models used in the VIPRE-01 computer code are described in Subsection 4.4.2.7c. Typical normalized core flow and enthalpy rise distributions are shown in Figure 4.4-2, Figure 4.4-3 and Figure 4.4-4 for the Cycle 1 core design. The distributions are also typical of those which would be found in later operating cycles.

4.4.2.5 Core Coolant Flow Distribution

Assembly average coolant mass velocity and enthalpy at various radial and axial core locations are given below. Typical coolant enthalpy rise and flow distributions are shown for the 4 foot elevation (1/3 of core height) in Figure 4.4-2, and 8 foot elevation (2/3 of core height) in Figure 4.4-3 and at the core exit in Figure 4.4-4. These distributions are for the full power conditions as given in Table 4.4-1 and for the radial power density distribution shown in Figure 4.3-7, which correspond to the Cycle 1 core design. The values are also typical for later operating cycles. The analysis for this case utilized a uniform core inlet enthalpy and inlet flow distribution. No orificing is employed in the reactor design.

4.4.2.6 Core Pressure Drops and Hydraulic Loads

a. Core Pressure Drops

The analytical model and experimental data used to calculate the pressure drops shown in Table 4.4-1 are described in Subsection 4.4.2.7. The core pressure drop includes the fuel assembly, lower core plate, and upper core plate pressure drops. The full power operation pressure drop values shown in Table 4.4-1 are the unrecoverable pressure drops across the vessel, including the inlet and outlet nozzles, and across the core. These pressure drops are based on a best estimate flow of 402,000 gpm, 3659 MWt and core inlet temperature of 556.8°F.

Uncertainties associated with the core pressure drop values are discussed in Subsection 4.4.2.9b.

b. Hydraulic Loads

The fuel assembly hold down springs, Figure 4.2-2, are designed to keep the fuel assemblies in contact with the lower core plate under all Condition I and II events, with the exception of the turbine overspeed transient associated with a loss of external load. The hold down springs are designed to tolerate the possibility of an over deflection associated with fuel assembly liftoff for this case, and provide contact between the fuel assembly and the lower core plate following this transient. More adverse flow conditions occur during a loss-of-coolant accident. These conditions are presented in Subsection 15.6.5.

Hydraulic loads at normal operating conditions are calculated considering the best estimate flow and accounting for the best estimate core bypass flow based on manufacturing tolerances. Core hydraulic loads at cold plant startup conditions are based on the cold best estimate flow, but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient, which could possibly create flow rates 18 percent greater than the best estimate flow, are evaluated to be approximately twice the fuel assembly weight. Applicable uncertainties are applied to these results.

4.4.2.7 Correlation and Physical Data

a. Surface Heat Transfer Coefficients

Forced convection heat transfer coefficients are obtained from the familiar Dittus-Boelter correlation (Reference 20), with the properties evaluated at bulk fluid conditions:

$$\frac{hDe}{K} = 0.023\left(\frac{D_e G}{\mu}\right)^{0.8}\left(\frac{C_p \mu}{K}\right)^{0.4} \qquad (4.4\text{-}8)$$

where:

h	=	heat transfer coefficient, (Btu/hr-ft^2°F)
D_e	=	equivalent diameter, (ft)
K	=	thermal conductivity, (Btu/hr-ft-°F)
G	=	mass velocity, (lb$_m$/hr-ft^2)
μ	=	dynamic viscosity, (lb$_m$/ft-hr)
C_P	=	heat capacity, (Btu/lb$_m$ -°F)

This correlation has been shown to be conservative (Reference 21) for rod bundle geometries with pitch-to-diameter ratios in the range used by PWRs.

The onset of nucleate boiling occurs when the clad wall temperature reaches the amount of superheat predicted by Thom's correlation, Reference 22. After this occurrence the outer clad wall temperature is determined by:

$$\Delta T_{sat} = (0.072 \exp(-P/1260))\,(q'')^{0.5} \qquad (4.4\text{-}9)$$

where:

ΔT_{sat}	=	wall superheat, $T_w - T_{sat}$, (°F)
q''	=	wall heat flux, (Btu/hr-ft^2)
P	=	pressure, (psia)
T_w	=	outer clad wall temperature, (°F)
T_{sat}	=	saturation temperature of coolant at P, (°F)

b. Total Core and Vessel Pressure Drop

Unrecoverable pressure losses occur as a result of viscous drag (friction) and/or geometry changes (form) in the fluid flow path. The flow field is assumed to be incompressible, turbulent, single- phase water. These assumptions apply to the core and vessel pressure drop calculations for the purpose of establishing the primary loop flow rate. Two-phase considerations are neglected in the vessel pressure drop evaluation because the core average void is negligible (see Table 4.4-2). Two-phase flow considerations in the core thermal subchannel analyses are considered and the models are discussed in Subsection 4.4.4.2c. Core and vessel pressure losses are calculated by equations of the form:

$$P_L = \left(K + \frac{FL}{D_e}\right)\frac{\rho V^2}{2g_c(144)} \qquad (4.4\text{-}10)$$

Where:

P_L	=	unrecoverable pressure drop, (lb$_f$/in^2)
ρ	=	fluid density, (lb$_m$ /ft^3)
L	=	length, (ft)
D_e	=	equivalent diameter, (ft)
V	=	fluid velocity, (ft/sec)
g_c	=	$32.174 \dfrac{\text{lb}_m\text{-ft}}{\text{lb}_f\text{-sec}^2}$
K	=	form loss coefficient, dimensionless

F = friction loss coefficient, dimensionless

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

Values are quoted in Table 4.4-1 for unrecoverable pressure loss across the reactor vessel, including the inlet and outlet nozzles and across the core. The results of full-scale tests of core components and fuel assemblies were utilized in developing the core pressure loss characteristic. The pressure drop for the vessel was obtained by combining the core loss with correlation of 1/7th scale model hydraulic test data on a number of vessels (References 23 and 24) and form loss relationships (Reference 25). Moody curves (Reference 26) were used to obtain the single-phase friction factors.

c. Void Fraction Correlation

Empirical correlations are used in VIPRE to model the void fraction in two-phase flow. The subcooled void correlation used to model the non-equilibrium transition from single phase to nucleate boiling is given in Reference 81. The bulk (saturated) void model relates flow quality with void fraction which can account for phase slip.

4.4.2.8 Thermal Effects of Operational Transients

DNB core safety limits are generated as a function of coolant temperature, pressure, core power and axial power imbalance. Steady state operation within these safety limits insures that the minimum DNBR is not less than the safety analysis limit. Figure 15.0-1 shows the safety analysis limit lines and the resulting Overtemperature ΔT trip lines (which become part of the Technical Specifications or Core Operating Limits Report), plotted as ΔT, versus T_{avg} for various pressures. This system provides adequate protection against anticipated operational transients that are slow with respect to fluid transport delays in the primary system. In addition, for fast transients, e.g., uncontrolled rod bank withdrawal at power incident (Subsection 15.4.2), specific protection functions are provided as described in Section 7.2 and the use of these protection functions is described in Chapter 15.

4.4.2.9 Uncertainties in Estimates

a. Uncertainties in Fuel and Clad Temperatures

As discussed in paragraph 4.4.2.11, the fuel temperature is a function of crud, oxide, clad, pellet-clad gap, and pellet conductances. Uncertainties in the fuel temperature calculation are essentially of two types: fabrication uncertainties, such as variations in the pellet and clad dimensions and the pellet density; and model uncertainties, such as variations in the pellet conductivity and the gap conductance. These uncertainties have been quantified by comparison of the thermal model to the inpile thermocouple measurements, (References 40 - 46) by out-of-pile measurements of the fuel and clad properties, (References 47 - 58) and by measurements of the fuel and clad dimensions during fabrication. The resulting uncertainties are then used in all evaluations involving the fuel temperature. The effect of densification on fuel temperature uncertainties is also included in the calculation of the total uncertainty.

In addition to the temperature uncertainty described above, the measurement uncertainty in determining the local power and the effect of density and enrichment variations on the local power are considered in establishing the heat flux hot channel factor. These uncertainties are described in paragraph 4.3.2.2.1. Reactor trip setpoints, as specified in the Technical Specifications, include allowance for instrument and measurement uncertainties such as calorimetric error, instrument drift and channel reproducibility, temperature measurement uncertainties, noise, and heat capacity variations.

Uncertainty in determining the cladding temperature results from uncertainties in the crud and oxide thicknesses. Because of the excellent heat transfer between the surface of the rod and the coolant, the film temperature drop does not appreciably contribute to the uncertainty.

b. Uncertainties in Pressure Drops

Core and vessel pressure drops based on a measured flow, as described in Section 5.1, are quoted in Table 4.4-1. The uncertainties quoted are based on the uncertainties in both the test results and the analytical extension of these values to the reactor application.

A major use of the core and vessel pressure drops was to determine the primary system coolant flow rates as discussed in Section 5.1. As discussed in Subsection 4.4.5.1, tests on the primary system prior to initial criticality were made to verify that conservative primary system coolant flow has been used in the mechanical design and safety analyses of the plant.

c. Uncertainties Due to Inlet Flow Maldistribution

The effects of uncertainties in the inlet flow maldistribution criteria used in the core thermal analyses are discussed in Subsection 4.4.4.2b.

d. Uncertainty in DNB Correlation

The uncertainty in the DNB correlation (Subsection 4.4.2.2) can be written as a statement on the probability of not being in DNB based on the statistics of the DNB data. This is discussed in Subsection 4.4.2.2b.

e. Uncertainties in DNBR Calculations

The uncertainties in the DNBRs calculated by VIPRE-01 analysis (see Subsection 4.4.4.5a) with the RTDP are accounted for as discussed in Section 4.4.1.1. For those transients that do not use RTDP, the uncertainties are applied directly to the VIPRE-01 input parameters. The results of a sensitivity study (Reference 18) show that the minimum DNBR in the hot channel is relatively insensitive to variations in the core-wide-radial power distribution (for the same value of F_H).

The ability of the VIPRE-01 computer code to accurately predict flow and enthalpy distributions in rod bundles is discussed in Subsection 4.4.4.5a and in Reference 81.

f. Uncertainties in Flow Rates

The uncertainties associated with loop flow rates are discussed in Section 5.1. For core thermal performance evaluations, a minimum loop flow is used which is less than the best estimate loop flow. In addition, up to 8.3 percent of the thermal design flow is assumed to be ineffective for core heat removal capability because it bypasses the core through the various available vessel flow paths described in Subsection 4.4.4.2a.

g. Uncertainties in Hydraulic Loads

As discussed in Subsection 4.4.2.6b, applicable uncertainties are applied to the hydraulic loads on the fuel assembly that are calculated using best estimate flows.

h. Uncertainty in Mixing Coefficient

The value of the mixing coefficient, TDC, used in VIPRE-01 analyses for this application is 0.038. The mean value of TDC obtained in the "R" grid mixing tests described in Subsection 4.4.2.2a was 0.042 (for 26 inch grid spacing). The value 0.038 is one standard deviation below the mean value; approximately 90 percent of the data give values of TDC greater than 0.038 (Reference 12).

The results of the mixing tests done on 17x17 geometry, as discussed in Subsection 4.4.2.2c, had a mean value of TDC of 0.059 and standard deviation of $\sigma = 0.007$. Hence, the current design value of TDC is almost 3 standard deviations below the mean for 26 inch grid spacing.

4.4.2.10 Flux Tilt-Considerations

Significant quadrant power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation. A dropped or misaligned Rod Cluster Control Assembly could cause changes in hot channel factors. However, these events are analyzed separately in Chapter 15.

Other possible causes for quadrant power tilts include X-Y xenon transients, inlet temperature mismatches, enrichment variations within tolerances and so forth.

In addition to unanticipated quadrant power tilts as described above, other readily explainable asymmetries may be observed during calibration of the excore detector quadrant power tilt alarm. During operation, incore maps are taken at least once per month and, periodically, additional maps are obtained for calibration purposes. Each of these maps is reviewed for deviations from the expected power distributions. Asymmetry in the core, from quadrant to quadrant, is frequently a consequence of the design when assembly and/or component shuffling and rotation requirements do not allow exact symmetry preservation. In each case, the acceptability of an observed asymmetry, planned or otherwise, depends solely on meeting the required accident analyses assumptions.

In practice, once acceptability has been established by review of the incore map, the quadrant power tilt alarms and related instrumentation are adjusted to a Quadrant Power Tilt Ratio of one (no excore tilt) as the final step in the calibration process. This action ensures that the instrumentation is correctly calibrated to alarm in the event an unexplained or unanticipated change occurs in the quadrant to quadrant relationships between calibration intervals. Proper functioning of the quadrant power tilt alarm is significant because no allowances, beyond accounting for the maximum tilt allowed by Technical Specifications, are made in the design for increased hot channel factors due to unexpected developing flux tilts, since all likely causes are prevented by design or procedures, or are specifically analyzed. Finally, in the event that unexplained flux tilts do occur, the Technical Specifications (Subsection 3/4.2.4) provide appropriate corrective actions to ensure continued safe operation of the reactor.

4.4.2.11 Fuel and Cladding Temperatures

Consistent with the thermal-hydraulic design bases described in Subsection 4.4.1, the following discussion pertains mainly to fuel pellet temperature evaluation. A discussion of fuel clad integrity is presented in Subsection 4.2.3.1.

The thermal-hydraulic design assures that the maximum fuel temperature is below the melting point of $U0_2$ (melting point of 5080°F (Reference 1) unirradiated and decreasing by 58°F per 10,000 MWd/Mtu). To preclude center melting, and as a basis for overpower protection system setpoints, cycle-specific values for the peak linear heat generation rate precluding centerline melt are determined as a function of fuel rod average exposure. These are observed as an overpower limit for Condition I and II events. They provide sufficient margin for uncertainties in the thermal evaluations described in Subsection 4.4.2.9a. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the $U0_2$ thermal conductivity. However, the computation of radial fuel temperature distributions combines crud, oxide, clad gap and pellet conductances. The factors which influence these conductances, such as gap size (or contact pressure), internal gas pressure, gas composition, pellet density, and radial power distribution within the pellet, etc., have been combined into a Westinghouse semi-empirical thermal model (see Subsection 4.2.3.3) with the model modifications for time dependent fuel densification given in Reference 2. This thermal model enables the determination of these factors and their net effects on temperature profiles. The temperature predictions have been compared to inpile fuel temperature measurements (References 40-46 and 59) and melt radius data (References 60 and 61).

As described in Reference 2, fuel rod thermal evaluations (fuel centerline, average and surface temperatures) are determined throughout the fuel rod life-time with consideration of time dependent densification. To determine the maximum fuel temperatures, various burnup rods, including the highest burnup rod, are analyzed over the rod linear power range of interest.

The principal factors which are employed in the determination of the fuel temperature are discussed below.

a. <u>$U0_2$ Thermal Conductivity</u>

The thermal conductivity of uranium dioxide was evaluated from data reported from a number of measurements.

At the higher temperatures, thermal conductivity is best obtained by utilizing the integral conductivity to melt which can be determined with more certainty. From an examination of the data, it has been concluded that the best estimate for the value of 2800°C Kdt is 93 watts/cm.

The design curve is in excellent agreement with the recommendation of the IAEA panel (Reference 36).

b. <u>Radial Power Distribution in $U0_2$ Fuel Rods</u>

An accurate description of the radial power distribution as a function of burnup is needed for determining the power level for incipient fuel melting and other important performance parameters such as pellet thermal expansion, fuel swelling and fission gas release rates.

Radial power distribution in UO_2 fuel rods is determined with the neutron transport code LASER. The LASER code has been validated by comparing the code predictions on radial burnup and isotopic distributions with measured radial microdrill data (References 62 and 63). A "radial power depression factor," f, is determined using radial power distributions predicted by LASER. The factor, f, enters into the determination of the pellet centerline temperature, T_C, relative to the pellet surface temperature, T_S, through the expression:

$$\int_{T_s}^{T_c} K(T)\, dT = \frac{q''f}{4\pi}$$

where:

K(T) = the thermal conductivity for $U0_2$ with a uniform density distribution

q' = the linear power generation rate.

c. Gap Conductance

The temperature drop across the pellet-clad gap is a function of the gap size and the thermal conductivity of the gas in the gap. The gap conductance model is selected such that when combined with the UO_2 thermal conductivity model, the calculated fuel centerline temperatures reflect the inpile temperature measurements. A more detailed discussion of the gap conductance model is presented in Reference 64.

d. Surface Heat Transfer Coefficients

The fuel rod surface heat transfer coefficients during subcooled forced convection and nucleate boiling are presented in Subsection 4.4.2.7a.

e. Fuel Clad Temperatures

The outer surface of the fuel rod at the hot spot operates at a temperature of approximately 660°F for steady operation at rated power throughout core life due to the onset of nucleate boiling. Initially (beginning-of-life), this temperature is that of the clad metal outer surface.

During operation over the life of the core, the buildup of oxides and crud on the fuel rod surface causes the clad surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise. Since the thermal-hydraulic design basis limits DNB, adequate heat transfer is provided between the fuel clad and the reactor coolant so that the core thermal output is not limited by considerations of clad temperature.

f. Treatment of Peaking Factors

The total heat flux hot channel factor, F_Q, is defined by the ratio of the maximum to core average heat flux. As presented in Table 4.3-2 and discussed in Subsection 4.3.2.2f, the design value of F_Q for normal operation is 2.50. This results in a peak linear power of 14.6 kW/ft at full power conditions.

The centerline temperature must be below the $U0_2$ melt temperature over the lifetime of the rod, including allowances for uncertainties. The fuel temperature design basis is discussed in Subsection 4.4.1.2. The centerline temperature resulting from overpower transients/operator errors is below that required to produce melting.

4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

4.4.3.1 Plant Configuration Data

Plant configuration data for the thermal hydraulic and fluid systems external to the core are provided in the appropriate Chapters 5, 6, and 9. Implementation of the Emergency Core Cooling System (ECCS) is discussed in Chapter 15. Some specific areas of interest are the following:

a. Total coolant flow rates for the Reactor Coolant System (RCS) and each loop are provided in Table 5.1-1. Flow rates employed in the evaluation of the core are presented in Section 4.4.

b. Total RCS volume including pressurizer and surge line, RCS liquid volume including pressurizer water at steady state power conditions are given in Table 5.1-1.

c. The flow path length through each volume may be calculated from physical data provided in the above referenced tables.

d. The height of fluid in each component of the RCS may be determined from the physical data presented in Section 5.4. The components of the RCS are water filled during power operation with the pressurizer being approximately 60 percent water filled.

e. Components of the ECCS are to be located so as to meet the criteria for net positive suction head described in Section 6.3.

f. Line lengths and sizes for the Safety Injection System are determined so as to guarantee a total system resistance which will provide, as a minimum, the fluid delivery rates assumed in the safety analyses described in Chapter 15.

g. The parameters for components of the RCS are presented in Section 5.4, component and subsystem design.

h. The steady state pressure drops and temperature distributions through the RCS are presented in Table 5.1-1.

4.4.3.2 Operating Restrictions on Pumps

The minimum net positive action head (NPSH) and minimum seal injection flow rate must be established before operating the reactor coolant pumps. With the minimum 6 gpm labyrinth seal injection flow rate established, the operator will have to verify that the system pressure satisfies NPSH requirements.

4.4.3.3 Power-Flow Operating Map (BWR)

Not applicable to pressurized water reactors.

4.4.3.4 Temperature-Power Operating Map

The relationship between Reactor Coolant System temperature and power is shown in Figure 4.4-6.

The effects of reduced core flow due to inoperative pumps is discussed in Subsections 5.4.1, 15.3.1, and 15.3.2. Natural circulation capability of the system is demonstrated in Subsection 15.2.6.

4.4.3.5 Load Following Characteristics

The Reactor Coolant System is designed on the basis of steady state operation at full power heat load. The reactor coolant pumps utilize constant speed drives. The reactor coolant pump assembly is described in Section 5.4. Reactor power is controlled to maintain average coolant temperature at a value which is a linear function of load, as described in Section 7.7.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal and hydraulic characteristics are given in Table 4.4-1 and Table 4.4-2.

4.4.4 Evaluation

4.4.4.1 Critical Heat Flux

The critical heat flux correlation utilized in the core thermal analysis is explained in detail in Subsection 4.4.2.

4.4.4.2 Core Hydraulics

a. Flow Paths Considered in Core Pressure Drop and Thermal Design

The following flow paths for core bypass flow are considered:

1. Flow through the spray nozzles into the upper head for head cooling purposes.
2. Flow entering into the RCC guide thimbles to cool the control rods.

3. Leakage flow from the vessel inlet nozzle directly to the vessel outlet nozzle through the gap between the vessel and the barrel.

4. Flow introduced between the baffle and the barrel for the purpose of cooling these components and which is not considered available for core cooling.

5. Flow in the gaps between the fuel assemblies on the core periphery and the adjacent baffle wall.

The above contributions are evaluated to confirm that the design value of the core bypass flow is met. The design value of core bypass flow for Seabrook Station is equal to 8.3 percent of the total vessel flow when all thimble plugs are deleted.

Of the total allowance, 4.0 percent is associated with the core, item 2 above, and the remainder is associated with the internals (items 1, 3, 4 and 5 above). Calculations have been performed using drawing tolerances on a worst case basis and accounting for uncertainties in pressure losses. Based on these calculations, the core bypass flow for the plant is < 8.3 percent when all thimble plugs are deleted.

b. Inlet Flow Distributions

Data have been considered from several 1/7th scale hydraulic reactor model tests (References 23, 24, and 37) and from sensitivity studies, Reference 18, in arriving at the core inlet flow maldistribution criteria to be used in the VIPRE-01 analyses (see Subsection 4.4.4.5a).

The effect of the total flow rate on the inlet velocity distribution was studied in the experiments of Reference 23. As was expected, on the basis of the theoretical analysis, no significant variation could be found in inlet velocity distribution with reduced flow rate.

c. Empirical Friction Factor Correlations

Two empirical friction factor correlations are used in the VIPRE-01 computer code (described in Subsection 4.4.4.5a).

The friction factor in the axial direction, parallel to the fuel rod axis, is evaluated using the correlations described in Reference 81).

The flow in the lateral directions, normal to the fuel rod axis, views the reactor core as a large tube bank. Thus, the lateral friction factor proposed by Idel'chik (Reference 25) is applicable. This correlation is of the form:

$$F_L = A\ Re_L^{-0.2} \qquad (4.4\text{-}12)$$

where:

A is a function of the rod pitch and diameter as given in Reference 25.

Re_L is the lateral Reynolds number based on the rod diameter.

4.4.4.3 Influence of Power Distribution

The core power distribution, which is largely established at beginning-of-life by fuel enrichment, loading pattern, and core power level is also a function of variables such as control rod worth and position, and fuel depletion throughout lifetime. Radial power distributions in various planes of the core are often illustrated for general interest; however, the core radial enthalpy rise distribution as determined by the integral of power up each channel is of greater importance for DNB analyses. These radial power distributions, characterized by F_H (defined in paragraph 4.3.2.2.1), as well as axial heat flux profiles are discussed in the following two paragraphs.

4.4.4.3.1 Nuclear Enthalpy Rise Hot Channel Factor, FH

Given the local power density q' (kW/ft) at a point x, y, z in a core with N fuel rods and height H,

$$F_{\Delta H}^{N} = \frac{\text{Hot rod power}}{\text{average rod power}} = \frac{MAX \int_0^H q'(x_0, y_0, z_0)dz}{\frac{1}{N}\sum_{\substack{all \\ rods}} \int_0^H q'(x, y, z)dz}$$

The way in which F_H is used in the DNB calculation is important. The location of minimum DNBR depends on the axial profile, and the value of DNBR depends on the enthalpy rise to that point. Basically, the maximum value of the rod integral is used to identify the most likely rod for minimum DNBR. An axial power profile is obtained which, when normalized to the design value of F_H, recreates the axial heat flux along the limiting rod. The surrounding rods are assumed to have the same axial profile with rod average powers which are typical distributions found in hot assemblies. In this manner, worst-case axial profiles can be combined with worst-case radial distributions for reference DNB calculations.

It should be noted again that F_H is an integral and is used as such in DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal power shapes throughout the core.

For operation at a fraction of full power, the design F_H used is given by:

$$F_{\Delta H}^{N} = F_{\Delta H}^{RTD}[1 + PF_{\Delta H}(1 - P)]$$

$F_{\Delta H}^{RTP}$ is the limit at rated thermal power (RTP) specified in the core Operating Limits Report (COLR).

$PF_{\Delta H}$ is the power factor multiplier for F_H specified in the COLR.

P is the fraction of rated thermal power.

The permitted relaxation of F_H is included in the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits, (Reference 84) thus allowing greater flexibility in the nuclear design.

4.4.4.3.2 Axial Heat Flux Distributions

As discussed in paragraph 4.3.2.2, the axial heat flux distribution can vary as a result of rod motion or power change or as a result of a spatial xenon transient which may occur in the axial direction. Consequently, it is necessary to measure the axial power imbalance by means of the excore nuclear detectors (as discussed in paragraph 4.3.2.2.7) and to protect the core form excessive axial power imbalance. The reference axial shape used in establishing core DNB limits (that is, overtemperature ΔT protection system setpoints) is a chopped cosine with a peak-to-average value of 1.55. The reactor trip system provides automatic reduction of the trip setpoints on excessive axial power imbalance. To determine the magnitude of the setpoint reduction, the reference shape is supplemented by other axial shapes skewed to the bottom and top of the core.

The course of those accidents in which DNB is a concern is analyzed in chapter 15 assuming that the protection setpoints have been set on the basis of these shapes. In many cases, the axial power distribution in the hot channel changes throughout the course of the accident due to rod motion, coolant temperature, and power level changes.

The initial conditions for the accidents for which DNB protection is required are assumed to be those permissible within the specified axial offset control limits described in paragraph 4.3.2.2. In the case of the loss-of-flow accident, the hot channel heat flux profile is very similar to the power density profile in normal operation preceding the accident.

4.4.4.4 Core Thermal Response

A general summary of the steady state thermal-hydraulic design parameters including thermal output, flow rates, etc., is provided in Table 4.4-1 for all loops in operation.

As stated in Subsection 4.4.1, the design bases of the application are to prevent DNB and to prevent fuel melting for Condition I and II events. The protective systems described in Chapter 7 are designed to meet these bases. The response of the core to Condition II transients is given in Chapter 15.

4.4.4.5 Analytical Techniques

a. Core Analysis

The objective of reactor core thermal design is to determine the maximum heat-removal capability in all flow subchannels and to show that the core safety limits are not exceeded using the most conservative power distribution. The thermal design takes into account local variations in dimensions, power generation, flow redistribution, and mixing. VIPRE-01 is a realistic three-dimensional matrix model which has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core. (Reference 81). The behavior of the hot assembly is determined by superimposing the power distribution among the assemblies upon the inlet flow distribution while allowing for flow mixing and flow distribution between assemblies. The local variations in power, fuel rod and pellet fabrication, and mixing within the hottest assembly are superimposed on the average conditions of the hottest assembly in order to determine the conditions in the hot channel.

b. Steady State Analysis

The VIPRE-01 computer program and subchannel analysis methodology, as approved by the NRC (Reference 81) is used to determine coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distributions within the reactor core hot subchannel under all expected operating conditions. The VIPRE-01 code is described in detail in Reference 81, including models and correlations used.

c. Experimental Verification

Experimental verification of VIPRE-01 is presented in References 11 and 81.

The VIPRE-01 analysis methodology is based on a knowledge and understanding of the heat transfer and hydrodynamic behavior of the coolant flow and the mechanical characteristics of the fuel elements. VIPRE-01 analysis provides a realistic evaluation of the core performance and is used in the thermal analyses as described above.

d. Transient Analysis

The approved VIPRE-01 methodology (Reference 81) was shown to be conservative for transient thermal-hydraulic analysis.

4.4.4.6 Hydrodynamic and Flow Power Coupled Instability

Boiling flows may be susceptible to thermohydrodynamic instabilities, (Reference 69). These instabilities are undesirable in reactors since they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition or to undesired forced vibrations of core components. Therefore, a thermohydraulic design criterion was developed which states that modes of operation under Conditions I and II events shall not lead to thermohydrodynamic instabilities.

Two specific types of flow instabilities are considered for Westinghouse PWR operation. These are the Ledinegg or flow excursion type of static instability and the density wave type of dynamic instability.

A Ledinegg instability involves a sudden change in flow rate from one steady state to another. This instability occurs (Reference 69) when the slope of the reactor coolant system pressure drop-flow rate curve ($\delta\Delta P/\delta G$ internal) becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve ($\delta\Delta P/\delta G$ external). The criterion for stability is thus ($\delta\Delta P/\delta G$ internal > $\delta\Delta P/\delta G$ external). The Westinghouse pump head curve has a negative slope ($\delta\Delta P/\delta G$external < 0) whereas the reactor coolant system pressure drop-flow rate curve has a positive slope ($\delta\Delta P/\delta G$ internal > O) over the Condition I and Condition II operational ranges. Thus, the Ledinegg instability will not occur.

The mechanism of density wave oscillations in a heated channel has been described by Lahey and Moody (Reference 70). Briefly, an inlet flow fluctuation produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single phase region and causes quality or void perturbations in the two phase regions which travel up the channel with the flow. The quality and length perturbations in the two-phase region create two-phase pressure drop perturbations. However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, then the two-phase pressure drop perturbation feeds back to the single phase region. These resulting perturbations can be either attenuated or self-sustained.

A simple method has been developed by Ishii (Reference 71) for parallel closed channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability. This method has been used to assess the stability of typical Westinghouse reactor designs (References 72, 73, 74), under Conditions I and II operation. The results indicate that a large margin to density wave instability exists, e.g., increases on the order of 150 to 200 percent of rated reactor power would be required for the predicted inception of this type of instability.

The application of the method of Ishii, Reference 71, to Westinghouse reactor designs is conservative due to the parallel open channel feature of Westinghouse PWR cores. For such cores, there is little resistance to lateral flow leaving the flow channels of high power density. There is also energy transfer from channels of high power density to lower power density channels. This coupling with cooler channels has led to the opinion that an open channel configuration is more stable than the above closed channel analysis under the same boundary conditions. Flow stability tests (Reference 75) have been conducted where the closed channel systems were shown to be less stable than when the same channels were cross connected at several locations. The cross connections were such that the resistance to channel-to-channel cross flow and enthalpy perturbations would be greater than that which would exist in a PWR core which has a relatively low resistance to cross flow.

Flow instabilities which have been observed have occurred almost exclusively in closed channel systems operating at low pressures relative to the Westinghouse PWR operating pressures. Kao, Morgan and Parker (Reference 76) analyze parallel closed channel stability experiments simulating a reactor core flow. These experiments were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

Additional evidence that flow instabilities do not adversely affect thermal margin is provided by the data from the rod bundle DNB tests. Many Westinghouse rod bundles have been tested over wide ranges of operating conditions with no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundle.

In summary, it is concluded that thermohydrodynamic instabilities will not occur under Condition I and II modes of operation for Westinghouse PWR reactor designs. A large power margin, greater than doubling rated power, exists to predicted inception of such instabilities. Analysis has been performed which shows that minor plant to plant differences in Westinghouse reactor design such as fuel assembly arrays, core power to flow ratios, fuel assembly length, etc., will not result in gross deterioration of the above power margins.

4.4.4.7 Fuel Rod Behavior Effects from Coolant Flow Blockage

Coolant flow blockages can occur within the coolant channels of a fuel assembly or external to the reactor core. The effects of fuel assembly blockage within the assembly on fuel rod behavior is more pronounced than external blockages of the same magnitude. In both cases, the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, where it occurs in the reactor, and how far along the flow stream the reduction persists are considerations which will influence the fuel rod behavior. The effects of coolant flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as computer programs. Inspection of the DNB correlation (Subsection 4.4.2.2 and Reference 8) shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

Thermal-hydraulic codes are capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on a subchannel basis, regardless of where the flow blockage occurs. In Reference 19, it is shown that for a fuel assembly similar to the Westinghouse design, the flow distribution within the fuel assembly when the inlet nozzle is completely blocked can be accurately predicted. Full recovery of the flow was found to occur about 30 inches downstream of the blockage. With the reference reactor operating at the nominal full power conditions specified in Table 4.4-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor reaching a minimum DNBR below the safety analysis limit.

From a review of the open literature it is concluded that flow blockage in "open lattice cores" similar to the Westinghouse cores cause flow perturbations which are local to the blockage. For instance, A. Ohtsubol, et al. (Reference 77), show that the mean bundle velocity is approached asymptotically about 4 inches downstream from a flow blockage in a single flow cell. Similar results were also found for 2 and 3 cells completely blocked. P. Basmer, et al. (Reference 78) tested an open lattice fuel assembly in which 41 percent of the subchannels were completely blocked in the center of the test bundle between spacer grids. Their results show the stagnant zone behind the flow blockage essentially disappears after 1.65 L/De or about 5 inches for their test bundle. They also found that leakage flow through the blockage tended to shorten the stagnant zone or, in essence the complete recovery length. Thus, local flow blockages within a fuel assembly have little effect on subchannel enthalpy rise. The reduction in local mass velocity is the main parameter which affects the DNBR. Westinghouse analysis results presented in the original Seabrook FSAR demonstrated that if the plant was operating at full power and nominal steady state conditions as specified in Table 4.4-1, a substantial reduction in local mass velocity would be required to reduce the DNBR close to the DNBR Safety Analysis Limits. The above mass velocity effect on the DNB correlation was based on the assumption of fully developed flow along the full channel length. In reality, a local flow blockage is expected to promote turbulence and thus would likely not effect DNBR at all.

Coolant flow blockages induce local crossflows as well as promote turbulence. Fuel rod behavior is changed under the influence of a sufficiently high crossflow component. Fuel rod vibration could occur, caused by this crossflow component, through vortex shedding or turbulent mechanisms. If the crossflow velocity exceeds the limit established for fluid elastic stability, large amplitude whirling results. The limits for a controlled vibration mechanism are established from studies of vortex shedding and turbulent pressure fluctuations. The crossflow velocity required to exceed fluid elastic stability limits is dependent on the axial location of the blockage and the characterization of the crossflow (jet flow or not). These limits are greater than those for vibratory fuel rod wear. Crossflow velocity above the established limits can lead to mechanical wear of the fuel rods at the grid support locations. Fuel rod wear due to flow induced vibration is considered in the fuel rod fretting evaluation (Section 4.2).

4.4.5 Testing and Verification

4.4.5.1 Tests Prior to Initial Criticality

A reactor coolant flow test was performed following fuel loading but prior to initial criticality. Elbow tap pressure drop data obtained in this test allowed determination of the coolant flow rates at reactor operating conditions. This test verified that conservative coolant flow rates have been used in the core thermal and hydraulic analysis.

4.4.5.2 Initial Power and Plant Operation

Core power distribution measurements are made at several core power levels (see Chapter 14). These tests are used to insure that conservative peaking factors are used in the core thermal and hydraulic analysis.

4.4.5.3 Component and Fuel Inspections

Inspections performed on the manufactured fuel are delineated in Subsection 4.2.4. Fabrication measurements critical to thermal and hydraulic analysis are obtained to verify that the engineering hot channel factors in the design analyses (Subsection 4.4.2.2d) are met.

4.4.6 Instrumentation Requirements

4.4.6.1 Incore Instrumentation

Instrumentation is located in the core so that radial, axial, and azimuthal core characteristics may be obtained for all core quadrants.

The incore detector assemblies enter the core from the bottom and are positioned in the full length instrumentation thimbles that are located in the center of the fuel assemblies. Figure 4.4-7 shows the location of the 58 instrumented assemblies in the core. Each detector assembly consists of the calibration tube for the moveable incore detectors, five fixed platinum detectors at various core heights, and a core-exit thermocouple at the tip of the detector assembly. The platinum detectors measure the gamma and neutron flux and are processed to determine the local power distribution. Each thermocouple measures the temperature of the fluid in the instrumentation thimble that is heated by conduction from the bulk core fluid and by gamma heating of the components in the instrumentation thimble. Replacement detector assemblies do not have a functional path for a moveable detector.

The Incore Instrumentation System is described in more detail in Subsections 4.4.6.5, 7.5 and 7.7.1.9.

The incore flux instrumentation is provided to obtain data from which fission power density distribution in the core, coolant enthalpy distribution in the core, and fuel burnup distribution may be determined. The thermocouples provide core exit temperature, input to accident monitor instrumentation for hot channel temperature subcooling margin, and reactor vessel level.

4.4.6.2 Overtemperature and Overpower ΔT Instrumentation

The Overtemperature ΔT trip protects the core against low DNBR. The Overpower ΔT trip protects against excessive power (fuel rod rating protection).

As discussed in Subsection 7.2.1.1b, factors included in establishing the Overtemperature ΔT and Overpower ΔT trip setpoints include the reactor coolant temperature in each loop and the axial distribution of core power through the use of the two section excore neutron detectors.

4.4.6.3 Instrumentation to Limit Maximum Power Output

The output of the three ranges (source, intermediate, and power) of detectors, with the electronics of the nuclear instruments, is used to limit the maximum power output of the reactor within their respective ranges.

There are six radial locations containing a total of eight neutron flux detectors installed around the reactor in the primary shield, two proportional counters for the source range installed on opposite "flat" portions of the core containing the primary startup sources at an elevation approximately one quarter of the core height. Two compensated ionization chambers for the intermediate range, located in the same instrument wells and detector assemblies as the source range detectors, are positioned at an elevation corresponding to one half of the core height; four dual section uncompensated ionization chamber assemblies for the power range installed vertically at the four corners of the core and located equidistant from the reactor vessel at all points and, to minimize neutron flux pattern distortions, within one foot of the reactor vessel. Each power range detector provides two signals corresponding to the neutron flux in the upper and in the lower sections of a core quadrant. The three ranges of detectors are used as inputs to monitor neutron flux from a completely shutdown condition to 120 percent of full power with the capability of recording overpower excursions up to 200 percent of full power.

The output of the power range channels is used for:

a. The rod speed control function

b. To alert the operator to an excessive power unbalance between the quadrants

c. To protect the core against rod ejection accidents

d. To protect the core against adverse power distributions resulting from dropped rods.

Details of the neutron detectors and nuclear instrumentation design and the control and trip logic are given in Chapter 7. The limits on neutron flux operation and trip setpoints are given in the Technical Specifications and Core Operating Limits Report.

4.4.6.4 Loose Parts Monitoring System (LPMS)

The LPMS is a system provided for the detection of loose metallic parts in the primary system during preoperational testing, startup and power operation modes. The LPMS, together with the associated programmatic and reporting procedures, comprise the Loose Part Detection Program described in Regulatory Guide 1.133, Rev. 1.

A detailed comparison of the LPMS with each of the specific positions of Section C of Regulatory Guide 1.133 is presented below with exceptions and clarifications.

Reg. Guide Position	Discussion
C.1.a C.1.b	A total of sixteen loose part sensors are provided to detect loose part impacts with a kinetic energy of 0.5 ft-lb. of parts weighing between .25 lb. and 30 lbs. in the vicinity of six natural collection regions in the nuclear steam supply system. a) Two sensors on the exterior of the reactor vessel in the vicinity of the lower plenum and two sensors on the reactor vessel head lifting lugs. b) Three sensors on the exterior of each steam generator in the vicinity of the reactor coolant inlet plenum. Two sensors are normally active and one is normally passive. The normally passive sensor may be switched into the system to replace a normally active sensor or to aid in the localization of a loose part in a steam generator.
C.1.c	Two or more independently monitored sensors are provided at each natural collection region. Each of these channels is physically separated from each other at the sensors up to and including the charge converters. From there, sensor signals are routed by individual shielded cables through seismically qualified conduit and tray associated with safety-related Train A up to penetration EDE-MM-126. Outside containment, all signal cabling is routed in seismically qualified tray associated with safety-related Train A up to the control room electronics.

Reg. Guide Position	Discussion
C.1.d C.4.e	The Automatic Data Acquisition System of the LPMS will be actuated (all active channels simultaneously) by the system electronics when the measured magnitude of the acoustic signal from any one channel exceeds the predetermined alert level for that channel. An audible alarm will alert control room personnel of any excursion above the predetermined alert level. To ensure that the data provided at the output of the system electronics is recorded to allow accurate offline analysis, the recorder is wide-band with respect to the bandwidth of the filtered data. The analog recorder provided will use the direct (as opposed to FM) recording mode. Two selectable tape speeds are provided, allowing selection of recording bandwidth. The Automatic Data Acquisition System has a manual override.
C.1.e C.2 C.3	The alert logic of the LPMS has the following features and capabilities
C.4.b	a) Minimization of false alarms due to flow or other disturbances not indicative of metallic loose part impacts. b) Maintenance of sensitivity to metallic loose part impacts under conditions of varying background noise. c) The signal filtering process attenuates the signals due to operational disturbances outside the filter system's bandwidth. d) The alert logic is capable of functioning satisfactorily in varying background noise levels. e) To differentiate between valid impacts and plant noise associated with one-time transient events (as opposed to steady state noise), the alert module common to each group of six electronic channels of the LPMS incorporates a variable timer circuit. The alert module will not perform its functions (alarm actuation and automatic recorder actuation) unless a predetermined number of impacts (excursions above the alert level) occur within a predetermined time period. This time circuit may be disabled, by use of a selector switch, so

Reg. Guide
Position Discussion

that any single excursion above the alert level will cause the alarm module to perform its functions.

f) To vary the alert level from one sensor to the other to compensate for various background noises at each sensor location.

The alert level for each channel is a function of the steady state background noises measured by that channel, according to the relation:

$$AL = (1 + K)\ BN$$

Where AL is the alert level, BN is the background noise level, and K is the fraction of the background noise level by which an impact must exceed the background in order to be detected. The K value was individually determined for each channel following initial system calibration. The K value for each channel was initially determined within two constraints:

a) The value (1 + K) BN shall be greater than the largest signal presented to the impact detection module when noise of magnitude BN is applied to the input terminals of the system electronics, as determined by factory acceptance testing of the LPMS and in situ monitoring of the signals presented to each impact detector.

b) The value (1 + K) BN (for the largest expected BN level) shall be less than the magnitude of the signal associated with the specified detectable loose part impact, as determined during initial LPMS calibration.

c) The minimum value of K consistent with the above criterion (a) was chosen and the satisfaction of criterion (b) was then verified. Satisfaction of these criteria will be periodically verified during operation in accordance with Regulatory Position 3.e of Regulatory Guide 1.133.

The alert level for power operation was submitted to the commission (in the startup report) following completion of the startup test program. If the alert level is exceeded, diagnostic steps will be taken within 72 hours to determine if a loose part is present. The safety significance of any identified loose part will be determined. During initial startup,

Reg. Guide Position	Discussion
	power operation and refueling, channel checks, monitoring audio channels, channel functional tests, background noise measurements, and channel calibrations will be performed as prescribed in the regulatory guide. A channel calibration includes the adjustments recommended by the vendor and an assessment of the overall channel response by observing the response to a known mechanical input or by comparing the background noise spectra to baseline background noise spectra. Calibration equipment and procedures are available for review at Seabrook Station.
C.1.f	The LPMS has the capability for periodic on-line channel checks and channel functional tests in addition to on-line and off-line channel calibration.
C.1.g C.4.k	The LPMS is designed to operate under normal environmental conditions The LPMS (excluding the recording equipment) has been seismically qualified to IEEE 344-75 to be functional up to and including the Operating Basis Earthquake (OBE). The LPMS sensor, charge converter, and system cabinet are seismically supported.
C.1.h	The LPMS will be included in the Seabrook surveillance and maintenance program. Maintenance and surveillance actions will be performed in accordance with approved procedures. The documented maintenance history will be maintained and evaluated over the life of the plant. Components will be of a quality that is consistent with minimal maintenance requirements and low failure rates. In lieu of the recommendations to replace components prior to end of service life, LPMS components will be maintained on a run to failure basis. This maintenance philosophy is considered to be acceptable since the system performs no active safety functions and a minimum of two diverse sensors will be provided for each collection region. A single random failure of one sensor in a collection region will not preclude monitoring of the region. The Seabrook maintenance and corrective action programs will trend LPMS equipment degradation and increases in failure rates will initiate augmented system maintenance.
C.1.i	Recognition of a faulty channel is easily identified by a blinking LED condition. All Control Room electronics are rack-mounted, designed

Reg. Guide Position	Discussion
	for the ease of replacement or repair in the event of a malfunctioning channel.
C.4.a	The loose parts monitoring sensors are piezoelectric accelerometers designed for use in high temperature and high radiation environments. Two accelerometers are mounted on the reactor vessel head. These accelerometers are mounted into two of the vessel head lifting lugs. Two accelerometers are threaded into clamps on the bottom-mounted instrumentation tubes. These locations allow monitoring of the reactor vessel upper and lower plenums and facilitated the mounting of the sensors. There are three accelerometers on each steam generator, two which are normally active and the third normally passive. The two normally active sensors are located in a vertical line approximately 16 inches above and below the centerline of the tube sheet, oriented 20° on the hot leg side of the tube lane centerline. The normally passive accelerometer is located on the tube sheet centerline 90° from the other sensors but still on the hot leg side of the tube lane centerline. These accelerometers are mounted on the side of the steam generator. All steam generator sensors are capable of monitoring the steam generator reactor coolant inlet plenum. They are dispersed to assist in localization of a loose part.
C.4.c	Anticipated major sources of external and internal noises are pump starts, reactor trip, and control rod stepping.
C.4.d	By meeting the criteria as defined in position C.3, the acquisition of quality data is ensured.
C.4.f C.5	Operability and surveillance requirements for the LPMS are included in Technical Requirements Manual.
C.4.g	Seabrook procedures provide a diagnostic program using information from other plant systems and operating history to confirm the presence of a loose part.

Reg. Guide Position	Discussion
C.4.h	The procedures for performing channel check, channel functional test, and background noise measurements are available at Seabrook Station.
C.4.i	Radiation protection procedures have been developed to provide guidance and direction to station personnel for minimizing radiation exposure during maintenance, calibration, and diagnostic work activities. The overall radiation protection program is described in the Updated FSAR Chapter 12.
C.4.j	Seabrook's non-licensed training program provides pertinent training for plant personnel involved with system operation, and maintenance. Loose part diagnosis is performed by an organization qualified to interpret loose part data.
C.6	If the presence of a loose part is confirmed and is evaluated to have safety significance, it will be reported to the NRC in accordance with 10 CFR 50.72.

4.4.6.5 Instrumentation for Detection of Inadequate Core Cooling

The Inadequate Core Cooling Monitoring System installed at Seabrook Station includes the following:

- Core Exit Thermocouple Monitoring
- Core Subcooling Margin Monitor
- Reactor Vessel Level Monitoring

The inadequate core cooling monitor provides improved information presentation and display to the plant operators on the status of core heat removal capability. The system monitors core exit thermocouples and wide-range reactor pressure and calculates core subcooling margin utilizing redundant channels of instrumentation and control room displays.

The monitoring system displays several levels of information including: (a) bulk average core exit thermocouple trending (b) a spatial map exhibiting the thermocouple temperature at its respective location in the core (c) a core map showing minimum, average, and maximum quadrant temperatures (d) subcooling margin (e) a detailed data list exhibiting thermocouple location, tag designation, temperature; and (f) hot channel core exit temperature. The Reactor Vessel Level Instrumentation System (RVLIS) consists of two redundant independent trains that monitor the reactor vessel water levels. Each train provides two vessel level indications: full range and dynamic head. The full range RVLIS reading provides an indication of reactor vessel water level from the bottom of the vessel to the top of the vessel during natural circulation conditions. The dynamic head reading provides an indication of reactor core, internals, and outlet nozzle pressure drop for any combination of operating reactor coolant pumps. Comparison of the measured pressure drop with the normal, single phase pressure drop provides an approximate indication of the relative void content of the circulating fluid.

4.4.6.6 Instrumentation for Mid-loop Operation

Generic Letter 88-17, "Loss of Decay Heat Removal," recommended that licensees implement certain actions prior to operation in a reduced Reactor Coolant System (RCS) inventory condition with irradiated fuel in the core. The concern stated in the Generic Letter is the potential consequences involved in preventing and recovering from loss of shutdown cooling while operating in a reduced inventory condition. The NRC recommended expeditious action and programmed enhancements to maintain sufficient equipment in an operable or available status so as to mitigate a loss of shutdown cooling or RCS inventory should they occur. Reduced inventory is defined by the NRC to be an RCS level lower than three feet below the reactor vessel flange.

In response to the NRC recommendations, the design includes (1) reliable indications of parameters that describe the state of the RCS and the performance of systems normally used to cool the RCS for both normal and accident conditions, (2) procedures to cover reduced inventory operation and (3) provisions for alternate sources of inventory for addition if necessary. The following is a brief description of the plant equipment, instrumentation and procedures that are used to comply with the recommendations of Generic Letter 88-17:

Reactor Coolant System Level Monitoring: At least two diverse RCS level indications are operational during reduced inventory conditions with irradiated fuel in the core. Continuous level indications are monitored in the Control Room and audible alarms sound on inadvertent transition in RCS level from the existing operating condition. The RCS level instrumentation consists of an RCS sight glass, wide range level indication provided by differential pressure measurement and three diverse narrow range level indicators provided by ultrasonic measurements (2) and differential pressure measurement (1). With exception of the sight glass, the RCS level instrumentation provides diverse indication, trend and low-level alarm capability in the control room via the Main Plant Computer System (MPCS) during all phases of operation under reduced inventory.

Reactor Coolant System Temperature Monitoring: When the reactor vessel head is located on the reactor vessel, two independent core exit temperature measurements are demonstrated to be operable prior to draining the RCS down to reduced inventory. The core exit temperature measurements are provided using the core exit thermocouple portion of the redundant Class 1E safety-related Inadequate Core Cooling Monitor.

Thermocouple readings are displayed on the Main Control Board and input into the MPCS. Mid-loop high temperature alarms are provided by the MPCS based on selection of the maximum reliable thermocouple temperature.

Residual Heat Removal System Performance: Continuous monitoring and trend capability of Residual Heat Removal System performance is provided in the Control Room by the MPCS. The RHR system parameters that are monitored include RHR loop flow, RHR heat exchanger inlet and outlet temperatures, RHR pump suction pressures and RHR pump motor current indications.

Administrative Controls: Controls are in place to implement specific actions to be taken when draining the RCS with irradiated fuel in the core. Required actions are based on the Westinghouse Owners Group reduced inventory project guidance and plant specific analyses. Plant procedures include the necessary information to determine equipment and/or operational requirements and limitations, including:

1. Prior to entry into a reduced inventory condition, controls are established to provide reasonable assurance that containment closure can be achieved before core is uncovered as a result of loss of decay heat removal. With the exception of penetrations that are in use or undergoing maintenance which are administratively controlled, at least one boundary of each containment penetration is maintained intact during reduced inventory operation. In the event of a loss of decay heat removal, administratively controlled penetrations are closed.
2. Prior to entering a reduced inventory condition, communication is established between the control room and a local nuclear systems operator in containment.
3. When operating at reduced inventory with steam generator nozzle dams in place, one centrifugal charging pump and one safety injection pump are available with a specified flow path to the reactor core. A gravity flow path from the Reactor Water Storage Tank (RWST) to the RCS is also made available as a secondary source. An adequate vent is provided to preclude RCS pressurization that could prevent gravity feed from the RWST and/or damage to the steam generator nozzle dams. Administrative controls assure availability of the redundant centrifugal charging and safety injection pumps upon unavailability of the operable pump.

4. When operating at reduced inventory with nozzle dams removed and the RCS vent closed for evacuation and fill, one centrifugal charging pump and one safety injection pump are available with specified flow paths to the reactor core. A gravity feed flow path from the RWST is also available for inventory addition as a secondary source. Administrative controls assure availability of the redundant centrifugal charging and safety injection pumps upon unavailability of the operable pump.

4.4.7 References

1. Christensen, J. A., Allio, R. J. and Biancheria, A., "Melting Point of Irradiated $U0_2$," WCAP-6065, February 1965.
2. Hellman, J. M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A (Proprietary), March 1975 and WCAP-8219-A, March 1975.
3. Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, TID-25887, 1972.
4. "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" Letter, E. P. Rahe, Jr., (Westinghouse) to J. R. Miller (NRC), NS-EPR-2515, dated October 9, 1981; "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" Letter, E. P. Rahe, Jr., (Westinghouse) to J. R. Miller (NRC), NS-EPR-2572, dated March 16, 1982.
5. Davidson, S. L. (Ed.), "VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610-P-A, April 1995.
6. Davidson, S. L. (Ed.), "Westinghouse Fuel Criteria Evaluation Report," WCAP-12488-A, October 1994.
7. Letter from J. Galloway (W) to R. J. Rodriguez (FPL) NF-NA-05-79, "Adjustment to DNB Margin Predictions from WRB-2M," dated November 18, 2005.
8. Tong, L. S., "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," J. Nucl. Energy, 21, 241-248 (1967).
9. Smith, L. D., et al, "Modified WRB-2 Correlation, WRB-2M, for predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," WCAP-15025-P-A, April 1999.
10. Letter from D. S. Collins (NRC) to J. A. Gresham (W), Subject: "Modified WRB-2 Correlation WRB-2M for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," February 3, 2006.

11. NP-2511-CCM Volumes 1-5, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores", Electric Power Research Institute.

12. Cadek, F. F., Motley, F. E. and Dominicis, D. P., "Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid," WCAP-7941-P-A (Proprietary), January 1975 and WCAP-7959-A, January 1975.

13. Rowe, D. S., Angle, C. W., "Crossflow Mixing Between Parallel Flow Channels During Boiling, Part II Measurements of Flow and Enthalpy in Two Parallel Channels," BNWL-371, part 2, December 1967.

14. Rowe, D. S., Angle, C. W., "Crossflow Mixing Between Parallel Flow Channels During Boiling, Part III Effect of Spacers on Mixing Between Two Channels," BNWL-371, part 3, January 1969.

15. Gonzalez-Santalo, J. M. and Griffith, P., "Two-Phase Flow Mixing in Rod Bundle Subchannels," ASME Paper 72-WA/NE-19.

16. Motley, F. E., Wenzel A. H., Cadek, F. F., "The Effect of 17x17 Fuel Assembly Geometry on Interchannel Thermal Mixing," WCAP-8298-P-A (Proprietary), January 1975 and WCAP-8299-A, January 1975.

17. Cadek, F. F., "Interchannel Thermal Mixing Vane Grids," WCAP-7667-P-A (Proprietary), January 1975 and WCAP-7755-A, January 1975.

18. Hochreiter, L. F., "Application of the THINC-IV Program to PWR Design," WCAP-8054 (Proprietary), October 1973, and WCAP-8195, October 1973.

19. Hochreiter, L. E., Chelemer, H. and Chu, P. T., "THINC-IV An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

20. Dittus, F. W., and Boelter, L. M. K., "Heat Transfer in Automobile Radiators of the Tubular Type," Calif. Univ. Publication In Eng., 2, No. 13, 443461 (1930).

21. Weisman, J., "Heat Transfer to Water Flowing Parallel to Tube Bundles," Nucl. Sci. Eng., 6, 78-79 (1959).

22. Thom, J. R. S., Walker, W. M., Fallon, T. A. and Reising, G. F. S., "Boiling in Sub-Cooled Water During Flow up Heated Tubes or Annuli," Prc. Instn. Mech. Engrs., 180, Pt. C, 226-46 (1955-66).

23. Hetsroni, G., "Hydraulic Tests of the San Onofre Reactor Model," WCAP-3269-8, June 1964.

24. Hetsroni, G., "Studies of the Connecticut-Yankee Hydraulic Model," NYO-3250-2, June 1965.

25. Idel'chik, I. E., "Handbook of Hydraulic Resistance," AEC-TR-6630, 1960.

26. Moody, L. F., "Friction Factors for Pipe Flow," Transaction of the American Society of Mechanical Engineers, 66, 671-684 (1944).

27. Not used

28. Not used

29. Not used

30. Hill, K. W., Motley, F. E., Catek, F. F., "Effect of Local Heat Flux Spikes on DNB in Non-Uniform Heated Rod Bundles," WCAP-8174, August 1973.

31. Not used

32. Not used

33. Not used

34. Not used

35. Not used

36. International Atomic Energy Agency, "Thermal Conductivity of Uranium Dioxide," Report of the Panel held in Vienna, April 1965, IAEA Technical Reports Series, No. 59, Vienna, The Agency, 1966.

37. Carter, F. D., "Inlet Orificing of Open PWR Cores," WCAP-9004, January 1969 (Proprietary) and WCAP-7836, January 1972 (Nonproprietary).

38. Not used

39. Not used

40. Kjaerheim, G., and Rolstad, E., "In-Pile Determination of $U0_2$ Thermal Conductivity, Density Effects, and Gap conductance," HPR-80, December 1967.

41. Kjaerheim, G., In-Pile Measurements of Center Fuel Temperatures and Thermal Conductivity Determination of Oxide Fuels, Paper IFA-175 Presented at the European Atomic Energy Society Symposium on Performance Experience of Water-Cooled Power Reactor Fuel, Stockholm, Sweden, October 1969.

42. Cohen, I., Lustman, B., and Eichenberg, D., "Measurement of the Thermal Conductivity of Metal-Clad Uranium Oxide Rods During Irradiation," WAPD-228, 1960.

43. Clough, D. J., and Sayers, J. B., "The Measurement of the Thermal Conductivity of $U0_2$ under Irradiation in the Temperature Range 150 to 1600C," AERE-R-4690, UKAEA Research Group, Harwell, December 1964.

44. Stora, J. P., et al., "Thermal Conductivity of Sintered Uranium Oxide under In-Pile Conditions," EURAEC-1095, 1964.

45. Devold, I., "A Study of the Temperature Distribution in $U0_2$ Reactor Fuel Elements," AE-318, Aktiebolaget Atomenergi, Stockholm, Sweden, 1968.

46. Balfour, M. G., Christensen, J. A., and Ferrari, H. M., "In-Pile Measurement of $U0_2$ Thermal Conductivity," WCAP-2923, 1966.

47. Howard, V. C., and Gulvin, T. G., "Thermal Conductivity Determinations on Uranium Dioxide by a Radial Flow Method," UKAEA IG-Report 51, November 1960.

48. Lucks, C. F., and Deem, H. W., "Thermal Conductivity and Electrical Conductivity of $U0_2$," in Progress Reports Relating to Civilian Applications, BMI-1448 (Revised) for June 1960, BNI-1489 (Revised) for December 1960, and BMI-1518 (Revised) for May 1961.

49. Daniel, J. L., Matolich, J., Jr., and Deem, H. W., "Thermal Conductivity of $U0_2$" HW-69945, September 1962.

50. Feith, A. D., "Thermal Conductivity of $U0_2$ by a Radial Heat Flow Method," TID-21668, 1962.

51. Vogt, J., Grandell, L., and Runfors, U., "Determination of the Thermal Conductivity of Unirradiated Uranium Dioxide," AB Atomenergi Report RMB-527, 1964, Quoted by IAEA Technical Report Series No. 59, "Thermal Conductivity of Uranium Dioxide."

52. Nishijima, T., Kawada, T., and Ishihata, A., "Thermal Conductivity of Sintered $U0_2$ and $A1_2/0_3$ at High Temperatures," Journal of the American Ceramic Society 48, pp 31-34, 1965.

53. Ainscough, J. B., and Wheeler, M. J., "Thermal Diffusivity and Thermal Conductivity of Sintered Uranium Dioxide, "Proceedings of the Seventh Conference of Thermal Conductivity, National Institute of Standards and Technology, Washington, p 467, 1968.

54. Godfrey, T. G., et al., "Thermal Conductivity of Uranium Dioxide and Armco Iron by an Improved Radial Heat Flow Technique," ORNL-3556, June 1964.

55. Stora, J. P., et al., "Thermal Conductivity of Sintered Uranium Oxide Under In-Pile Conditions," EURAEC-1095, August 1964.

56. Bush, A. J., "Apparatus for Measuring Thermal Conductivity to 2500C," Report 64-1P6-401-43 (Proprietary), Westinghouse Research Laboratories, February 1965.

57. Asamoto, R. R., Anselin, F. L., and Conti, A. E., "The Effect of Density on the Thermal Conductivity of Uranium Dioxide," GEAP-5493, April 1968.

58. Kruger, O. L., Heat Transfer Properties of Uranium and Plutonium Dioxide, Paper 11-N-68F, Presented at the Fall Meeting of Nuclear Division of American Ceramic Society, Pittsburgh, September 1968.

59. Leech, W. J., et al., "Revised PAD Code Thermal Safety Model, WCAP-8720, Addendum 2, October 1982.

60. Duncan, R. N., "Rabbit Capsule Irradiation of $U)_2$," CVTR Project, CVNA-142, June 1962.

61. Nelson, R. C., et al., "Fission Gas Release from $U0_2$ Fuel Rods with Gross Central Melting," GEAP-4572, July 1964.

62. Poncelet, C. G., "Burnup Physics of Heterogeneous Reactor Lattices," WCAP-6069, June 1965.

63. Nodvick, R. J., "Saxton Core II Fuel Performance Evaluation," WCAP-3385-56, Part II, "Evaluation of Mass Spectrometric and Radiochemical Analyses or Irradiated Saxton Plutonium Fuel," July 1970.

64. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design-and Safety Evaluation," WCAP-10851-P-A, August 1988.

References 65 through 68 are not used.

69. J. A. Boure, A. E. Bergles, and L. S. Tong, "Review of Two-Phase Flow Instability," Nucl. Engr. Design 25 (1973) p. 165-192.

70. R. T. Lahey and F. J. Moody, "The Thermal Hydraulics of a Boiling Water Reactor," American Nuclear Society, 1977.

71. P. Saha, M. Ishii, and N. Zuber, "An Experimental Investigation of the Thermally Induced Flow Oscillations in Two-Phase Systems," J. of Heat Transfer, Nov. 1976, pp. 616-662.

72. Virgil C. Summer FSAR, Docket #50-395.

73. Byron/Braidwood FSAR, Docket #50-456.

74. South Texas FSAR, Docket #50-498.

75. S. Kakac, T. N. Veziroglu, K. Akyuzlu, 0. Berkol, "Sustained and Transient Boiling Flow Instabilities in a Cross-Connected Four- Parallel-Channel Upflow System," Proc. of 5th International Heat Transfer Conference, Tokyo, Sept. 3-7, 1974.

76. H. S. Kao, C. D. Morgan, and W. B. Parker, "Prediction of Flow Oscillation in Reactor Core Channel," Trans. ANS, Vol. 16, 1973, pp. 212-213.

77. Ohtsubo A., and Uruwashi, S., "Stagnant Fluid due to Local Flow Blockage," J. Nucl. Sci. Technol., 9, No. 7, 433-434, (1972).

78. Basmer, P., Kirsh, D. and Schultheiss, G. F., "Investigation of the Flow Pattern in the Recirculation Zone Downstream of Local Coolant Blockages in Pin Bundles," Atomwirtschaft, 17, No. 8, 416-417, (1972). (In German).

79. Letter from H. A. Sepp (Westinghouse) to T. E. Collins (NRC), "Notification of FCEP Application for WRB-1 and WRB-2 Applicability to the 17x17 Modified LPD Grid Design for Robust Fuel Assembly Application," NSD-NRC-98-5618, March 25, 1998.

80. Davidson, S. L. (Editor), "Reference Core Report - VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985, "VANTAGE 5H Fuel Assembly," WCAP-10444-P-A, Addendum 2-A, April 1988.

81. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y X Sung, et al., April 1997.

82. Not used

83. Skaritka, J., ed, "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1, July 1979.

84. McFarlane, A. F., "Power Peaking Factors," WCAP-7912-P-A (Proprietary) and WCAP-7912-A (Nonproprietary), January 1975.

85. Letter From C. Berlinger (NRC) to E. P. Rahe Jr. (W), Subject: "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1986.

86. Not used

87. Not used

88. Not used

89. Not used

90. Not used

91. Not used

92. Generic Letter 88-17, "Loss of Decay Heat Removal," dated October 17, 1988.

93. NAESCO letter NYN-93154 from T. C. Feigenbaum (NAESCO) to USNRC, "Revised Commitment Related to Generic Letter 88-17," November 5, 1993.

94. PSNH letter NYN-89001 from G. S. Thomas (PSNH) to USNRC, "Response to Generic Letter 88-17," January 3, 1989.

95. PSNH letter NYN-89012 from G. S. Thomas (PSNH) to USNRC, "Response to Generic Letter 88-17," February 3, 1989.

4.5 REACTOR MATERIALS

4.5.1 Control Rod Drive System Structural Materials

4.5.1.1 Materials Specifications

All parts of the Control Rod Drive System exposed to the reactor coolant are made of metals which resist the corrosive action of the water. Three types of metals are used exclusively: stainless steels, nickel-chromium-iron and cobalt-based alloys. In the case of stainless steels, only austenitic and martensitic stainless steels are used. The martensitic stainless steels are not used in the heat-treated conditions which cause susceptibility to stress corrosion cracking or accelerated corrosion in the Westinghouse pressurized water reactor water chemistry. Materials with yield strength greater than 90,000 psi are 410 stainless steel, Haynes 25 and Inconel X-750; their usage and properties are presented in the following subsections.

a. Pressure Vessel

All pressure-containing parts of the CRDM comply with Section III of the ASME Boiler and Pressure Vessel Code, and are fabricated from austenitic (Type 304) stainless steel.

b. Coil Stack Assembly

The coil housings require a magnetic material. Both low carbon cast steel and ductile iron have been successfully tested for this application, with ductile iron eventually specified for the control rod drive mechanism (CRDM). The finished housings are zinc plated or flame sprayed to provide corrosion resistance.

Coils are wound on bobbins of molded Dow Corning 302 material, with double glass insulated copper wire. Coils are then vacuum impregnated with silicon varnish. A wrapping of mica sheet is secured to the coil outside diameter. The result is a well-insulated coil capable of sustained operation at 200°C.

c. Latch Assembly

Magnetic pole pieces are fabricated from Type 410 stainless steel. All nonmagnetic parts, except pins and springs, are fabricated from Type 304 stainless steel. Haynes 25 is used to fabricate link pins. Springs are made from nickel-chromium-iron alloy (Inconel X-750). Latch arm tips are clad with Stellite-6 to provide improved wearability. Hard chrome plate and Stellite-6 are used selectively for bearing and wear surfaces.

d. Drive Rod Assembly

The major portion of the drive rod assembly is a Type 410 stainless steel. The coupling is machined from Type 403 stainless steel. Other parts are Type 304 stainless steel with the exception of the springs which are nickel-chromium-iron alloy and the locking button which is Haynes 25.

4.5.1.2 Fabrication and Processing of Austenitic Stainless Steel Components

The discussions provided in Subsection 5.2.3 concerning the processes, inspections and tests on austenitic stainless steel components to assure freedom from increased susceptibility to intergranular corrosion caused by sensitization, and the discussions provided in Subsection 5.2.3 on the control of welding of austenitic stainless steels, especially control of delta ferrite, are applicable to the austenitic stainless steel pressure housing components of the CRDM.

4.5.1.3 Contamination Protection and Cleaning of Austenitic Stainless Steel

The CRDMs are cleaned prior to delivery in accordance with the guidance of ANSI 45.2.1. Process specifications in packaging and shipment are discussed in Subsection 5.2.3. Westinghouse personnel do conduct surveillance to ensure that manufacturers and installers adhere to appropriate requirements, as discussed in Subsection 5.2.3.

4.5.2 Reactor Internals Materials

4.5.2.1 Materials Specifications

The structural material for the reactor internals is Type 304 stainless steel. Parts not fabricated from Type 304 stainless steel include bolts and dowel pins which are fabricated from Type 316 stainless steel and radial support key bolts which are fabricated of Inconel X-750. There are no other materials used in the reactor internals or core support structures which are not otherwise included in the ASME Code, Section III, Appendix I.

4.5.2.2 Controls on Welding

The discussions provided in Subsection 5.2.3 are applicable to the welding of reactor internals and core support components.

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products and Fittings

The nondestructive examination of wrought seamless tubular products and fittings is in accordance with Section III of the ASME Code.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel Components

The discussions provided in Subsection 5.2.3 and Section 1.8 verify conformance of reactor internals and core support structures with Regulatory Guide 1.31. The discussion provided in Section 1.8 verifies conformance of reactor internals with Regulatory Guide 1.34.

The discussion provided in Section 1.8 verifies conformance of reactor internals and core support structures with Regulatory Guide 1.71.

4.5.2.5 Contamination Protection and Cleaning of Austenitic Stainless Steel

The discussions provided in Subsection 5.2.3 and Section 1.8 are applicable to the reactor internals and core support structures, and verify conformance with ANSI 45 specifications and Regulatory Guide 1.37.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

4.6.1 Information for Control Rod Drive System (CRDS)

The CRDS controls the power to the rod drive mechanism for rod movement in response to signals received from the Reactor Control System or from signals generated by reactor operator action.

The control rod drive mechanism is described in Subsection 3.9(N).4.1. The instrumentation and controls for the Reactor Trip System are described in Section 7.2 and the Reactor Control System is described in Section 7.7.

4.6.2 Evaluation of the CRDS

The analysis used to evaluate the CRDS is that known as a failure mode and effects analysis. Failure modes of several systems of the CRDS are identified, failure mechanisms attributable to identified failure modes are postulated, the methods used for failure detection are determined, and the effects of a failure on the CRDS operation are analyzed. This analysis is presented in tabular form in Reference 1. This study, and the analyses presented in Chapter 15, demonstrate that the CRDS performs its intended safety function by putting the reactor in a subcritical condition when a safety system setting is approached, with any assumed credible failures of a single active component.

Despite the extremely low probability of a common mode failure impairing the ability of the Reactor Trip System to perform its safety function, analyses have been performed in accordance with the requirements of WASH-1270. These analyses, documented in References 2 and 3, have demonstrated that acceptable safety criteria would not be exceeded even if the CRDS were rendered incapable of functioning during a reactor transient for which their function would normally be expected.

4.6.3 Testing and Verification of the CRDS

The tests performed on the CRDS are: (1) prototype tests of components of the CRDS prior assembly, (2) prototype tests of the CRDS in a simulated reactor environment, (3) tests of components following manufacturing, (4) onsite preoperational and initial startup tests and (5) periodic in-service tests. The test methods and acceptance criteria are discussed in Sections 4.2, 14.2, Subsection 3.9(N).4, and Technical Specification 3/4.1.3.

4.6.4 Information for Combined Performance of Reactivity Systems

As is indicated in Chapter 15, the only postulated events which assume credit for reactivity control systems, other than a reactor trip, to render the plant subcritical are the steam line break, feedwater line break, and loss-of-coolant accident. The Reactivity Control Systems for which credit is taken in these accidents are the Reactor Trip System and the Safety Injection System (SIS). Additional information on the CRDS is presented in Subsection 3.9(N).4 and on the SIS in Section 6.3. No credit is taken for the boration capabilities of the Chemical and Volume Control System (CVCS) as a system in the analysis of transients presented in Chapter 15. Information on the capabilities of the CVCS is provided in Subsection 9.3.4. The adverse boron dilution possibilities due to the operation of the CVCS are investigated in Subsection 15.4.6. Prior proper operation of the CVCS has been presumed as an initial condition to evaluate transients, and appropriate Technical Requirements have been prepared to ensure the correct operation or remedial action.

4.6.5 Evaluation of Combined Performance

The evaluation of the steam line break, feedwater line break, and the loss-of-coolant accident, which presume the combined actuation of the Reactor Trip System to the CRDS and the SIS, are presented in Subsections 15.1.5, 15.2.8 and 15.6.5. Reactor trip signals and safety injection signals for these events are generated from functionally diverse sensors and actuate diverse means of reactivity control, i.e., control rod insertion and injection of soluble poison.

Nondiverse but redundant types of equipment are utilized only in the processing of the incoming sensor signals into appropriate logic which initiates the protective action. This equipment is described in detail in Sections 7.2 and 7.3. In particular, note that protection from equipment failures is provided by redundant equipment and periodic testing. Effects of failures of this equipment have been extensively investigated as reported in Reference 4. This failure mode and effects analysis verifies that any single failure will not have a deleterious effect on the Engineered Safety Features Actuation System. Adequacy of the Emergency Core Cooling System and SIS performance under faulted conditions is verified in Section 6.3.

4.6.6 References

1. Shopsky, W. E., "Failure Mode and Effects Analysis (FMEA) of the Solid State Full Length Rod Control System," WCAP-8976, September 1977.
2. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.
3. Gangloff, W. C. and Loftus, W. D., "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706-L (Proprietary) and WCAP-7706 (Nonproprietary), July 1971.
4. Eggleston, F. T., Rawlins, D. H. and Petrow, J. R., "Failure Mode and Effects Analysis (FMEA) of the Engineering Safeguard Features Actuation System," WCAP-8584 (Proprietary) and WCAP-8760 (Nonproprietary), April 1976.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 4
REACTOR

TABLES

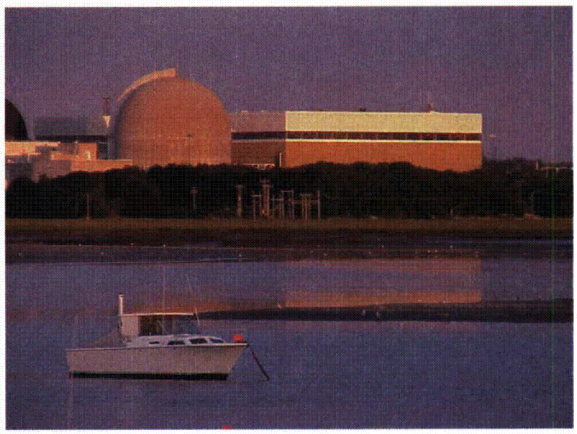


Table 4.1-1 **REACTOR DESIGN COMPARISON TABLE INITIAL CORE**

Thermal and Hydraulic Design Parameters	Seabrook Unit 1	W. B. McGuire Units 1 & 2
Reactor core heat output (MWt)	3411	3411
Reactor core heat output (10^6 Btu/hr)	11,641	11,641
Heat generated in fuel (%)	97.4	97.4
System pressure, nominal (psia)	2250	2250
System pressure, minimum steady state (psia)	2200	2220
Coolant Flow		
Total thermal flow rate (10^6 lbm/hr)	142.1	140.3
Effective flow rate for heat transfer 10^6 lbm/hr)	133.9	134.0
(Effective flow area for heat transfer ft^2)	51.1	51.1
Average velocity along fuel rods (ft/sec)	16.7	16.7
Average mass velocity (10^6 lbm/hr-ft^2)	2.62	2.62
Coolant Temperature		
Nominal inlet (°F)	558.8	558.1
Average rise in vessel (°F)	59.4	60.2
Average rise in core (°F)	62.6	62.7
Average in core (°F)	591.8	589.4
Average in vessel (°F)	588.5	588.2
Heat Transfer		
Active heat transfer surface area (ft^2)	59,700	59,700
Average heat flux, (Btu/hr-ft^2)	189,800	189,800
Maximum heat flux for normal operation, (Btu/hr-ft^2)	474,400[a]	440,300[a]
Average linear power (kW/ft)	5.44	5.44
Peak linear power for normal operation (kW/ft)	13.6[a]	12.6[a]
Heat flux hot channel factor, F_Q	2.50[b]	2.32[b]
Peak fuel central temperature at peak linear power for prevention of centerline melt (°F)	4700	4700
Core Mechanical Design Parameters		

Thermal and Hydraulic Design Parameters	Seabrook Unit 1	W. B. McGuire Units 1 & 2
Design	RCC canless, 17x17	RCC canless, 17x17
Number of fuel assemblies	193	193
$U0_2$ rods per assembly	264	264
Rod pitch (in.)	0.496	0.496
Overall dimensions (in.)	8.426x8.426	8.426x8.426
Fuel weight, as $U0_2$ (lb.)	222,739	222,739
Clad weight (lb.)	45,234	45,234
Number of grids per assembly	8 - Type R	8 - Type R
Loading technique	3 region nonuniform	3 region nonuniform
Fuel Rods		
Number	50,952	50,952
Outside diameter (in.)	0.374	0.374
Diametral gap (in.)	0.0065	0.0065
Clad thickness (in.)	0.0225	0.0225
Clad material	Zirconium Alloy	Zircaloy-4
Fuel Pellets		
Material	$U0_2$ sintered	$U0_2$ sintered
Density (% of Theoretical)	95	95
Diameter (in.)	0.3225	0.3225
Length (in.)	0.387	0.530
Rod Cluster Control Assemblies		
Neutron absorber		
Full length	Ag-In-Cd	Ag-In-Cd
Part length	---	Ag-In-Cd
Cladding Material	Austenitic SS	Type 304 SS-cold worked

Thermal and Hydraulic Design Parameters	Seabrook Unit 1	W. B. McGuire Units 1 & 2
Clad thickness		
Ag-In-Cd (in.)	0.0185	---
Number of clusters, full length/part length	57/0	53/8
Number of absorber rods per cluster	24	24
Core Structure		
Core barrel, I.D./O.D. (in.)	148.0/152.5	148.0/152.5
Thermal shield	Neutron pad design	Neutron pad design
Structure Characteristics		
Core diameter, equivalent (in.)	132.7	132.7
Core height, active fuel (in.)	144.0	144.0
Reflector Thickness and Composition		
Top, water plus steel (in.)	~10	~10
Bottom, water plus steel (in.)	~10	~10
Side, water plus steel (in.)	~15	~15
H_20/U molecular ratio lattice (cold)	2.41	2.41

a. This limit is associated with the maximum value of F_Q for normal operation.

b. This is the maximum value of F_Q for normal operation.

TABLE 4.1-2 **ANALYTICAL TECHNIQUES IN CORE DESIGN**

Analysis	Technique	Computer Code	Section Reference
Mechanical design core internals loads, deflections, and stress analysis	Static and dynamic modeling	Blowdown code, FORCE, finite element, structural analysis code, and others	3.7(N) 3.9(N) 3.9(N)
Fuel rod design			
Fuel performance characteristics (temperature, internal pressure, clad stress, etc.)	Semi-empirical thermal model of fuel rod with consideration of fuel changes, heat transfer, fission gas release, etc.	Westinghouse fuel rod design model	4.2 4.3 4.4
Nuclear design			
1. Cross sections	40 Group 2D neutron transport theory	CASMO-3 Phoenix – P	4.3
2. 3D power distributions, boron concentrations, reactivity coefficients, kinetic parameters, control rod worths, reactor and fuel assembly criticality	3D 2 Group advanced	SIMULATE-3 ANC	4.3
3. Steam line break, rod ejection doppler flattening factor	3D 2 Group advanced	ANC	15.0
Thermal-hydraulic design			
1. Steady state	Subchannel analysis of local fluid conditions in rod bundles, including inertial and crossflow resistance terms	VIPRE-01	4.4
2. Transient departure from nucleate boiling	Subchannel analysis of local fluid conditions in rod bundles during transient	VIPRE-01	4.4

TABLE 4.1-3 DESIGN LOADING CONDITIONS FOR REACTOR CORE COMPONENTS

1. Fuel assembly weight
2. Fuel assembly spring forces
3. Internals weight
4. Control rod trip (equivalent static load)
5. Differential pressure
6. Spring preloads
7. Coolant flow forces (static)
8. Temperature gradients
9. Differences in thermal expansion
   a. Due to temperature differences
   b. Due to expansion of different materials
10. Interference between components
11. Vibration (mechanically or hydraulically induced)
12. One or more loops out of service
13. All operational transients listed in Table 3.9(N)-1
14. Pump overspeed
15. Seismic loads (Operating Basis Earthquake and Safe Shutdown Earthquake)
16. Blowdown forces (due to cold and hot leg reactor coolant pipe breaks)

TABLE 4.3-1 REACTOR CORE DESCRIPTION (Typical Low Leakage Cycle Design)

Active Core	
Equivalent diameter (in.)	132.7
Active fuel height, first core (in.)	144.0
Height-to-diameter ratio	1.08
Total cross section area (ft^2)	96.06
H_20/U molecular ratio, lattice (Cold)	2.41
Reflector Thickness and Composition	
Top, water plus steel (in.)	~10
Bottom, water plus steel (in.)	~10
Side, water plus steel (in.)	~15
Fuel Assemblies	
Number	193
Rod array	17x17
Rods per assembly	264
Rod pitch (in.)	0.496
Overall transverse dimensions (in.)	8.426x8.426
Fuel weight (as UO_2) (lb.)	~220,000
Zirlo/Zircaloy weight (lb.)	46,920 – 53,300
Number of grids per assembly	8 – Structural 3 – IFM 1 – P-Grid
Composition of grids	Inconel or Zirconium Alloy
Weight of grids, effective in core (lb.)	2324 – 3150
Number of guide thimbles per assembly	24
Composition of guide thimbles	Zirconium Alloy
Diameter of guide thimbles, upper part (in.)	
(17x17 RFA)	0.442 I.D. x 0.482 O.D.
Diameter of guide thimbles, lower part (in.)	
(17x17 RFA)	0.397 I.D. x 0.439 O.D.
Diameter of instrument guide thimbles (in.)	
(17x17 RFA)	0.442 I.D. x 0.482 O.D.
Fuel Rods	
Number	50,952
Outside diameter (in.)	0.374
Diameter gap (in.)	0.0065
Clad thickness (in.)	0.0225
Clad material	Zirconium Alloy

Fuel Pellets	
Material	UO_2 Sintered
Density (percent of theoretical)	95
Fresh Fuel enrichments w/o	
Typical Low Enrichment in Split	3.6-4.4
Typical High Enrichment in Split	4.0-4.8
Diameter (in.)	0.3225
Length (in.)	0.387
Mass of UO_2 per foot of fuel rod (lb./ft)	~0.36
Rod Cluster Control Assemblies	
Neutron absorber	Ag-In-Cd
Composition	80%-15%-5%
Diameter (in.)	0.341 Ag-In-Cd
Density (lb./in.^{3})	0.367 Ag-In-Cd
Cladding material	Type 304, Cold Worked Stainless Steel
Clad thickness (in.)	0.0185
Number of clusters - full length	57
Number of absorber rods per cluster	24
Integral Fuel Burnable Absorbers (IFBA)	
Number	6,000 – 12,000 (typical)
Material	Zr_2B
Coating Thickness (in.)	0.0002 – 0.0004
Boron loading (mg/in)	1.57 - 3.14
Initial reactivity worth (%$\Delta\rho$)	Dependent on Number in Assembly
Excessive Reactivity	
Maximum fuel assembly k_∞ (cold clean, unborated water)	1.430
Maximum core reactivity (cold, zero power, beginning of cycle)	1.210

TABLE 4.3-2 NUCLEAR DESIGN PARAMETERS (Typical Low Leakage Cycle Design)

Core Average Linear Power, kW/ft, including densification effects	5.84	
Total Heat Flux Hot Channel Factor, F_Q	2.50	
Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}$	1.65	
Reactivity Coefficients[+]	Design Limits	Best Estimate
Doppler-only Power, Upper Curve Coefficients, pcm/% power[++], H2P to HFP	-19.4 to -12.6	-16 to -9
(See Figure 15.1-3, Sh. 1), Lower Curve	-9.55 to -6.05	-13 to -8.5
Doppler Temperature Coefficient pcm/°F[++]	-3.2 to –0.9	-2.1 to –1.3
Moderator Temperature Coefficient, pcm/°F[++]	+5. to –55	+2. to –43.
Boron Coefficient, pcm/ppm[++]	-16 to –5	-14.5 to –6.5
Rodded Moderator Density, pcm/gm/cc[++]	$\leq 0.54 \times 10^5$	$\leq 0.41 \times 10^5$
Delayed Neutron Fraction and Lifetime		
β_{eff} BOL, (EOL)	0.0075, (0.0044)	0.0065, (0.0048)
Control Rods		
Rod Requirements	See Table 4.3-3	
Maximum Bank Worth, pcm	< 2000	<1300
Maximum Ejected Rod Worth	See Chapter 15	

Radial Factor Peak Pin FΔh (BOL to EOL)		
Unrodded		1.44 to 1.40
D bank		1.44 to 1.40
D + C		1.55 to 1.44
Boron Concentrations		
Zero Power, $k_{eff} = 0.978$, Cold Rod Cluster Control Assemblies Out		2110
Zero Power, $k_{eff} = 0.987$, Hot Rod Cluster Control Assemblies Out		2133
Refueling Boron Concentration (Lower Limit)		2100 (or $K_{eff} \leq 0.95$)
Zero Power, $k_{eff} \leq 0.95$, Cold Rod Cluster Control Assemblies In		1809
Zero Power, $k_{eff} = 1.00$, Hot Rod Cluster Control Assemblies Out		1936
Full Power, No Xenon, $k_{eff} = 1.0$, Hot Rod Cluster Control Assemblies Out		1736
Full Power, Equilibrium Xenon, $k_{eff} = 1.0$, Hot Rod Cluster Control Assemblies Out		1338
Reduction with Fuel Burnup Cycle ppm/GWd/Mtu++++		See Figure 4.3-3

\+ Uncertainties are given in Subsection 4.3.3.3

++ Note: 1 pcm = (percent mille) 10^{-5} Δρ where Δρ is calculated from two statepoint values of k_{eff} by $\ln(K_2/K_1)$.

++++ Gigawatt Day (GWd) = 1000 Megawatt Day (1000 MWd).

Table 4.3-3 REACTIVITY REQUIREMENTS FOR ROD CLUSTER CONTROL ASSEMBLIES

Reactivity Effects, Percent	Beginning-of-Life (First Cycle)	End-of-Life (First Cycle)	End-of-Life (Typical Low Leakage Cycle)
1. Control requirements			
Fuel temperature, Doppler (%Δρ)	1.36	1.12	1.53
Moderator temperature** (%Δρ)	0.15	1.22	1.20
Redistribution (%Δρ)	0.50	0.85	***
Rod insertion allowance (%Δρ)	0.50	0.50	0.45
2. Total control (%Δρ)	2.51	3.69	3.18

Reactivity Effects, Percent	Beginning-of-Life (First Cycle)	End-of-Life (First Cycle)	End-of-Life (Typical Low Leakage Cycle)
3. Estimated Rod Cluster Control Assembly worth (57 rods, Ag-In-Cd)			
a. All full length assemblies inserted (%Δρ)	8.73	8.83	7.42
b. All but one (highest worth) assemblies inserted (%Δρ)	7.69	7.76	6.58
4. Estimated Rod Cluster Control Assembly credit with 10 percent adjustment to accommodate uncertainties, 3b - 10 percent (%Δρ)	6.92	6.98	5.93
5. Shutdown margin available, 4-2 (%Δρ)	4.41	3.29	2.75****

**** The design basis minimum shutdown is 1.3%.

TABLE 4.3-4 **AXIAL STABILITY INDEX PRESSURIZED WATER REACTOR CORE WITH A 12 FOOT HEIGHT**

Burnup (MWd/Mtu)	F_Z	C_B (ppm)	Stability Index (hr^{-1}) Exp	Calc
1550	1.34	1065	-0.041	-0.032
7700	1.27	700	-0.014	-0.006
		Difference:	+0.027	+0.026

TABLE 4.3-5 TYPICAL NEUTRON FLUX LEVEL (N/CM2-SEC) AT FULL POWER

	E>1.0MeV	**0.111MeV<E <1.0MeV**	**0.3eV≤E <0.111MeV**	**E<0.3EV**
Core Center	9.79×10^{13}	9.82×10^{13}	1.91×10^{14}	1.98×10^{13}
Core Outer Radius At Mid-Height	2.47×10^{13}	2.61×10^{13}	5.29×10^{13}	5.19×10^{13}
Core Top, on Axis	5.20×10^{13}	5.35×10^{13}	1.10×10^{14}	1.30×10^{13}
Pressure Vessel Inner Diameter Azimuthal Peak, Core Mid-Height	1.93×10^{10}	2.05×10^{10}	3.55×10^{10}	1.67×10^{10}

TABLE 4.3-6

(DELETED)

TABLE 4.3-7

(DELETED)

TABLE 4.3-8

(DELETED)

TABLE 4.3-9

(DELETED)

TABLE 4.3-10

(DELETED)

TABLE 4.4-1 THERMAL AND HYDRAULIC COMPARISON TABLE

Design Parameters	Seabrook Cycle 10 Design	Seabrook Uprate
Reactor core heat output (MWt)	3411	3659 (analyzed)
Reactor core heat output (10^6 Btu/hr)	11,641	12,485
Heat generated in fuel (%)	97.4	97.4
System pressure, nominal (psia)	2250	2250
System pressure, minimum steady state (psia)	2200	2200
DNB Correlation	WRB-2[1]	WRB-2M[8]
Correlation Limit Value	1.17[1]	1.14[8]
Design Limit Value		
Typical flow channel	1.26	1.22
Thimble (cold wall) flow channel	1.24	1.22
Safety Analysis Limit Value		
Typical flow channel	1.91	1.47
Thimble (cold wall) flow channel	1.91	1.47
Minimum DNBR at nominal conditions		
Typical flow channel	3.02[2]	2.73[10]
Thimble (cold wall) flow channel	2.88[2]	2.67[10]
Coolant Flow		
Total thermal flow rate (10^6 lb_m /hr)	145.7[3]	142.75[9]
Effective flow rate for heat transfer (10^6 lb_m /hr)	138.7[3]	133.0[9]
Effective flow area for heat transfer (ft^2)	51.3	51.1[9]
Average velocity along fuel rods (ft/sec)	17.1[3]	15.6[9]
Average mass velocity (10^6 lb_m/hr-ft^2)	2.71[3]	2.46[9]
Coolant Temperature		
Nominal inlet (°F)	559.5[3]	557.5[4]
Average rise in vessel (°F)	58.0[3]	63.2[4]
Average rise in core (°F)	60.6[3]	67.2[4]
Average in core (°F)	591.4[3]	593.1[4]
Average in vessel (°F)	588.5	589.1

Design Parameters	Seabrook Cycle 10 Design	Seabrook Uprate
Heat Transfer		
Active heat transfer, surface area (ft^2)	59,700	59,700
Average heat flux (Btu/hr-ft^2)	189,800	203,500
Maximum heat flux for normal operation (Btu/hr-ft^2)	474,500[5]	508,800[5]
Average linear power (Kw/ft)	5.445	5.84[5]
Peak linear power for normal operation (Kw/ft)	13.6[5]	14.6[5]
Pressure Drop		
Across core (psi)	28.5±2.85[7]	28.6
Across vessel, including nozzle (psi)	48.7±7.3[7]	48.7

1 For conditions outside the range of applicability of WRB-2, the W-3 correlation is used with a correlation limit of 1.45 in the pressure range of 500 to 1000 psia and 1.30 for pressures above 1000 psia.

2 This value is associated with the current design power distribution at 100 % rated power: a 1.60/1.04 = 1.54 FΔH value for V5H and RFA (w/IFMs) chopped cosine axial power shape. Values for the most adverse power distribution within the Axial Flux Difference LCO band are cycle dependent and may be slightly lower.

3 At minimum measured flow conditions.

4 At thermal design flow conditions

5 This limit is associated with the current design value of $F_Q = 2.50$.

6 Based on the original best estimate reactor flow rate as discussed in Section 5.1, and with thimble plug assemblies inserted.

7 For RFA (w/IFMs) based on a measured flow of 404,000 GPM Thimble Plugs Inserted.

8 For conditions outside the range of applicability of WRB-2M, the WRB-2 or W-3 correlation is used. The W-3 correlation limits are 1.45 in the pressure range of 500 to 1000 psia and 1.30 for pressures above 1000 psia.

9 Based on minimum measured flow = 383,000 gpm, best estimate bypass flow = 6.8%, 2250 psia, vessel average temperature = 589.1°F.

10 This value associated with $F_{\Delta H}$ = 1.587 = 1.67/1.04, 100% power and 1.55 cosine axial shape.

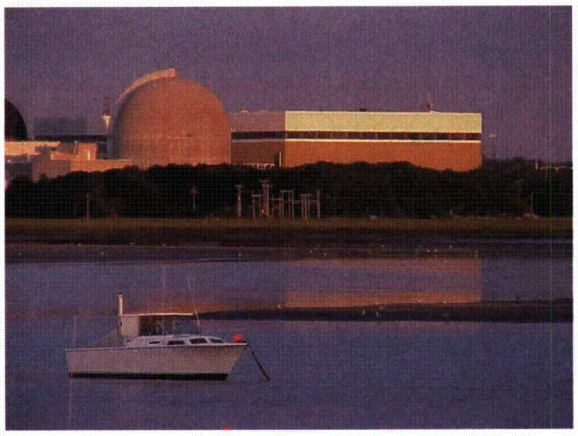
TABLE 4.4-2 VOID FRACTIONS AT NOMINAL REACTOR CONDITIONS

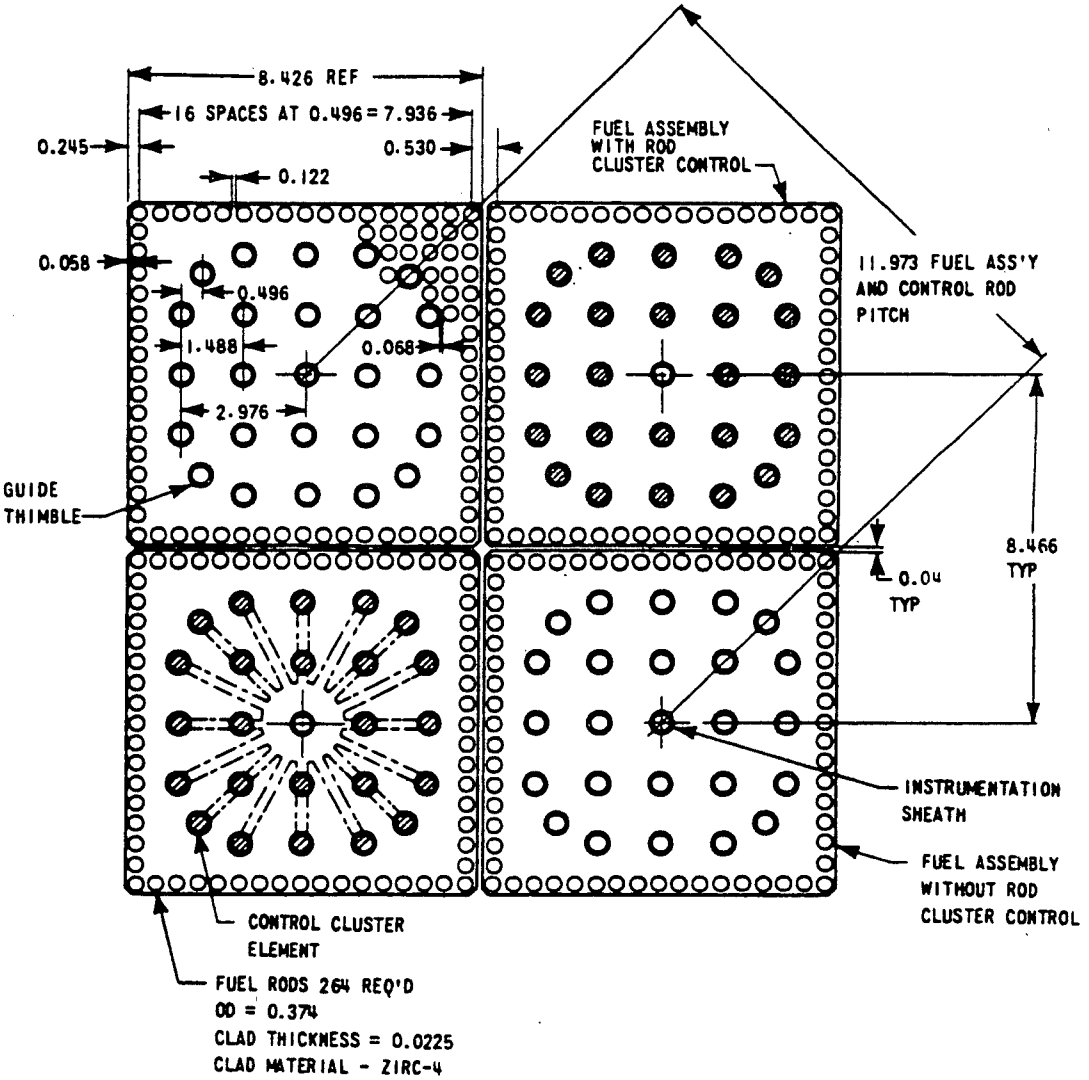
	Average (%)	Maximum (%)
Core	0.0	-
Hot Subchannel	0.3	7.0

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

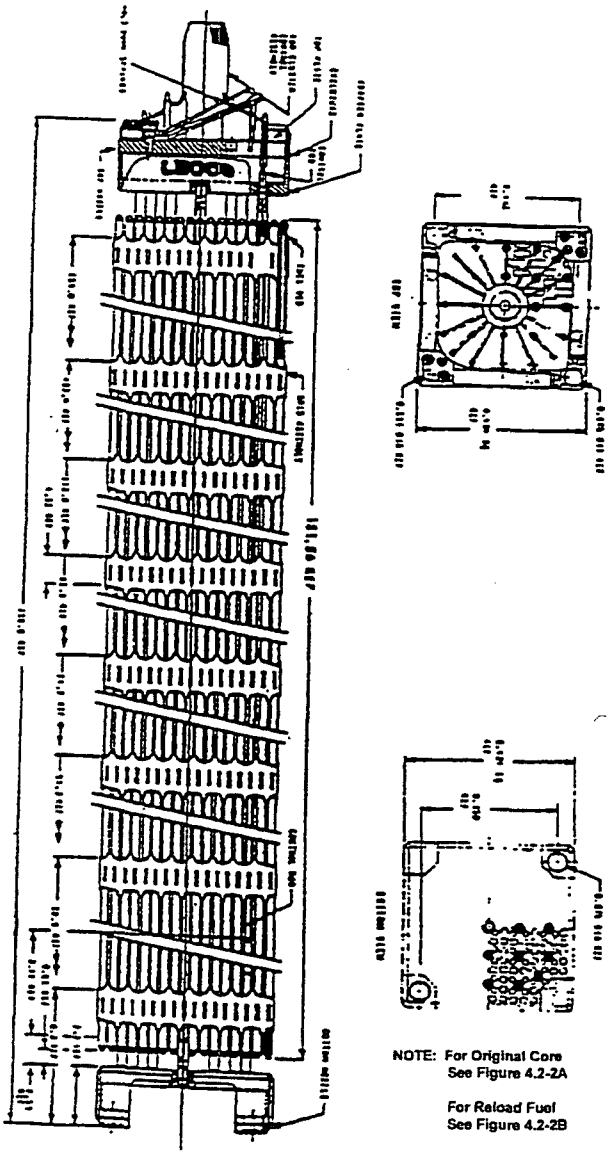
CHAPTER 4
REACTOR

FIGURES



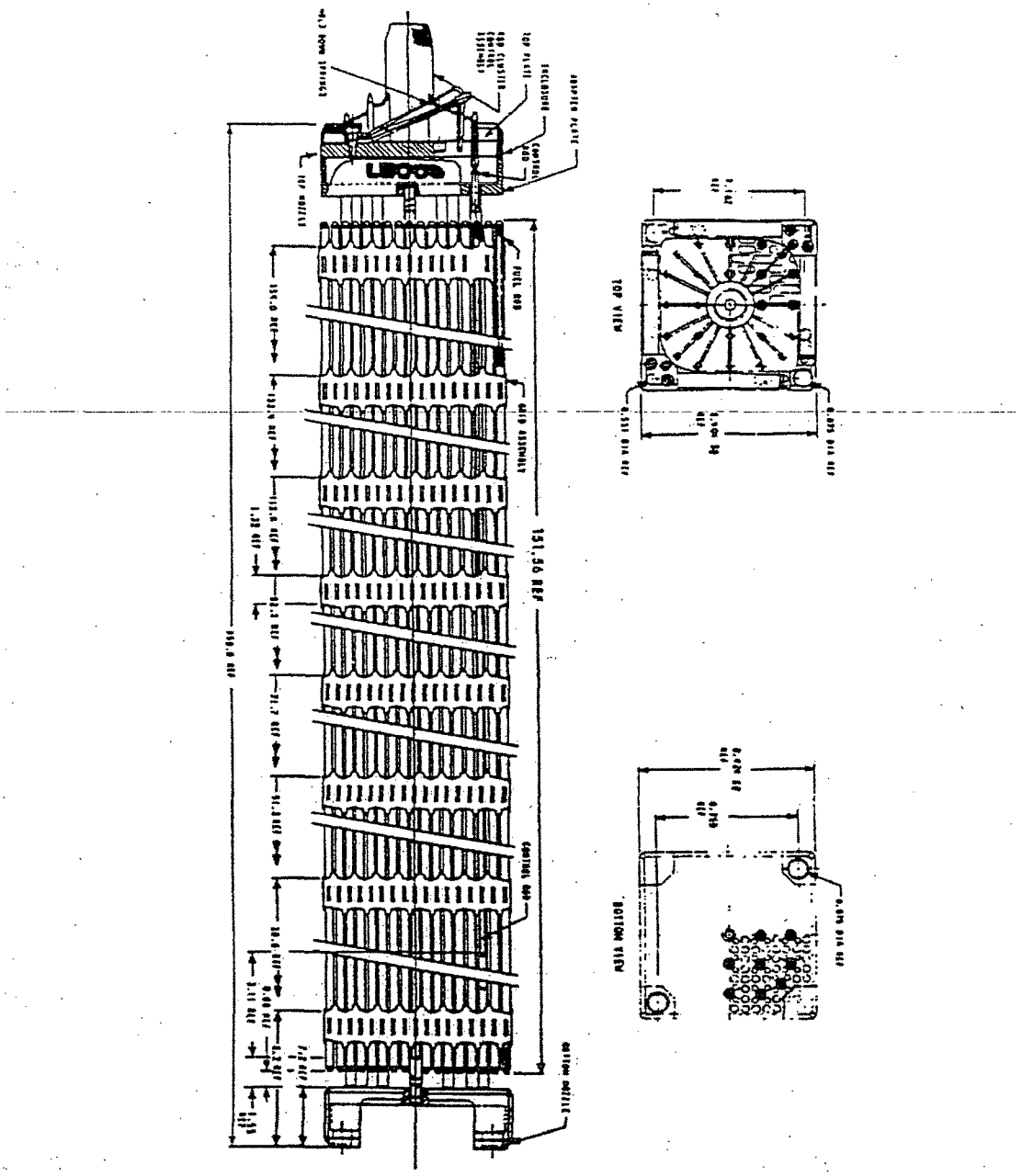


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Assembly Cross Section - 17x17	
		Figure 4.2-1

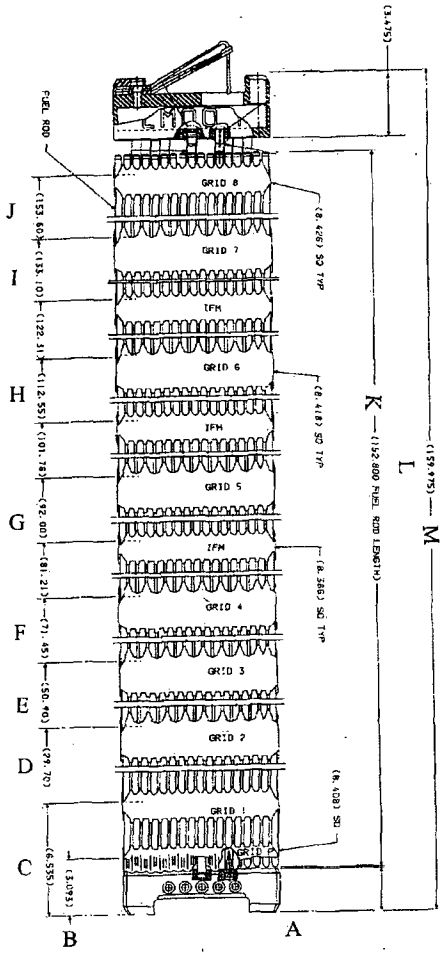


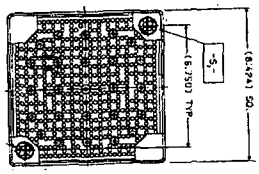
G:\Word\Images_P\Images_P\UFSAR\422.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Assembly Outline – 17X17	
		Figure 4.2-2



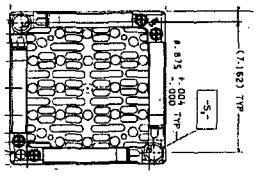
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Assembly Outline - 17x17 Original Core	
		Figure 4.2-2A





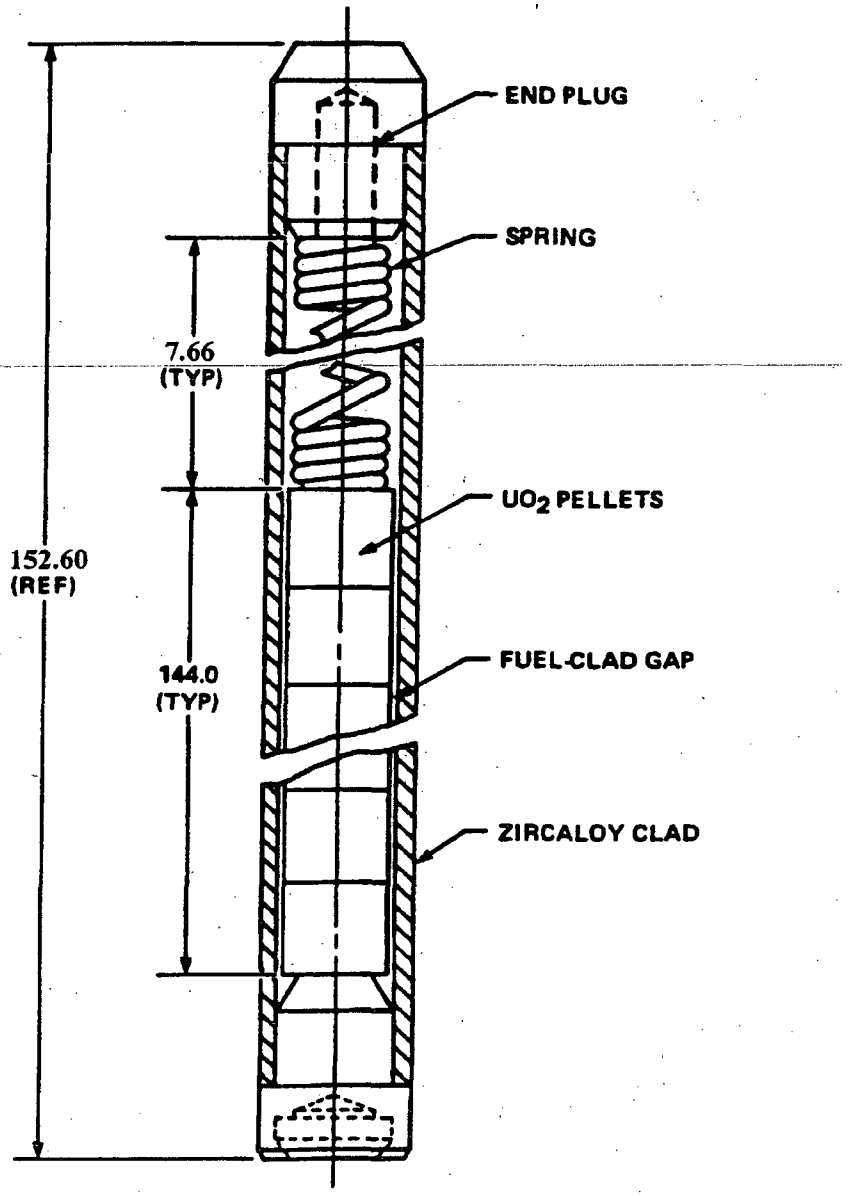
Dim.	17 X 17 STD	V5H* w/P-grid	RFA
A	2.383	2.383	2.383
B	N/A	3.093	3.093
C	5.84	6.535	6.535
D	30.26	29.70	29.70
E	50.81	50.90	50.90
F	71.36	71.45	71.45
G	91.91	92.00	92.00
H	112.46	112.55	112.55
I	133.01	133.10	133.10
J	153.60	153.40	153.60
K	152.20	152.60	152.61
L	156.50	156.30	156.50
M	159.98	159.775	159.98

*Dimensions also apply to V5H with IFMs



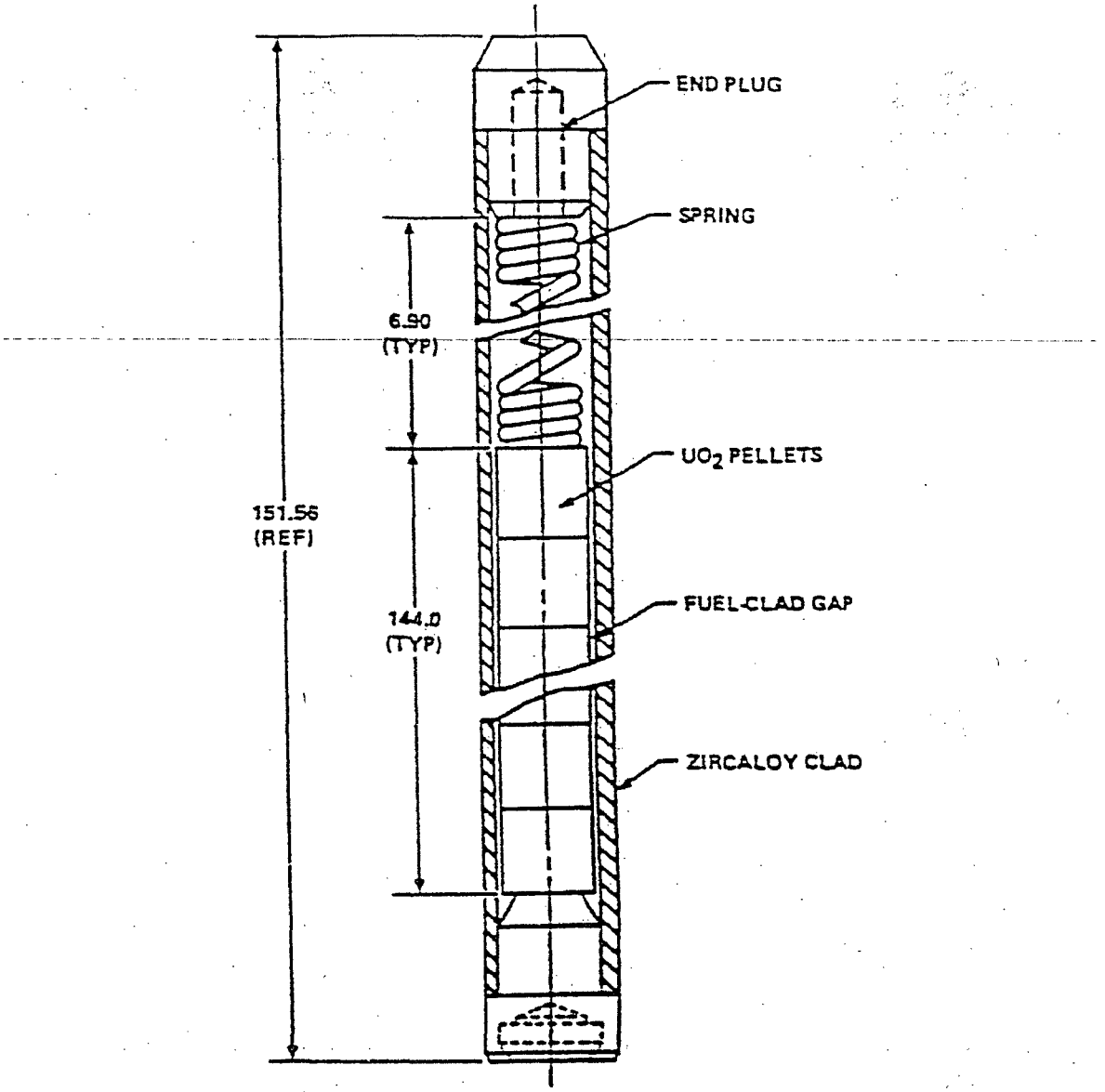
G:\Word\Images_P\UFSAR\422b.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Assembly Outline - 17X17 Reload Fuel	
		Figure 4.2-2B



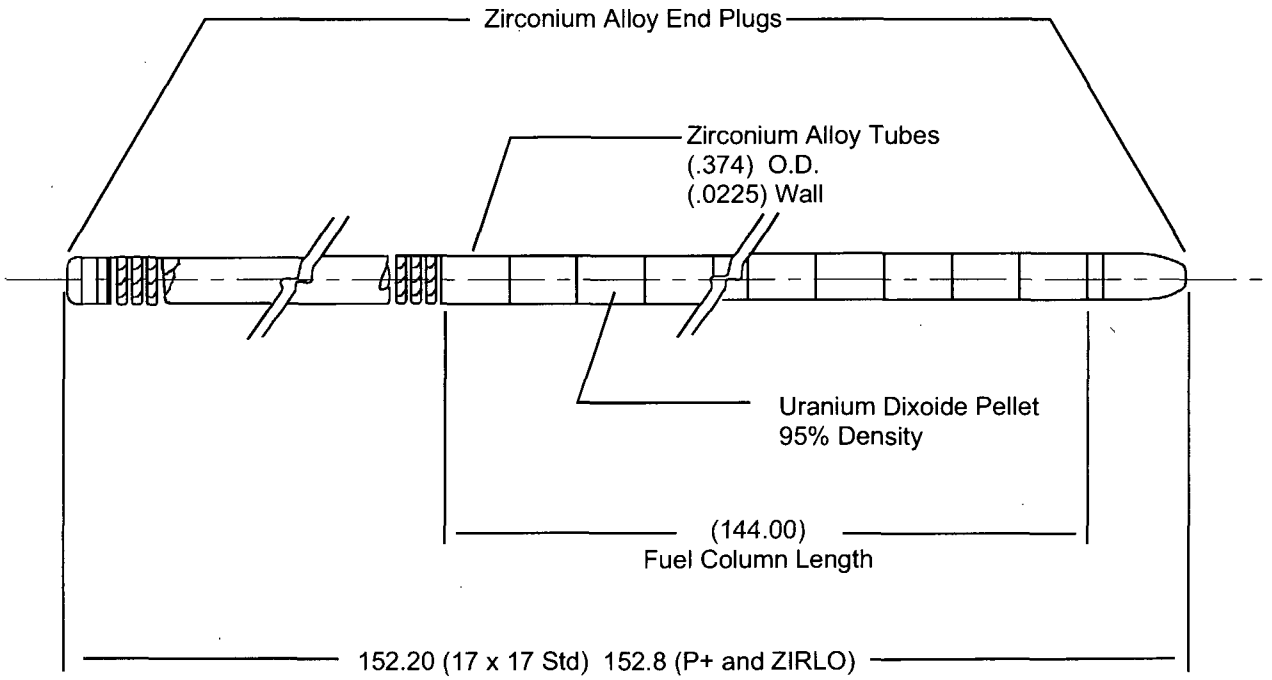
SPECIFIC DIMENSIONS DEPEND ON DESIGN VARIABLES SUCH AS PRE-PRESSURIZATION, POWER HISTORY, AND DISCHARGE BURNUP

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Rod Schematic	
		Figure 4.2-3



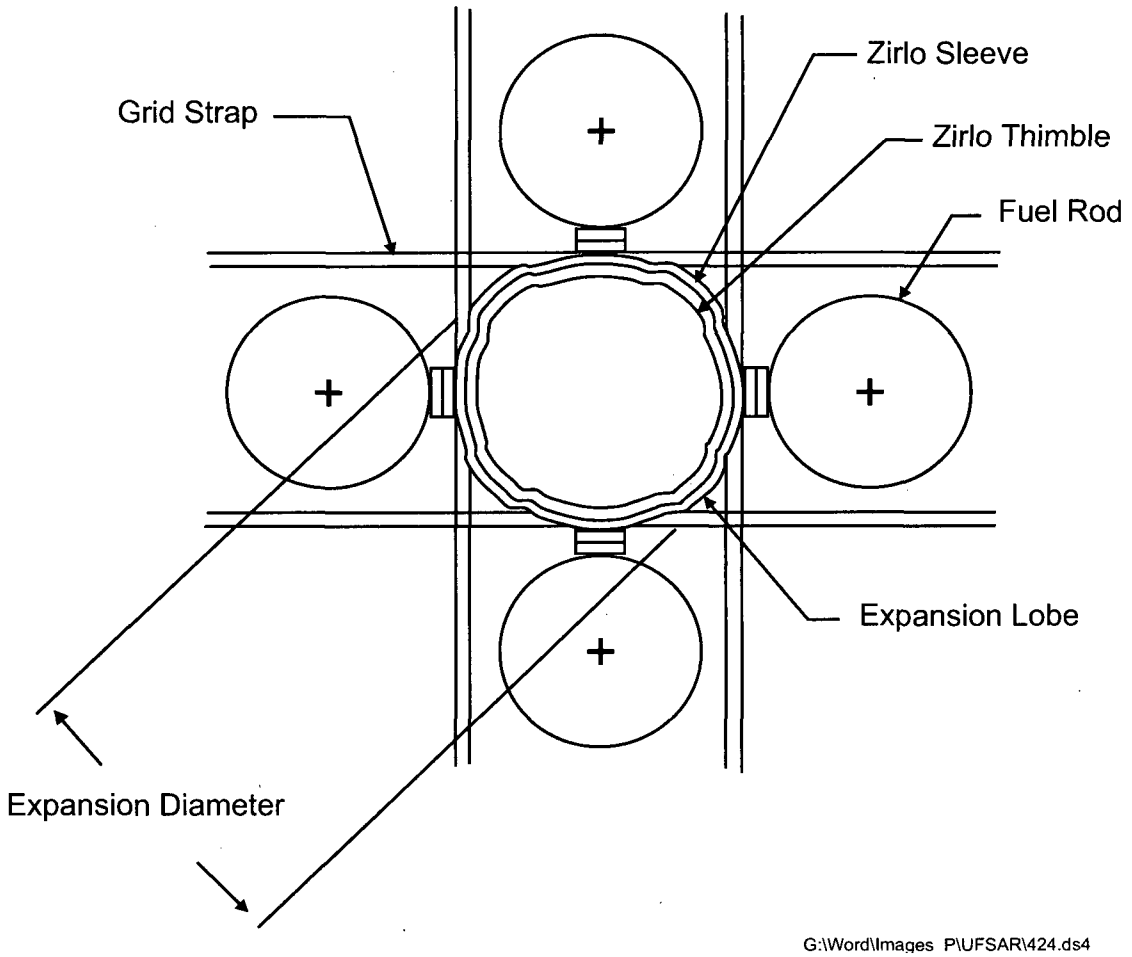
SPECIFIC DIMENSIONS DEPEND ON DESIGN VARIABLES SUCH AS PRE-PRESSURIZATION, POWER HISTORY, AND DISCHARGE BURNUP

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Rod Schematic - Original Core	
		Figure 4.2-3A

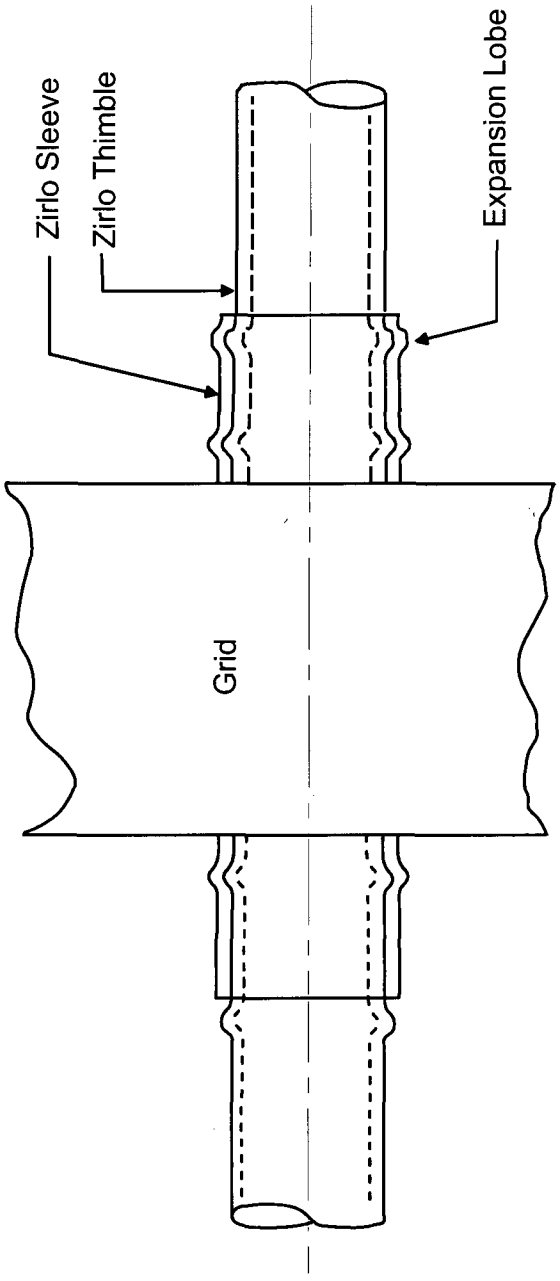


G:\Word\Images_P/UFSAR\423b.dsf

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Rod Schematic - Reload Fuel	
	Rev. 12	Figure 4.2-3B

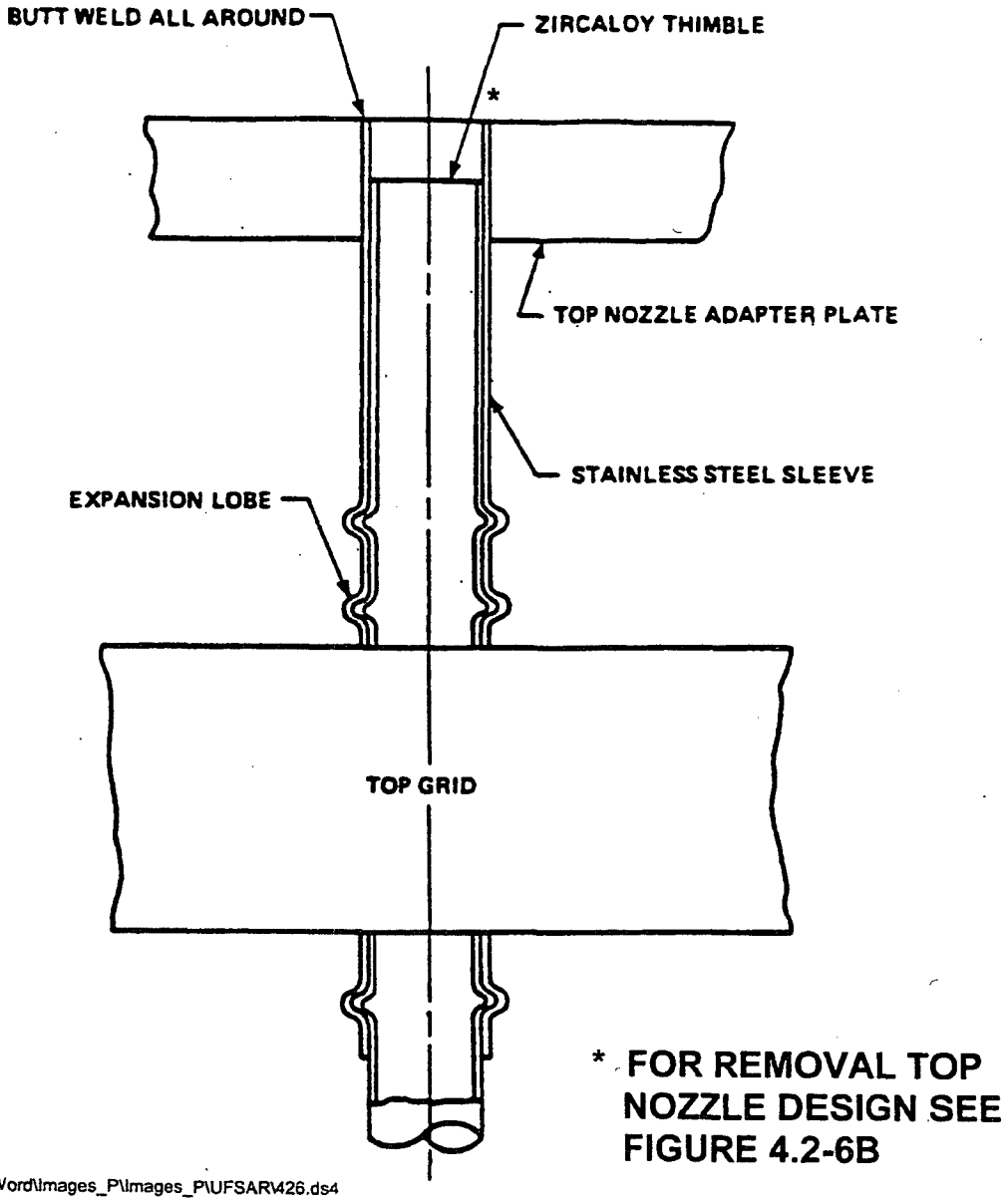


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Plan View	
		Figure 4.2-4

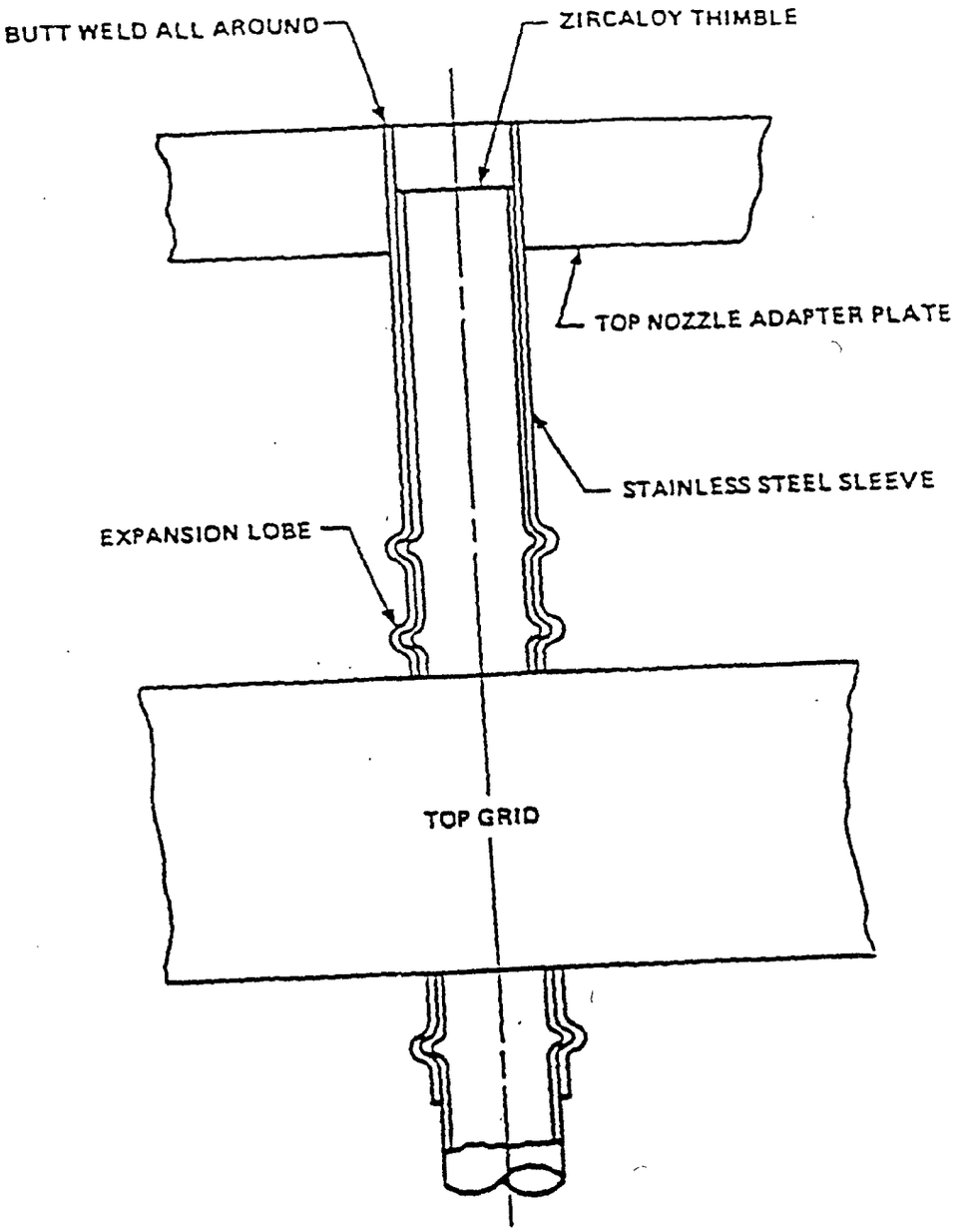


G:\Word\Images_P\UFSAR\425.ds4

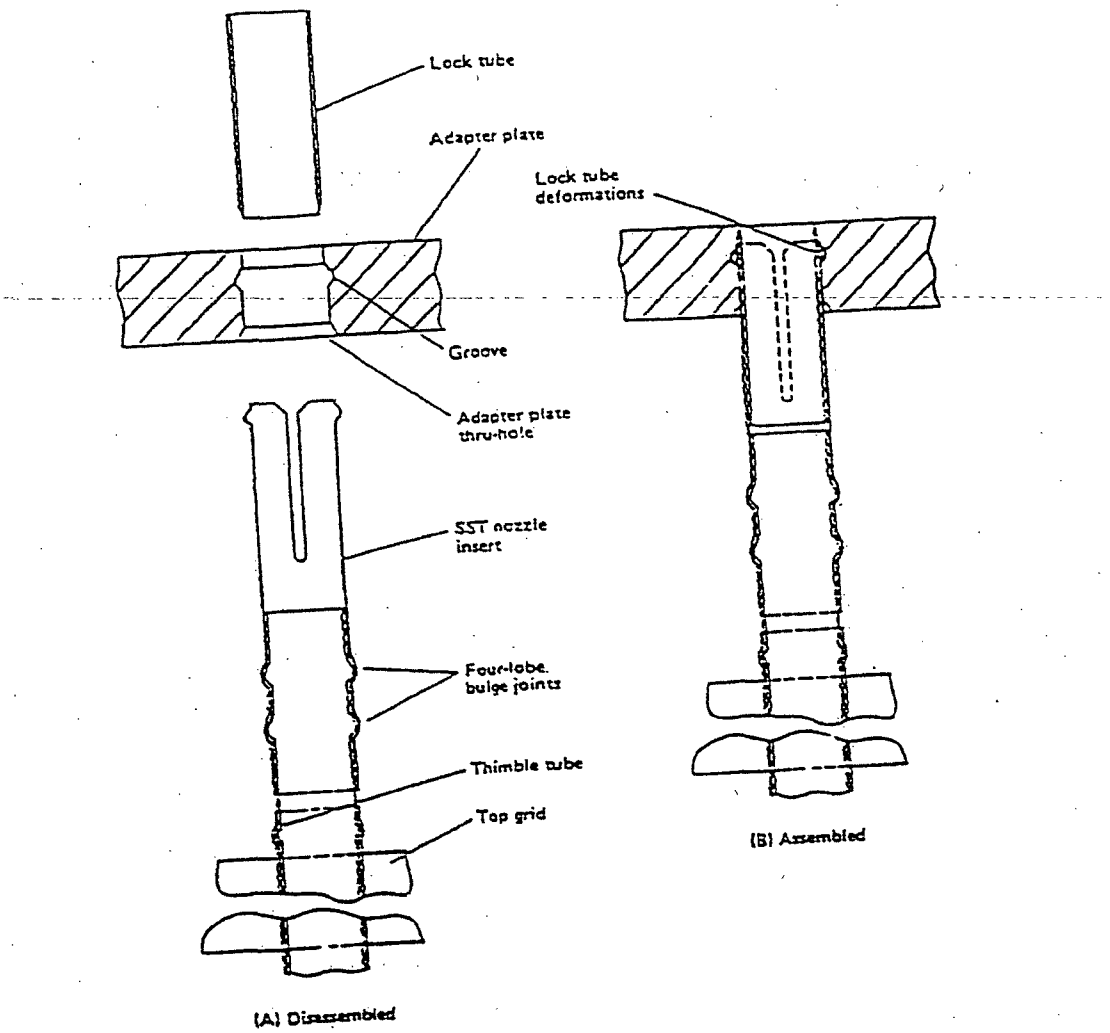
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Elevation View	
		Figure 4.2-5



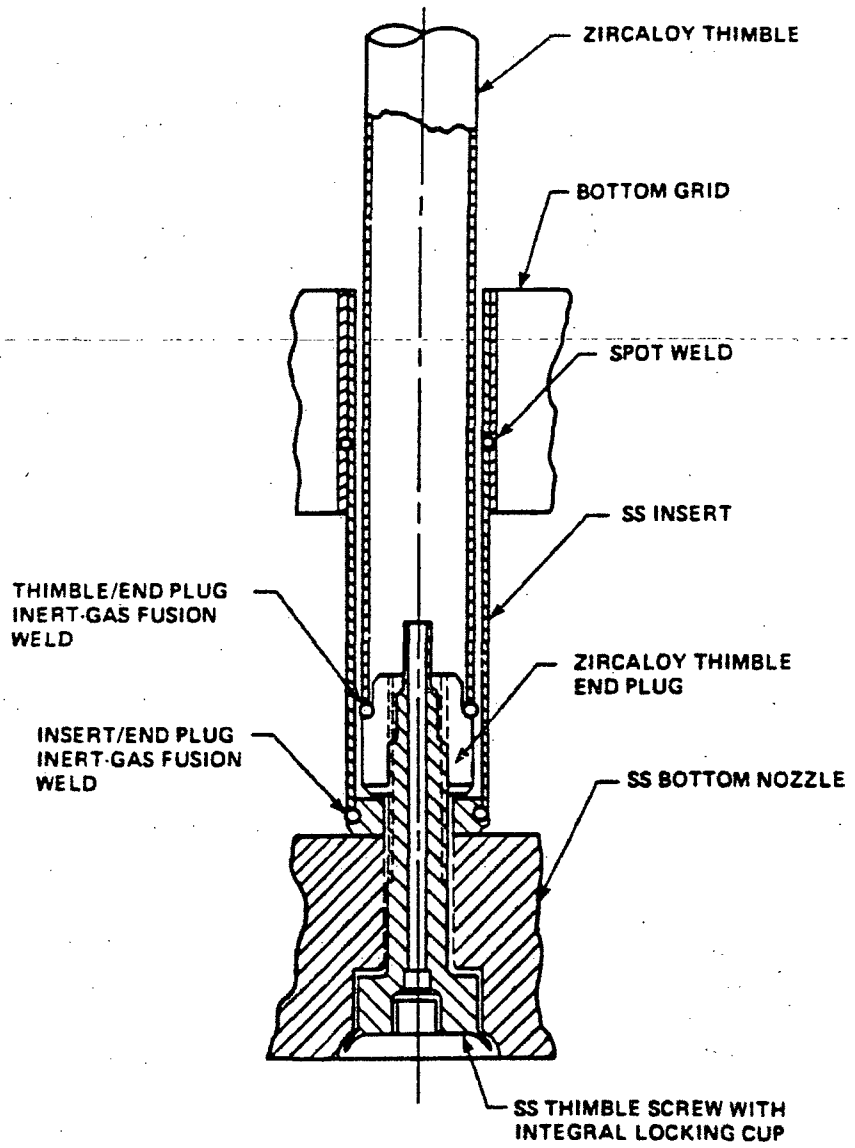
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Top Grid to Nozzle Attachment	
		Figure 4.2-6



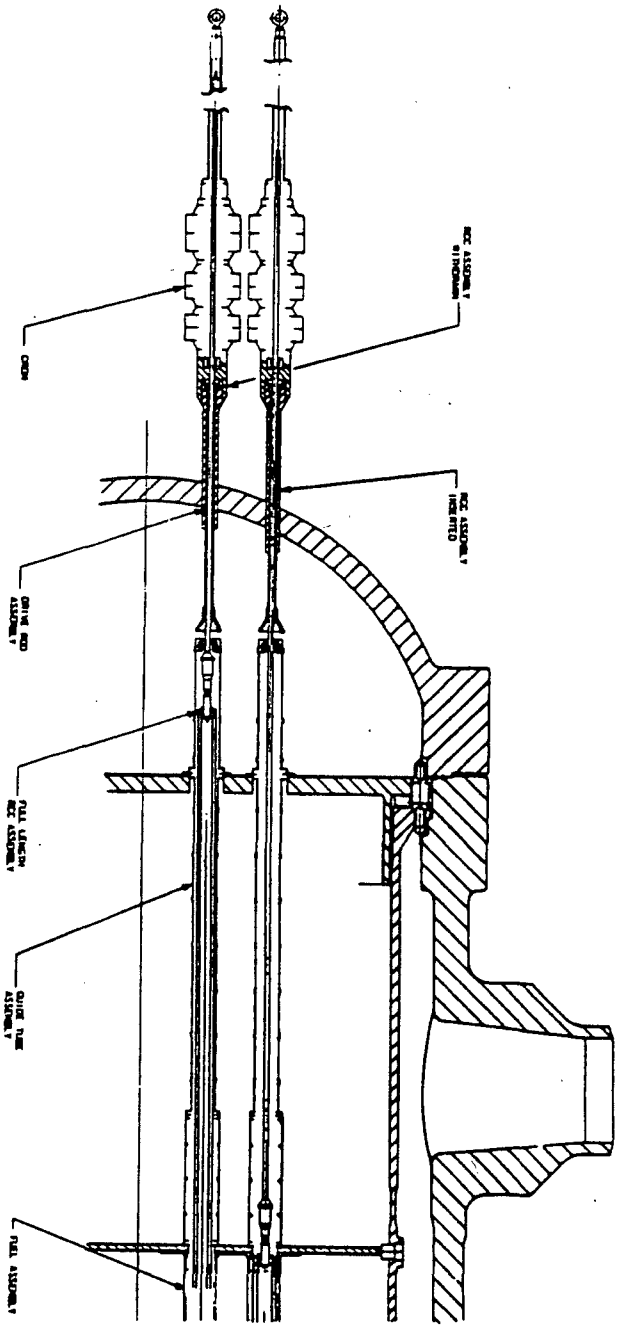
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Top Grid to Nozzle Attachment - Original Core	
		Figure 4.2-6A



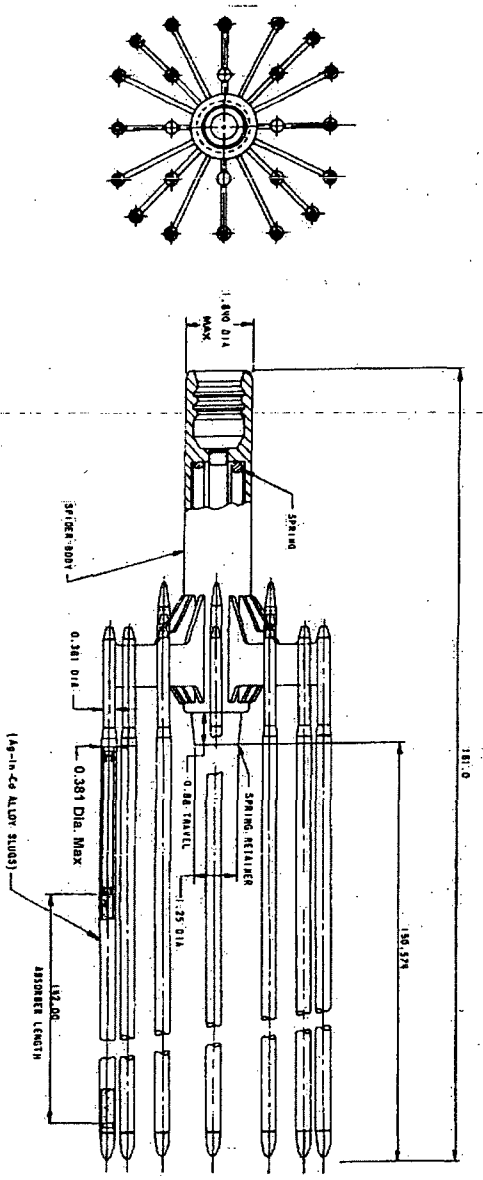
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Removable Top Nozzle (RTN) Top Grid to Nozzle Attachment - Reload Fuel	
		Figure 4.2-6B



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Guide Thimble to Bottom Nozzle Joint	
		Figure 4.2-7

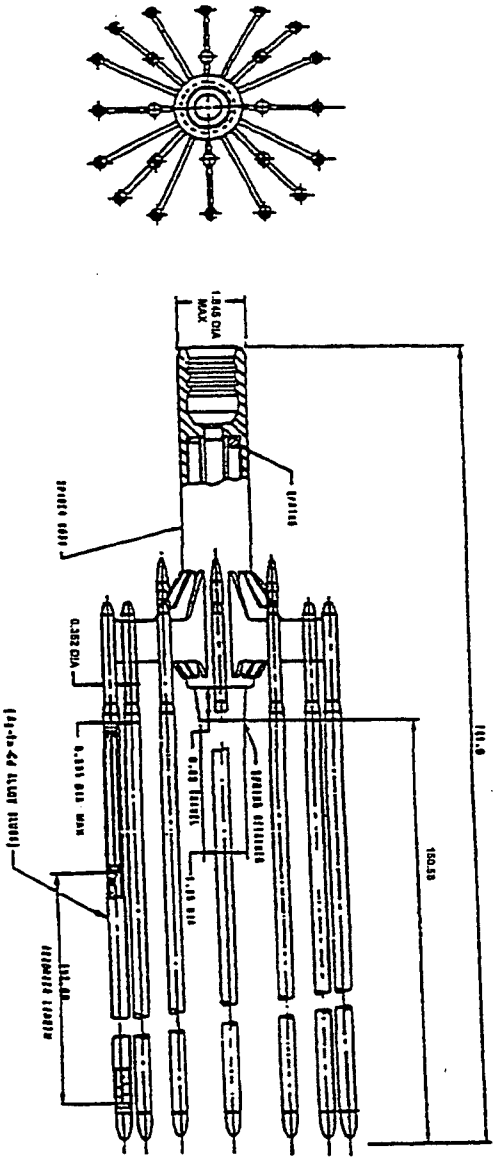


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Rod Cluster Control and Drive Rod Assembly with Interfacing Components	
		Figure 4.2-8



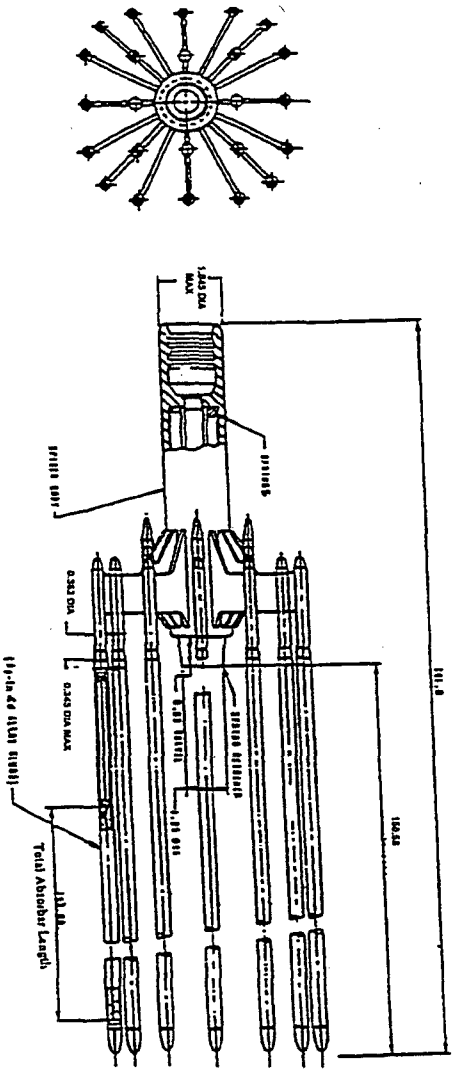
G:\Word\Images_P\UFSAR\429.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Rod Cluster Control Assembly Outline	
		Figure 4.2-9



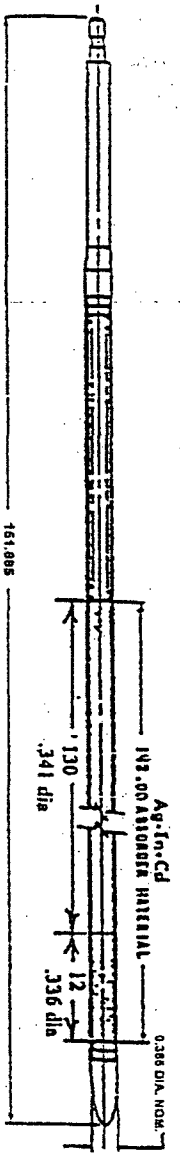
G:\Word\Images_P\Images_P\UFSAR\429A.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Rod Cluster Control Assembly Outline - Original RCCAs	
		Figure 4.2-9A



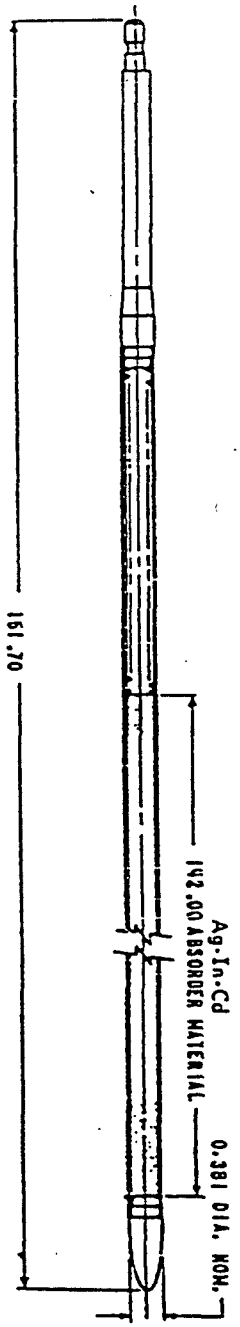
G:\Word\Images_P\Images_P\UFSAR\429B.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Rod Cluster Control Assembly Outline - Replacement RCCAs	
		Figure 4.2-9B

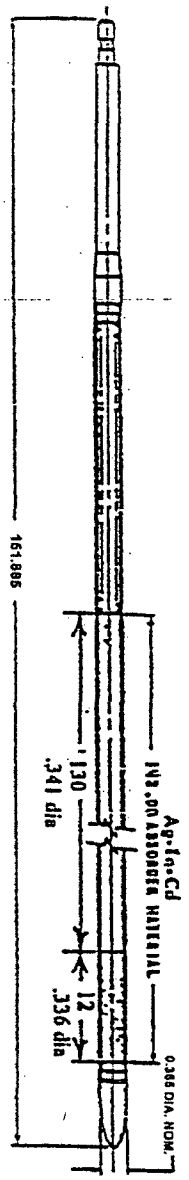


G:\Word\Images_P\Images_P\UFSAR\4210B.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Ag-In-Cd Absorber Rod	
		Figure 4.2-10

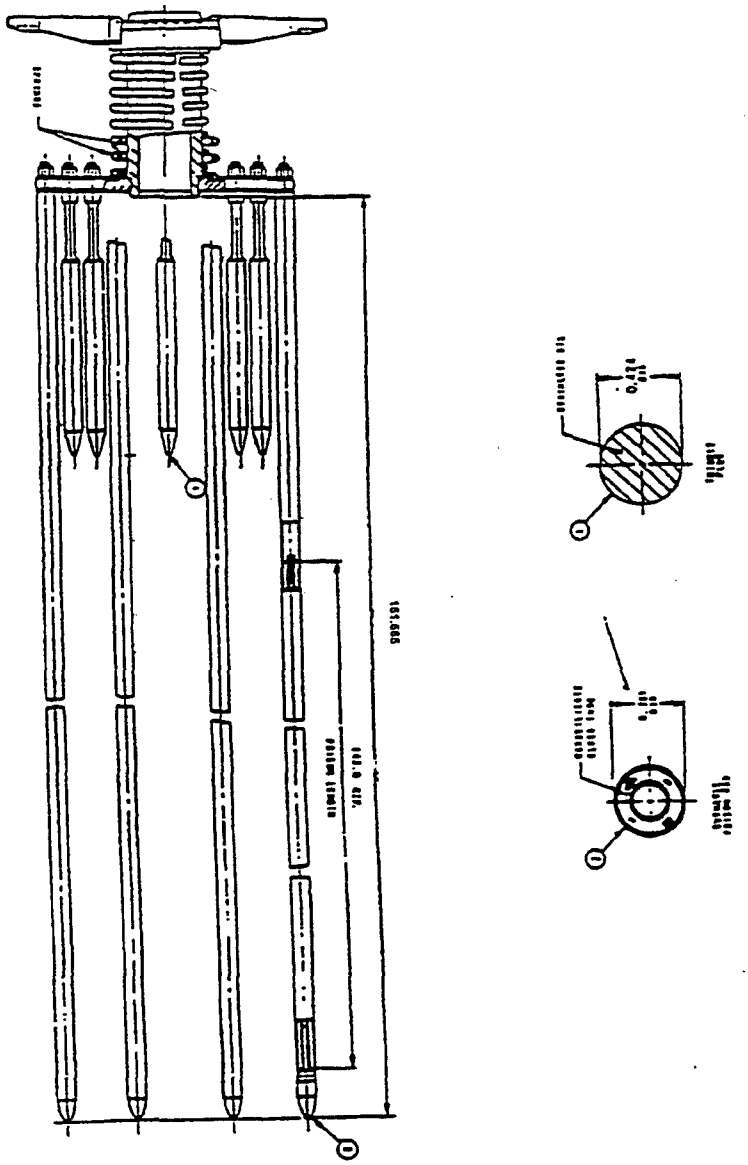


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Ag-In-Cd Absorber Rod - Original RCCAs	
		Figure 4.2-10A



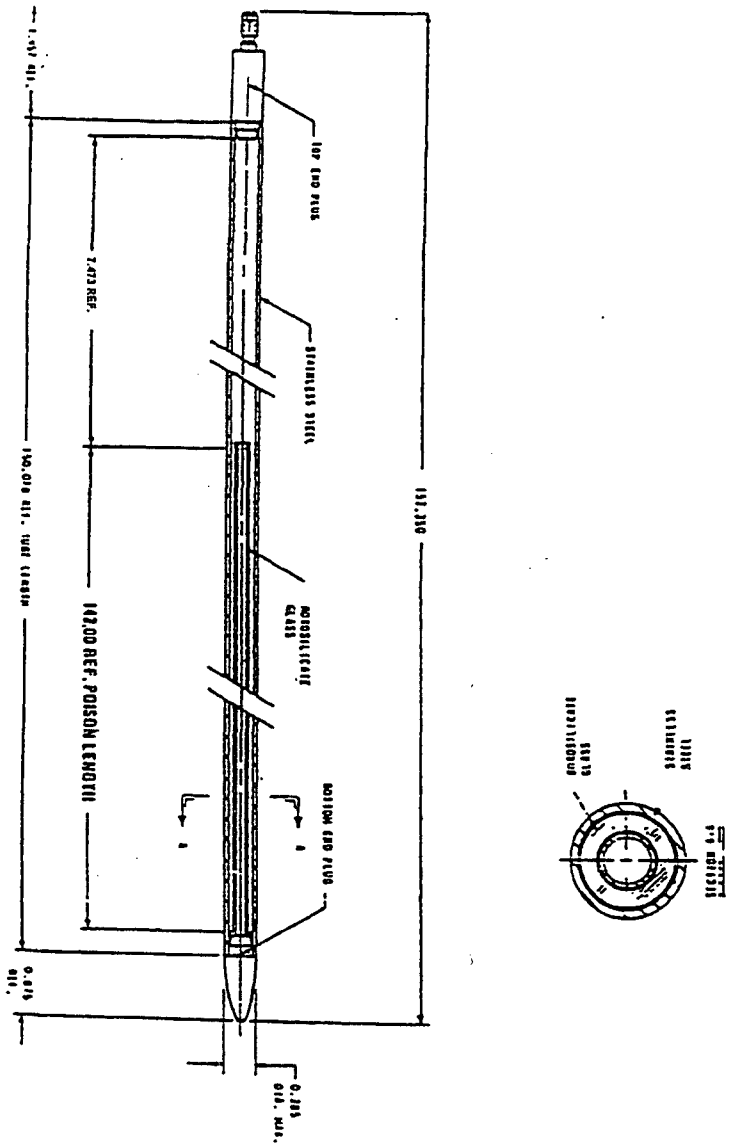
G:\Word\Images_P\Images_P\UFSAR\4210B.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Ag-In-Cd Absorber Rod - Replacement RCCAs	
		Figure 4.2-10B



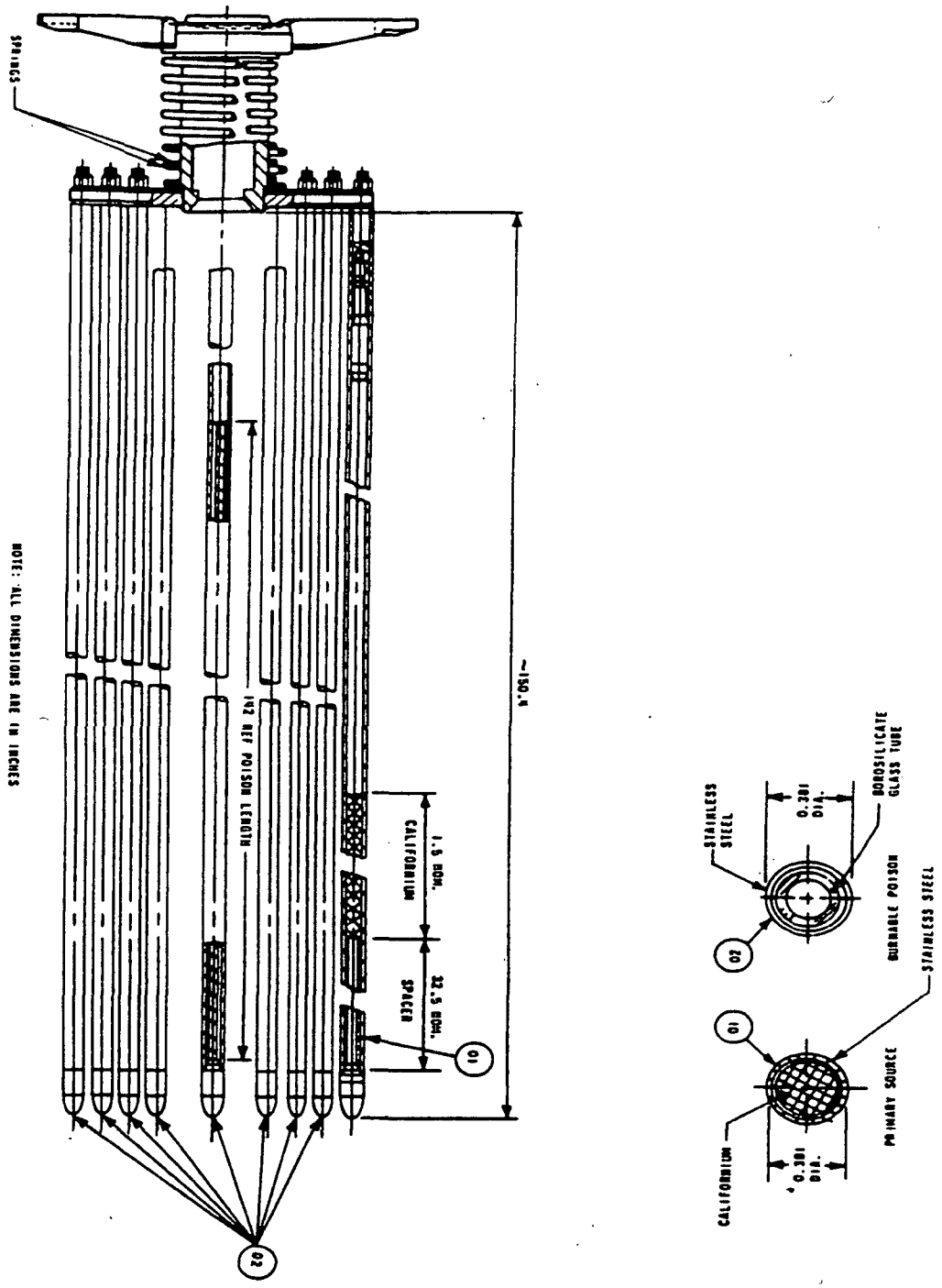
G:\Word\Images_P\Images_P\UFSAR\4211.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Burnable Poison Assembly	
		Figure 4.2-11

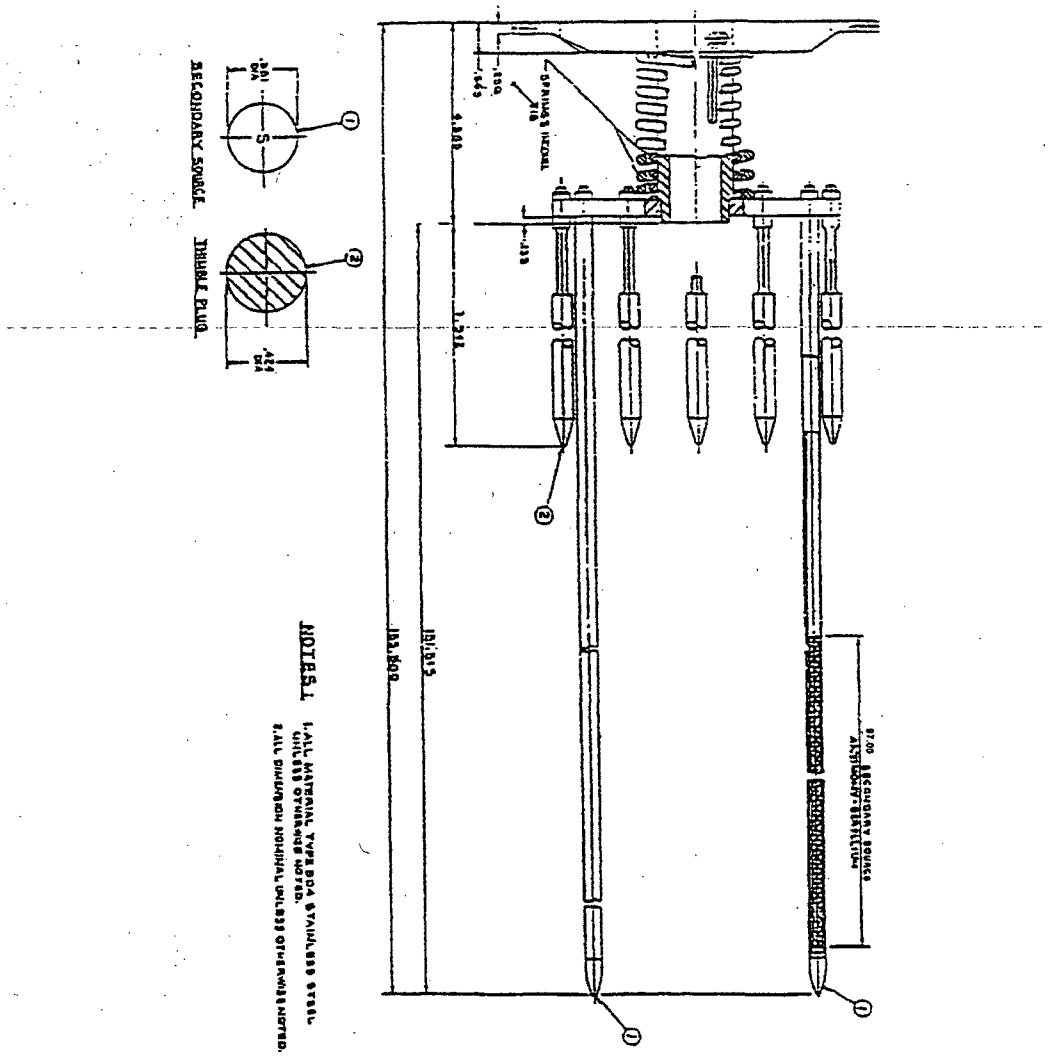


G:\Word\Images_P\Images_P\UFSAR\4212.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Burnable Poison Rod Cross Section	
		Figure 4.2-12

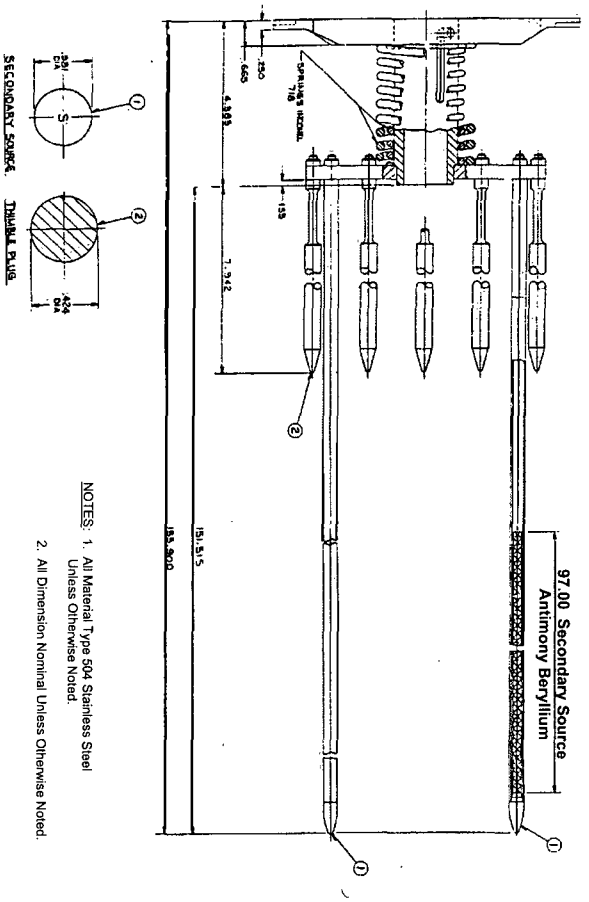


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Primary Source Assembly	
		Figure 4.2-13



G:\Word\Images_P\Images_P\UFSAR\4214.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Secondary Source Assembly	
		Figure 4.2-14

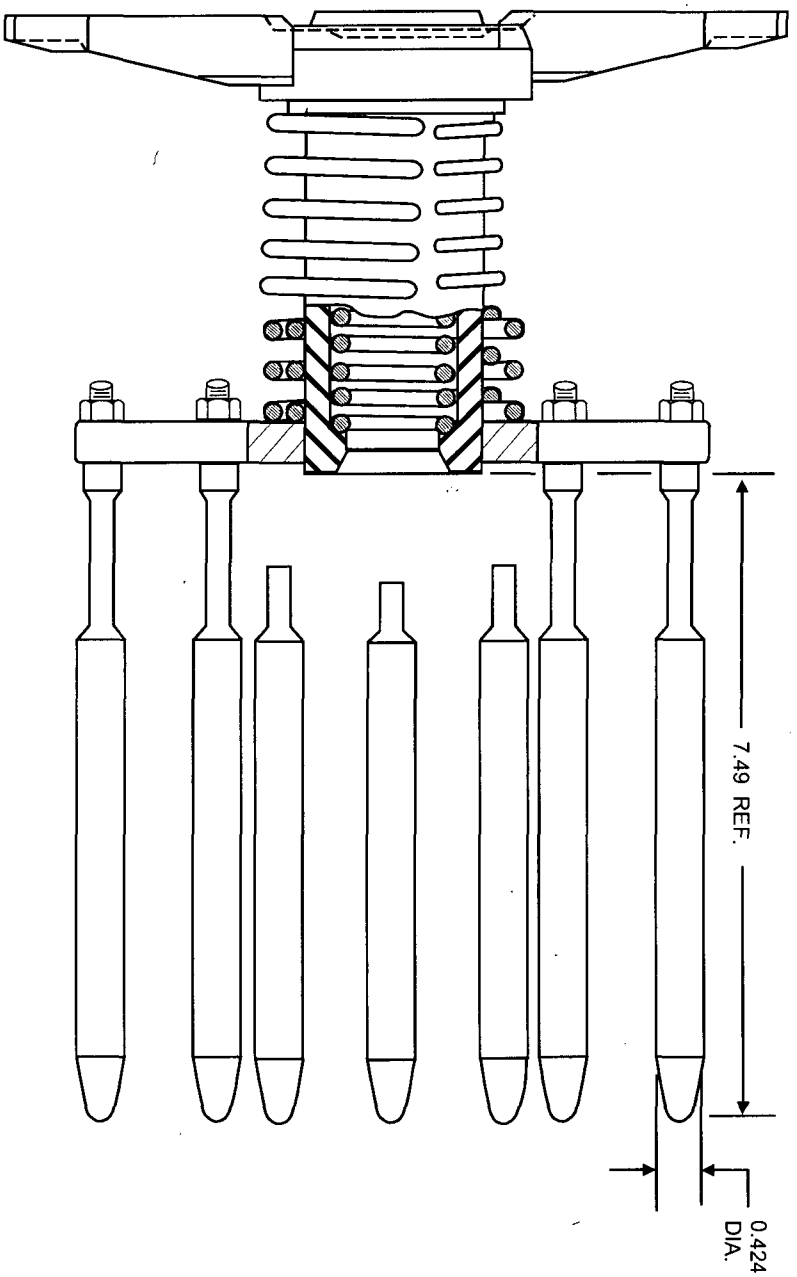


G:\Word\Images_P\UFSAR\4214a.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Secondary Source Assembly - Original Core	
		Figure 4.2-14A

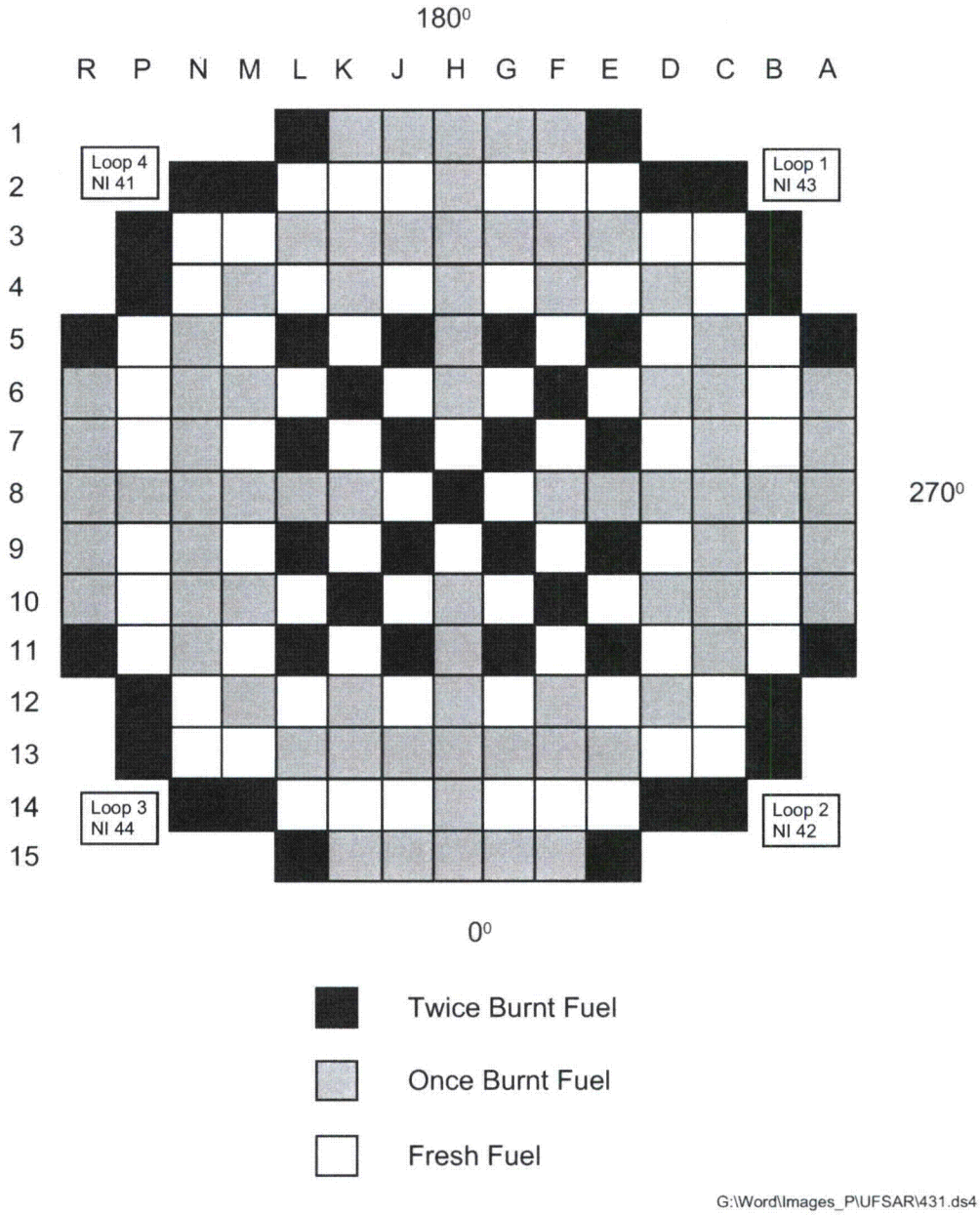
Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Secondary Source Assembly - Additional Secondary Sources	
		Figure 4.2-14B



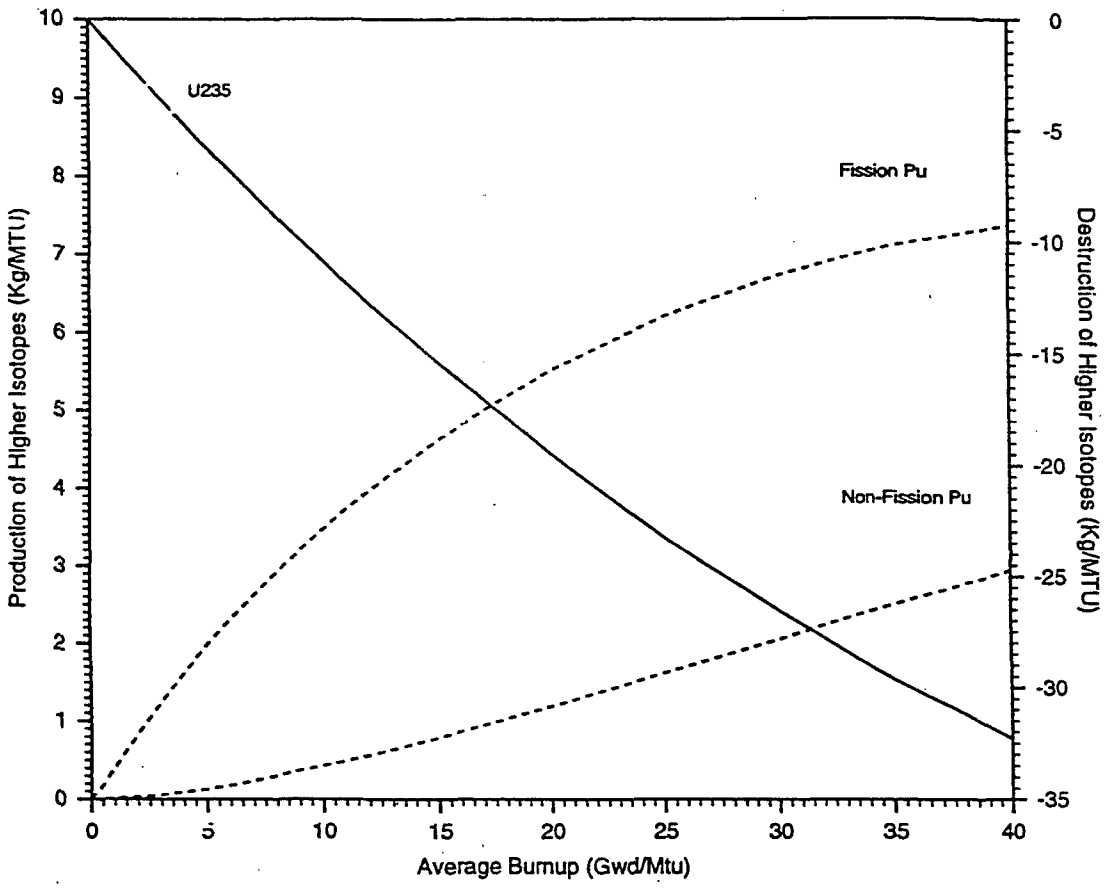
G:\Word\Images_P\UFSAR\4215.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Thimble Plug Assembly	
		Figure 4.2-15

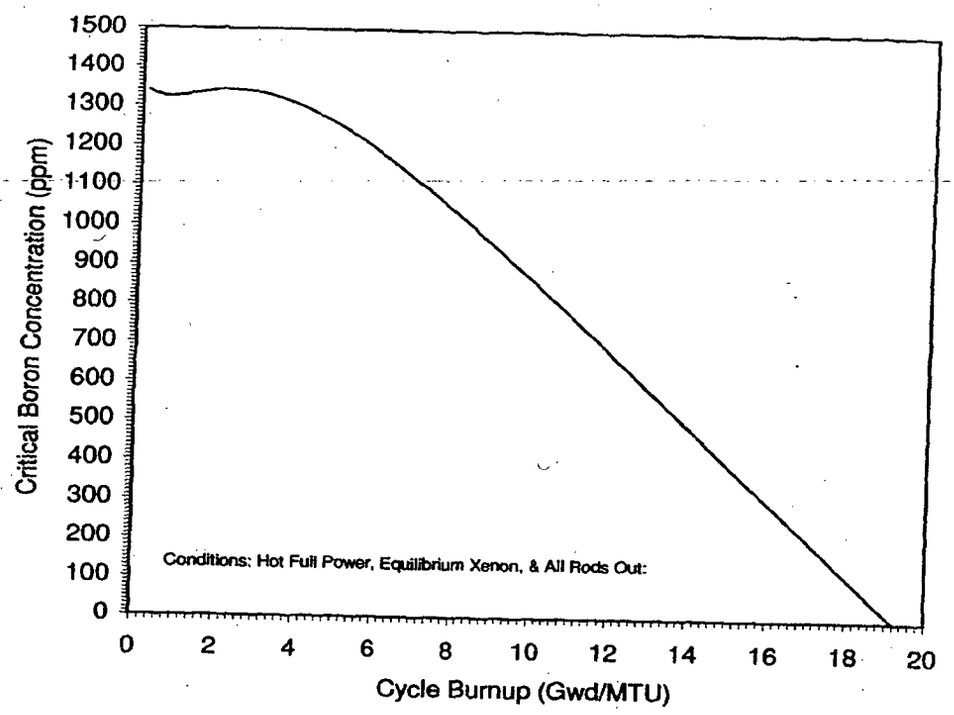


G:\Word\Images_P\UFSAR\431.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Fuel Loading Arrangement	
		Figure 4.3-1

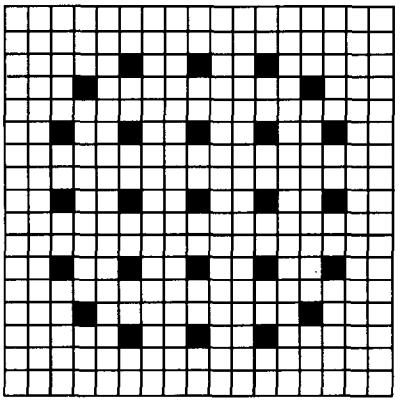


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Production and Destruction of Higher Isotopes for Typical Low Leakage Cycle Design Fuel	
		Figure 4.3-2

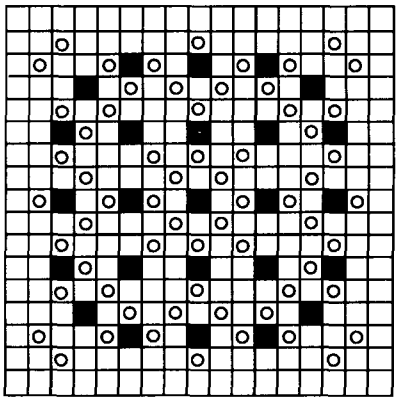


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Boron Concentration vs. Cycle Burnup for Typical Low Leakage Cycle Design	
		Figure 4.3-3

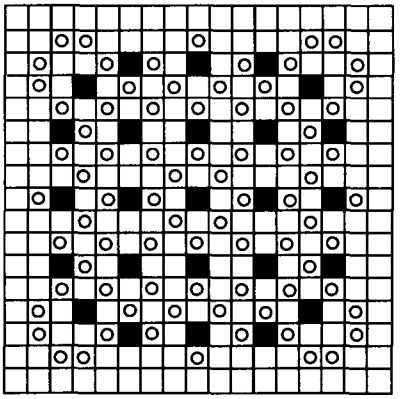
Standard Assembly Pin Layout



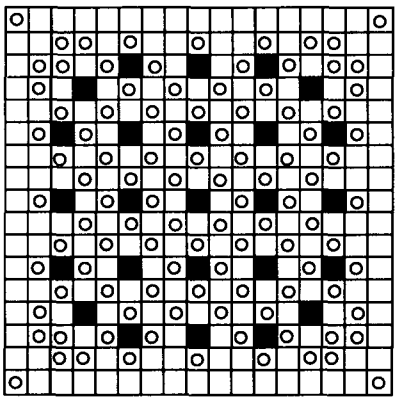
64 IFBA Pin Assembly Layout



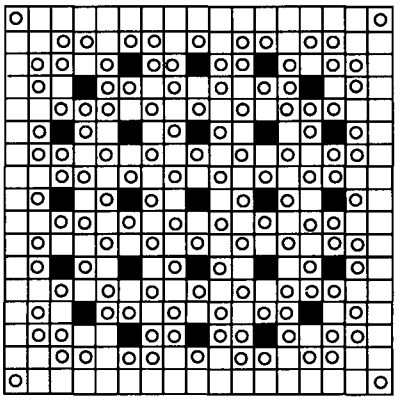
80 IFBA Pin Assembly Layout



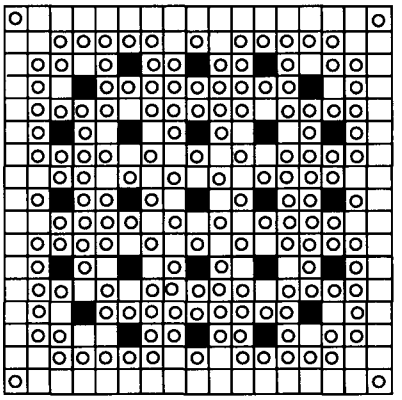
104 IFBA Pin Assembly Layout



128 IFBA Pin Assembly Layout



156 IFBA Pin Assembly Layout



O= IFBA Rod

G:\Word\Images_P\UFSAR\434.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Integral Fuel Burnable Absorber Arrangement within an Assembly	
		Figure 4.3-4

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2	Loop 4 NI 41				64	128	128		128	128	64			Loop 1 NI 43	
3			64	104								104	64		
4			104		128		128		128		128		104		
5		64		128		128				128		128		64	
6		128			128		128		128		128			128	
7		128		128		128		104		128		128		128	
8							104		104						
9		128		128		128		104		128		128		128	
10		128			128		128		128		128			128	
11		64		128		128				128		128		64	
12			104		128		128		128		128		104		
13			64	104								104	64		
14	Loop 3 NI 44				64	128	128		128	128	64			Loop 2 NI 42	
15															

270°

0°

G:\Word\Images_P\UFSAR\435.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical IFBA Placement in Low Leakage Fuel Loading Arrangement	
		Figure 4.3-5

	H	G	F	E	D	C	B	A
8	0.714	1.007	1.040	1.019	1.149	1.310	1.184	0.631
9	1.007	0.861	1.012	0.929	1.217	1.246	1.226	0.634
10	1.040	1.017	0.892	1.110	1.190	1.346	1.226	0.616
11	1.019	0.930	1.112	1.019	1.280	1.345	1.108	0.304
12	1.149	1.219	1.194	1.284	1.236	1.250	0.550	
13	1.310	1.252	1.352	1.349	1.250	0.942	0.273	
14	1.184	1.238	1.233	1.112	0.549	0.265		
15	0.631	0.640	0.621	0.306				

Value Represents Assembly Relative Power

G:\Word\Images_P\UFSAR\436.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical BOL Power Density Distribution Low Leakage Fuel Loading Arrangement - Conditions: BOL, ARO, HZP, Eq Xenon	
		Figure 4.3-6

	H	G	F	E	D	C	B	A
8	0.858	1.162	1.160	1.100	1.165	1.255	1.123	0.629
9	1.162	1.001	1.136	1.007	1.228	1.199	1.164	0.631
10	1.160	1.141	0.992	1.171	1.182	1.268	1.156	0.615
11	1.100	1.008	1.172	1.051	1.248	1.256	1.046	0.314
12	1.165	1.230	1.185	1.250	1.181	1.177	0.551	
13	1.255	1.203	1.272	1.258	1.177	0.909	0.286	
14	1.123	1.173	1.160	1.048	0.550	0.277		
15	0.629	0.636	0.618	0.315				

Value Represents Assembly Relative Power

G:\Word\Images_P\UFSAR\437.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical BOL Power Density Distribution Low Leakage Fuel Loading Arrangement - Conditions: BOL, ARO, HFP, Eq Xenon	
		Figure 4.3-7

	H	G	F	E	D	C	B	A
8	0.812	1.280	1.176	1.066	1.085	1.128	1.049	0.625
9	1.280	1.052	1.304	1.021	1.282	1.106	1.221	0.644
10	1.176	1.308	1.062	1.316	1.129	1.163	1.221	0.636
11	1.066	1.022	1.316	1.051	1.273	1.134	1.037	0.329
12	1.085	1.282	1.130	1.274	0.979	1.143	0.540	
13	1.128	1.107	1.164	1.134	1.142	0.900	0.293	
14	1.049	1.227	1.223	1.037	0.539	0.285		
15	0.625	0.648	0.637	0.329				

Value Represents Assembly Relative Power

G:\Word\Images_P\UFSAR\438.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical BOL Power Density Distribution Low Leakage Fuel Loading Arrangement - Conditions: BOL, Group 35% Inserted, HFP, Eq Xenon	
		Figure 4.3-8

	H	G	F	E	D	C	B	A
8	0.921	1.298	1.161	1.041	1.055	1.095	1.017	0.603
9	1.298	1.052	1.283	1.000	1.252	1.078	1.187	0.625
10	1.161	1.287	1.045	1.301	1.123	1.153	1.200	0.621
11	1.041	1.000	1.302	1.065	1.318	1.161	1.039	0.326
12	1.055	1.253	1.125	1.320	1.140	1.210	0.556	
13	1.095	1.080	1.155	1.161	1.209	0.951	0.307	
14	1.017	1.193	1.203	1.040	0.555	0.298		
15	0.603	0.628	0.623	0.326				

Value Represents Assembly Relative Power

G:\Word\Images_P\UFSAR\439.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical MOL power Density Distribution Low Leakage Fuel Loading Arrangement - Conditions: MOL, ARO, HFP, Eq Xenon	
		Figure 4.3-9

	H	G	F	E	D	C	B	A
8	0.908	1.235	1.108	1.019	1.034	1.083	1.043	0.682
9	1.235	1.019	1.234	0.984	1.216	1.063	1.214	0.701
10	1.108	1.236	1.018	1.248	1.080	1.123	1.219	0.696
11	1.019	0.984	1.248	1.030	1.256	1.129	1.075	0.389
12	1.034	1.215	1.080	1.257	1.100	1.208	0.620	
13	1.083	1.063	1.123	1.129	1.207	1.006	0.371	
14	1.043	1.217	1.220	1.074	0.619	0.361		
15	0.682	0.703	0.697	0.389				

Value Represents Assembly Relative Power

G:\Word\Images_P\UFSAR\4310.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical EOL Power Density Distribution Low Leakage Fuel Loading Arrangement - Conditions: EOL, ARO, HFP, Eq Xenon	
		Figure 4.3-10

	H	G	F	E	D	C	B	A
8	0.787	1.210	1.122	1.046	1.065	1.119	1.080	0.709
9	1.210	1.016	1.253	1.006	1.246	1.092	1.251	0.726
10	1.122	1.255	1.035	1.261	1.085	1.133	1.241	0.715
11	1.046	1.006	1.261	1.014	1.207	1.100	1.073	0.395
12	1.065	1.246	1.085	1.208	0.930	1.136	0.603	
13	1.119	1.092	1.133	1.100	1.135	0.951	0.356	
14	1.080	1.254	1.242	1.073	0.602	0.346		
15	0.709	0.728	0.716	0.395				

Value Represents Assembly Relative Power

G:\Word\Images_P\UFSAR\4311.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical EOL Power Density Distribution Low Leakage Fuel Loading Arrangement - Conditions: EOL, Group D 35% Inserted, HFP, Eq Xenon	
		Figure 4.3-11

1.054	1.112	1.124	1.140	1.163	1.177	1.191	1.201	1.200	1.207	1.206	1.200	1.194	1.177	1.166	1.157	1.094
1.101	1.095	1.052	1.080	1.161	1.157	1.194	1.209	1.181	1.217	1.211	1.182	1.195	1.119	1.096	1.143	1.143
1.100	1.041	1.070	1.185	1.166		1.174	1.240		1.248	1.192		1.204	1.232	1.119	1.091	1.146
1.106	1.058	1.174		1.234	1.176	1.189	1.153	1.235	1.162	1.209	1.206	1.276		1.232	1.114	1.157
1.119	1.129	1.145	1.223	1.159	1.231	1.147	1.193	1.177	1.203	1.168	1.264	1.201	1.297	1.205	1.192	1.176
1.124	1.116		1.156	1.221		1.229	1.176		1.187	1.252		1.267	1.212		1.182	1.185
1.131	1.144	1.135	1.161	1.130	1.220	1.144	1.190	1.174	1.201	1.165	1.255	1.174	1.217	1.200	1.214	1.195
1.134	1.152	1.192	1.118	1.167	1.160	1.182	1.147	1.230	1.158	1.205	1.194	1.213	1.173	1.261	1.224	1.199
1.130	1.121		1.192	1.144		1.159	1.223		1.233	1.181		1.190	1.251		1.190	1.195
1.134	1.152	1.191	1.117	1.166	1.158	1.181	1.146	1.228	1.156	1.203	1.192	1.211	1.171	1.259	1.222	1.198
1.130	1.143	1.134	1.159	1.128	1.218	1.141	1.187	1.170	1.197	1.161	1.251	1.170	1.213	1.196	1.210	1.191
1.122	1.114		1.154	1.217		1.224	1.171		1.180	1.244		1.260	1.205		1.176	1.179
1.117	1.126	1.141	1.210	1.154	1.224	1.141	1.185	1.168	1.194	1.158	1.253	1.191	1.268	1.196	1.184	1.168
1.104	1.055	1.170		1.227	1.168	1.180	1.143	1.224	1.150	1.196	1.193	1.262		1.219	1.103	1.147
1.098	1.037	1.065	1.179	1.158		1.162	1.226		1.233	1.176		1.187	1.215	1.104	1.077	1.133
1.098	1.091	1.046	1.072	1.152	1.145	1.180	1.193	1.164	1.198	1.191	1.162	1.174	1.099	1.077	1.125	1.126
1.052	1.108	1.116	1.131	1.151	1.162	1.175	1.182	1.179	1.185	1.182	1.174	1.168	1.152	1.141	1.135	1.075

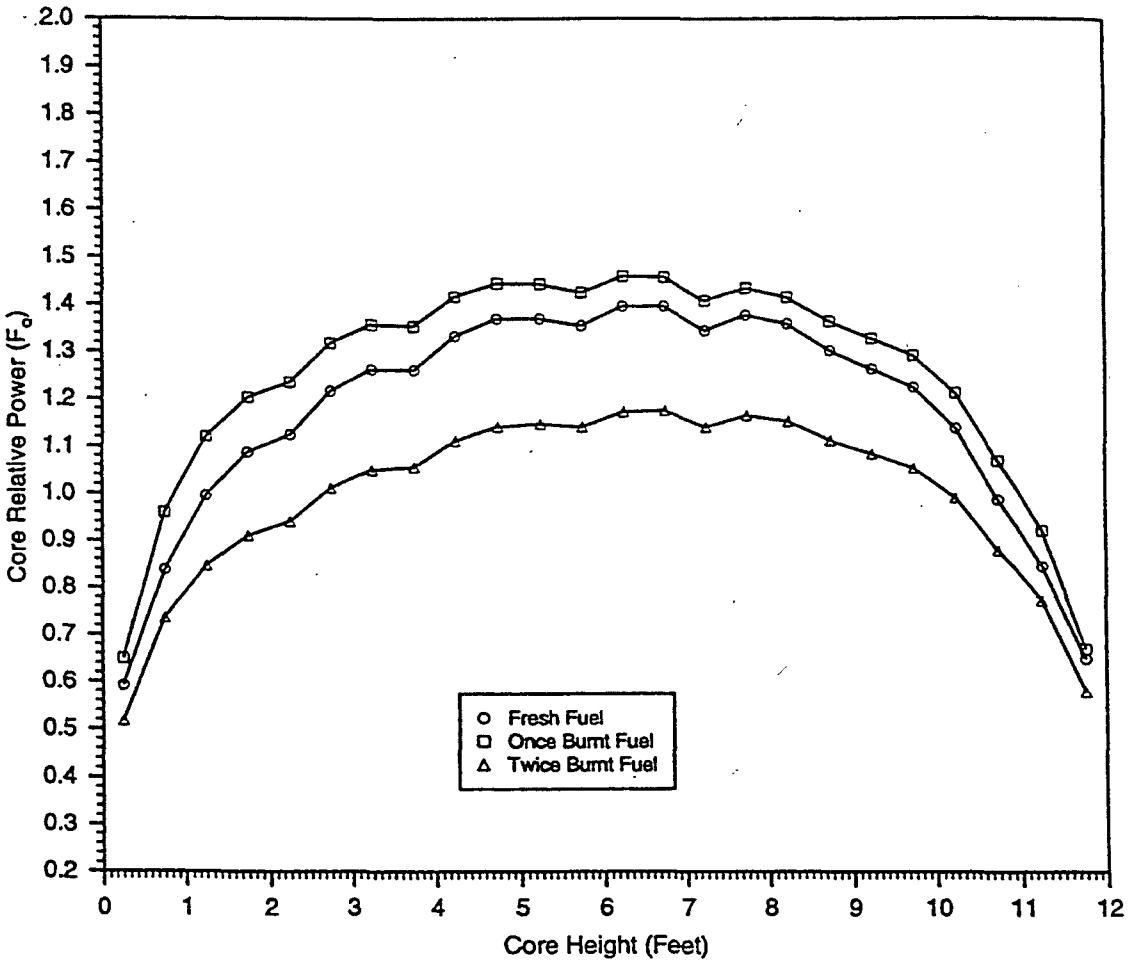
G:\Word\Images P\UFSAR\4312.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Assembly Power Density Distribution Low Leakage Fuel Loading Arrangement - Conditions: BOL, ARO, HFP, Eq Xenon	
		Figure 4.3-12

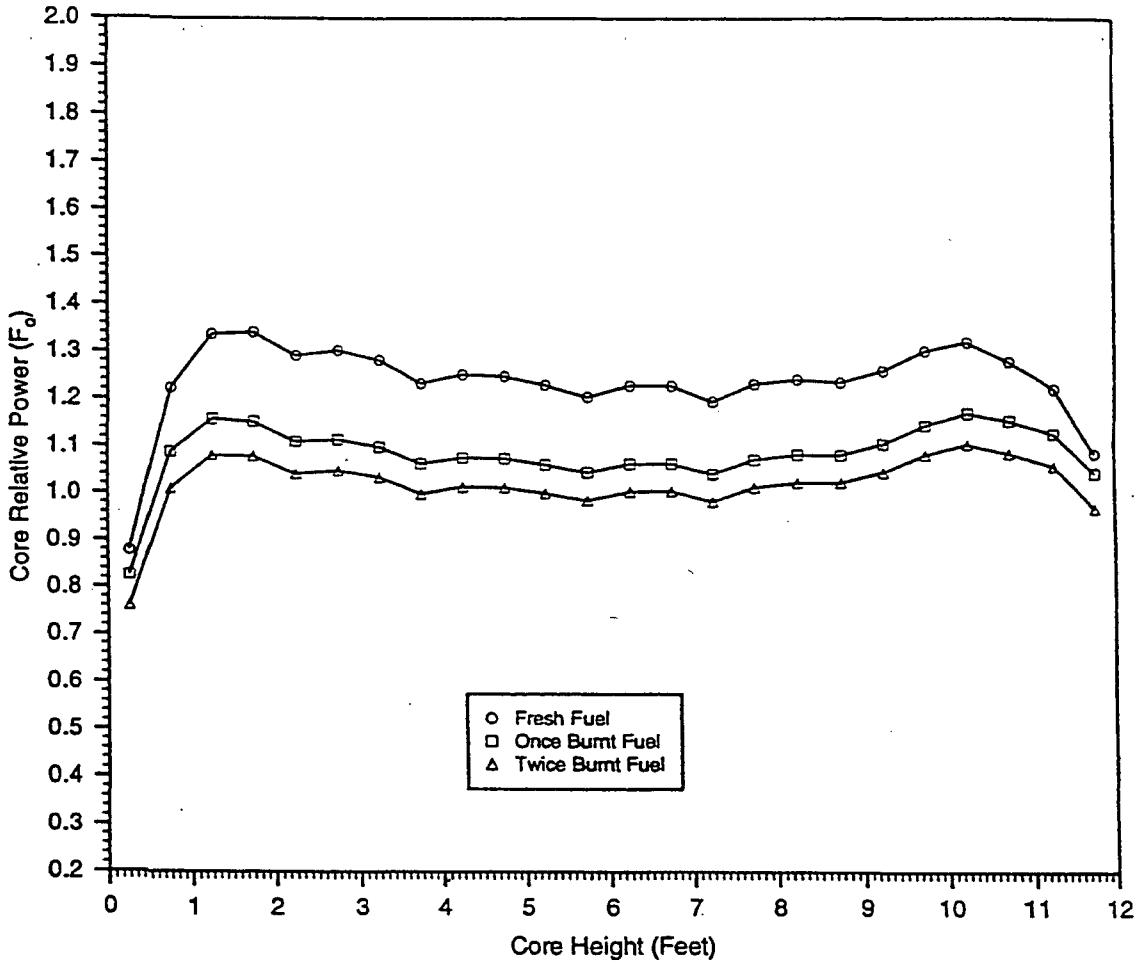
1.171	1.173	1.180	1.190	1.201	1.210	1.213	1.215	1.220	1.219	1.219	1.220	1.215	1.208	1.201	1.197	1.192
1.170	1.164	1.167	1.183	1.218	1.234	1.228	1.231	1.240	1.234	1.234	1.244	1.233	1.202	1.189	1.187	1.188
1.171	1.162	1.191	1.240	1.256		1.253	1.264		1.266	1.259		1.272	1.260	1.215	1.186	1.190
1.176	1.173	1.234		1.280	1.266	1.255	1.247	1.273	1.249	1.261	1.277	1.296		1.260	1.199	1.196
1.181	1.202	1.245	1.274	1.259	1.280	1.251	1.262	1.271	1.265	1.257	1.291	1.276	1.297	1.272	1.230	1.204
1.186	1.213		1.255	1.274		1.283	1.276		1.278	1.289		1.292	1.279		1.243	1.211
1.185	1.203	1.233	1.241	1.242	1.279	1.255	1.268	1.275	1.270	1.262	1.291	1.260	1.265	1.263	1.234	1.211
1.185	1.204	1.241	1.230	1.251	1.269	1.265	1.259	1.285	1.262	1.272	1.281	1.270	1.255	1.272	1.236	1.212
1.187	1.211		1.255	1.259		1.272	1.284		1.286	1.278		1.276	1.279		1.243	1.213
1.184	1.203	1.241	1.229	1.250	1.268	1.264	1.258	1.284	1.260	1.270	1.279	1.268	1.252	1.270	1.234	1.210
1.184	1.202	1.232	1.239	1.241	1.277	1.253	1.266	1.273	1.268	1.259	1.288	1.258	1.262	1.260	1.232	1.208
1.185	1.212		1.254	1.273		1.280	1.273		1.274	1.285		1.288	1.275		1.239	1.207
1.180	1.200	1.243	1.272	1.256	1.276	1.247	1.258	1.266	1.260	1.252	1.286	1.271	1.292	1.267	1.225	1.199
1.175	1.171	1.232		1.276	1.262	1.250	1.241	1.267	1.242	1.254	1.270	1.289		1.253	1.193	1.190
1.170	1.160	1.189	1.237	1.252		1.248	1.257		1.259	1.251		1.263	1.252	1.207	1.178	1.183
1.169	1.163	1.165	1.180	1.214	1.229	1.222	1.224	1.232	1.225	1.225	1.235	1.223	1.193	1.180	1.178	1.180
1.171	1.173	1.178	1.187	1.197	1.204	1.206	1.208	1.212	1.211	1.211	1.211	1.206	1.199	1.193	1.188	1.184

G:\Word\Images P\UFSAR\4313.ds4

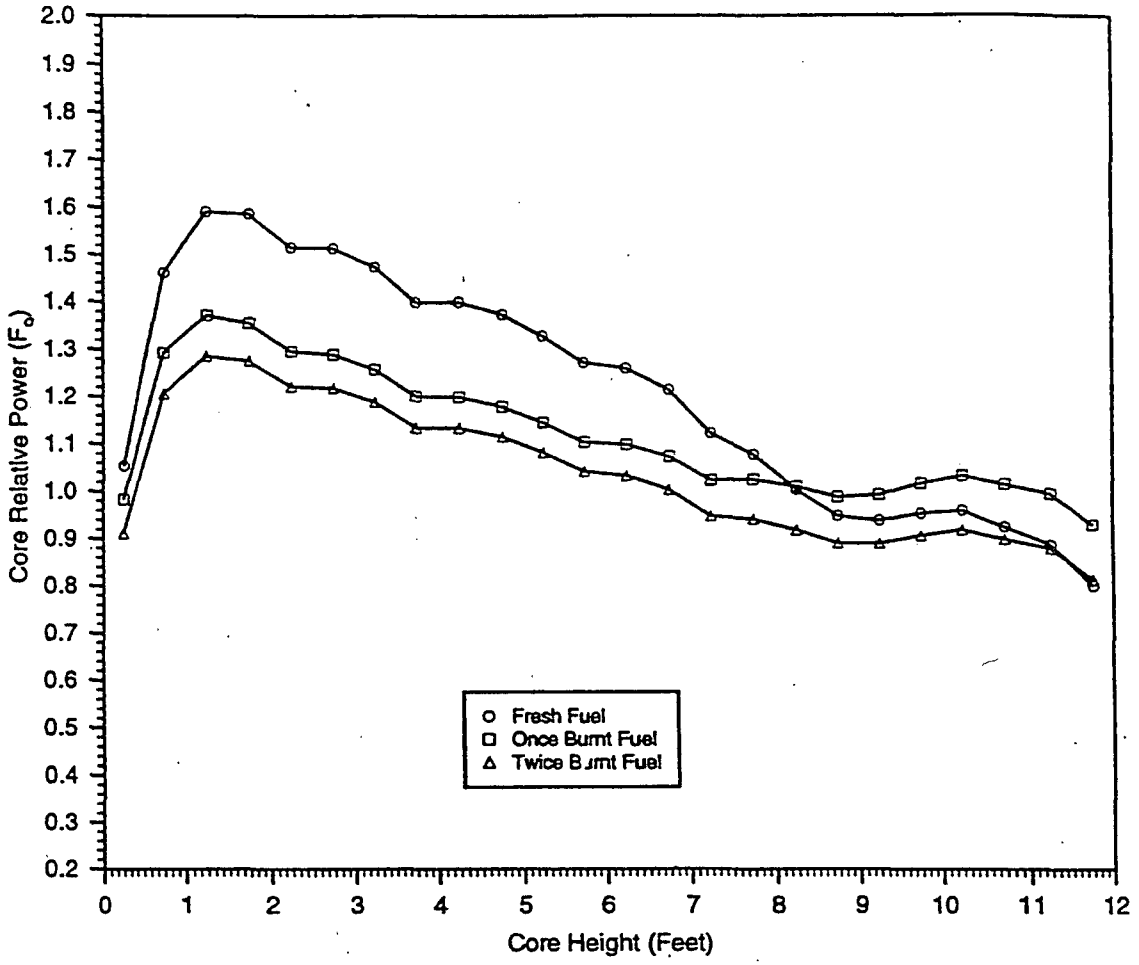
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Assembly Power Density Distribution Low Leakage Fuel Loading Arrangement - Conditions: EOL, ARO, HFP, Eq Xenon	
		Figure 4.3-13



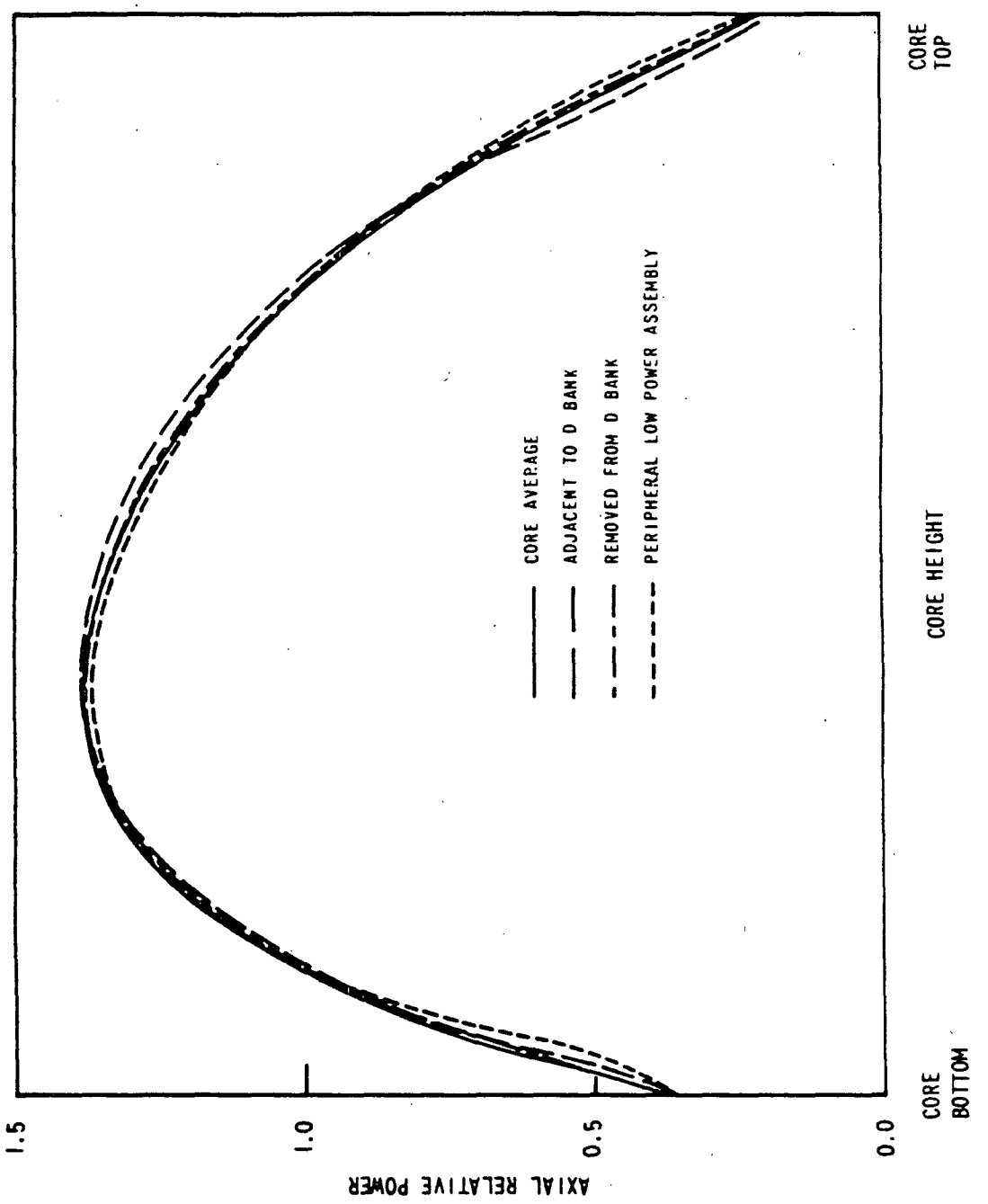
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Axial Power Shapes at BOL During Normal Operation for Typical Low Leakage Cycle Design	
		Figure 4.3-14



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Axial Power Shapes at EOL During Normal Operation for Typical Low Leakage Cycle Design	
		Figure 4.3-15



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Axial Power Shapes During Rodded Operation for Typical Low Leakage Cycle Design	
		Figure 4.3-16



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Comparison of Assembly Axial Power Distribution with Core Average Axial Distribution - D Bank Slightly Inserted	
		Figure 4.3-17

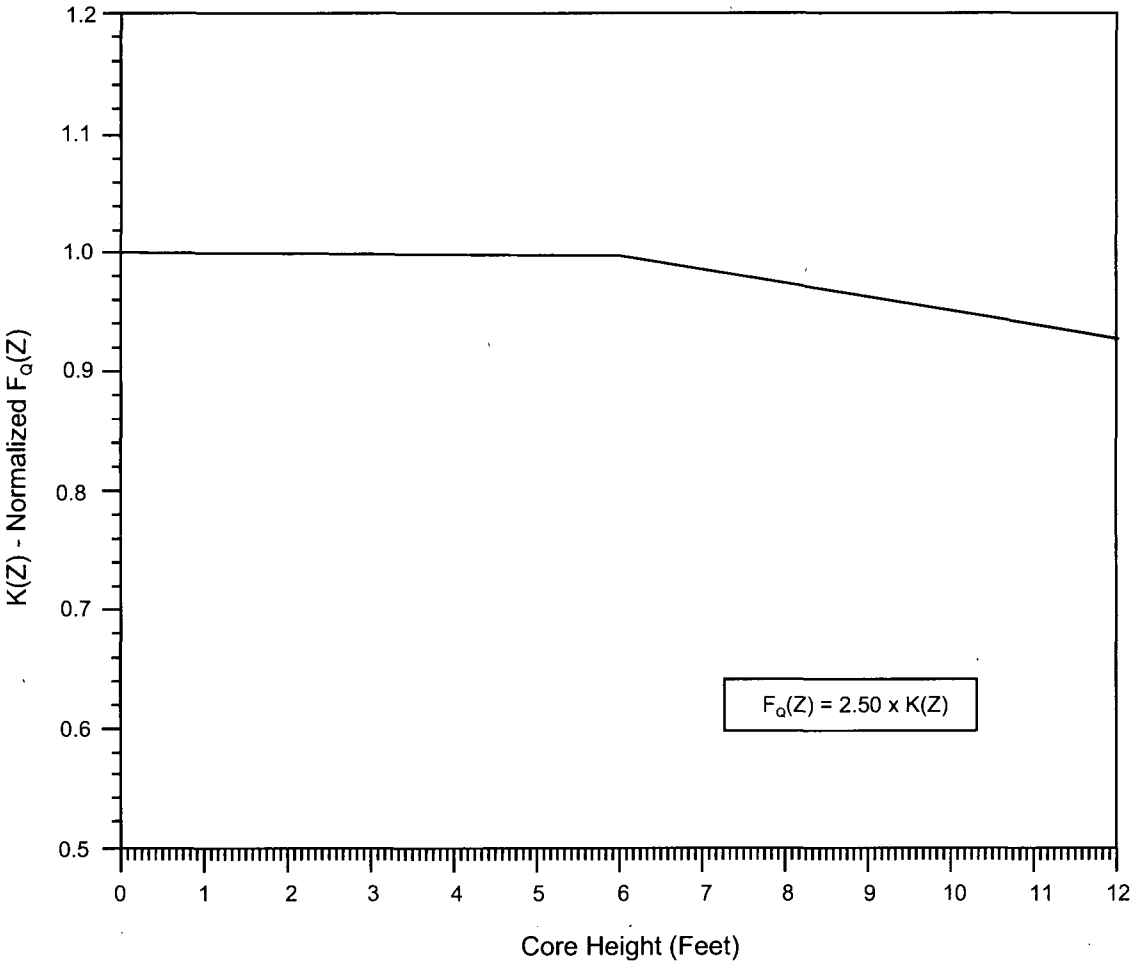
DELETED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 4.3-18

DELETED

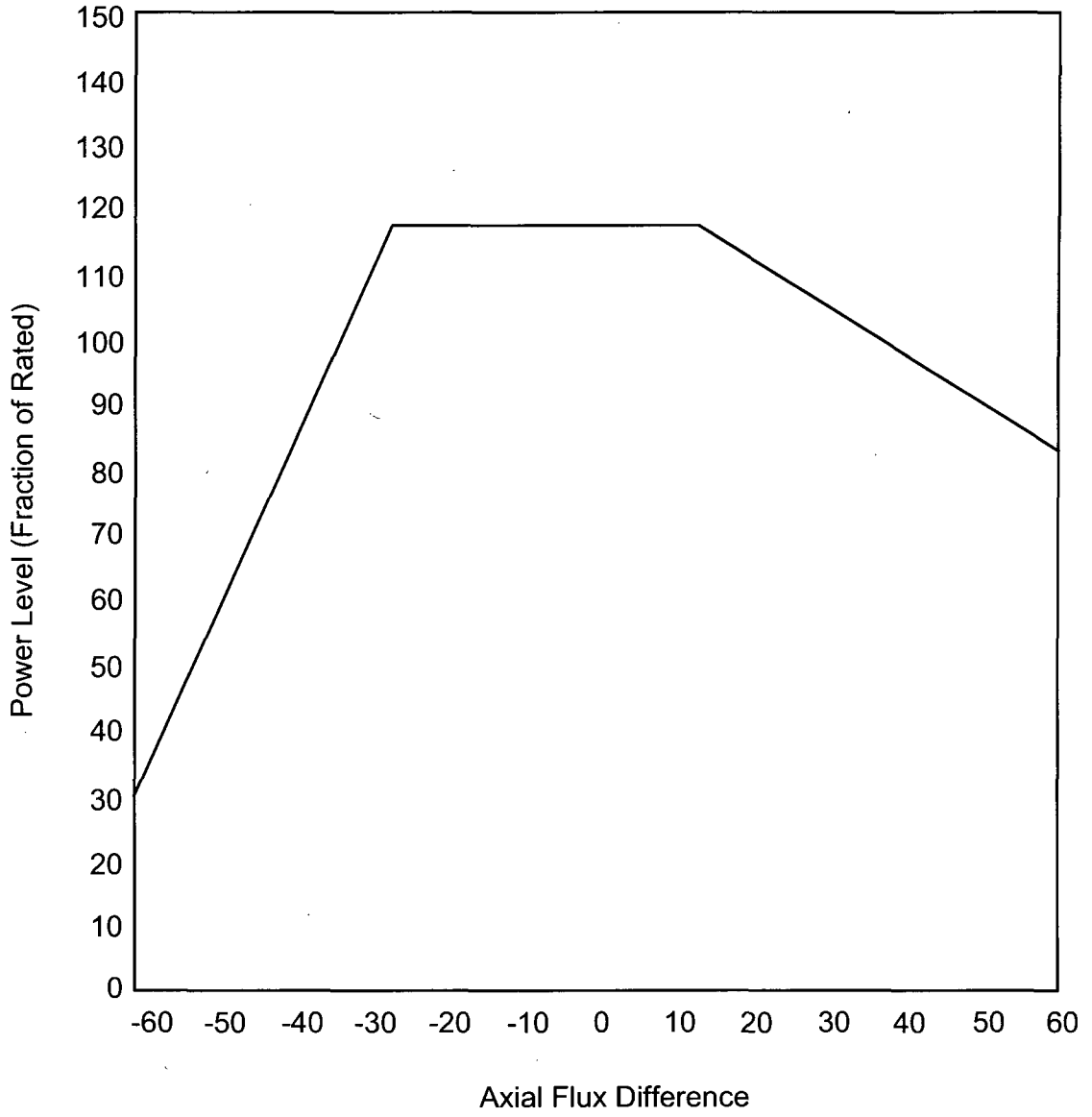
DELETED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 4.3-20



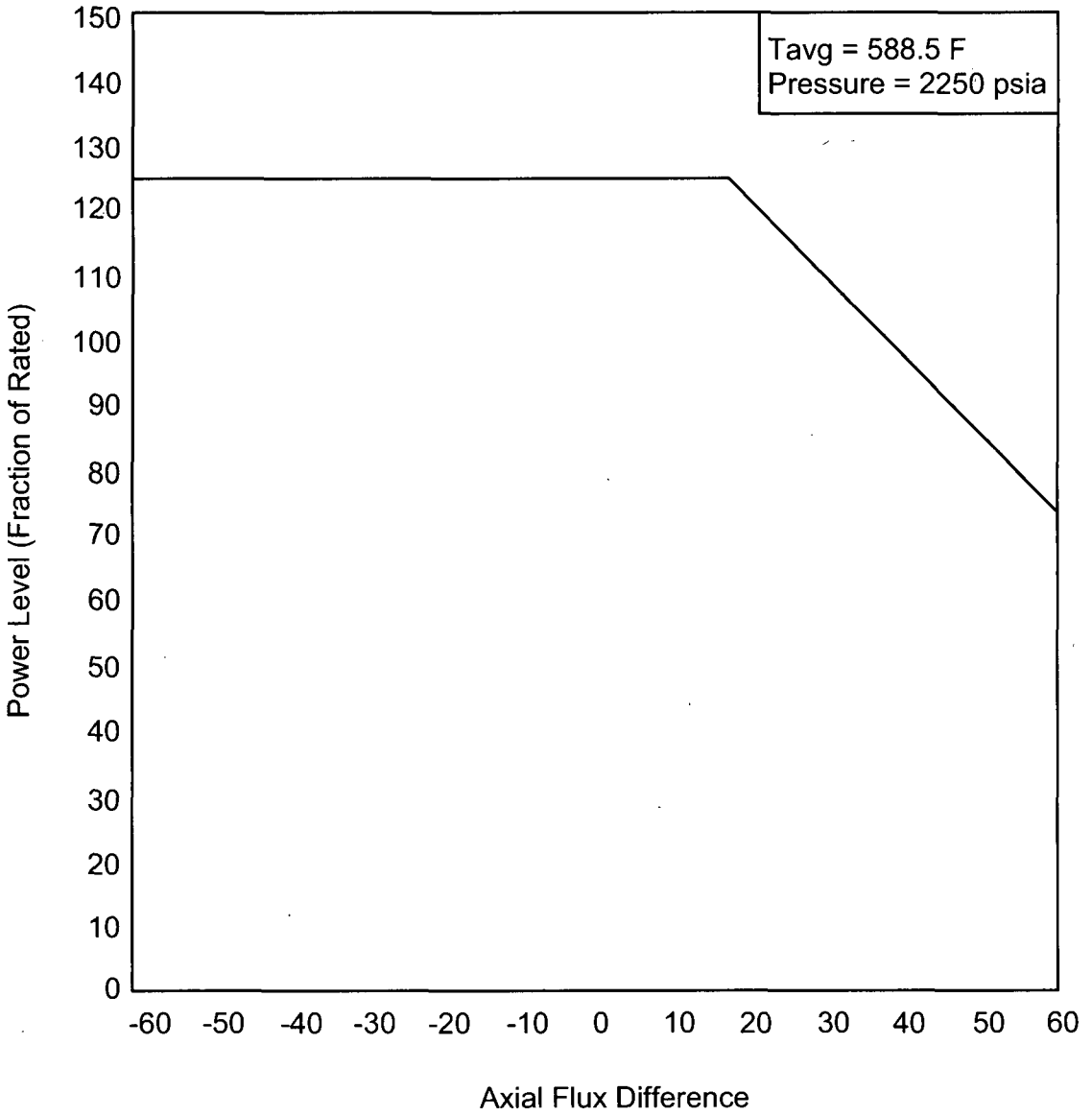
G:\Word\Images_P\UFSAR\4321 ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized Maximum F_Q versus Axial Height during Normal Operation for Typical Low Leakage Cycle Design	
		Figure 4.3-21



G:\Word\Images P\UFSAR\4322.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Illustration of Required Overpower ΔT Setpoint for a Typical Reload core Versus Axial Flux Difference	
		Figure 4.3-22



G:\Word\Images P\UFSAR\4323.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Illustration of Required Overtemperature ΔT Setpoint at Nominal conditions for a Typical Reload Core versus Axial Flux Difference	
		Figure 4.3-23

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					0.313 0.311 0.687	0.615 0.609 0.924	0.631 0.624 1.157	0.620 0.612 1.384	0.614 0.619 -0.829	0.587 0.606 -3.247	0.300 0.310 -3.278				
2	Loop 4 NI 41		0.277 0.277 0.022	0.544 0.544 0.050	1.047 1.039 0.777	1.176 1.165 0.939	1.187 1.171 1.348	1.103 1.085 1.575	1.164 1.162 0.176	1.143 1.161 -1.511	1.010 1.037 -2.649	0.518 0.544 -5.012	0.273 0.285 -4.316		Loop 1 NI 43
3		0.286 0.285 0.298	0.914 0.911 0.335	1.187 1.186 0.130	1.245 1.236 0.781	1.252 1.242 0.772	1.178 1.169 0.787	1.218 1.208 0.825	1.173 1.165 0.625	1.241 1.239 0.137	1.205 1.234 -2.353	1.129 1.186 -4.987	0.874 0.911 -4.262	0.267 0.277 -3.605	
4		0.547 0.544 0.462	1.191 1.186 0.449	1.192 1.178 1.114	1.312 1.283 2.165	1.203 1.177 2.162	1.247 1.244 0.244	1.143 1.138 0.494	1.248 1.242 0.456	1.171 1.174 -0.293	1.252 1.281 -2.319	1.154 1.178 -2.100	1.158 1.186 -2.368	0.536 0.544 -1.357	
5	0.313 0.310 0.755	1.049 1.037 1.189	1.250 1.234 1.347	1.304 1.281 1.786	1.081 1.063 1.658	1.236 1.216 1.590	1.017 1.009 0.789	1.096 1.087 0.832	1.016 1.008 0.849	1.209 1.215 -0.525	1.039 1.063 -2.275	1.275 1.283 -0.616	1.219 1.236 -1.396	1.021 1.039 -1.693	0.298 0.311 -4.538
6	0.611 0.606 0.762	1.183 1.161 1.921	1.249 1.239 0.790	1.187 1.174 1.041	1.220 1.215 0.429	1.026 1.011 1.480	1.207 1.187 1.657	1.191 1.170 1.767	1.188 1.182 0.541	0.999 1.011 -1.192	1.189 1.216 -2.329	1.172 1.177 -0.443	1.228 1.242 -1.134	1.150 1.165 -1.377	0.583 0.609 -4.501
7	0.620 0.619 0.220	1.175 1.162 1.059	1.172 1.165 0.586	1.251 1.242 0.694	1.010 1.008 0.213	1.190 1.182 0.671	1.030 1.022 0.848	1.221 1.211 0.872	1.032 1.022 1.031	1.190 1.187 0.188	1.014 1.009 0.549	1.248 1.244 0.300	1.161 1.169 -0.667	1.157 1.171 -1.198	0.610 0.624 -2.286
8	0.603 0.612 -1.493	1.077 1.085 -0.822	1.193 1.208 -1.266	1.139 1.138 0.138	1.087 1.087 0.031	1.176 1.170 0.503	1.209 1.211 -0.175	0.886 0.880 0.697	1.225 1.211 1.166	1.184 1.170 1.169	1.096 1.087 0.829	1.144 1.138 0.556	1.211 1.208 0.188	1.088 1.085 0.229	0.615 0.612 0.578
9	0.614 0.624 -1.634	1.159 1.171 -0.995	1.157 1.169 -1.093	1.250 1.244 0.452	1.012 1.009 0.313	1.182 1.187 -0.446	1.009 1.022 -1.224	1.212 1.211 0.112	1.050 1.022 2.718	1.209 1.182 2.233	1.022 1.008 1.393	1.250 1.242 0.618	1.169 1.165 0.272	1.166 1.162 0.356	0.623 0.619 0.653
10	0.613 0.609 0.681	1.171 1.165 0.448	1.207 1.242 -2.961	1.179 1.177 0.134	1.217 1.216 0.068	1.009 1.011 -0.225	1.181 1.182 0.095	1.179 1.170 0.804	1.208 1.187 1.692	1.022 1.011 1.128	1.220 1.215 0.383	1.179 1.174 0.392	1.243 1.239 0.291	1.176 1.161 1.291	0.614 0.606 1.263
11	0.324 0.311 3.861	1.080 1.039 3.859	1.243 1.236 0.551	1.286 1.283 0.230	1.070 1.063 0.713	1.219 1.215 0.363	1.013 1.008 0.568	1.080 1.087 -0.671	1.001 1.009 -0.782	1.200 1.216 -1.336	1.071 1.063 0.754	1.292 1.281 0.851	1.251 1.234 1.428	1.049 1.037 1.227	0.314 3.310 1.184
12		0.544 0.544 0.070	1.140 1.186 -3.952	1.176 1.178 -0.201	1.302 1.281 1.652	1.194 1.174 1.655	1.259 1.242 1.360	1.137 1.138 -0.017	1.241 1.244 -0.201	1.161 1.177 -1.391	1.298 1.283 1.134	1.194 1.178 1.289	1.210 1.186 2.022	0.552 0.544 1.323	
13		0.275 0.277 -0.477	0.906 0.911 -0.553	1.190 1.186 0.339	1.256 1.234 1.782	1.236 1.239 -0.245	1.159 1.165 -0.519	1.189 1.208 -1.639	1.154 1.169 -1.286	1.209 1.242 -2.777	1.231 1.236 -0.387	1.196 1.186 0.915	0.921 0.911 1.096	0.289 0.285 1.411	
14	Loop 3 NI 44		0.285 0.285 -0.102	0.546 0.544 0.237	1.044 1.037 0.722	1.147 1.161 -1.166	1.137 1.162 -2.181	1.064 1.085 -1.974	1.154 1.171 -1.448	1.130 1.165 -3.113	1.019 1.039 -1.926	0.539 0.544 -0.790	0.277 0.277 0.273		Loop 2 NI 42
15					0.309 0.310 -0.404	0.591 0.606 -2.520	0.596 0.619 -3.815	0.590 0.612 -3.698	0.606 0.624 -2.924	0.591 0.609 -3.150	0.305 0.311 -1.990	Measured Power Predicted Power % Difference			

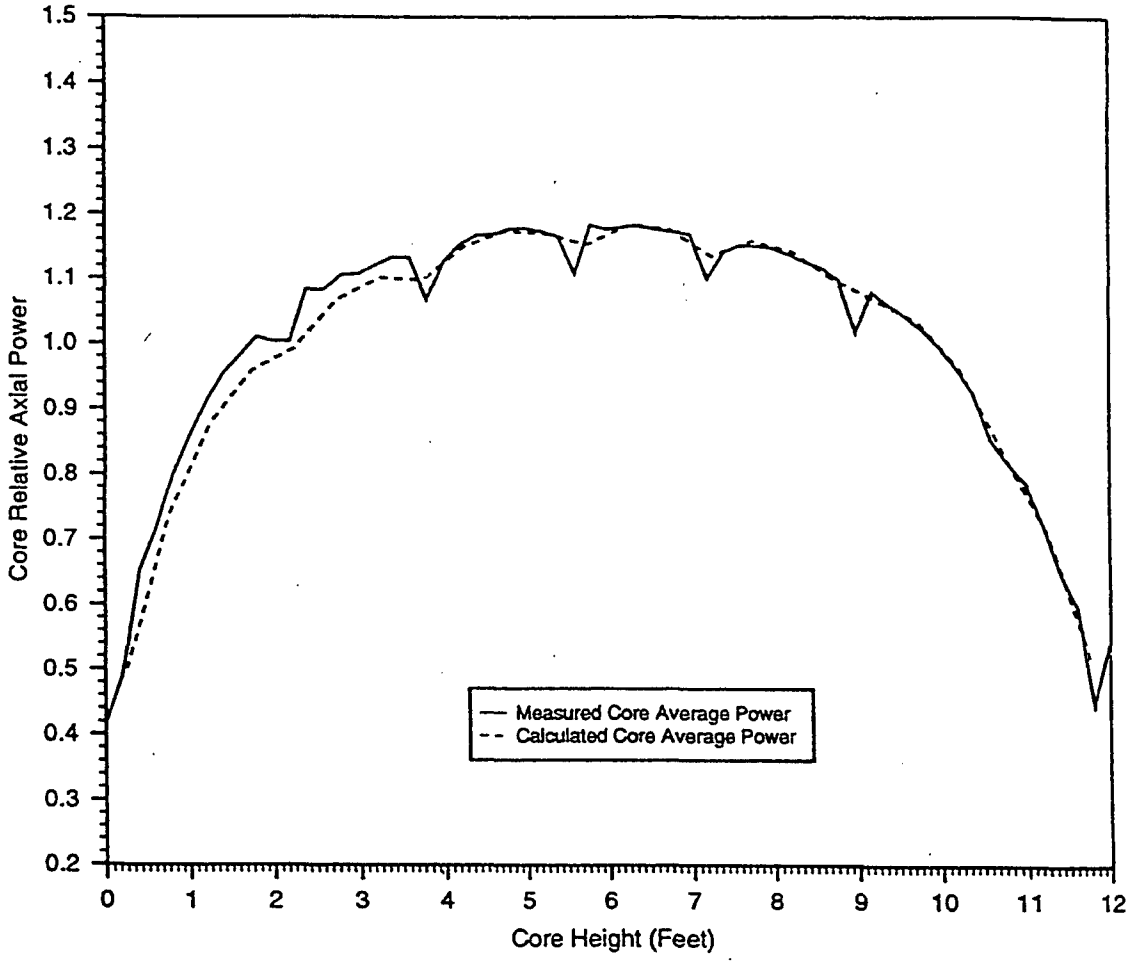
270°

0°

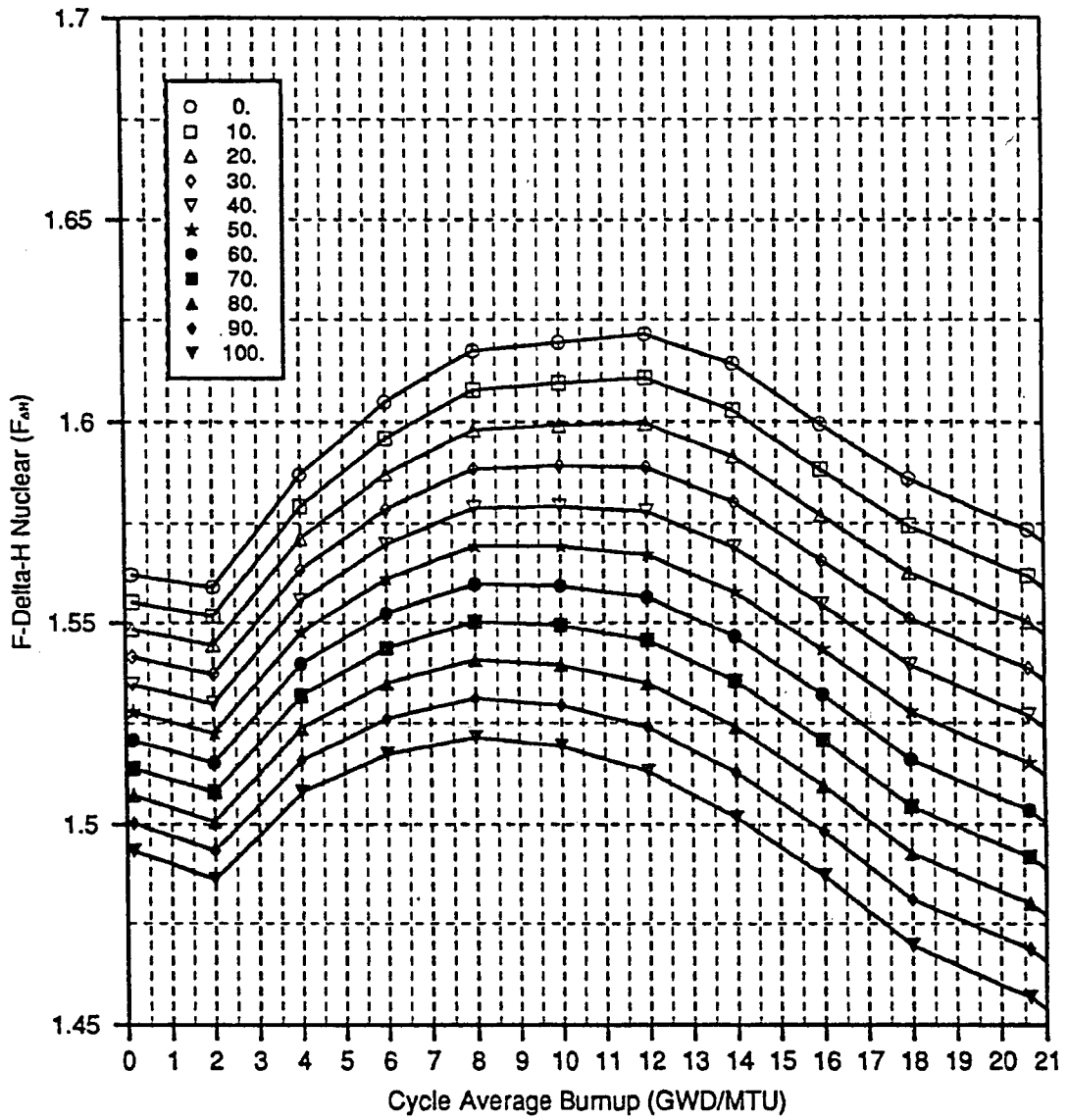
-1.0500 Axial Offset
1.6025 Maximum F_{XY}
1.4239 Maximum F_a
1.8524 Maximum F_Q
-5.0124 Maximum F_a Assembly Difference

G:\Word\Images_P\UFSAR\4324.ds4

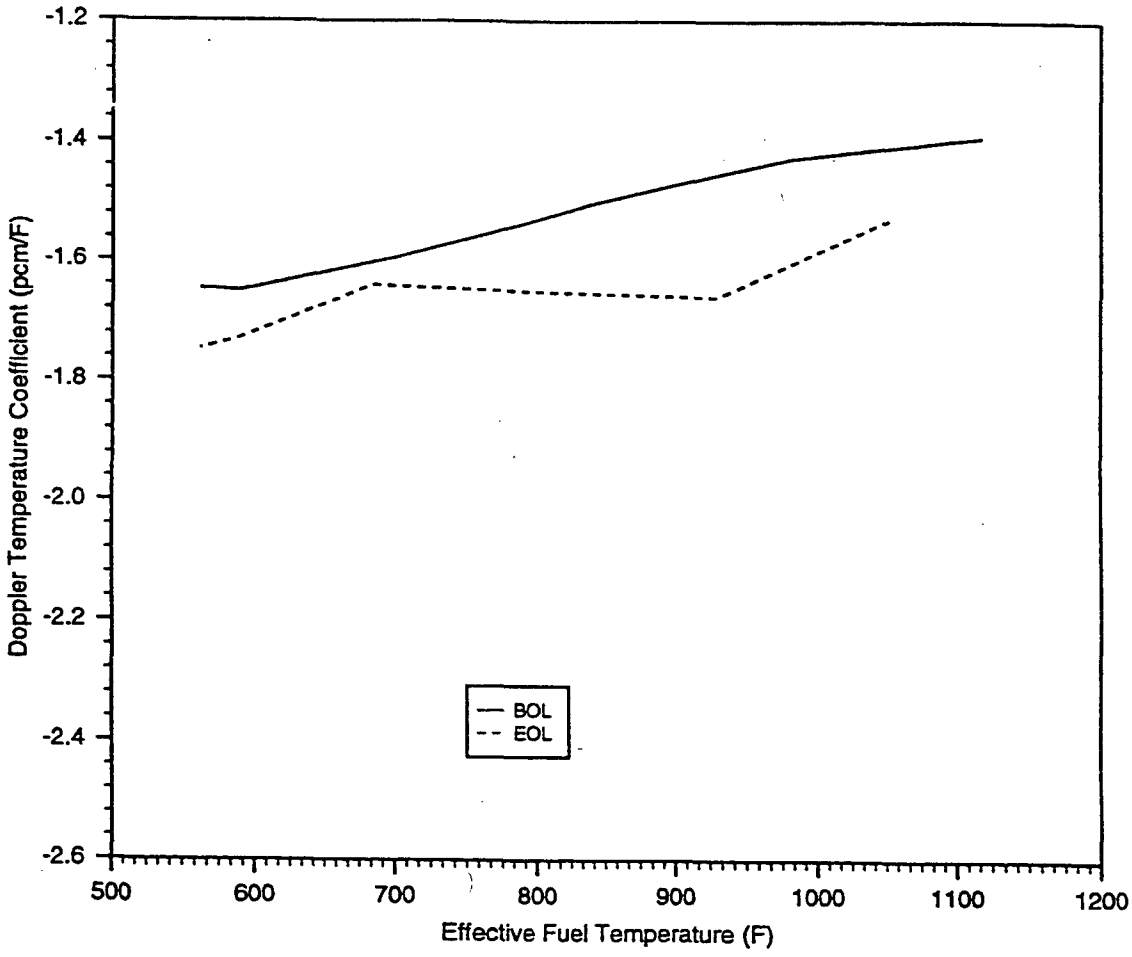
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Example of a Typical Comparison between Calculated and Measured Relative Fuel Assembly Power Distribution - Conditions: BOL, HFP, ARO	
		Figure 4.3-24



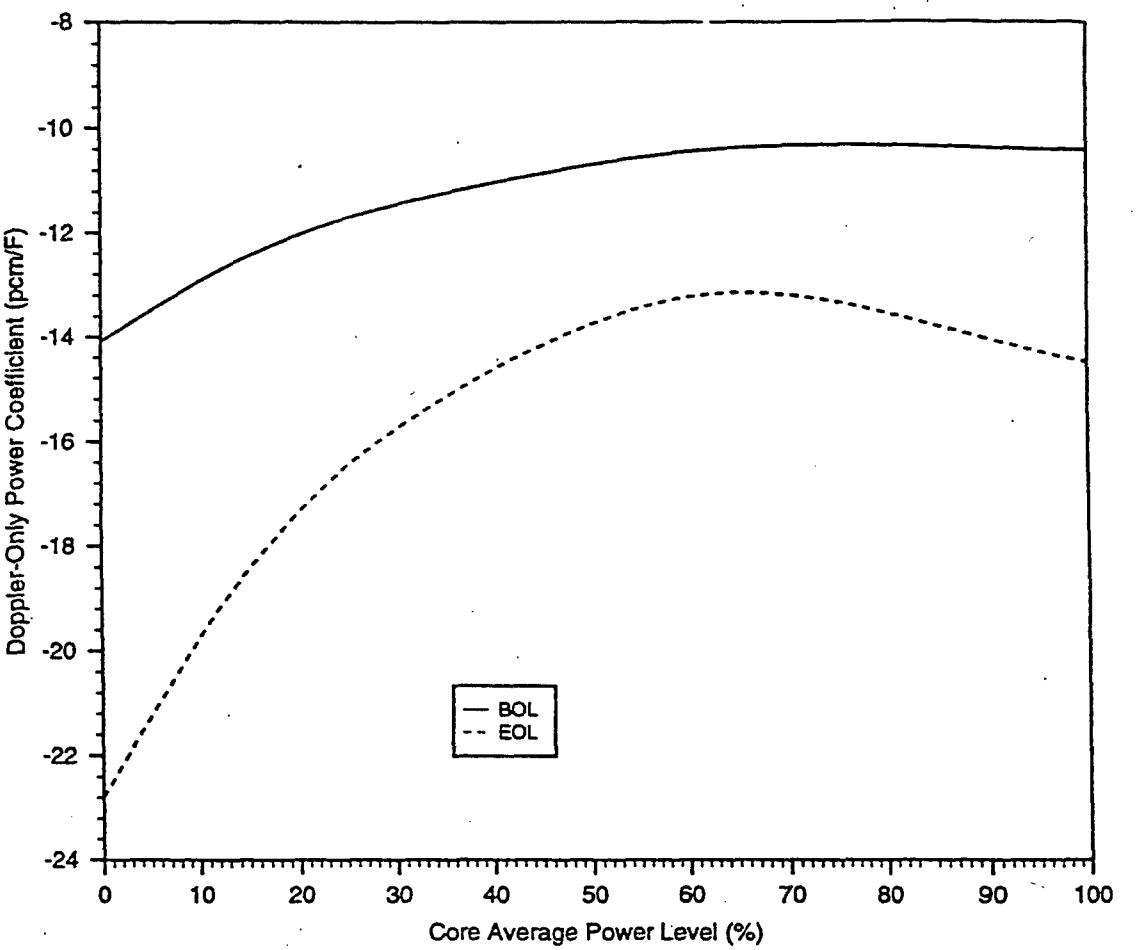
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Example of a Typical Comparison between Calculated and Measured Core Average Axial Power - Conditions; BOL, HFP, ARO	
		Figure 4.3-25



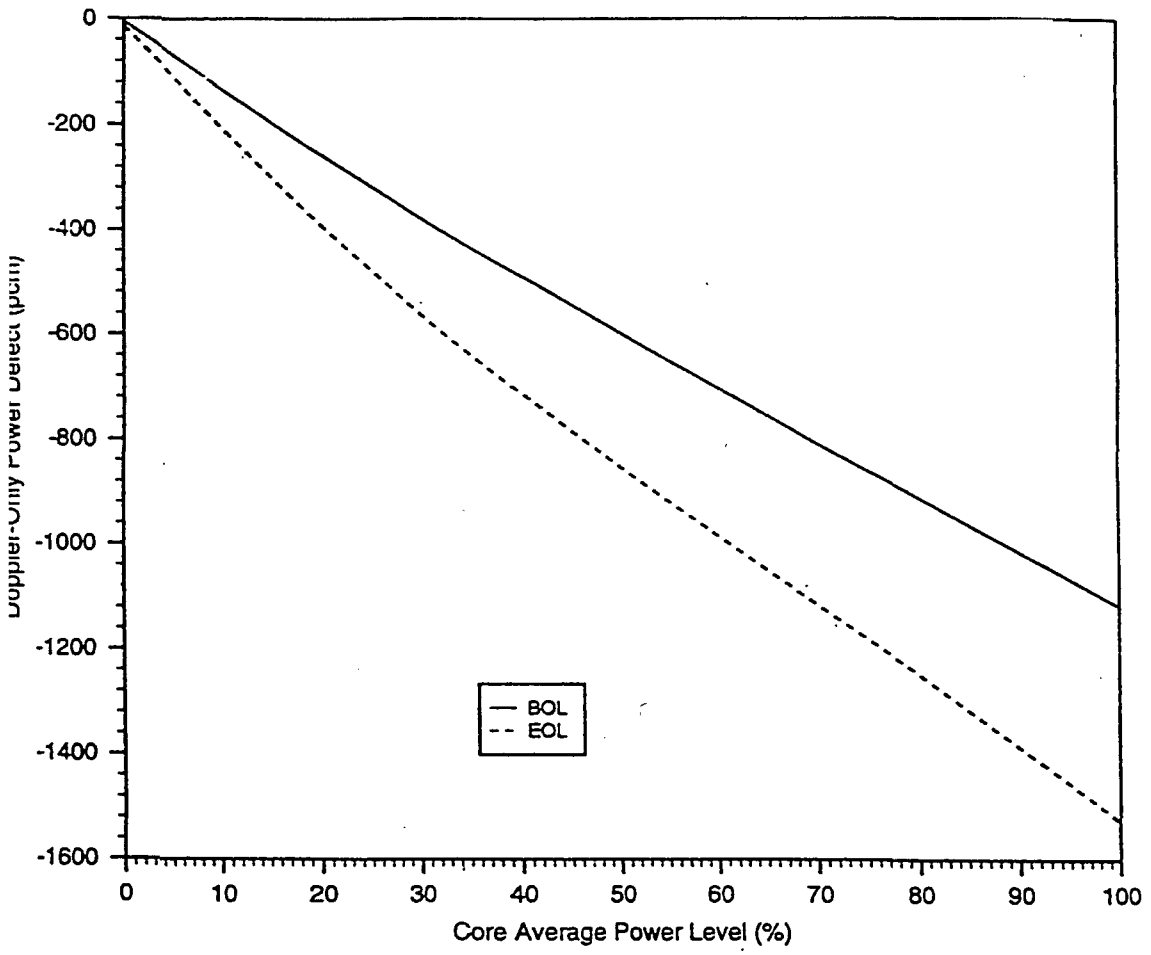
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	All-Rods-Out Nuclear Enthalpy Rise Hot Channel Factor versus Power Level	
		Figure 4.3-26



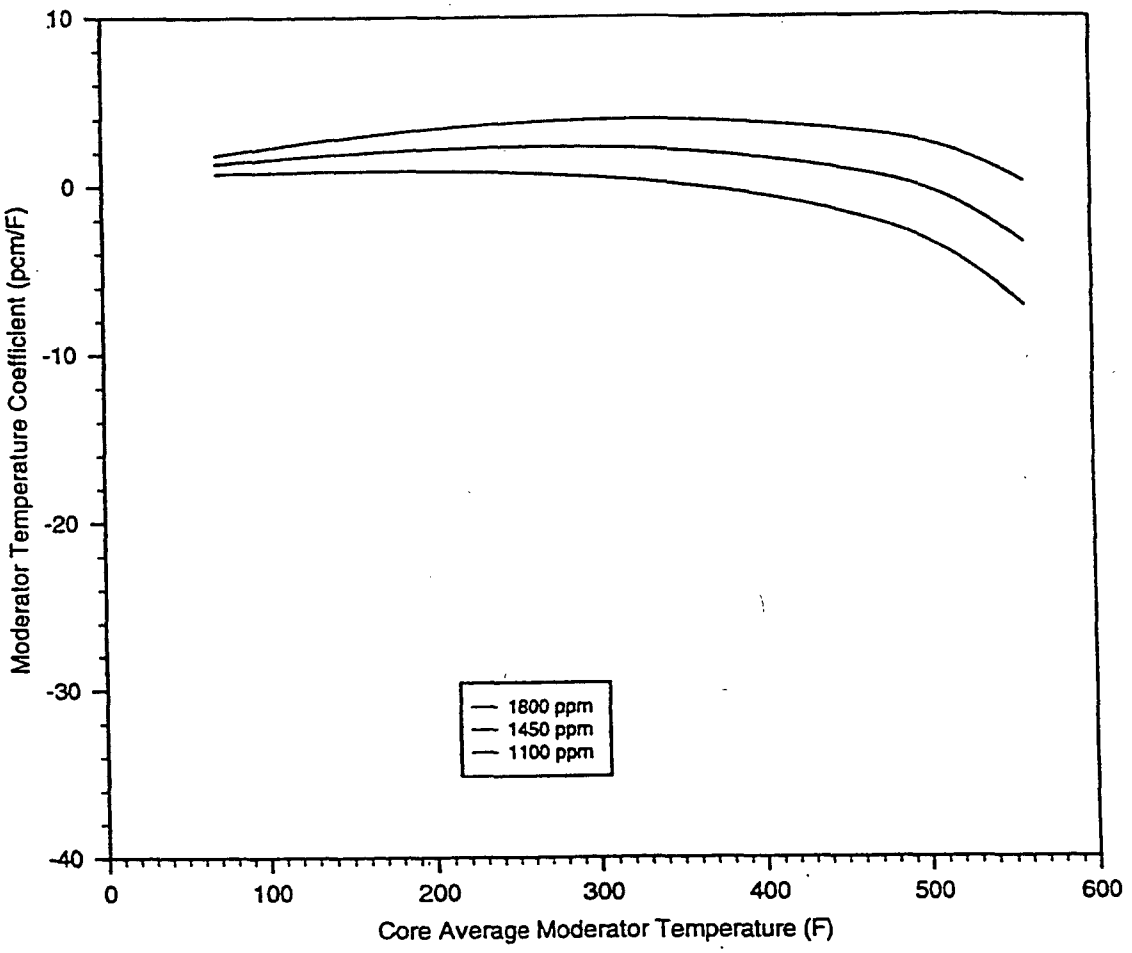
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Doppler Temperature Coefficient at BOL and EOL	
		Figure 4.3-27



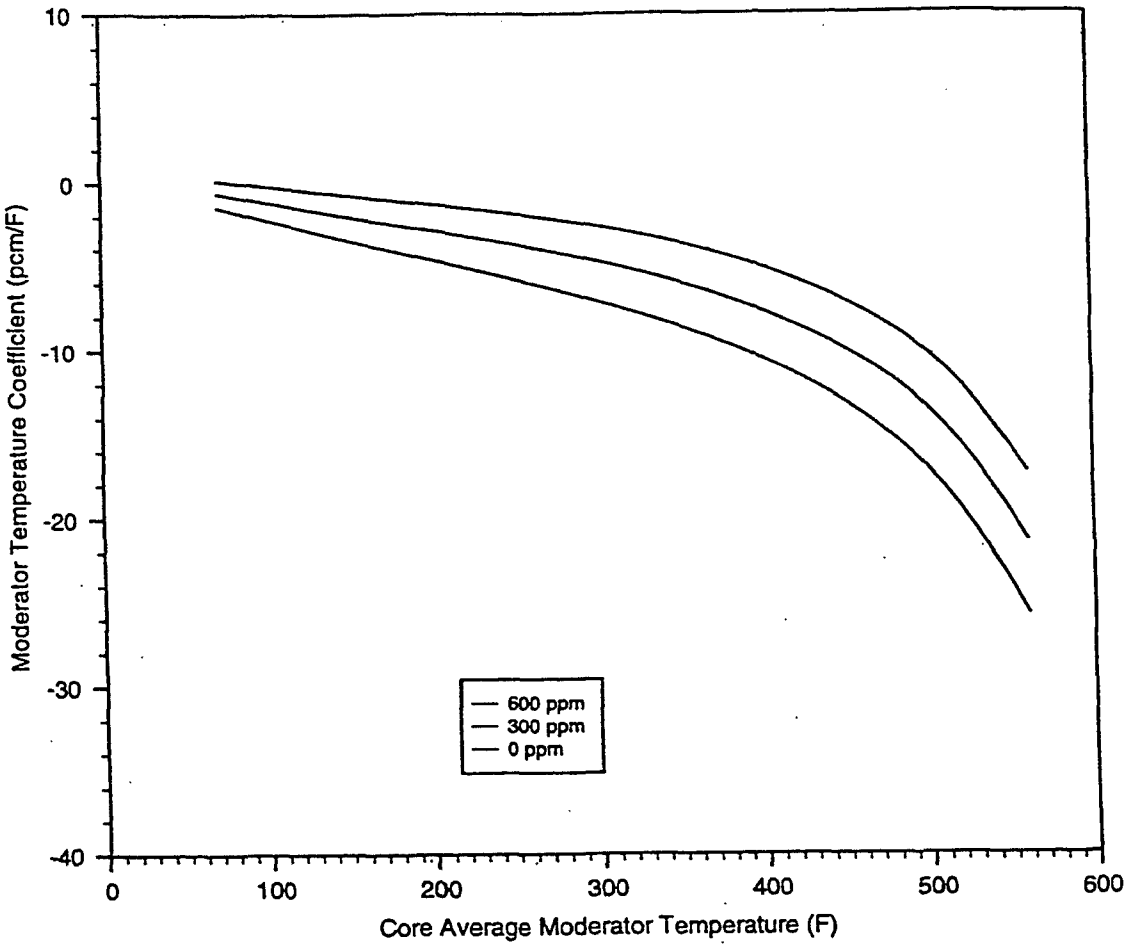
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Doppler-Only Power Coefficient - BOL, & EOL	
		Figure 4.3-28



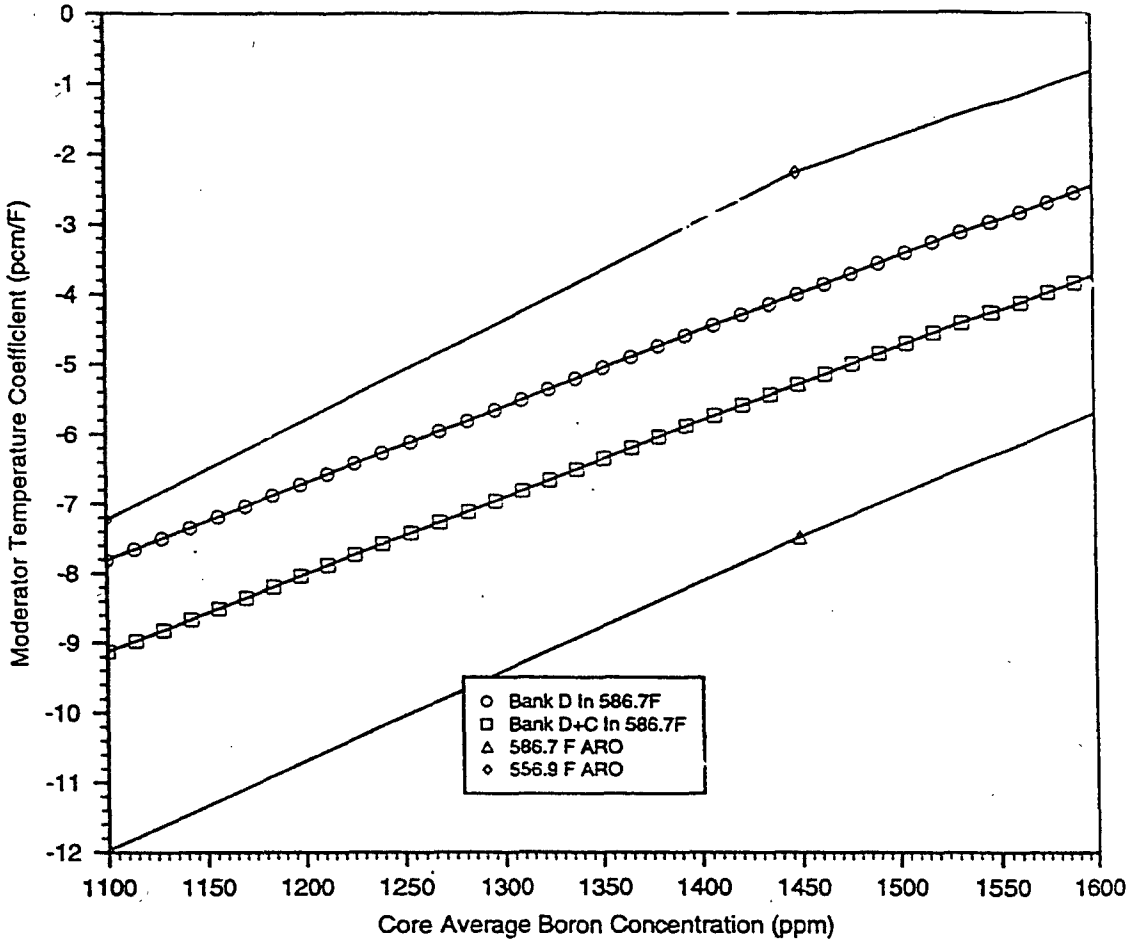
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Doppler-Only Power Defect - BOL, & EOL	
		Figure 4.3-29



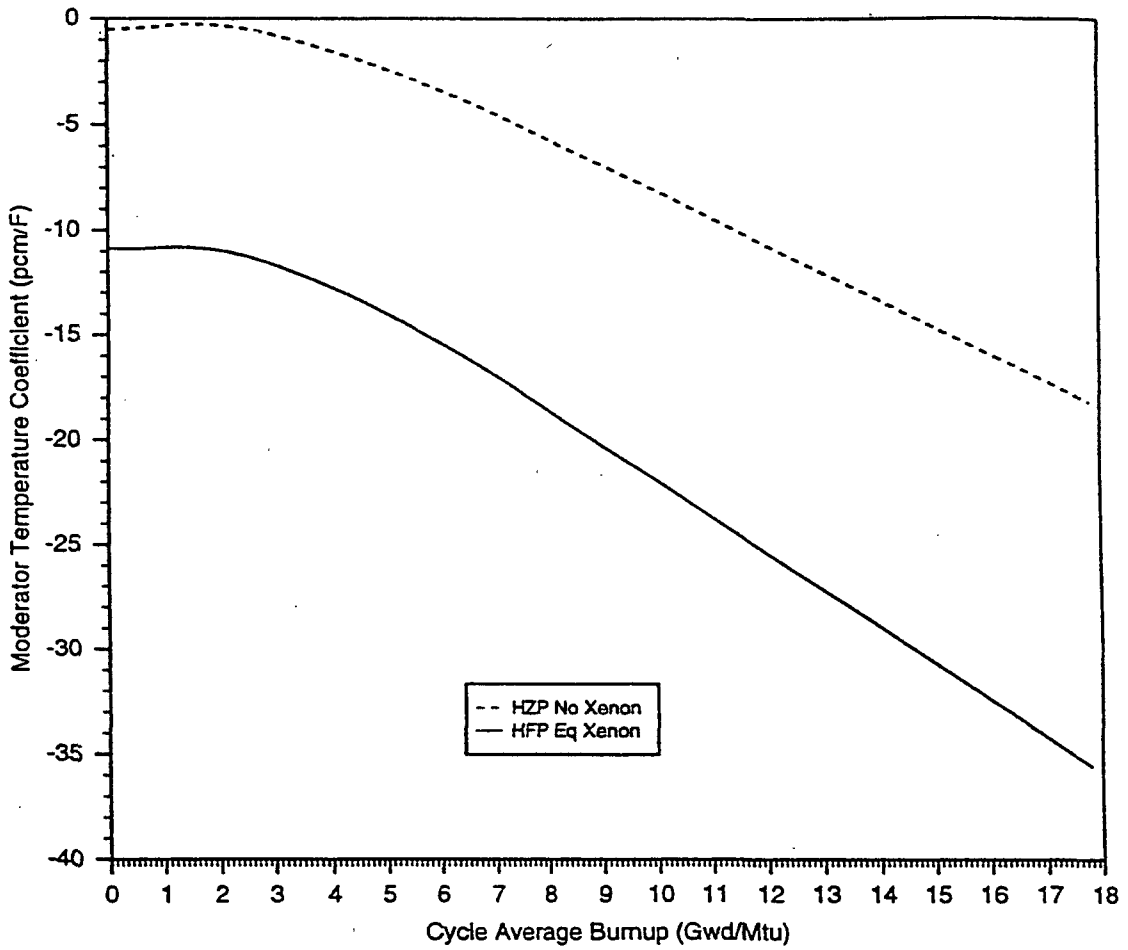
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Moderator Temperature Coefficient at BOL, ARO	
		Figure 4.3-30



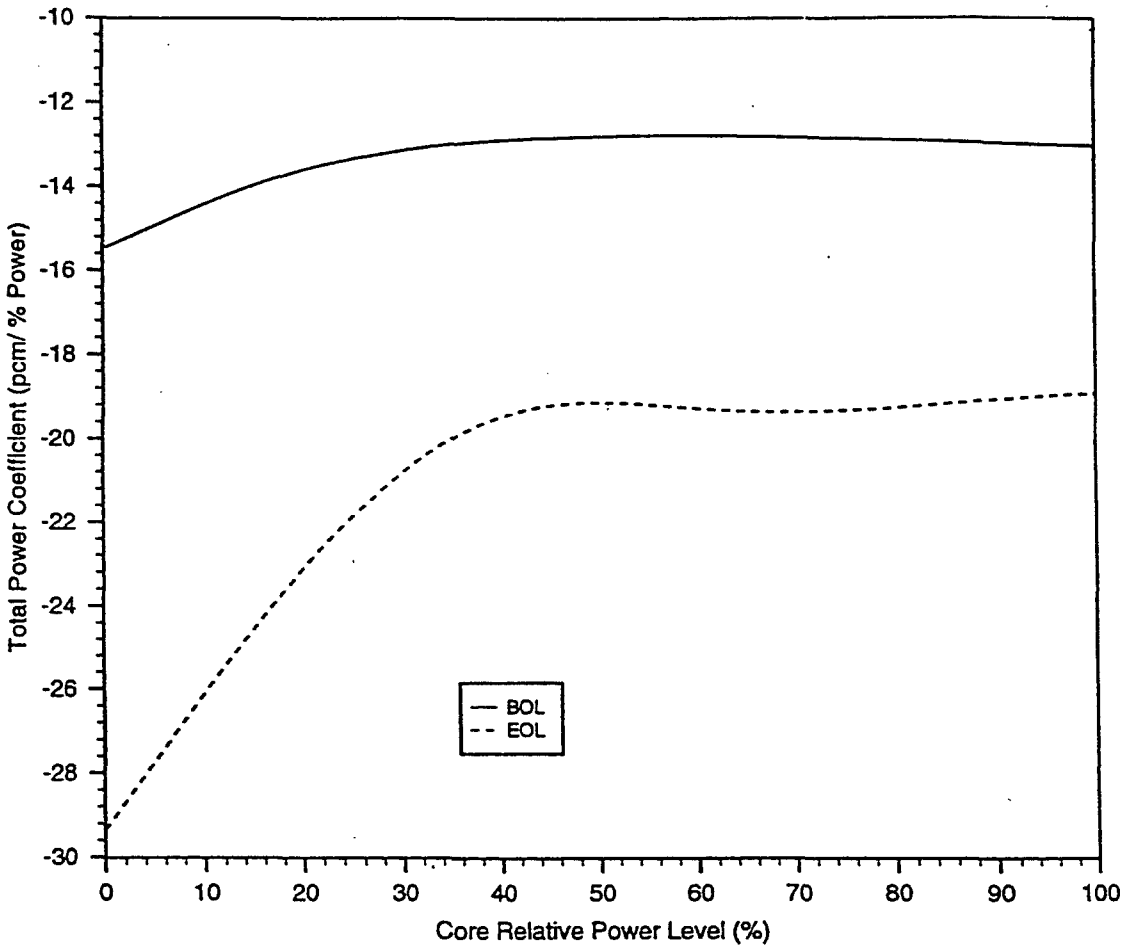
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Moderator Temperature Coefficient at EOL, ARO	
		Figure 4.3-31



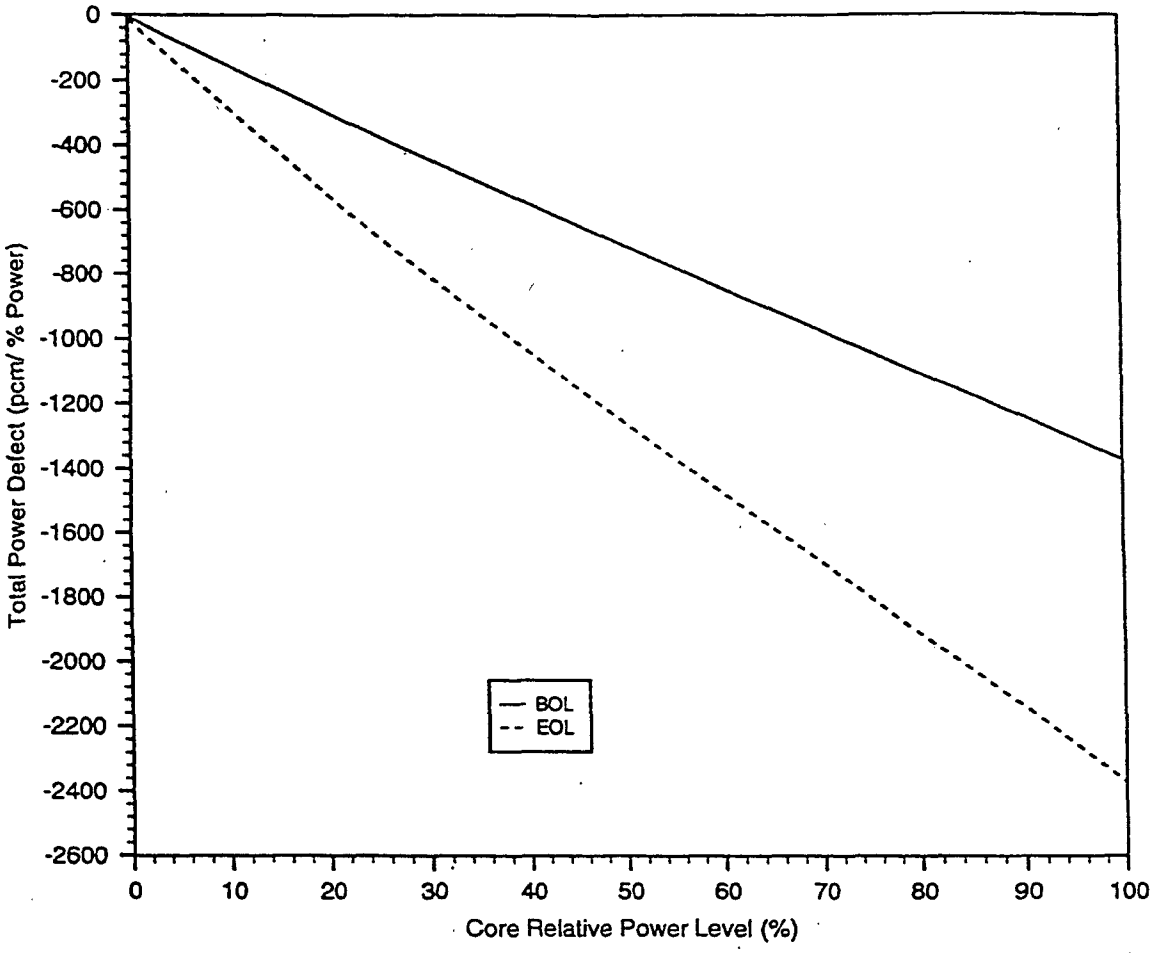
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Moderator Temperature Coefficient at BOL versus Boron Concentration	
		Figure 4.3-32



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Moderator Temperature Coefficient versus cycle Burnup	
		Figure 4.3-33



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Total Power Coefficient at BOL & EOL	
		Figure 4.3-34



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Total Power Defect at BOL & EOL	
		Figure 4.3-35

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
								180°								
1							S_P									
2				S_A		B		C		B		S_A				
3					S_D		S_B		S_B		S_C					
4		S_A		D				S_E				D		S_A		
5			S_C		A						A		S_D			
6		B				C		A		C				B		
7			S_B										S_B		S_P	
8	90°	C		S_E		A		D		A		S_E		C		270°
9	S_P		S_B										S_B			
10		B				C		A		C				B		
11			S_D		A						A		S_C			
12		S_A		D				S_E				D		S_A		
13					S_C		S_B		S_B		S_D					
14				S_A		B		C		B		S_A				
15									S_P							
								0°								

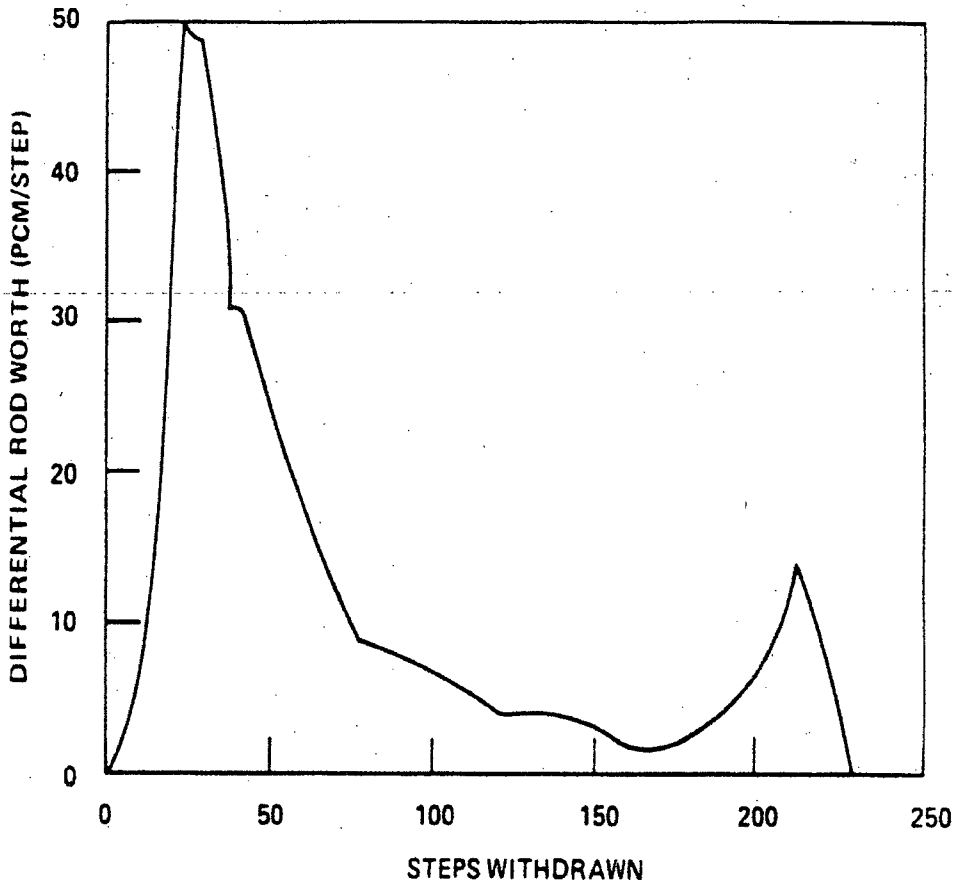
Control Bank	Number of Rods
A	8
B	8
C	8
D	5
Total	29

S_P - Spare Locations

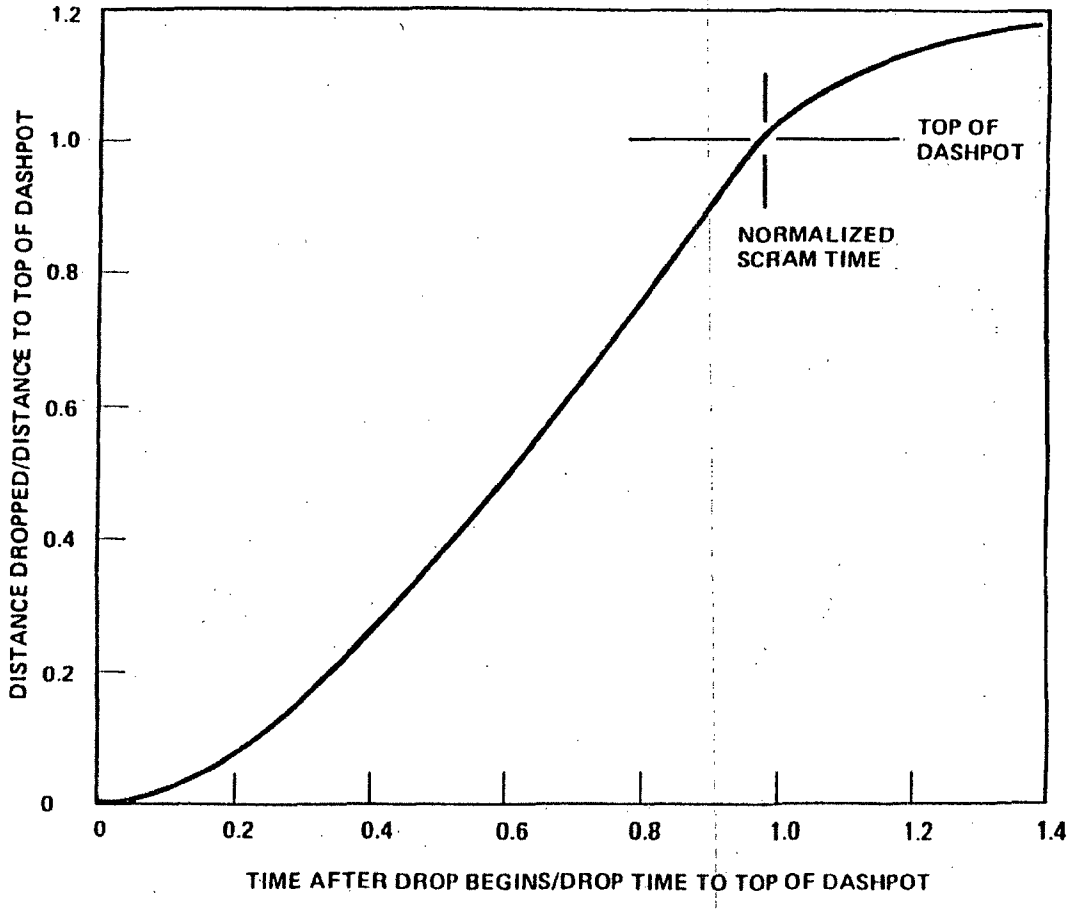
Shutdown Bank	Number of Rods
SA	8
SB	8
SC	4
SD	4
SE	4
Total	28

G:\Word\Images_P\UFSAR\4336.ds4

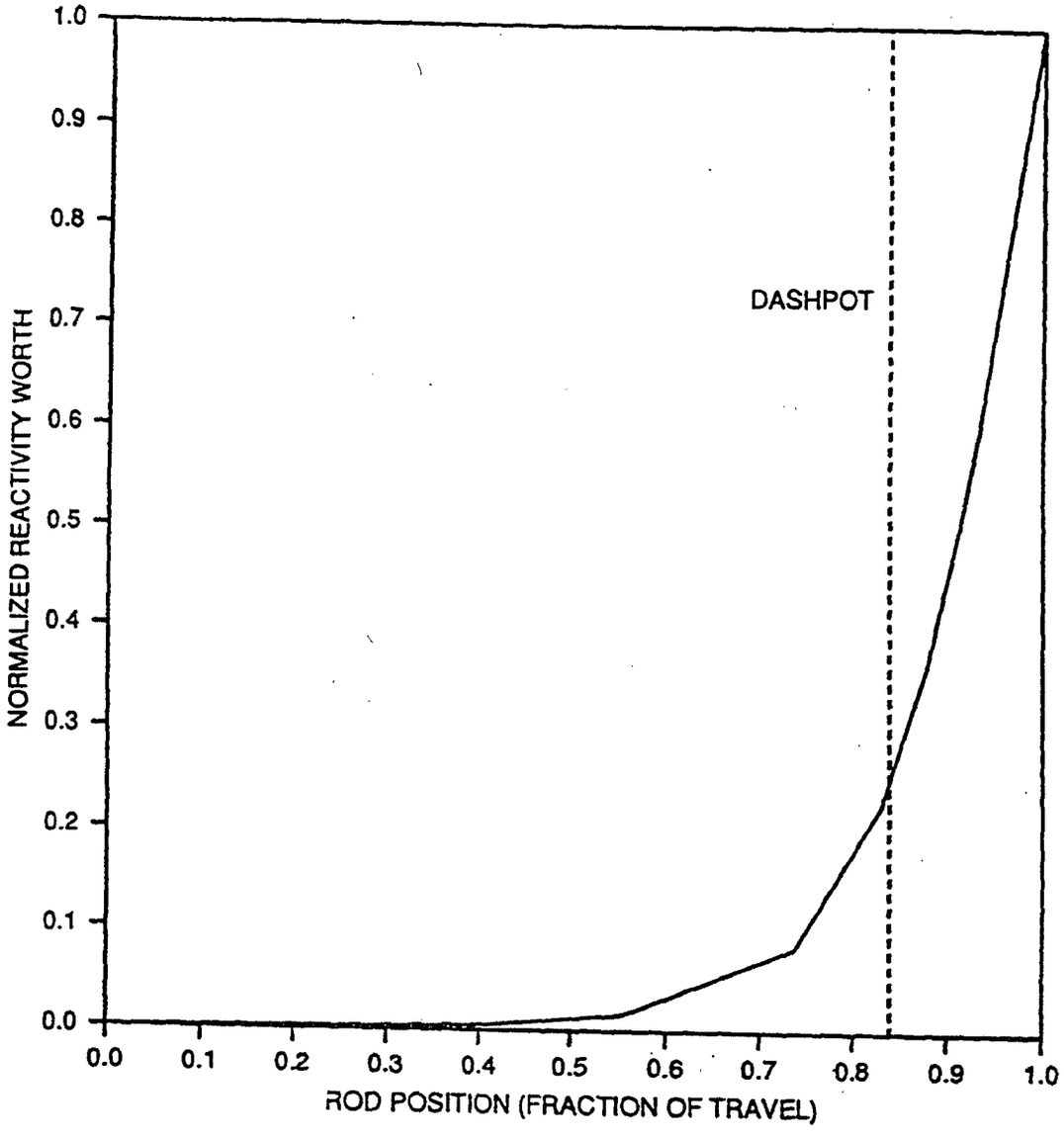
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Rod Cluster Control Assembly Pattern	
		Figure 4.3-36



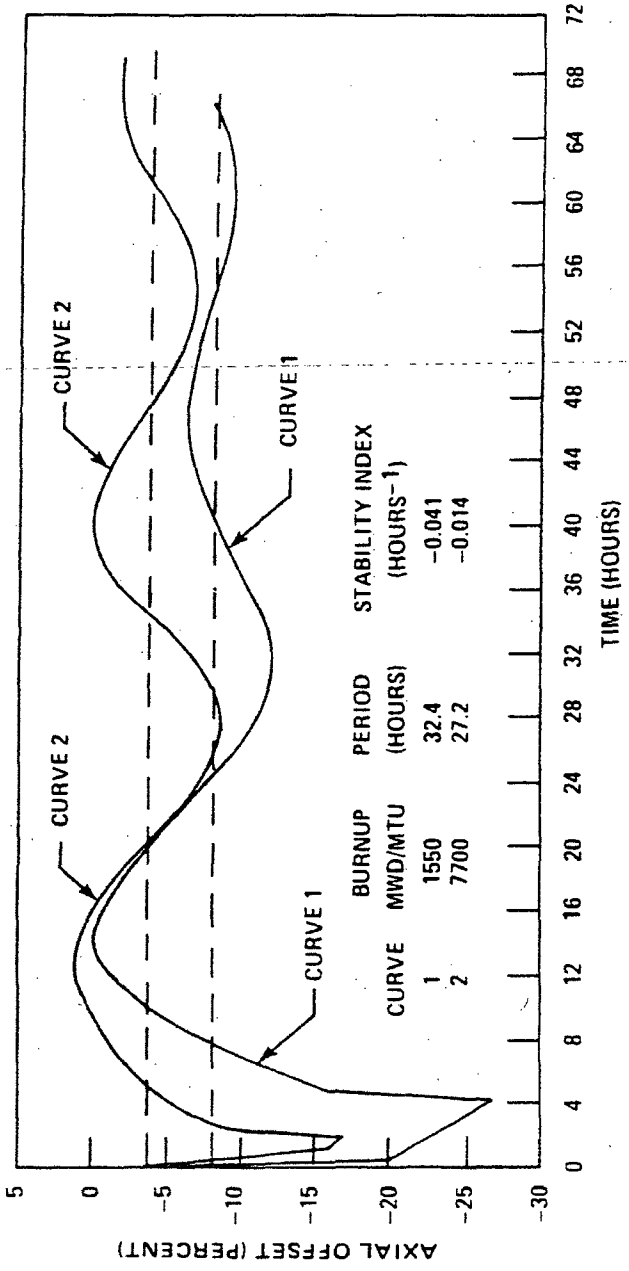
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Accident Simulated Withdrawal of Two Control Banks - EOL, HZP, Banks C and B Moving in Same Plane	
		Figure 4.3-37



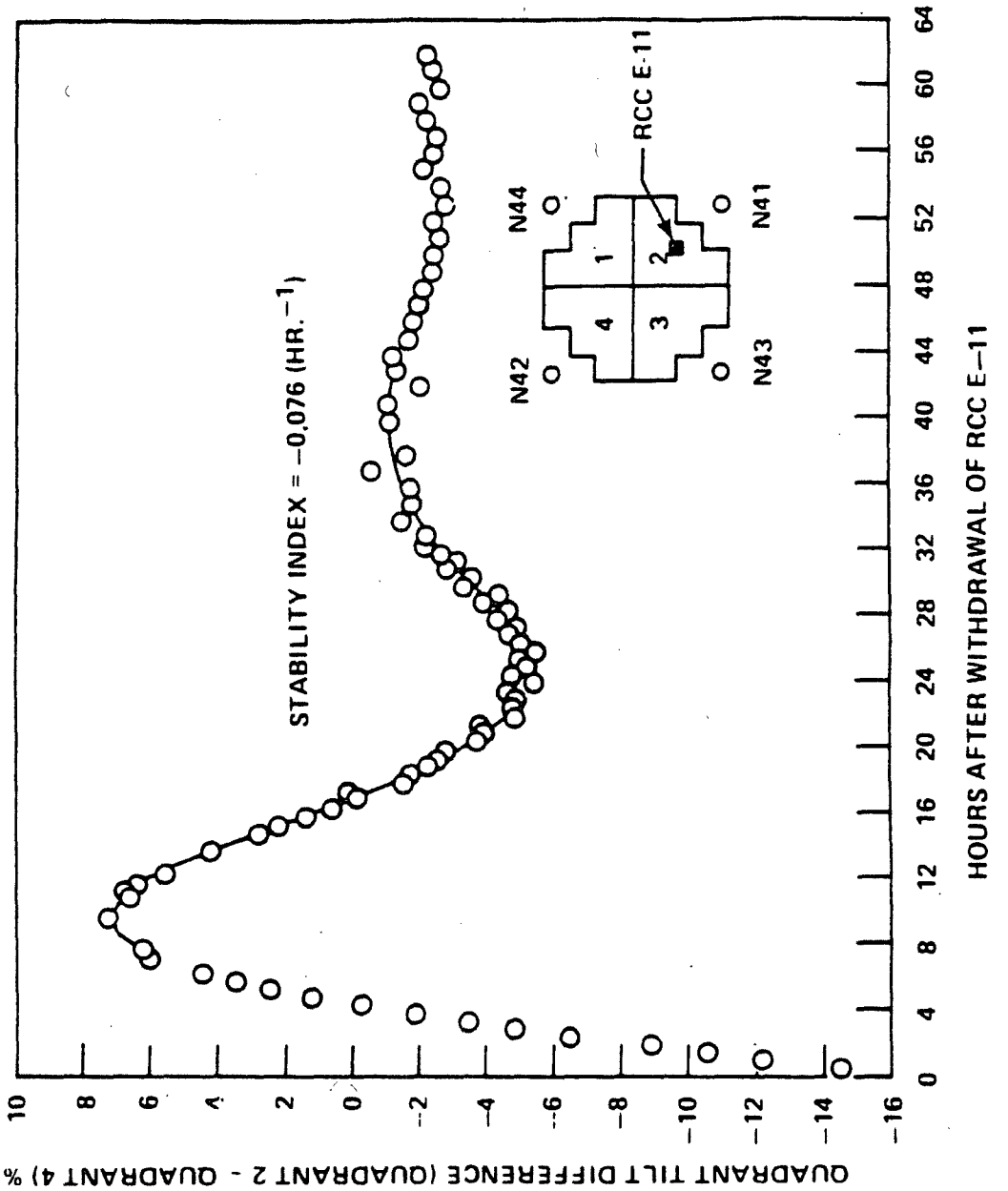
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Design Trip Curve	
		Figure 4.3-38



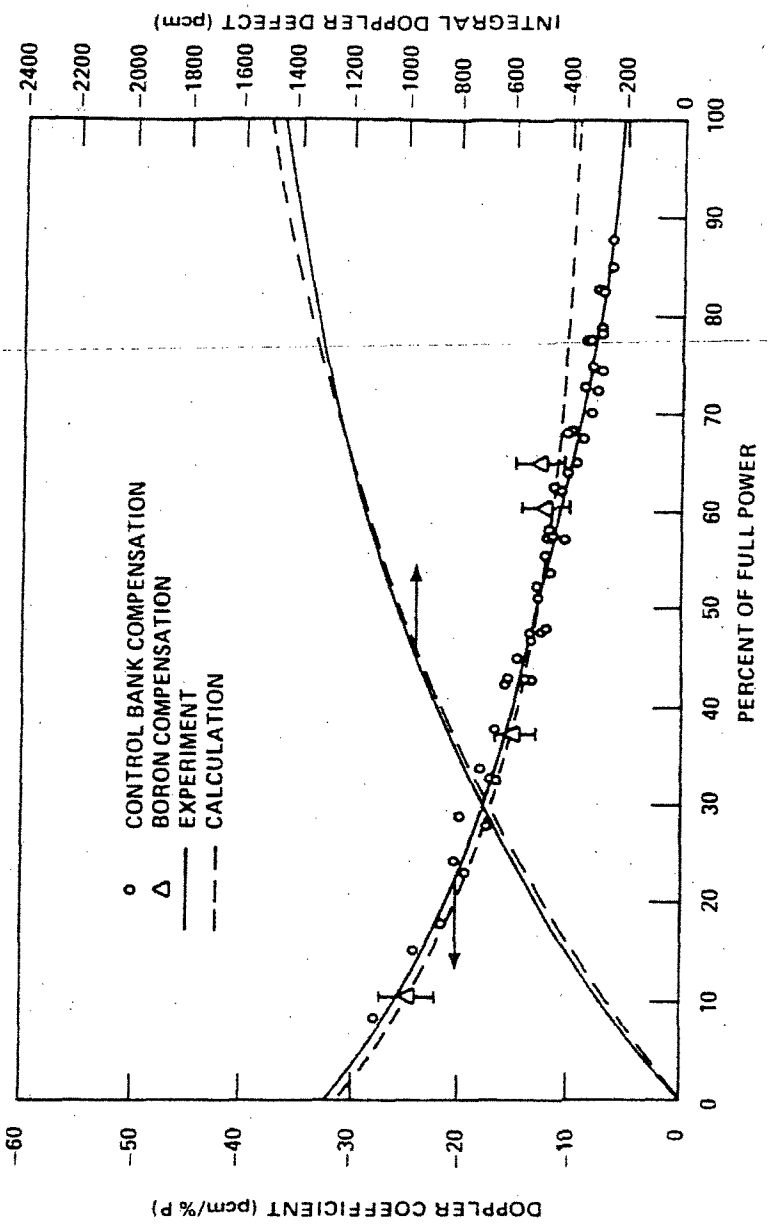
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized RCCA Reactivity Worth versus Fraction of Travel	
		Figure 4.3-39



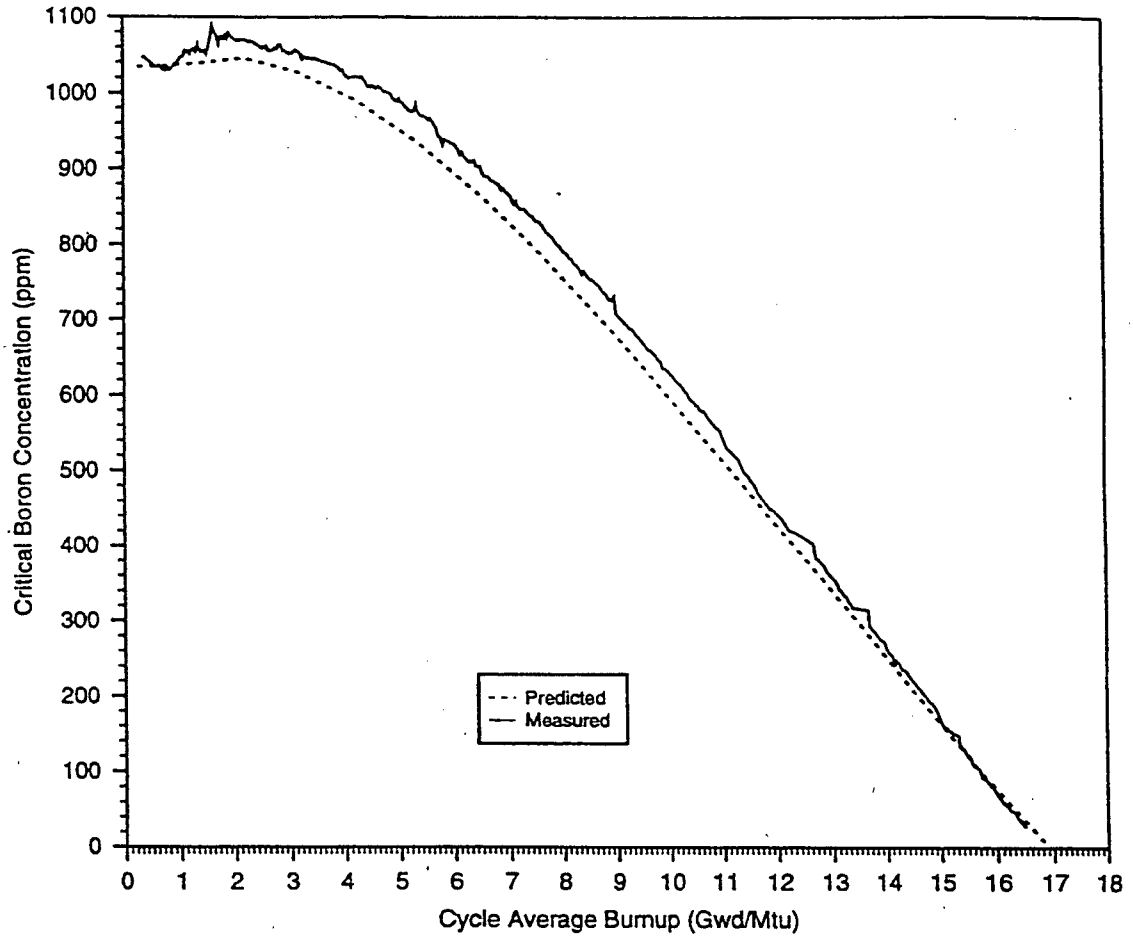
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Axial Offset vs. Time - PWR Core with a 12-Ft Height and 121 Assemblies	
		Figure 4.3-40



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	XY Xenon Test Thermocouple Response - Quadrant Tilt Difference vs. Time	
		Figure 4.3-41



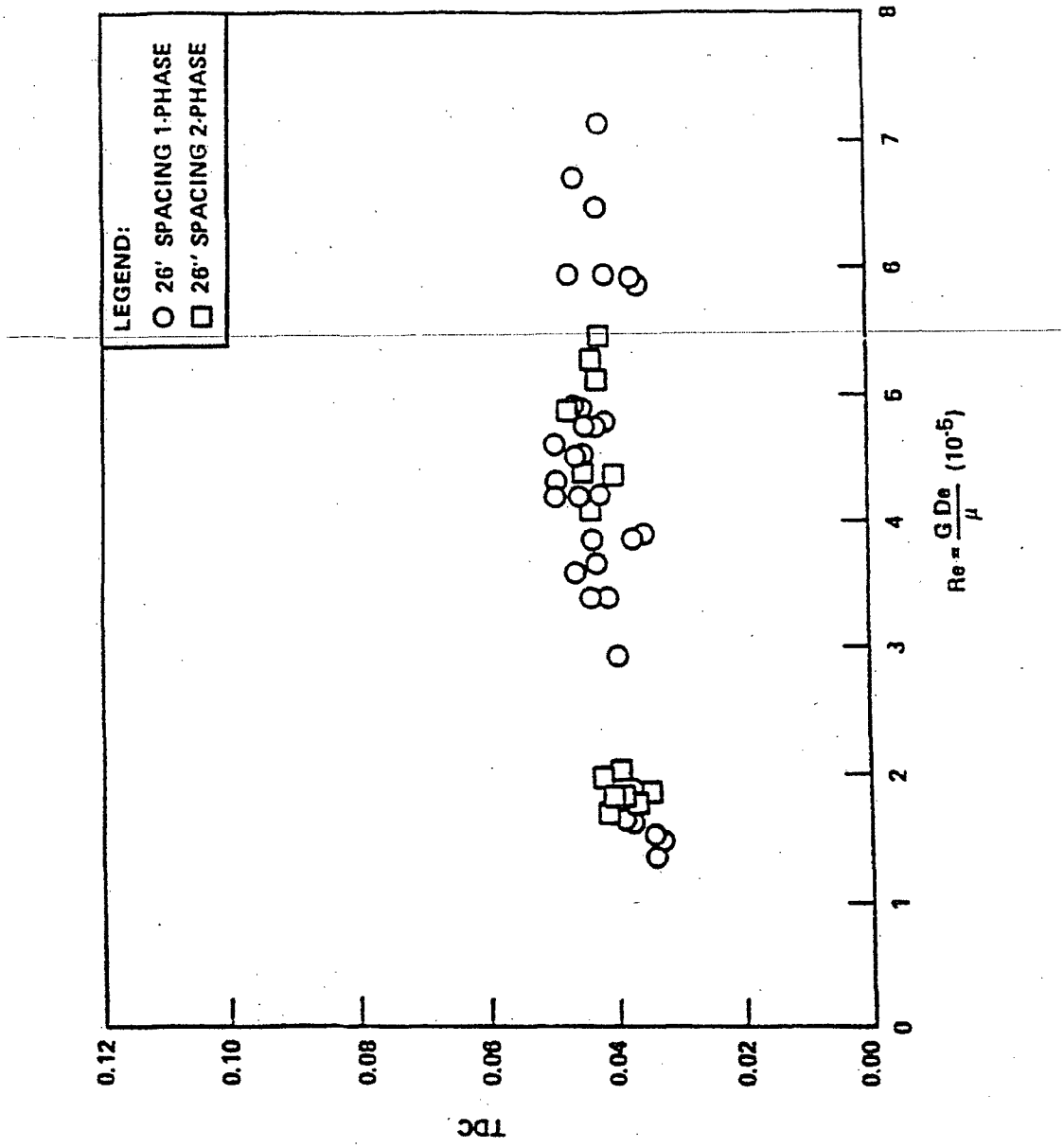
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Calculated and Measured Doppler Defect and Coefficient at BOL, 2-Loop Plant, 121 Assemblies, 12-Ft Core	
		Figure 4.3-42



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Typical Low Leakage Core Design Comparison of Calculated and Measured Boron Concentration	
		Figure 4.3-43

DELETED

DELETED



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	TDC vs. Reynolds Number for 26 Inch Grid Spacing	
		Figure 4.4-1

1.096 1.001						
1.029 1.001	1.120 1.001					
1.153 1.000	1.074 1.001	1.185 1.000				
1.166 1.000	1.209 1.000	1.162 1.001	1.185 1.000			
1.223 1.000	1.170 1.001	1.188 1.000	1.065 1.002	1.238 1.000		
1.126 1.001	1.161 1.001	1.093 1.002	1.086 1.002	0.916 1.001	0.967 1.001	
1.025 1.002	1.025 1.002	0.990 1.002	0.975 1.002	0.823 1.000	0.466 0.997	
0.717 0.999	0.780 0.999	0.664 0.998	0.563 0.997			

Key: $\Delta h/\overline{\Delta h}$ $G/\overline{G}$

For Radial Power Distribution Near Beginning of Life, Hot Full Power, Equilibrium Xenon

Calculated $F^N_{\Delta H} = 1.34$

G:\Word\Images_P\UFSAR\442.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized Radial Flow and Enthalpy Distribution at 4-Ft Elevation for Cycle 1	
		Figure 4.4-2

1.096 0.996							Key: $\Delta h/\overline{\Delta h}$ $G/\overline{G}$
1.026 1.002	1.120 0.994						
1.155 0.991	1.072 0.998	1.188 0.988					
1.169 0.990	1.212 0.986	1.165 0.990	1.189 0.998				
1.228 0.985	1.173 0.989	1.191 0.988	1.065 0.998	1.242 0.981			
1.129 0.992	1.165 0.990	1.095 0.995	1.086 0.996	0.916 1.009	0.963 1.006		
1.024 1.000	1.025 1.000	0.989 1.003	0.973 1.004	0.819 1.018	0.468 1.018		
0.715 1.022	0.777 1.019	0.663 1.021	0.562 1.019				

For Radial Power Distribution Near Beginning of Life, Hot Full Power, Equilibrium Xenon

Calculated $F_{\Delta H}^{N} = 1.34$

G:\Word\Images_P\UFSAR\443.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized Radial Flow and Enthalpy Distribution at 8-Ft Elevation for Cycle 1	
		Figure 4.4-3

1.097 0.995							Key: $\Delta h/\overline{\Delta h}$ $G/\overline{G}$
1.026 0.999	1.121 0.993						
1.157 0.991	1.073 0.996	1.189 0.989					
1.170 0.990	1.215 0.980	1.166 0.991	1.190 0.990				
1.231 0.987	1.175 0.990	1.193 0.989	1.066 0.997	1.243 0.987			
1.130 0.993	1.165 0.991	1.095 0.995	1.087 0.996	0.914 1.005	0.961 1.003		
1.023 1.000	1.024 1.000	0.987 1.002	0.971 1.003	0.817 1.011	0.469 1.030		
0.711 1.016	0.774 1.013	0.660 1.019	0.560 1.025				

For Radial Power Distribution Near Beginning of Life, Hot Full Power, Equilibrium Xenon

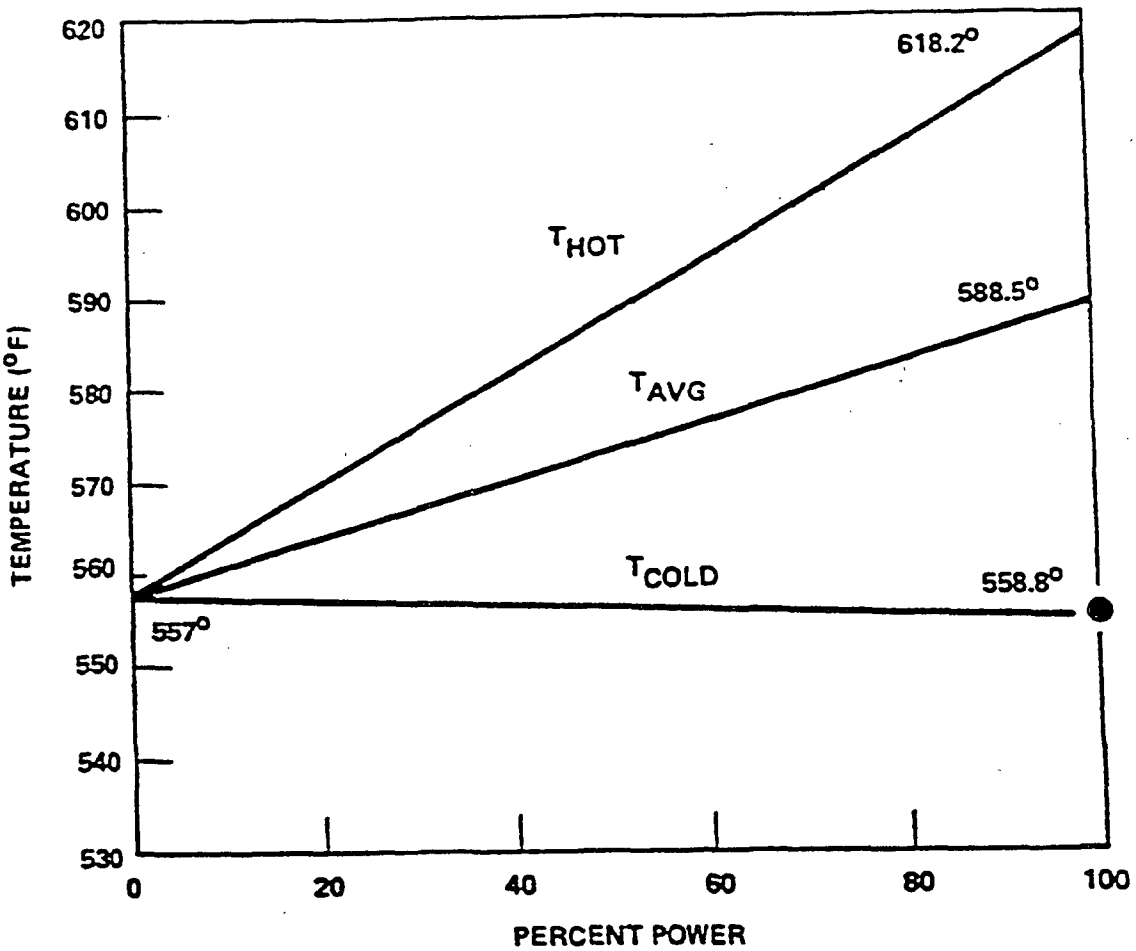
Calculated $F^N_{\Delta H} = 1.34$

G:\Word\Images_P\UFSAR\444.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized Radial Flow and Enthalpy Distribution at 12-Ft Elevation - Core Exit for Cycle 1	
		Figure 4.4-4

DELETED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Void Fraction vs. Thermodynamic Quality $H-H_{Sat}/Hg-H_{Sat}$	
		Figure 4.4-5



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant System Temperature - Percent Power Map	
		Figure 4.4-6

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							TD			TD					
2			TD			TD		TD							
3								TD		TD		TD		TD	
4		TD	TD					TD							
5					TD				TD		TD		TD		
6	TD		TD			TD		TD						TD	
7				TD			TD			TD			TD		
8	TD		TD		TD		TD			TD		TD	TD	TD	
9		TD							TD		TD				TD
10					TD		TD					TD			
11	TD				TD			TD			TD				TD
12						TD			TD			TD			
13			TD		TD			TD						TD	
14			TD				TD			TD		TD			
15					TD			TD							

004_Figure 04-07 Jan. 13, 2010

T = Thermocouple (58) At Tip of ICDA
D = Incore Detector Assembly (ICDA) (58 Locations) Replacement ICDAs **do not** have a Functional Path for a Movable Detector

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Distribution of Incore Instrumentation	
	Rev. 13	Figure 4.4-7

DELETED

DELETED

DELETED

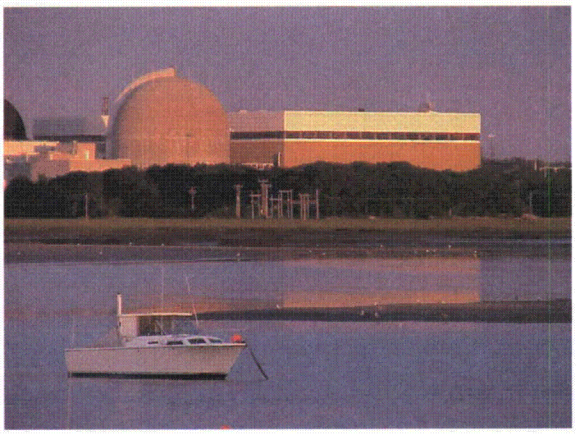
DELETED

DELETED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 4-4-12

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 5
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS



5.1 SUMMARY DESCRIPTION

The Reactor Coolant System (RCS), shown in Figure 5.1-1, Figure 5.1-2, Figure 5.1-3 and Figure 5.1-4, consists of four similar heat transfer loops connected in parallel to the reactor pressure vessel. Each loop contains a reactor coolant pump, steam generator and associated piping and valves. In addition, the system includes a pressurizer, pressurizer relief tank, pressurizer relief and safety valves, interconnecting piping and instrumentation necessary for operational control. All the above components are located in the Containment Building.

During operation, the RCS transfers the heat generated in the core to the steam generators where steam is produced to drive the turbine generator. Borated demineralized water is circulated in the RCS at a flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance. The water also acts as a neutron moderator and reflector and as a solvent for the neutron absorber used in chemical shim control.

The RCS pressure boundary provides a barrier against the release of radioactivity generated within the reactor, and is designed to ensure a high degree of integrity throughout the life of the plant.

RCS pressure is controlled by the use of the pressurizer, where water and steam are maintained in equilibrium by electrical heaters and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant. Spring-loaded safety valves and power-operated relief valves from the pressurizer provide for steam discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

a. Extent of the RCS

1. The reactor vessel, including control rod drive mechanism housings and incore instrumentation guide tubes
2. The reactor coolant side of the steam generators
3. Reactor coolant pumps
4. A pressurizer attached to one of the reactor coolant loops
5. Safety and relief valves
6. The interconnecting piping, valves and fittings between the principal components listed above
7. The piping, fittings and valves leading to connecting auxiliary or support systems up to and including the second isolation valve (from the high pressure side) on each line.

b. Reactor Coolant System Components

1. Reactor Vessel

The reactor vessel is cylindrical, with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the core, core supporting structures, control rods and other parts directly associated with the core.

The vessel has inlet and outlet nozzles located in a horizontal plane just below the reactor vessel flange but above the top of the core. Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom and flows up through the core to the outlet nozzles.

2. Steam Generators

The steam generators are vertical shell and U-tube evaporators with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

3. Reactor Coolant Pumps

The reactor coolant pumps are identical single speed centrifugal units driven by water/air cooled, three phase induction motors. The internal parts of the motor are cooled by air which is routed through external water/air heat exchangers. The shaft is vertical with the motor mounted above the pump. A flywheel on the shaft above the motor provides additional inertia to extend pump coastdown. The inlet is at the bottom of the pump and the discharge on the side.

4. Piping

The reactor coolant loop piping is specified in sizes consistent with system requirements.

The hot leg inside diameter is 29 inches and the inside diameter of the cold leg return line to the reactor vessel is 27½ inches. The piping between the steam generator and the pump suction is increased to 31 inches in inside diameter to reduce pressure drop and improve flow conditions to the pump suction.

5. Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. Electrical heaters are installed through the bottom head of the vessel while the spray nozzle, relief and safety valve connections are located in the top head of the vessel.

6. Safety and Relief Valves

The pressurizer safety valves are of the totally enclosed pop-type. The valves are spring loaded, self-activated with back pressure compensation. The power-operated relief valves limit system pressure for a large power mismatch. They are operated automatically or by remote manual control. Remotely operated valves are provided to isolate the inlet to the power-operated valves if excessive leakage occurs.

Steam from the pressurizer safety and relief valves is discharged into the pressurizer relief tank, where it is condensed and cooled by mixing with water near ambient temperature.

7. Pressurizer Relief Tank

Refer to Subsection 5.4.11.2a.

8. Pressurizer Relief Tank Pump

The PRT pump is an end suction centrifugal pump with TEFC Motor. The pump is used to circulate water through the PRT heat exchanger to cool the PRT following a discharge by the pressurizer SRVs or PORVS. The pump is also used to transfer the cooled fluid to the Liquid Waste Processing System.

9. Pressurizer Relief Tank Heat Exchanger

This heat exchanger is a horizontal shell and tube type. It is cooled by primary component cooling water to remove heat from the PRT following a discharge by the SRVs or PORVs.

c. Reactor Coolant System Performance Characteristics

Tabulations of important design and performance characteristics of the RCS are provided on Table 5.1-1.

d. Reactor Coolant Flow

The reactor coolant flow, a major parameter in the design of the system and its components, is established with a detailed design procedure supported by operating plant performance data, by pump model tests and analysis, and by pressure drop tests and analyses of the reactor vessel and fuel assemblies. Data from all operating plants have indicated that the actual flow has been well above the flow specified for the thermal design of the plant. By applying the design procedure described below, it is possible to specify the expected operating flow with reasonable accuracy.

Three reactor coolant flow rates are identified for the various plant design considerations. The definitions of these flows are presented in the following paragraphs.

1. Best Estimate Flow

The best estimate flow is the most likely value for the actual plant operating condition. This flow is based on the best estimate of the reactor vessel, steam generator and piping flow resistance, and on the best estimate of the reactor coolant pump head-flow capacity, with no uncertainties assigned to either the system flow resistance or the pump head. System pressure drops, based on best estimate flow, are presented on Table 5.1-1. Although the best estimate flow is the most likely value to be expected in operation, more conservative flow rates are applied in the thermal and mechanical designs.

2. Thermal Design Flow

Thermal design flow is the basis for the reactor core thermal performance, the steam generator thermal performance, and the nominal plant parameters used throughout the design. To provide the required margin, the thermal design flow accounts for the uncertainties in reactor vessel, steam generator and piping flow resistances, reactor coolant pump head, and the methods used to measure flow rate. The thermal design flow is approximately 4.5 percent less than the best estimate flow. The thermal design flow is confirmed when the plant is placed in operation. Tabulations of important design and performance characteristics of the RCS as provided on Table 5.1-1 are based on the thermal design flow.

3. Mechanical Design Flow

Mechanical design flow is the conservatively high flow used in the mechanical design of the reactor vessel internals and fuel assemblies. To assure that a conservatively high flow is specified, the mechanical design flow is based on a reduced system resistance and on increased pump head capability. For Seabrook, the mechanical design flow is approximately 4.0 percent greater than the best estimate flow.

Pump overspeed, due to a turbine generator overspeed of 20 percent, results in a peak reactor coolant flow of 120 percent of the mechanical design flow. The overspeed condition is applicable only to operating conditions when the reactor and turbine generator are at power.

e. Interrelated Performance and Safety Functions

The interrelated performance and safety functions of the RCS and its major components are listed below:

1. The RCS provides sufficient heat transfer capability to transfer the heat produced during power operation and when the reactor is subcritical, including the initial phase of plant cooldown, to the Steam and Power Conversion System.
2. The system provides sufficient capability to transfer the heat produced during the subsequent phase of plant cooldown and cold shutdown to the Residual Heat Removal System.
3. The system heat removal capability under power operation and normal operational transients, including the transition from forced to natural circulation, assures no fuel damage within the operating bounds permitted by the reactor control and protection systems.
4. The RCS provides the water used as the core neutron moderator and reflector and as a solvent for chemical shim control.
5. The system maintains the homogeneity of soluble neutron poison concentration and rate of change of coolant temperature, so that uncontrolled reactivity changes do not occur.
6. The reactor vessel is an integral part of the RCS pressure boundary and is capable of accommodating the temperatures and pressures associated with the operational transients. The reactor vessel functions to support the reactor core and control rod drive mechanisms.

7. The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, reactor coolant volume changes are accommodated in the pressurizer via the surge line.

8. The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generator.

9. The steam generators provide high quality steam to the turbine. The tube and tubesheet boundary are designed to prevent the transfer of activity generated within the core to the secondary system.

10. The RCS piping serves as a boundary for containing the coolant under operating temperature and pressure conditions and for limiting leakage (and activity release) to the containment atmosphere. The RCS piping contains demineralized borated water which is circulated at the flow rate and temperature consistent with achieving the reactor core thermal and hydraulic performance.

5.1.1 Schematic Flow Diagram

The RCS is shown schematically in Figure 5.1-5. Included in this figure are typical values for principal parameters of the system under normal steady state full power operating conditions. These values are based on the best estimate flow at the pump discharge. RCS volume under the above conditions is presented on Table 5.1-1.

5.1.2 Piping and Instrumentation Diagram

A piping and instrumentation diagram of the RCS is shown in Figure 5.1-1, Figure 5.1-2, Figure 5.1-3, and Figure 5.1-4. The diagrams show the extent of the systems located within the Containment, and the points of separation between the RCS and the secondary (heat utilization) system.

5.1.3 Elevation Drawing

Figure 1.2-2, Figure 1.2-3, Figure 1.2-5 and Figure 1.2-6 are plan and elevation drawings which present the principal dimensions of the Reactor Coolant System in relation to the supporting or surrounding concrete structures. These drawings show a measure of the protection afforded by the arrangement and the safety considerations incorporated in the layout.

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant's design lifetime. The RCPB is as defined in Section 50.2 of 10 CFR 50. In that definition, the RCPB extends to the outermost containment isolation valve in the system piping which penetrates the Containment and is connected to the Reactor Coolant System (RCS). Since other sections describe the components of these auxiliary fluid systems, this section will be limited to the components of the RCS as defined in Section 5.1, unless otherwise noted.

The reactor coolant pressure boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. The system is protected from overpressure by means of pressure-relieving devices, as required by applicable codes. Materials of construction are specified to minimize corrosion and erosion and to provide a structural system boundary throughout the life of the plant. Fracture prevention measures are taken to prevent brittle fracture. Inspection is in accordance with applicable codes, and provisions are made for surveillance of critical areas to enable periodic assessment of the boundary integrity, as described in Subsection 5.2.4.

For additional information on the RCS, and for components which are part of the RCPB but are not described in this section, refer to the following sections:

a. Section 6.3 - for the RCPB components which are part of the Emergency Core Cooling System

b. Subsection 9.3.4 - for the RCPB components which are part of the Chemical and Volume Control System

c. Subsection 3.9(N).1 - for the design loadings, stress limits, and analyses applied to the RCS and ASME Code Class 2 components

d. Subsection 3.9(N).3 - for the design loadings, stress limits and analyses applied to ASME Code Class 2 and 3 components.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR Section 50.55a

RCS components are designed and fabricated in accordance with 10 CFR 50, Section 50.55a, "Codes and Standards." The addenda of the ASME Code applied in the design of each component is listed in Table 5.2-1.

5.2.1.2 Applicable Code Cases

Compliance with Regulatory Guides 1.84 and 1.85 is discussed in Section 1.8. The following discussion addresses only unapproved or conditionally approved code cases (per Regulatory Guides 1.84 and 1.85) used on Class 1 primary components.

Code Case 1528 (SA 508 Class 2a) material has been used in the manufacture of the Seabrook pressurizer. It should be noted, that the purchase order for this equipment was placed prior to the original issue of Regulatory Guide 1.85 (June 1974); Regulatory Guide 1.85 presently reflects a conditional NRC approval of Code Case 1528. Westinghouse has conducted a test program which demonstrates the adequacy of Code Case 1528 material. The results of the test program are documented in Reference 1. Reference 1 and a request for approval of the use of Code Case 1528 have been submitted to the NRC (via letter NS-CE-1730 dated March 17, 1978, to Mr. J.F. Stolz, NRC Office of Nuclear Reactor Regulation, from Mr. C. Eicheldinger, Westinghouse Nuclear Safety Department).

In addition to Code Case 1528, the following ASME Code Cases have been used in the construction of Class 1 components:

NSSS

Component	Code Case	Title
CRDM	N-228	Alternate Rules for Sequence of Completion of Code Data Report Forms and Stamping for Section III, Classes 1, 2, 3 and MC Construction
Steam Generator	1528-3	High Strength Steel SA-508, Class 2, and SA-541, Class 2 Forgings, Section III, Class 1 Components
	1484-3	SB-163 Nickel-Chromium Iron Tubing (Alloy 600 and 690) and Nickel-Iron Chromium Alloy 800 at a Specified Minimum Yield Strength of 40.0 Ksi, Section III, Division 1, Class 1
RC Pipe	1423-2	Wrought Type 304 and 316 with Nitrogen Added, Sections I, III, VIII, Divisions 1 and 2
Valves	1553-1	Upset Heading and Roll Threading of SA-453 for High-Temperature Bolting, Section III, Classes 1, 2, 3 and MC Construction

Component	Code Case	Title
	1649	Modified SA-453 - Gr 660 for Classes 1, 2 and 3 and CS Construction

The reactor coolant pumps for Seabrook Station will be ASME certified following the cold hydrotest at the site. Following the testing, pumps will be N-stamped, ASME Code Data Forms will be signed, and appropriate Code Cases, if any, will be noted.

BOP

Component	Code Case	Title
Piping	N-228	Alternative Rules for Sequence of Completion of Code Case Data Report Forms and Stamping for Section III, Classes 1, 2, 3 and MC Construction.
	N-237	Hydrostatic Testing of Internal Piping, Section III; Division 1.
Small Bore	1621	Line Valve Internal and External Valves Items, Section III, Classes 1, 2 and 3, April 29, 1974.

5.2.2 Overpressure Protection

RCS overpressure protection is accomplished by the utilization of pressurizer safety valves along with the Reactor Protection System and associated equipment. Combinations of these systems provide compliance with the overpressure requirements of the ASME Boiler and Pressure Vessel Code, Section III, paragraphs NB-7300 and NC-7300, for pressurized water reactor systems.

Auxiliary or emergency systems connected to the RCS are not utilized during power operation for prevention of RCS overpressurization protection. During plant shutdowns, when RHR is in service, low temperature overpressure protection may be provided by RC-V24 and/or RC-V89. These are relief valves in the suction line to the RHR pumps and are outside the reactor coolant pressure boundary.

5.2.2.1 Design Bases

Overpressure protection is provided for the RCS by the pressurizer safety valves which discharge to the pressurizer relief tank by a common header. The transient which sets the design requirements for the primary system overpressure protection is a complete loss of steam flow to the turbine, with credit taken for steam generator safety valve operation. For the sizing of the pressurizer safety valves, main feedwater is maintained and no credit is taken for reactor trip nor the operation of the following:

a. Pressurizer power-operated relief valves

b. Steam line relief valve

c. Steam Dump System

d. Reactor Control System

e. Pressurizer Level Control System

f. Pressurizer spray valve.

For this transient, the peak RCS and peak steam system pressures must be limited to 110 percent of their respective design values.

Assumptions for the initial overpressure protection system design analysis include: (1) the plant is operating at the power level corresponding to the engineered safeguards design rating plus 2 percent uncertainty and (2) the RCS average temperature and pressure are at their maximum values. These are the most limiting assumptions with respect to system overpressure.

The assumed instrument and control errors were as follows:

2% Power

5.8°F Temperature

30 psi Pressure

Overpressure protection for the steam system is provided by steam generator safety valves. The steam system safety valve capacity is based on providing enough relief to remove 105 percent of the engineered safeguards design steam flow. This must be done by limiting the maximum steam system pressure to less than 110 percent of the steam generator shell side design pressure.

Blowdown and heat dissipation systems of the Nuclear Steam Supply System connected to the discharge of these pressure relieving devices are discussed in Subsection 5.4.11.

Steam Generator Blowdown Systems for the balance-of-plant are discussed in Chapter 10.

Postulated events and transients on which the design requirements of the Overpressure Protection System are based, are discussed in Reference 2.

5.2.2.2 Design Evaluation

The relief capacities of the pressurizer and steam generator safety valves are determined from the postulated overpressure transient conditions in conjunction with the action of the Reactor Protection System. An evaluation of the functional design of the system and an analysis of the capability of the system to perform its function is presented in Reference 2. Reference 2 describes in detail the types and number of pressure relief devices employed, relief device description, locations in the systems, reliability history, and the details of the methods used for relief device sizing based on typical worst case transient conditions and analysis data for each transient condition. The description of the analytical model used in the initial design analysis of the Overpressure Protection System and the basis for its validity are discussed in Reference 3.

A comparison of Seabrook parameters with those in Reference 2 is provided in Table 5.2-8.

A description of the pressurizer safety valves performance characteristics is provided in Chapter 5.

The redesign of the pressurizer safety and relief valve piping has eliminated the water seals from the safety valve piping. The new piping configuration has been analyzed through the use of the RELAP-5 computer program to determine the loadings to which the piping will be subjected under steam discharge and also water discharge. The associated effects are then factored into the piping and support analysis. The liquid water relief rates used in the analysis were: 344,644 lb./hr for the PORVs and 420,000 lb./hr for the safety valves. These values were extracted from the EPRI reports as typical values for this application.

a. Results of Design Evaluation

A plant-specific overpressurization protection study (Reference 8), based upon WCAP-7769 methodology, has been performed to demonstrate that the sizing criteria employed in the design of the Seabrook safety valves is as conservative as that recommended in SRP 5.2.2.

Pressurizer safety valve sizing is a two-step process. Initially, assumptions are made as to the worst anticipated transient, and a valve size in excess of that required is chosen. Secondly, all of the anticipated overpressure transients are analyzed, using the selected valve capacity. If the results from all the transients show that ASME Code allowables are not exceeded, then the selected valve capacity is accepted.

The sizing transient is the loss of load transient with feedwater maintained. The same transient conducted with loss of feedwater does not result in as large a valve capacity required. It is observed that the maximum safety valve capacity (0.86, normalized) is less for the loss of feedwater transient than that required for the transient with feedwater maintained (0.90, normalized). In short, more safety valve capacity is required for the loss of load transient with feedwater maintained.

It is also observed that the Reactor Coolant System (RCS) is protected from overpressure regardless of reactor trip, assuming all safety valves function properly. Should the reactor trip at the first protection grade trip (high pressurizer pressure), then only 40 percent of the total valve capacity is required. This 40 percent capacity readily falls within that provided by two of the three safety valves. In conclusion, the RCS is adequately protected from overpressure with only two (of the three) safety valves if the reactor trips at the first protection grade trip setpoint.

A major change that shows up in the plant-specific analysis is the use of a rod motion delay-time of 2 seconds, instead of the 1 second used in the generic analysis. As the reactor heat exists for an additional second, the pressurizer pressure goes higher and, consequently, more valve capacity is used. This increased capacity is displayed in the plant specific overpressure protection report.

For example, the overpressure protection analysis for Seabrook shows that with a 2 second rod motion delay time following a loss of load transient tripping on high pressurizer pressure, a peak pressurizer pressure of approximately 2550 psia is reached. The valve capacity used is approximately 67 percent of capacity. (RCS pressure peaks at approximately 2640 psia.)

Rod motion delay time has less effect following the second reactor protection system trip setpoint. Hence, peak pressurizer pressure has already occurred and the pressure is declining. Any additional delays in tripping the reactor at this stage have no effect on the overpressure protection afforded the RCS. However, should feedwater be lost, the steam system may be overpressurized. This takes a considerable amount of time, however, and can be prevented by reactor trip from any of the trip functions.

Should pressurizer safety valve capacity available be less than 67 percent (2 of 3 valves), then RCS pressure will go higher. A large margin exists between the calculated peak RCS pressure and the ASME code allowable of 2750 psia (110 percent). The limiting cases of 2750 psia were calculated as part of the EPRI safety and relief valve test program and found to lie within 40-50 percent of total valve capacity. Thus, RCS pressure should not exceed 2750 psia unless the valve capacity available fell below 40-50 percent.

Chapter 15 presents the results of more recent analyses confirming the continuing capability of the pressurizer and steam generator safety valves to provide overpressure protection for an increased pressurizer pressure control error of 50 psi, positive Moderator Temperature Coefficient, 8% steam generator tube plugging, and increased safety valve setpoint tolerances.

Note: The Loss of Load/Turbine Trip analysis performed at SPU conditions demonstrates that the pressurizer safety valves were adequately sized and that peak RCS pressure remains <110% design.

5.2.2.3 Piping and Instrumentation Diagrams

Overpressure protection for the RCS is provided by pressurizer safety valves shown in Figure 5.1-1. These discharge to the pressurizer relief tank by common header.

The steam system safety valves are discussed in Chapter 10 and are shown on Figure 10.3-1.

5.2.2.4 Equipment and Component Description

The operation, significant design parameters, number and types of operating cycles, and environmental qualification of the pressurizer safety valves are discussed in Subsection 5.4.13.

A discussion of the equipment and components of the steam system overpressure system is provided in Section 10.3.

5.2.2.5 Mounting of Pressure Relief Devices

Design and installation details pertinent to the mounting of pressure relief devices are discussed in Subsection 3.9(B).3.3.

5.2.2.6 Applicable Codes and Classification

The requirements of ASME Boiler and Pressure Vessel Code, Section III, paragraphs NB-7300 (Overpressure Protection Report) and NC-7300 (Overpressure Protection Analysis), are followed and complied with for pressurized water reactor systems.

Piping, valves and associated equipment used for overpressure protection are classified in accordance with ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." These safety class designations are delineated on Table 3.2-2 and shown on Figure 5.1-1.

For further information, refer to Section 3.9(N).

5.2.2.7 Material Specifications

Refer to Subsection 5.2.3 for a description of material specifications.

5.2.2.8 Process Instrumentation

Instrumentation is provided in the control room to give the open/closed status of the pressurizer safety and Power-Operated Relief Valves (PORVs). Each PORV is monitored by limit switches that operate red and green indicating lights on the main control board. The safety valves are monitored by an acoustic monitor that senses the acoustic emissions associated with flow in the discharge line that is common to the PORV and the three safety valves.

The above instrumentation is environmentally and seismically qualified and will actuate VAS alarms for PORV and/or safety valve open. The indication will not be redundant; therefore, backup indication and alarms are provided by temperature indication on the discharge of each safety valve and common discharge from the PORVs and by primary relief tank temperature, pressure, and level.

The instrumentation has been integrated into the Emergency Response Procedures (ERPs) and operator training. A determination of the needed characteristics of these displays to support the ERPs was made a part of the Detailed Control Room Design Review (DCRDR). A comparison was made of the needed characteristics against the available instrumentation, and no deficiencies were found.

5.2.2.9 System Reliability

The reliability of the pressure relieving devices is discussed in Section 4 of Reference 2.

5.2.2.10 Testing and Inspection

Testing and inspection of the overpressure protection components are discussed in Subsection 5.4.13.4 and Chapter 14.

5.2.2.11 RCS Pressure Control During Low Temperature Operation

Administrative procedures have been developed to aid the operator in controlling RCS pressure during low temperature operation. However, to provide a backup to the operator, the opening of any one of four valves would mitigate postulated inadvertent pressure excursions during water-solid (worst case) operations. Technical Specification 3.4.9.3, Overpressure Protection Systems, requires that two of these overpressure protections devices be operable at RCS cold leg temperatures below 290°F when the RCS is not depressurized and vented.

Two of the devices that provide protection against such postulated over-pressurization events are the two PORVs. The other two devices are the residual heat removal (RHRS) suction relief valves. Analyses have shown that one PORV or one RHRS suction relief valve is sufficient to prevent violation of pressure limits due to anticipated mass and heat input transients. The PORVs, when used to meet the technical specification requirements, are in automatic and armed. The RHRS suction relief valves are self-actuated.

The PORVs are powered by separate DC power sources; therefore, a single failure resulting in the loss of one DC bus will not disable both PORVs. Separate auctioneering circuits are provided for both the arming and actuating signals for each train. No single failure in the PORV power supply, the LTOP circuitry power supply, or the Cold Overpressure Mitigation System (COMS) circuitry itself will disable the automatic opening of both PORVs.

a. System Operation

Prior to alignment of the RCS to the RHRS, the two PORVs are utilized. The two pressurizer power-operated relief valves are each supplied with actuation logic to ensure that a redundant and independent RCS pressure control backup capability is available to the operator during low temperature operations. This system provides the capability for RCS inventory letdown, thereby maintaining RCS pressure within allowable limits. Refer to Section 7.6 and Subsections 5.4.7, 5.4.10, 5.4.13, and 9.3.4 for additional information on RCS pressure and inventory control during other modes of operation.

The basic function of the LTOP/COMS logic is to continuously monitor RCS temperature and pressure conditions whenever plant operation is at low temperatures. An auctioneered system temperature will be continuously converted to an allowable pressure and then compared to the actual RCS pressure. The system logic will first annunciate a main control board alarm whenever the measured pressure increases to within a predetermined amount of the allowable pressure, thereby indicating that a pressure transient is occurring, and on a further increase in measured pressure, an actuation signal will be transmitted to the power-operated relief valves and the PORV isolation valves when required to mitigate the pressure transient.

Once the RCS is aligned to the RHRS, the associated steam relief valves(s) are available for low temperature overpressure protection. These valves are spring-loaded and self-actuated. When both RHRS trains are aligned for shutdown cooling, redundancy is provided by the RHRS suction relief valves, and neither of the PORVs is required.

b. Operating Basis Earthquake (OBE) Evaluation

A fluid systems evaluation has been performed to analyze the potential for overpressure transients following on OBE. The basis of the evaluation assumes that the plant air system is inoperable since it is not seismically qualified. The results of the evaluation follow and demonstrate that overpressure transients following an OBE are not a concern.

1. A loss of plant air during the first part of plant cooldown, just prior to placing the RHRS on line at 350°F (i.e. decay heat removal via the steam generators, normal CVCS letdown), would cause the valves in the normal CVCS letdown path to fail closed and the charging flow control valve to fail open. These conditions would create a net mass addition to the RCS, thereby causing the pressure to increase. However, the pressure increase would be acceptable, since the pressurizer safety valves would limit system pressure to within allowable values.

2. A loss of plant air during the second part of plant cooldown (i.e. decay heat removal via the RHRS and a temperature less than 350°F) would cause the low pressure letdown valve in the residual heat removal to CVCS letdown path to fail closed and the charging flow control valve to fail open, similar to that discussed above. These conditions would create a net mass addition to the system which would be relieved by the solenoid-operated PORVs or the RHR relief valves which are set at 450 psig and, thus, maintain the pressure within allowable values.

For the various modes described above, the pressurizer PORVs, safety valves and RHRS relief valves provide pressure relief for the postulated transients following an OBE and, thus, maintain the primary system within the allowable pressure/temperature limits.

The Seabrook power-operated relief valves have been designed in accordance with the ASME Code to provide the integrity required for the reactor coolant pressure boundary. They have been analyzed for accident loads and for loads imposed by seismic events and have been shown to maintain their integrity.

c. Administrative Procedures

Although the pressure relieving devices described in Subsection 5.2.2.11a are designed to mitigate the pressure excursion and to address the allowable pressure limits, administrative procedures have been provided for minimizing the potential for any transient that could actuate the pressure relieving devices. The following discussion highlights these procedural controls, listed in hierarchy of their function in mitigating RCS cold overpressurization transients.

Of primary importance is the basic method of operation of the plant. Normal plant operating procedures will maximize the use of a pressurizer cushion (steam bubble) during periods of low pressure, low temperature operation. Water-solid operation is limited to mode 5 operation. It may be used for RCS fill and vent evolutions, or for crud bust cleanup operation prior to refueling evolutions. A pressurizer cushion dampens the plant's response to potential transient-generating inputs, thereby providing easier pressure control with the slower response rates.

An adequate cushion substantially reduces the severity of some potential pressure transients, such as reactor coolant pump-induced heat input, and slows the rate of pressure rise for others. In conjunction with the previously discussed alarms, this provides reasonable assurance that most potential transients can be terminated by operator action before the overpressure relief system actuates.

However, for those modes of operation when water-solid operation may still be possible, the following procedures will further highlight precautions to minimize the potential for developing an overpressurization transient:

1. Whenever the plant is water-solid and the reactor coolant pressure is being maintained by the low pressure letdown control valve, both high pressure letdown valves must be maintained open.

2. If all reactor coolant pumps have stopped for more than 5 minutes during plant heatup, and the reactor coolant temperature is greater than the charging and seal injection water temperature, no attempt shall be made to restart a pump unless a steam bubble is formed in the pressurizer. This precaution will minimize the pressure transient when the pump will be started and the cold water previously injected by the charging pumps will be circulated through the warmer reactor coolant components. The steam bubble will accommodate the resultant expansion as the cold water is rapidly warmed.

3. If all reactor coolant pumps are stopped and the RCS is being cooled down by the residual heat exchangers, a nonuniform temperature distribution may occur in the reactor coolant loops and the secondary side of the steam generators. No attempt shall be made to restart a reactor coolant pump unless (1) a steam bubble is formed in the pressurizer or (2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures when the cold leg temperatures are less than or equal to 350°F.

4. During plant cooldown, all steam generators shall be connected to the steam header to assure a uniform cooldown of the reactor coolant loops.

5. At least one reactor coolant pump shall be maintained in service until the reactor coolant temperature is reduced to 160°F.

These special precautions back up the normal operational mode of maximizing periods of steam bubble operation so that cold overpressure transient prevention is continued during periods of transitional operations.

The specific plant configurations of emergency core cooling system testing and alignment will also highlight procedures required to prevent developing cold overpressurization transients. During these limited periods of plant operation, the following procedures shall be followed:

1. To preclude inadvertent emergency core cooling system actuation during heatup and cooldown, blocking is required of the low pressurizer pressure and low steam line pressure safety injection signal actuation logic at 1900 psig.

2. When RCS pressure has decreased below 1000 psig and approximately 425°F during plant cooldown, the SI accumulator isolation valves are closed to prevent injection of the accumulator's volume into the RCS as RCS pressure is reduced.

This action involves energizing the MCCs powering the accumulators' MOV and then closing the valves. These actions are all performed in the control room.

Should a single failure disable the power supply to one or more of the SI accumulator isolation valves, solenoid operated vents are provided on each SI accumulator to allow relieving of the nitrogen overpressure gas to the Containment. These solenoids are Class 1E, powered by the emergency electrical train opposite that powering the SI accumulator isolation valve, and are operable from the control room and the remote shutdown location.

Additionally, during plant cooldown, one centrifugal charging pump, the positive displacement charging pump and both SI pumps will be made inoperable to preclude overpressurization events at low temperatures. This action can also be performed in the control room and the remote shutdown location.

Prior to decreasing RCS temperature below 350°F, the safety injection pumps and the nonoperating charging pumps are made inoperable. It should be noted that the high containment pressure safety injection actuation logic cannot be blocked. It should be noted that the high containment pressure safety injection actuation logic cannot be blocked.

3. Periodic emergency core cooling system pump performance testing requires the testing of the pumps during normal power operation or at hot shutdown conditions, to preclude any potential for developing a cold overpressurization transient.

During shutdown conditions charging pump and SI pump operation are restricted in accordance with the Technical Specifications and their supporting bases.

4. "S" signal circuitry testing, if performed during cold shutdown, will also require RHRS alignment and power lockout of both SI pumps and nonoperating charging pump to preclude developing cold overpressurization transients.

The above procedures, which will be followed for normal operations with a steam bubble, transitional operations where the Reactor Coolant System is potentially water-solid, and during specific testing operations, will provide in-depth cold overpressure prevention or reduction, thereby augmenting the installed Overpressure Relief System.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications

Material specifications used for the principal pressure retaining applications in each component of the RCPB are listed in Table 5.2-2 for ASME Class 1 primary components and Table 5.2-3 for ASME Class 1 and 2 auxiliary components. Table 5.2-2 and Table 5.2-3 also include the unstabilized austenitic stainless steel material specifications used for components in systems required for reactor shutdown and for emergency core cooling.

The unstabilized austenitic stainless steel materials for the reactor vessel internals, which are required for emergency core cooling for any mode of normal operation or under postulated accident conditions and for core structural load bearing members, are listed in Table 5.2-4.

In some cases, the tables may not be totally inclusive of the material specifications used in the listed applications. However, the listed specifications are representative of those materials used.

The materials used conform with the requirements of the ASME Code, Section III, plus applicable addenda and Code cases.

The welding materials used for joining the ferritic base materials of the RCPB conform to, or are equivalent to, ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18 and 5.20. They are qualified to the requirements of the ASME Code, Section III.

The welding materials used for joining austenitic stainless steel base materials of the RCPB conform to ASME Material Specifications SFA 5.4 and 5.9. They are qualified to the requirements of the ASME Code, Section III.

The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Material Specifications SFA 5.11 and 5.14. They are qualified to the requirements of the ASME Code, Section III.

5.2.3.2 Compatibility with Reactor Coolant

a. Chemistry of Reactor Coolant

The Reactor Coolant System (RCS) chemistry specifications are identified in Technical Specification 3.9.1 and Technical Requirement TR30.

The RCS water chemistry is selected to minimize corrosion. A routinely scheduled analysis of the coolant chemical composition is performed to verify that the reactor coolant chemistry meets the specifications.

The Chemical and Volume Control System provides a means for adding chemicals to the RCS which control the pH of the coolant during pre-startup and subsequent operation, scavenge oxygen from the coolant during heatup, and control radiolysis reactions involving hydrogen, oxygen and nitrogen during all power operations subsequent to startup. The limits specified for chemical additives and reactor coolant impurities for power operation are described in the EPRI PWR Primary Water Chemistry Guidelines and implemented in the Chemistry Manual.

The pH control chemical employed is lithium hydroxide monohydrate, enriched in ^{7}Li isotope to 99.9 percent. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/Inconel systems. In addition, ^{7}Li is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The lithium-7 hydroxide is introduced into the RCS via the charging flow. The solution is prepared in the laboratory and transferred to the chemical additive tank. Reactor makeup water is then used to flush the solution to the suction header of the charging pumps. The concentration of lithium-7 hydroxide in the RCS is maintained in the range specified for pH control. If the concentration exceeds this range, the cation bed demineralizer is employed in the letdown line in series operation with the mixed bed demineralizers.

During a reactor startup from a cold condition (i.e., following a refueling outage), hydrazine may be added to the coolant as an oxygen scavenging agent. The hydrazine is typically added prior to formation of a steam bubble in the pressurizer. This allows an excess of hydrazine to be present in the system for improved reaction kinetics to take place at higher RCS temperature. Oxygen limits are described in the EPRI PWR Primary Water Chemistry Guidelines and implemented in the Chemistry Manual.

The reactor coolant is treated with dissolved hydrogen to control the products formed by the decomposition of water by radiolysis. The hydrogen reacts with the oxygen to form water and prevent the oxygen from reacting with the nitrogen and forming nitric acid (HNO_3). The hydrogen overpressure is accomplished by a self-contained pressure control valve in the vapor space of the volume control tank. This valve can be adjusted to maintain the correct equilibrium hydrogen concentration.

Boron, in the chemical form of boric acid, is added to the RCS to accomplish long-term reactivity control of the core. The mechanism for the process involves the absorption of neutrons by the ^{10}B isotope of naturally occurring boron.

Suspended solids (corrosion product particulates) and other impurity concentrations are maintained below specified limits by controlling the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the CVCS mixed bed demineralizers.

b. Compatibility of Construction Materials with Reactor Coolant

All of the ferritic low alloy and carbon steels which are used in principal pressure-retaining applications are provided with corrosion resistant cladding on all surfaces that are exposed to the reactor coolant. The corrosion resistance of this cladding material is at least equivalent to the corrosion resistance of Types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron, martensitic stainless steel and precipitation-hardened stainless steel. The cladding on ferritic type base materials receives a post-weld heat treatment, as required by the ASME Code.

Ferritic low alloy and carbon steel nozzles are safe-ended with either stainless steel wrought materials, stainless steel weld metal analysis A-7 (designated A-8 in the 1974 Edition of the ASME Code), or nickel-chromium-iron alloy weld metal F-Number 43. The latter buttering material requires further safe-ending, with austenitic stainless steel base material after completion of the post-weld heat treatment, when the nozzle is larger than a 4 inch nominal inside diameter and/or the wall thickness is greater than 0.531 inches.

All of the austenitic stainless steel and nickel-chromium-iron alloy base materials with primary pressure-retaining applications are used in the solution-annealed heat treat condition. These heat treatments are as required by the material specifications.

During subsequent fabrication, these materials are not heated above 800°F, other than locally by welding operations. The solution-annealed surge line material is subsequently formed by hot bending, followed by a re-solution annealing heat treatment.

Components with stainless steel, sensitized in the manner expected during component fabrication and installation, will operate satisfactorily under normal plant chemistry conditions in pressurized water reactor systems because chlorides, fluorides and oxygen are controlled to very low levels.

c. Compatibility With External Insulation and Environmental Atmosphere

In general, all of the materials listed in Table 5.2-2 and Table 5.2-3, which are used in principal pressure-retaining applications and are subject to elevated temperature during system operation, are in contact with thermal insulation that covers their outer surfaces.

The thermal insulation used on the RCPB is either reflective stainless steel type or made of compounded materials that yield low leachable chloride and/or fluoride concentrations. The compounded materials in the form of blocks, boards, cloths, tapes, adhesives, cements, etc., are silicated to provide protection of austenitic stainless steels against stress corrosion which may result from accidental wetting of the insulation by spillage, minor leakage or other contamination from the environmental atmosphere. Section 1.8 includes a discussion which indicates the degree of conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is anticipated from service experience, such as valve packing, pump seals, etc., only materials that are compatible with the coolant are used. These are shown in Table 5.2-2 and Table 5.2-3. Ferritic materials exposed to coolant leakage can be readily observed as part of the in-service visual and/or nondestructive inspection program to assure the integrity of the component for subsequent service.

5.2.3.3 Fabrication and Processing of Ferritic Materials

a. Fracture Toughness

The fracture toughness properties of the primary components meet the requirements of the ASME Code, Section III, paragraph NB-2300.

The fracture toughness properties of the reactor vessel materials are discussed in Section 5.3.

Limiting steam generator and pressurizer RT_{NDT} temperatures are guaranteed at 60°F for the base materials and the weldments. These materials at 120°F will meet the 50 ft-lbs. absorbed energy and 35 mils lateral expansion requirements of the ASME Code Section III. The actual results of these tests are provided in the ASME material data reports which are supplied for each component and submitted to Public Service Company of New Hampshire at the time of shipment of the component.

Calibration of temperature and charpy impact test machines are performed to meet the requirements of the ASME Code, Section III, paragraph NB-2360.

Westinghouse has conducted a test program to determine the fracture toughness of low alloy ferritic materials with specified minimum yield strengths greater than 50,000 pounds per square inch (psi), to demonstrate compliance with Appendix G of the ASME Code, Section III. In this program, fracture toughness properties were determined and shown to be adequate for base metal plates and forgings, weld metal and heat-affected zone metal for higher strength ferritic materials used for components of the RCPB. The results of the program are documented in Reference 1 which was submitted to the NRC via Reference 4.

b. Control of Welding

All welding is conducted using procedures qualified according to the rules of Sections III and IX of the ASME Code. Control of welding variables, as well as examination and testing during procedure qualification and production welding, is performed in accordance with ASME Code requirements.

Section 1.8 includes discussions which indicate the degree of conformance of the ferritic materials components of the RCPB with Regulatory Guides 1.34, "Control of Electroslag Properties"; 1.43, "Control of Stainless Steel Weld Cladding of Low Alloy Steel Components"; 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"; and 1.71, "Welder Qualification for Areas of Limited Accessibility."

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

Subsections 5.2.3.4a to 5.2.3.4e address Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and present the methods and controls used by Westinghouse to avoid sensitization and to prevent intergranular attack of austenitic stainless steel components. The subject of austenitic stainless steel for use in ESF applications, is addressed by UE&C in Subsection 6.1(B).1, Metallic Materials. Also, Section 1.8 includes a discussion which indicates the degree of conformance with Regulatory Guide 1.44.

a. Cleaning and Contamination Protection Procedures

It is required that all austenitic stainless steel materials used in fabrication, installation, and testing of nuclear steam supply components and systems be handled, protected, stored, and cleaned according to recognized and accepted methods which are designed to minimize contamination which could lead to stress corrosion cracking. The rules covering these controls are stipulated in Westinghouse process specifications. As applicable, these process specifications supplement the equipment specifications and purchase order requirements for austenitic stainless steel components or systems which Westinghouse procures for the Seabrook NSSS, regardless of the ASME Code classification. They are also given to United Engineers and Constructors and to Public Service Company of New Hampshire for their information.

The process specifications which define these requirements and which follow the guidance of ANSI N-45 committee specifications are as follows:

82560HM	Requirements for Pressure Sensitive Tapes for Use on Austenitic Stainless Steel
83336KA	Requirements for Thermal Insulation Used on Austenitic Stainless Steel Piping and Equipment
83860LA	Requirements for Marking of Reactor Plant Components and Piping
84350HA	Site Receiving Inspection and Storage Requirements for Systems, Material and Equipment
84351NL	Determination of Surface Chloride and Fluoride on Austenitic Stainless Steel Materials
85310QA	Packaging and Preparing Nuclear Components for Shipment and Storage
292722	Cleaning and Packaging Requirements of Equipment for Use in the NSSS
597756	Pressurized Water Reactor Auxiliary Tanks Cleaning Procedures
597760	Cleanliness Requirements During Storage Construction, Erection and Start-Up Activities of Nuclear Power System.

Section 1.8 includes a discussion which indicates the degree of conformance of the austenitic stainless steel components of the RCPB with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

b. Solution Heat Treatment Requirements

The austenitic stainless steels listed in Table 5.2-2, Table 5.2-3 and Table 5.2-4 are utilized in the final heat treated condition required by the respective ASME Code Section II materials specification.

c. Material Inspection Program

The Westinghouse practice is that austenitic stainless steel materials of product forms with simple shapes need not be corrosion tested provided that the solution heat treatment is followed by water quenching. Simple shapes are defined as all plates, sheets, bars, pipe and tubes, as well as forgings, fittings, and other shaped products which do not have inaccessible cavities or chambers that would preclude rapid cooling when water quenched. When testing is required, the tests are performed in accordance with ASTM A262, Practice A or E, as amended by Westinghouse Process Specification 84201 MW.

d. Prevention of Intergranular Attack of Unstabilized Austenitic Stainless Steels

Unstabilized austenitic stainless steels are subject to intergranular attack (IGA), provided that three conditions are present simultaneously. These are:

1. An aggressive environment, e.g., an acidic aqueous medium containing chlorides or oxygen
2. A sensitized steel
3. A high temperature.

If any one of the three conditions described above is not present, intergranular attack will not occur. Since high temperatures cannot be avoided in all components in the NSSS, Westinghouse relies on the elimination of conditions 1 and 2 to prevent intergranular attack on wrought stainless steel components.

The water chemistry in the RCS of a Westinghouse pressurized water reactor is rigorously controlled to prevent the intrusion of aggressive species. In particular, the maximum permissible oxygen and chloride concentrations are 0.005 ppm and 0.15 ppm, respectively. Reference 5 describes the precautions taken to prevent the intrusion of chlorides into the system during fabrication, shipping, and storage. The use of hydrogen over-pressure precludes the presence of oxygen during operation. The effectiveness of these controls has been demonstrated by both laboratory tests and operating experience. The long time exposure of severely sensitized stainless steel in early plants to PWR coolant environments has not resulted in any sign of intergranular attack. Reference 5 describes the laboratory experimental findings and the Westinghouse operating experience. The additional years of operations since the issuing of Reference 5 have provided further confirmation of the earlier conclusions.

Severely sensitized stainless steels do not undergo any intergranular attack in Westinghouse pressurized water reactor coolant environments. In spite of the fact there never has been any evidence that PWR coolant water attacks sensitized stainless steels, Westinghouse considers it good metallurgical practice to avoid the use of sensitized stainless steels in the NSSS components. Accordingly, measures are taken to prohibit the purchase of sensitized stainless steels and to prevent sensitization during component fabrication. Wrought austenitic stainless steel stock used for components that are part of (1) the RCPB, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals (relied upon to permit adequate core cooling for normal operation or under postulated accident conditions) is used in one of the following conditions:

1. Solution-annealed and water quenched, or
2. Solution-annealed and cooled through the sensitization temperature range within less than approximately five minutes.

It is generally accepted that these practices will prevent sensitization. Westinghouse has verified this by performing corrosion tests on as-received wrought material.

Westinghouse recognizes that the heat-affected zones of welded components must, of necessity, be heated into the sensitization temperature range, 800 to 1500°F. However, severe sensitization, i.e., continuous grain boundary precipitates of chromium carbide, with adjacent chromium depletion, can still be avoided by control of welding parameters and welding processes. The heat input* and associated cooling rate through the carbide precipitation range are of primary importance. Westinghouse has demonstrated this by corrosion testing a number of weldments.

Of 25 production and qualification weldments tested, representing all major welding processes and a variety of components, and incorporating base metal thicknesses from 0.10 to 4.0 inches, only portions of two were severely sensitized. Of these, one involved a heat input of 120,000 joules, and the other involved a heavy socket weld in relatively thin-walled material. In both cases, sensitization was caused by high heat inputs relative to the section thickness. However, in only the socket weld did the sensitized condition exist at the surface, where the material is exposed to the environment; a material change has been made to eliminate this condition.

Westinghouse controls the heat input in all austenitic pressure boundary weldments by:

1. Prohibiting the use of block welding
2. Limiting the maximum interpass temperature to 350°F
3. Exercising approval rights on all welding procedures.

To further assure that these controls are effective in preventing sensitization, Westinghouse will, if necessary, conduct additional intergranular corrosion tests of qualification mockups of primary pressure boundary and core internal component welds, including the following:

1. Reactor Vessel Safe Ends

---

* NOTE: Heat input is calculated according to the formula:

$$H = \frac{(E)(I)(60)}{S}$$

Where:

H = joules/in
E = volts
I = amperes
S = travel speed (in/min)

2. Pressurizer Safe Ends
3. Surge Line and Reactor Coolant Pump Nozzles
4. Control Rod Drive Mechanisms Head Adaptors
5. Control Rod Drive Mechanisms Seal Welds
6. Control Rod Extensions
7. Lower Instrumentation Penetration Tubes.

To summarize, Westinghouse has a four point program designed to prevent intergranular attack of austenitic stainless steel components.

1. Control of primary water chemistry to ensure a benign environment.
2. Utilization of materials in the final heat-treated condition, and the prohibition of subsequent heat treatments in the 800 to 1500°F temperature range.
3. Control of welding processes and procedures to avoid heat-affected zone (HAZ) sensitization.
4. Confirmation that the welding procedure used for the manufacture of components in the primary pressure boundary and of reactor internals do not result in the sensitization of heat-affected zones.

Both operating experience and laboratory experiments in primary water have conclusively demonstrated that this program is 100 percent effective in preventing intergranular attack in Westinghouse NSSS's utilizing unstabilized austenitic stainless steel.

e. Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitization Temperatures

It is not normal Westinghouse practice to expose unstabilized austenitic stainless steels to the sensitization range of 800 to 1500°F during fabrication into components. If, during the course of fabrication, the steel is inadvertently exposed to the sensitization temperature range, 800 to 1500°F, the material may be tested in accordance with ASTM A262, Practice A or E, as amended by Westinghouse Process Specification 84201 MW, to verify that it is not susceptible to intergranular attack, except that testing is not required for:

1. Cast metal or weld metal with a ferrite content of five percent or more

2. Material with carbon content of 0.03 percent or less that is subjected to temperatures in the range of 800 to 1500°F for less than one hour

3. Material exposed to special processing, provided the processing is properly controlled to develop a uniform product and provided that adequate documentation exists of service experience and/or test data, to demonstrate that the processing will not result in increased susceptibility to intergranular stress corrosion.

If it is not verified that such material is not susceptible to intergranular attack, the material will be re-solution annealed and water quenched or rejected.

f. Control of Welding

The following paragraphs address Regulatory Guide 1.31, "Control of Stainless Steel Welding," and present the methods used, and the verification of these methods, for austenitic stainless steel welding.

The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring or hot cracking in the weld. Although published data and experience have not confirmed that fissuring is detrimental to the quality of the weld, it is recognized that such fissuring is undesirable in a general sense. Also, it has been well documented in the technical literature that the presence of delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. However, there is insufficient data to specify a minimum delta ferrite level below which the material will be prone to hot cracking. It is assumed that such a minimum lies somewhere between 0 and 3 percent delta ferrite.

The scope of these controls discussed here encompasses welding processes used to join stainless steel parts in components designed, fabricated or stamped in accordance with ASME Code, Section III, Class 1 and 2, and core support components. Delta ferrite control is appropriate for the above welding requirements, except where no filler metal is used or for other reasons such control is not applicable. These exceptions include electron beam welding, autogenous gas-shielded tungsten arc welding, explosive welding, and welding using fully austenitic welding materials.

The fabrication and installation specifications require welding procedure and welder qualification in accordance with the ASME Code, Section III, and include the delta ferrite determinations for the austenitic stainless steel welding materials that are used for welding qualification testing and for production processing. Specifically, the undiluted weld deposits of the "starting" welding materials are required to contain a minimum of 5 percent delta ferrite (the equivalent ferrite number may be substituted for percent delta ferrite), as determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams. When new welding procedure qualification tests are evaluated for these applications, including repair welding of raw materials, they are performed in accordance with the requirements of Section III and Section XI of the ASME Code.

The results of all the destructive and nondestructive tests are reported in the procedure qualification record, in addition to the information required by ASME Code, Section III.

The "starting" welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of ASME Code, Section III. The austenitic stainless steel welding material conforms to ASME weld metal analysis A-7, (designated A-8 in the 1974 Edition of the ASME Code), Type 308 or 308L for all applications. Bare weld filler metal, including consumable inserts, used in inert gas welding processes conform to ASME SFA-5.9, and are procured to contain not less than 5 percent delta ferrite according to ASME Code, Section III. Weld filler metal materials used in flux shielded welding processes conform to ASME SFA-5.4 or SFA-5.9, and are procured in a wire-flux combination to be capable of providing not less than 5 percent delta ferrite in the deposit according to the ASME Code, Section III. Welding materials are tested using the welding energy inputs to be employed in production welding.

Combinations of approved heat and lots of "starting" welding materials are used for all welding processes. The welding quality assurance program includes identification and control of welding material by lots and heats, as appropriate. All of the weld processing is monitored according to approved inspection programs which include review of "starting" materials, qualification records, and welding parameters. Welding systems are also subject to quality assurance audit including calibration of gages and instruments, identification of "starting" and completed materials, welder and procedure qualifications, availability and use of approved welding and heat treating procedures, and documentary evidence of compliance with materials, welding parameters and inspection requirements. Fabrication and installation welds are inspected using nondestructive examination methods according to the ASME Code, Section III, rules.

To assure the reliability of these controls, Westinghouse has completed a delta ferrite verification program, described in Reference 6, that has been approved as a valid approach to verify the Westinghouse hypothesis and is considered an acceptable alternative for conformance with the NRC Interim Position of Regulatory Guide 1.31. The Regulatory Staff's acceptance letter and topical report evaluation were received on December 30, 1974. The program results, which do support the hypothesis presented in Reference 6, are summarized in Reference 7.

Section 1.8 includes discussions which indicate the degree of conformance of the austenitic stainless steel components of the RCPB with Regulatory Guides 1.34, "Control of Electroslag Properties," and 1.71, "Welder Qualification for Areas of Limited Accessibility."

5.2.4 In-Service Inspection and Testing of Reactor Coolant Pressure Boundary

Components of the reactor coolant pressure boundary are designed, fabricated and erected in such a way as to comply with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code.

5.2.4.1 System Boundary Subject to Inspection

The system boundary (ASME Section III Class 1 components) subject to in-service inspection includes the following:

a. Reactor pressure vessel; including shell, heads, cladding, nozzles, penetrations and supports

b. Steam generators, primary side; including heads, nozzles and supports

c. Reactor coolant pumps; including pump bodies, nozzles and supports

d. Pressurizer vessel; including heads, body shell, nozzles and skirt

e. Reactor coolant piping; including the loop piping, pressurizer surge line, pressurizer spray line, valves and supports. (See Table 5.2-6 for the list of lines and Table 5.2-7 for the list of valves that form the reactor coolant pressure boundary, along with items a through d above.)

5.2.4.2 Accessibility

All components have been arranged to provide the maximum possible clearances for access for in-service inspection, consistent with the design and function of the plant, and in accordance with the requirements of IWA-1500 of Section XI of the Code.

In-service inspection of the reactor vessel is described in Subsections 5.3.1.6 and 5.3.3.7. In-service inspection of the steam generator is described in Subsection 5.4.2.5 and for the pressurizer in Subsection 5.4.10.4. Piping systems requiring volumetric (ultrasonic) in-service inspection are designed to provide access for inspection, and pipe welds are smoothed and contoured to permit effective use of the inspection equipment (see Figure 5.2-1).

Access for in-service inspection of major NSSS equipment, other than the reactor pressure vessel, has been facilitated as follows:

a. Manways have been provided in the steam generator channel head to permit internal inspection of the steam generator.

b. A manway has been provided in the pressurizer top head to allow access for internal inspection of the pressurizer.

c. The insulation covering all inspectable equipment and piping welds is designed for easy removal and replacement to facilitate weld inspection and repair, if necessary, and to provide adequate access for surface and volumetric examination.

d. The floor above the reactor coolant pumps has been provided with removable plugs to permit removal and installation of the pump motor, and to allow internal inspection of the pumps.

e. The reactor coolant loop compartments have been designed to permit personnel access during refueling outages, to permit direct inspection of the external sections of the piping and components.

f. Platforms and ladders have been strategically located to provide access to valves and other components for convenience in operation and maintenance. These platforms and ladders will facilitate in-service inspection.

5.2.4.3 Examination Techniques and Procedures

In general, except for the reactor pressure vessel, in-service inspection is performed using manual techniques wherever possible. In those few cases where plant configuration and radiation level make manual inspection undesirable, automated techniques are used. The baseline inspection uses the same procedures and type of equipment used for in-service inspection to verify the adequacy of procedures and equipment.

The reactor pressure vessel shell is inspected using an automated inspection device, as are the shell-to-nozzle welds and the inner surface of the nozzles. The upper head welds are examined using manual techniques. For details of reactor pressure vessel inspection, refer to Subsection 5.3.3.7 and Chapter 16. The remote inspection equipment is tested and proven adequate prior to its use for baseline inspection.

For details of inspection of the steam generators, refer to Subsection 5.4.2.5.

Laydown areas are provided for storage of removed components and racks; stands are provided to facilitate examination of removed components. Hoists and other handling equipment are provided in strategic locations for use during in-service inspection.

The visual examination employed as a basis for the report on the general condition of the parts, components or surfaces includes such conditions as scratches, wear, cracks, corrosion or erosion of surfaces, misalignment, deformation or leakage.

5.2.4.4 Inspection Intervals

Inspection intervals are ten years, and are in accordance with IWA-2400 of Section XI of the ASME Code, Plan B. See Chapter 16 for details of the in-service inspection plan. All inspections required by IWA-2500 will be completed by the end of each 10-year interval.

It is planned that the required in-service inspection will be accomplished during normal plant maintenance and refueling shutdown periods.

During the first of the inspection intervals, at least 16 percent of the required inspections, and not more than 34 percent, will be completed. By the end of the second of the interval, at least 50 percent and not more than 67 percent of the required inspections will be completed. The remaining inspections will be completed by the end of the 10-year interval.

5.2.4.5 Examination Categories and Requirements

In-service inspection categories and requirements are defined in Section XI of the Code for specific code classes. A detailed description of the means used to meet these requirements is given in Chapter 16. In general, Seabrook categories are identical with IWB-2500 of the Code, and requirements are the same as IWB-2000 of the Code.

5.2.4.6 Evaluation of Examination Results

All records of preservice and subsequent inspections are filed and maintained. Data from the subsequent examinations is compared with the baseline data from the preservice examination, and the comparisons are evaluated in accordance with IWB-3000 of the Code.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

System hydrostatic and leakage tests are performed and the test records are stored and maintained for the service life of the component or system, in accordance with IWA-1400 of Section XI, to provide a current status of each component. If repairs are necessary, a new baseline will be performed for the repair. The system leakage test will normally be conducted concurrently with the system hydrostatic test.

The requirements of IWB-5200 will be met. If test temperatures are required to be above 100°F, then system test pressure reduction will conform to the table in IWB-5200.

5.2.5 Detection of Leakage through Reactor Coolant Pressure Boundary

The leakage detection systems comply with applicable parts of NRC General Design Criterion 30 and Regulatory Guide 1.45. These systems provide a means of detecting, to the extent practical, leakage from the reactor coolant pressure boundary (RCPB).

5.2.5.1 Collection of Identified Leakage

The identified leakage within the Containment that is expected from the components of the RCPB, such as valve stem packing glands, reactor coolant pump shaft seals and other equipment that cannot practically be made completely leak-tight, is piped off to tanks as described below. The leakage rates from these components are monitored by temperature and flow instruments during plant operation. The sources of leakages are identified through the specific instrumentation provided in each line. Refer to Figure 5.2-2 Sheets 1 and 2 for details.

a. Reactor Coolant Drain Tank (RCDT)

The leakage directed to the RCDT includes the following sources:

1. Valve stem leakoffs
2. Reactor coolant pump seal (No. 2) leakage
3. Reactor flange leakoff
4. Excess letdown.

b. Pressurizer Relief Tank (PRT)

The PRT condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from the smaller relief valves located inside the Containment is also piped to the PRT.

5.2.5.2 Unidentified Leakage to Containment

The majority of leakage from sources within the reactor containment is ultimately collected in the containment drainage sumps. Drainage trenches on the floor of the containment channel leakages and condensation to the sump. The leakage rates can be established and monitored during plant operation. Unidentified leakage to the containment atmosphere is kept to a minimum to permit the leakage detection systems to detect positively and rapidly a small increase in the leakage. Identified and unidentified leakages are separated so that a small unidentified leakage will not be masked by a comparatively larger identified leakage.

5.2.5.3 Leakage Detection Methods

The RCPB leakage detection is implemented by using continuous monitoring methods and/or a periodic RCS water inventory balance method. These methods provide a means for detection of both identified and unidentified leakage. Figure 5.2-2 identifies the various monitoring instruments employed for this purpose.

a. Design Bases

The following design bases were established to satisfy the requirements of General Design Criterion 30 for design diversity and redundancy for the RCPB leak detection systems.

1. Leakage to the atmosphere from systems containing radioactive fluid, which would result in an increase in overall containment radioactivity levels, is detected by the use of airborne radioactivity monitors.
2. Indications of an increase in local temperature from any source releasing hot liquid to the atmosphere are provided by air temperature monitors.
3. Temperature monitors are provided to indicate temperature flux vs. flow of leakage in drainage and relief lines and tanks, e.g., reactor vessel flange, pump seal and primary valve leakoffs which discharge to the RCDT and the PRT.
4. Liquid level monitors are provided for drainage sumps and tanks to monitor the leakage.
5. The systems are designed to reliably annunciate increasing leakages. The radiation monitors are provided with failure alarms that will indicate any instrument troubles.

6. The monitors provided shall supply sufficient information to enable deduction of leakage rates, differentiation, identification and general location of leaks.

b. Monitoring System

The Reactor Coolant Leakage Monitoring Systems consist of the following instrumentation.

1. Containment Drainage Sump Liquid Inventory Monitor

As indicated in Subsection 5.2.5.2, leakage is collected in containment drainage sumps. Sump level monitoring is provided to inventory the drainage handled. Level switches are used to maintain the sump level between predetermined levels by cycling the sump pump. Leaks are indicated by the computer log and trend of the sump level and pump operation. Continuous sump level monitoring is available in the main control room.

2. Containment Airborne Radioactivity Monitors

Channel 6526 monitors a sample drawn from the containment atmosphere for particulate and gaseous radioactivity. Iodine monitoring may be done in batch mode by analyzing the charcoal cartridge periodically in the laboratory. A sample which is representative of the containment atmosphere is drawn by an integral pumping system, from containment to a moving paper particulate filter, an iodine cartridge and a noble gas chamber. The air sample is then discharged back to the Containment. One radiation detector is used to monitor the particulate filter and the second radiation detector monitors the noble gas. Also provided is a backup gaseous radiation monitor, 6548, which is located within the Containment at zero feet elevation to monitor containment atmosphere for noble gas. The detectors are of Beta Scintillator type. The detector outputs are converted into microcuries per cubic centimeter by the microprocessor.

3. DELETED

4. Reactor Vessel Flange Leakoff

   The reactor flange and head are sealed by two metallic 0-rings. Leakoff connections are provided between the 0-rings and beyond the outer 0-ring. The leakage is piped to the RCDT. A high temperature measurement by an RTD mounted in the piping indicates the reactor coolant leakage.

5. Reactor Coolant Drain Tank

   The various sources of leakage to the RCDT are identified in Subsection 5.2.5.1a. The RCDT is provided with temperature, pressure and level indications by which leakage is determined.

6. Reactor Coolant Pump Seal Leakoff

   Refer to Subsection 5.4.1.3 for a complete discussion of the reactor coolant pump shaft seal leakage. Seal water enters the pumps through a connection on the thermal barrier flange, and is directed to a plenum between the thermal barrier housing and the shaft. Here the flow splits: a portion flows down the shaft to cool the bearing and enters the RCS; the remainder flows up the shaft through the No. 1 seal, a controlled leakage seal. After passing through the seal, most of the flow leaves the pump via the No. 1 seal leakoff line. Minor flow passes through the No. 2 seal and leakoff line. This flow is indicated by the Main Plant Computer. A back flush injection from a head tank flows into the No. 3 seal between its "double dam" seal area. At this point, the flow divides with half flushing through one side of the seal and out the No. 2 seal leakoff while the remaining half flushes through the other side and out the No. 3 seal leakoff which uses a standpipe. Excessive leakage is piped to containment sump A. A high level of the upper standpipe is alarmed, indicating excess leakage.

7. Pressurizer Relief Tank (PRT)

   The leakages directed to PRT are identified in Subsection 5.2.5.1b. During normal operation, the leakage to the PRT is expected to be negligible, since all the valves are designed to minimize leakage at the normal system operating pressure. Temperature detectors are provided in the discharge piping of each valve to indicate possible leakage. PRT level, temperature and pressure indications and alarms are provided to indicate the leakage.

8. Containment Ambient Temperature Monitors

   Platinum resistance temperature detectors are strategically located throughout the Containment to detect local temperature changes and will assist in localizing a leak.

c. RCS Water Inventory Balance

The periodic RCS water inventory balance is designed to be conducted during steady state conditions with minimal T_{avg} variance. In the course of this inventory, the following parameters are monitored:

1. Time
2. T_{avg}
3. Pressurizer Level
4. VCT Level
5. PRT Level
6. RCDT Level
7. BA Flow Totalizer.

Changes in inventory due to sampling, draining, and steam generator tube leakage are accounted for separately. During the conduct of this inventory, every effort is made to avoid additions to the RCS, pump down of the RCDT, or diversion of letdown from the VCT.

Changes in the parameters are calculated over a convenient time period (the longer the period the more accurate the results). The inventory change rate is determined by summing the volume change associated with each parameter and dividing this value by the time interval. The difference between the containment sump leakage rate and the inventory change rate will indicate leakage from sources other than the primary system.

5.2.5.4 Intersystem Leakage Detection

The following three types of detection methods are employed to monitor systems connected with the RCPB for signs of intersystem leakage:

a. Primary Component Cooling Water System Radiation Monitors

These are gamma sensitive scintillation detectors. Liquid sample is drawn from the discharge side of the primary component cooling water pumps and returned back to the suction side. This system monitors primary component cooling water for radioactivity indicative of a leak from the Reactor Coolant System or from one of the radioactive systems which exchanges heat with the Primary Component Cooling System. These detectors are provided with the relevant flow information to obtain the radioactivity in terms of microcuries per cubic centimeter.

b. Condenser Air Evacuation Monitors

This method is employed for detection of steam generator tube leaks. Noble gases present in the steam generator tube or tube sheet coolant leakage leave solution in the steam generator and are ultimately vented along with other noncondensables. This detector is a gross beta scintillator. The detector is directly mounted in the discharge line of the three air evacuation pumps.

c. Steam Generator Blowdown Sample Monitors

These monitors provide indications of primary to secondary leaks in the steam generators by analyzing the liquid phases of the four steam generator secondary sides. These are of the gamma sensitive scintillation type. Pressures and temperatures are reduced for detection purposes.

5.2.5.5 Sensitivity and Response Time of Detectors

The sensitivity and response time of each leakage detection method provided to comply with the guidance in Regulatory Guide 1.45, position C.5 for detecting unidentified leakage to the Containment is discussed below.

a. Containment Drainage Sump Inventory Monitoring

Normal leakage from all the unidentified sources within the Containment is estimated to be in the range of 20 to 40 gallons per day. RCPB leaks on the order of 1 gpm are very large in comparison and are easily detected by log and trend of containment sump level.

Additionally, the level transmitters have sufficient resolutions to detect change in level due to a flow of as little as 1 gpm. Leakages of the order of 0.05 gpm are detected in a few hours with the expected background leakage. Leakage of 1 gpm can be detected in less than 60 minutes. The drainage sump instrumentation system has a sensitivity of 1 inch and an accuracy of ±5 percent.

b. Containment Air Particulate Monitor

The containment air particulate monitor is one of the most sensitive instruments available for detection of reactor coolant leakage into the Containment. The containment air particulate monitor has a sensitivity of 10^{-10} μCi/cc and an accuracy of ±20 percent. The measuring range is 10^{-10} to 10^{-6} μCi/cm^3. The response time is the shortest where the baseline leakage is low. The baseline airborne activity is kept low by adjusting the valve packings and pump seals properly. The air particulate monitor is capable of detecting leakage of 1 gpm in less than 60 minutes, if the reactor is operating with 0.12 percent fuel defects (reference Subsection 11.1.7.1) and a coolant corrosion product level of 2.3×10^{-2} μCi/gm (reference Table 11.1-1), assuming baseline leakage activity of 1 percent per day of reactor coolant mass. However, during the initial period of operation, and following refueling when the coolant activity is low, response time will be longer than those of other types of monitors which do not rely on coolant activity for detection.

c. Containment Radioactive Gas Monitor

Gaseous activity in the containment atmosphere results from fission gases (various Kr and Xe isotopes) in coolant leakage and from Ar-41 produced by activation of air around the reactor vessel. The regular and backup gas monitors have a sensitivity of 10^{-6} μCi/cc and an accuracy of ± 20 percent. The range of the regular monitor is 10^{-6} to 10^{-2} μCi/cm^3, while the range of the backup monitor is 10^{-6} to 10^{-2} μCi/cc. The gas monitor is able to detect a leakage of 1 gpm in less than 60 minutes if the reactor is operated with 0.12 percent fuel defects (reference Subsection 11.1.7.1).

5.2.5.6 Seismic Capability

The containment airborne radioactivity monitor and containment drainage sump level instrumentation is classified seismic Category I, and satisfies the requirement of NRC Regulatory Guide 1.45, for seismic qualification.

5.2.5.7 Indicators and Alarms

Positive indication of RCPB leakage is provided in the main control room by the instruments located there, in association with the RCPB leakage detection subsystems. All indicators, recorders, annunciators and computer logs are readily available to the main control room operators. The operators are provided with the procedures for interpreting the indications to identify the leakage source, and with criteria for plant operation under leakage conditions.

Continuous sump level monitor is available to the plant computer. The computer monitors the sump level as well as the running of the sump pump in order to determine the leakage. Sump level high and low level alarms are also available.

For PRT and RCDT, temperature and level detectors are provided so that high and low level indications and high temperatures are alarmed. Temperature detectors are provided in the discharge piping of each safety and relief valve, reactor vessel flange leakoff piping, pressurizer vent, and reactor vessel head vent, to indicate possible leakage. Temperature detectors at various areas of containment monitor any high ambient temperature. These detectors provide the capability to indirectly detect RCS leakage and aid in locating the leakage source. Reactor vessel flange inner and outer seal leaks will be identified by a high temperature alarm in the main control room. High flow indication in Seal No. 2 leakoff and high level indication in stand pipe connected to Seal No. 3 will alert the control room operator to reactor coolant pump seal leaks.

For all radiation monitoring systems, local indication of activity and high level alarm are provided locally at the monitor. By means of the RDMS, indication and alarms are provided at the main control room. Equipment failure alarms are also available.

5.2.5.8 Testing

Calibration and functional testing of the leak detection systems, i.e., sump level detection, containment air and particulate monitoring, will be performed prior to initial plant startup.

During normal plant operation, periodic readings of leakage detection system instrumentation will indicate leakage trends. Periodic inspection and calibration of leak detection instruments can be performed during normal operation and will ensure accuracy and dependability.

For all radiation monitors, a primary calibration is performed on a one-time basis, utilizing typical isotopes of interest to determine proper detector response. Secondary standard calibrations are performed with multiple radiation sources to confirm the channel sensitivity obtained on primary calibration. This single point calibration confirms the channel sensitivity. Each monitor has a diagnostic program built into its microprocessor. This program continuously conducts a diagnostic routine within the monitor and provides an alarm input to the RDMS when a failure is detected. Each monitor is equipped with a "check source" that is inserted upon the command of the RDMS. Each time the check source is inserted, the microprocessor measures and stores the effect of the check source and compares it to the previous reading to obtain an indication of calibration trends.

5.2.5.9 Technical Specification

The Technical Specification is provided in Section 3/4.4.6.

5.2.6 Reactor Coolant Vent System

5.2.6.1 Design Basis

The Reactor Coolant Vent System is designed to allow venting the large quantities of noncondensible gases that can be generated within the reactor following core damage. It provides a vent path to the containment atmosphere via the pressurizer relief tank to insure that noncondensible gases cannot accumulate in the core to the point where core cooling would be interrupted and further core damage occur.

The design temperature and pressure is the same as the Reactor Coolant System, i.e., 650°F and 2485 psig. Piping and valve material is stainless steel, Type 316. All material is compatible with the reactor coolant chemistry and will be fabricated and tested in accordance with SRP Subsection 5.2.3, "Reactor Coolant Pressure Boundary Materials."

5.2.6.2 System Description

This system (see Figure 5.1-2 and Figure 5.1-4) provides the capability to vent the Reactor Coolant System from two locations: the reactor vessel head and the pressurizer steam space. The vent valves will be manually operated from the control room. The function of these vents is to vent any noncondensible gases that may collect in the reactor vessel head and in the pressurizer following core damage.

a. Reactor Vessel Head Vent

The reactor vessel head vent consists of a single solenoid valve and a motor-operated valve in series. This vent ties into the reactor head vent line which is normally used to vent the vessel for vessel fill. A flow restricting orifice is provided immediately downstream of the tie-in to the normal vent line. Both valves are powered from the same train emergency power source.

b. Pressurizer Vent

The vent for the pressurizer steam space uses the two parallel, redundant, safety-grade PORVs. The PORVs are 3"x6" pilot-operated solenoid valves with redundant, direct position indication. Each PORV has its own motor-operated isolation valve.

The PORV and its associated isolation valve are both powered from the same emergency power electrical train. However, each PORV and its associated isolation valve are supplied by opposite train emergency power sources.

5.2.6.3 Safety Evaluation

The RCS vessel head vent piping and valves are Safety Class 1 and 2, seismic Category 1 up to, and including, the second isolation valve. A temperature detector is located immediately downstream of the second isolation valve for leakage detection.

The reactor vessel head vent line has two normally closed valves in series; therefore, a single failure which results in an inadvertent opening of one valve does not initiate venting. The line also contains an orifice on the vessel side of the isolation valves that restricts the flow rate from a pipe break downstream of the orifice to within the makeup capacity of the charging system. Therefore, a break in this line, downstream of the orifice, or an inadvertent actuation of the vent during normal operation does not constitute a LOCA, and does not require ECCS actuation. All piping and components downstream of the flow restricting orifice are Safety Class 2, and seismic Category I. The valve, piping and components downstream of the motor-operated valve are classified Non-Nuclear Safety (NNS).

This valve is designed to withstand the safe shutdown earthquake. In addition, there is no piping which might be affected by spray from a postulated break in the NNS portion of the piping (which is routed to the pressurizer relief tank).

The pressurizer vent consists of the normal pressurizer PORVs and their normally open, motor-operated isolation valves. While inadvertent operation of a PORV would result in RCS depressurization, the effects have been analyzed and are bounded by the analysis presented in Chapter 15 and do not represent an unreviewed safety question. All piping and components upstream of, and including, the PORVs are Safety Class 1 and seismic Category I. All other piping and components downstream of the PORVs are classified Non-Nuclear Safety.

All electrical equipment for both the reactor vessel vent and the pressurizer vent is Class 1E. Motive and control power supplies for these valves are also Class 1E. Equipment within the containment atmosphere is environmentally qualified to insure operability in a hostile environment resulting from an accident.

5.2.6.4 Instrumentation Requirements

Vent valve position indication is provided on the main control board. Inadvertent opening of the vent valves is detected by means of temperature detectors in the downstream piping, with alarm indication for the pressurizer vent path at the main control board.

5.2.6.5 Testing and Inspection

The safety class portions of the Reactor Coolant Vent System will receive the same inspection program as outlined for the reactor coolant loop piping, as defined in Updated FSAR Subsection 5.4.14.4.

5.2.6.6 Technical Specifications

The Technical Specifications for the RCS vent system include RCS vent system limiting conditions for operation as well as surveillance requirements.

5.2.7 References

1. WCAP-9292, "Dynamic Fracture Toughness of ASME SA508 Class 2a and ASME SA533 Grade A Class 2 Base and Heat Affected Zone Material and Applicable Weld Metal," March 1978.
2. Cooper, L., Miselis, V. and Starek, R.M., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972 (also letter NS-CE-622, dated April 16, 1975, C. Eicheldinger (Westinghouse) to D.B. Vassallo (NRC), Additional Information on WCAP-7769, Revision 1).
3. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907, October 1972.
4. Letter NS-CE-1730, dated March 17, 1978, C. Eicheldinger (Westinghouse) to J.F. Stolz (NRC).
5. Golik, M.A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7735, August 1971.
6. Enrietto, J.F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1975.
7. Enrietto, J.F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976.
8. Brown, R.J., and Osborne, M.P., "Overpressure Protection Report for Seabrook Nuclear Power Plant Units 1 and 2," March 1981.

5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

5.3.1.1 Material Specifications

Material specifications are in accordance with the ASME Code requirements and are given in Subsection 5.2.3.

5.3.1.2 Special Processes Used For Manufacturing and Fabrication

a. The vessel is Safety Class 1. Design and fabrication of the reactor vessel is carried out in strict accordance with ASME Code, Section III, Class 1, requirements. The head flanges and nozzles are manufactured as forgings. The cylindrical portion of the vessel is made up of several shells, each consisting of formed plates joined by full penetration longitudinal weld seams. The hemispherical heads are made from dished plates. The reactor vessel parts are joined by welding, using the single or multiple wire submerged arc.

b. The use of severely sensitized stainless steel as a pressure boundary material has been prohibited, and has been eliminated by either a select choice of material or by programming the method of assembly.

c. The control rod drive mechanism head adaptor threads and surfaces of the guide studs are chrome-plated to prevent possible galling of the mated parts.

d. At all locations in the reactor vessel where stainless steel and Inconel are joined, the final joining beads are Inconel weld metal in order to prevent cracking.

e. Core region shells fabricated of plate material have longitudinal welds which are angularly located away from the peak neutron exposure experienced in the vessel, where possible.

f. The location of full penetration weld seams in the upper closure head and vessel bottom head are restricted to areas that permit accessibility during in-service inspection.

g. The stainless steel clad surfaces are sampled to assure that composition requirements are met.

h. Minimum preheat requirements have been established for pressure boundary welds using low alloy material. The preheat must be maintained either until (at least) an intermediate post-weld heat treatment is completed, or until the completion of welding. In the latter case, upon completion of welding, a low temperature (400°F minimum) post-weld heat treatment is applied for four hours, then the weldment is allowed to cool to ambient temperature. This practice is specified for all pressure boundary welds except for the installation of nozzles. For the primary nozzle to shell welds, the preheat temperature is maintained until a high temperature (greater than 800°F) post-weld heat treatment is applied in accordance with the requirements of the ASME Code, Section III. This method is followed because higher restraint stresses may be present in the nozzle-to-shell weldments.

i. The procedure qualification for cladding low alloy steel (SA-508, Class 2) requires a special evaluation to assure freedom from underclad cracking.

5.3.1.3 Special Methods for Nondestructive Examination

The nondestructive examination of the reactor vessel and its appurtenances is conducted in accordance with ASME Code, Section III, requirements; also numerous examinations are performed in addition to ASME Code, Section III, requirements. Nondestructive examination of the vessel is discussed in the following paragraphs and the reactor vessel quality assurance program is given in Table 5.3-2.

a. Ultrasonic Examination

1. In addition to the design code straight beam ultrasonic test, angle beam inspection over 100 percent of one major surface of plate material is performed during fabrication to detect discontinuities that may be undetected by the straight beam examination.

2. In addition to the ASME Code, Section III, nondestructive examination, all full penetration ferritic pressure boundary welds in the reactor vessel are ultrasonically examined during fabrication. This test is performed upon completion of the welding and intermediate heat treatment but prior to the final post-weld heat treatment.

3. After hydrotesting, all full penetration ferritic pressure boundary welds in the reactor vessel, as well as the nozzle to safe end welds, are ultrasonically examined. These inspections are also performed in addition to the ASME Code, Section III, nondestructive examinations.

b. Penetrant Examinations

The partial penetration welds for the control rod drive mechanism head adaptors and the bottom instrumentation tubes are inspected by dye penetrant after the root pass, in addition to code requirements. Core support block attachment welds are inspected by dye penetrant after the first layer of weld metal and after each ½ inch of weld metal. All clad surfaces and other vessel and head internal surfaces are inspected by dye penetrant after the hydrostatic test.

c. Magnetic Particle Examination

The magnetic particle examination requirements below are in addition to the magnetic particle examination requirements of Section III of the ASME Code.

All magnetic particle examinations of materials and welds are performed in accordance with the following:

Prior to the final post-weld heat treatment - only by the Prod, Coil or Direct Contact Method.

After the final post-weld heat treatment - only by the Yoke Method.

The following surfaces and welds are examined by magnetic particle methods. The acceptance standards are in accordance with Section III of the ASME Code.

1. Surface Examinations

   (a) Magnetic particle examination of all exterior vessel and head surfaces after the hydrostatic test.

   (b) Magnetic particle examination of all exterior closure stud surfaces and all nut surfaces after threading. Continuous circular and longitudinal magnetization is used.

   (c) Magnetic particle examination of inside diameter surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling. This inspection is performed after forming and machining (if performed) and prior to cladding.

2. Weld Examination

   Magnetic particle examination or ultrasonic examination of the weld metal buildup for vessel supports, welds attaching the closure head lifting lugs and refueling seal ledge to the reactor vessel after the first layer and each ½ inch of weld metal is deposited. All pressure boundary welds are examined after back chipping or back grinding operations.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steel

Welding of ferrite steels and austenitic stainless steels is discussed in Subsection 5.2.3. Subsection 5.2.3 includes discussions which indicate the degree of conformance with Regulatory Guides 1.44, "Control of the Use of Sensitized Stainless Steel" and 1.31, "Control of Stainless Steel Welding." Section 1.8 discusses the degree of conformance with Regulatory Guides 1.34, "Control of Electroslag Weld Properties," 1.43 "Control of Stainless Steel Weld Cladding of Low Alloy Steel Components," 1.50 "Control of Preheat Temperature for Welding of Low Alloy Steels," 1.71 "Welder Qualification for Areas of Limited Accessibility," and 1.99 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

5.3.1.5 Fracture Toughness

Assurance of adequate fracture toughness of ferritic materials in the reactor coolant pressure boundary (ASME Code, Section III, Class 1 components) is provided by compliance with the requirements for fracture toughness testing included in NB-2300 to Section III of the ASME Code and Appendix G of 10 CFR 50.

The initial Charpy V-notch minimum upper shelf fracture energy levels for the reactor vessel beltline (including welds) are 75 foot-pounds, as required per Appendix G of 10 CFR 50. Materials having a section thickness greater than 10 inches with an upper shelf of less than 75 foot-pounds are evaluated with regard to effects of chemistry (especially copper content), initial upper shelf energy and fluence to assure that a 50 foot-pound shelf energy as required by Appendix G of 10 CFR 50 is maintained throughout the life of the vessel. The specimens are oriented as required by NB-2300 of Section III of the ASME Code. Fracture toughness data for the Seabrook Unit 1 reactor vessel materials is presented in Table 5.3-3. Details of the beltline material fracture toughness are given in Table 5.3-5 and Table 5.3-6.

5.3.1.6 Material Surveillance

In the surveillance program, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch, tensile and ½ T (thickness) compact tension (CT) fracture mechanics test specimens. The program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach. The program will conform with ASTM E185-79, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and 10 CFR 50, Appendix H.

The reactor vessel surveillance program uses six specimen capsules. The capsules are located in guide baskets welded to the outside of the neutron shield pads, and are positioned directly opposite the center portion of the core. The capsules can be removed when the vessel head is removed and can be replaced when the internals are removed. The six capsules contain reactor vessel steel specimens, oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting base material located in the core region of the reactor vessel and associated weld metal and weld heat-affected zone metal. The six capsules contain 54 tensile specimens, 360 Charpy V-notch specimens (which include weld metal and weld heat-affected zone material), and 72 CT specimens. Archive material sufficient for two additional capsules will be retained.

Dosimeters, including Ni, Cu, Fe, Co-Al, Cd shielded Co-Al, Cd shielded Np-237, and Cd shielded U-238, are placed in filler blocks drilled to contain them. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low melting point alloys are included to monitor the maximum temperature of the specimens. The specimens ensure good thermal conductivity. The complete capsule is helium leak tested. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent will be made for surveillance material and as-deposited weld metal.

Each of the six capsules contains the following specimens:

Material	Number of Charpys	Number of Tensiles	Number of CTs
Limiting Base Material*	15	3	4
Limiting Base Material**	15	3	4
Weld Metal***	15	3	4
Heat-Affected Zone	15	-	-

The following dosimeters and thermal monitors are included in each of the six capsules:

Dosimeters

Iron

Copper

Nickel

Cobalt-Aluminum (0.15 percent Co)

Cobalt-Aluminum (Cadmium shielded)

U-238 (Cadmium shielded)

Np-237 (Cadmium shielded)

Thermal Monitors

97.5 percent Pb, 2.5 percent Ag (579°F melting point)

97.5 percent Pb, 1.75 percent Ag, 0.75 percent Sn (590°F melting point)

---

* Specimens oriented in the major rolling or working direction.

** Specimens oriented normal to the major rolling or working direction.

*** Weld metal to be selected per ASTM E185.

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the vessel wall, with the specimens being located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition temperature shift measurements are representative of the vessel at a later time in life. Data from CT fracture toughness specimens are expected to provide additional information for use in determining allowable stresses for irradiated material.

Correlations between the calculations and the measurements of the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Subsection 5.3.1.6a. They have indicated good agreement. The anticipated degree to which the specimens will affect the fast neutron flux and energy distribution will be considered in the evaluation of the surveillance specimen data. In accordance with Regulatory Guide RG-1.190 (Reference 1), validation of the calculated neutron exposures will be made using data from the withdrawn capsules. The schedule for removal of the capsules for post-irradiation testing will conform with ASTM E185-79 and Appendix H of 10 CFR 50.

a. Measurement of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

The use of passive neutron sensors such as those included in the internal surveillance capsule dosimetry sets does not yield a direct measure of the energy dependent neutron flux level at the measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average flux level, and hence, time integrated exposure (fluence) experienced by the sensors may be developed from the measurements only if the sensor characteristics and the parameters of the irradiation are well known. In particular, the following variables are of interest:

1) the measured specific activity of each sensor;
2) the physical characteristics of each sensor;
3) the operating history of the reactor;
4) the energy response of each sensor; and
5) the neutron energy spectrum at the sensor location.

In this subsection, the procedures used to determine sensor specific activities, to develop reaction rates for individual sensors from the measured specific activities and the operating history of the reactor, and to derive key fast neutron exposure parameters from the measured reaction rates are described.

Determination of Sensor Reaction Rates

The specific activity of each of the radiometric sensors is determined using established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor is determined by means of a high purity germanium gamma spectrometer. In the case of the surveillance capsule multiple foil sensor sets, these analyses are performed by direct counting of each of the individual wires; or, as in the case of U-238 and Np-237 fission monitors, by direct counting preceded by dissolution and chemical separation of cesium from the sensor.

The irradiation history of the reactor over its operating lifetime is determined from plant power generation records. In particular, operating data is extracted on a monthly basis from the reactor startup to the end of the capsule irradiation period. For the sensor sets utilized in the surveillance capsule irradiations, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation are determined

$$R = \frac{A}{N_0 F Y \Sigma_j \frac{P_j}{P_{ref}} C_j \left[1 - e^{-\lambda t_j}\right] e^{-\lambda t_d}}$$

from the following equation:

where:

A = measured specific activity provided in terms of disintegrations per second per gram of target material (dps/gm).

R = reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} expressed in terms of reactions per second per nucleus of target isotope (rps/nucleus).

N_0 = number of target element atoms per gram of sensor.

F = weight fraction of the target isotope in the sensor material.

Y = number of product atoms produced per reaction.

P_j = average core power level during irradiation period j (MW).

P_{ref} = maximum or reference core power level of the reactor (MW).

C_j = calculated ratio of $\phi(E > 1.0 \text{ MeV})$ during irradiation period j to the time weighted average $\phi(E = 1.0 \text{ MeV})$ over the entire irradiation period.

λ = decay constant of the product isotope (sec^{-1}).

t_j = length of irradiation period j (sec).

t_d = decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the total irradiation period.

In the above equation, the ratio P_j/P_{ref} accounts for month-by-month variation of power level within a given fuel cycle. The ratio C_j is calculated for each fuel cycle and accounts for the change in sensor reaction rates caused by variations in flux level due to changes in core power spatial distributions from fuel cycle to fuel cycle. Since the neutron flux at the measurement locations within the surveillance capsules is dominated by neutrons produced in the peripheral fuel assemblies, the change in the relative power in these assemblies from fuel cycle to fuel cycle can have a significant impact on the activation of neutron sensors. For a single-cycle irradiation, $C_j = 1.0$. However, for multiple-cycle irradiations, particularly those employing low leakage fuel management, the additional C_j correction must be utilized in order to provide accurate determinations of the decay corrected reaction rates for the dosimeter sets contained in the surveillance capsules.

Corrections to Reaction Rate Data

Prior to using measured reaction rates in the least squares adjustment procedure discussed ahead, additional corrections are made to the U-238 measurements to account for the presence of U-235 impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation.

In addition to the corrections made for the presence of U-235 in the U-238 fission sensors, corrections are also made to both the U-238 and Np-237 sensor reaction rates to account for gamma ray induced fission reactions occurring over the course of the irradiation.

Least Squares Adjustment Procedure

Least squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a Best Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as neutron fluence (E > 1.0 MeV) or iron atom displacements (dpa) along with their uncertainties are then easily obtained from the adjusted spectrum. The use of measurements in combination with the analytical results reduces the uncertainty in the calculated spectrum and acts to remove biases that may be present in the analytical technique.

In general, the least squares methods, as applied to pressure vessel fluence evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross-section, σ_{ig}, each with an uncertainty δ.

The use of least squares adjustment methods in LWR dosimetry evaluations is not new. The American Society for Testing and Materials (ASTM) has addressed the use of adjustment codes in ASTM Standard E944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance" and many industry workshops have been held to discuss the various applications. For example, the ASTM-EURATOM Symposia on Reactor Dosimetry holds workshops on neutron spectrum unfolding and adjustment techniques at each of its bi-annual conferences.

The primary objective of the least squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement. The analytical method alone may be deficient because it inherently contains uncertainty due to the input assumptions to the calculation. Typically these assumptions include parameters such as the temperature of the water in the peripheral fuel assemblies, by-pass region, and downcomer regions, component dimensions, and peripheral core source. Industry consensus indicates that the use of calculation alone results in overall uncertainties in the neutron exposure parameters in the range of 15-20% (1σ).

The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.
2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.
3. The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For a given application, the calculated neutron spectrum is obtained from the results of plant specific neutron transport calculations applicable to the irradiation period experienced by the dosimetry sensor set. The calculation is performed using the benchmarked transport calculational methodology described in Subsection 5.3.1.6b. The sensor reaction rates are derived from the measured specific activities obtained from the counting laboratory using the specific irradiation history of the sensor set to perform the radioactive decay corrections. The dosimetry reaction cross-sections and uncertainties that are utilized in LWR evaluations comply with ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)."

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum are input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties also follows the guidance provided in ASTM Standard E 944.

b. Calculation of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

A generalized set of guidelines for performing fast neutron exposure calculations within the reactor configuration, and procedures for analyzing measured irradiation sample data that can be correlated to these calculations, has been promulgated by the Nuclear Regulatory Commission (NRC) in Regulatory Guide RG-1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Reference 1]. Since different calculational models exist and are continuously evolving along with the associated model inputs, e.g., cross-section data, it is worthwhile summarizing the key models, inputs, and procedures that the NRC staff finds acceptable for use in determining fast neutron exposures within the reactor geometry. This material is highlighted below.

Calculation and Dosimetry Measurement Procedures

The selection of a particular geometric model, the corresponding input data, and the overall methodology used to determine fast neutron exposures within the reactor geometry are based on the needs for accurately determining a solution to the problem that must be solved and the data/resources that are currently available to accomplish this task. Based on these constraints, engineering judgment is applied to each problem based on an analyst's thorough understanding of the problem, detailed knowledge of the plant, and due consideration to the strengths and weaknesses associated with a given calculational model and/or methodology. Based on these conditions, Regulatory Guide RG-1.190 does not recommend using a singular calculational technique to determine fast neutron exposures. Instead, RG 1.190 suggests that one of the following neutron transport tools be used to perform this work.

- Discrete Ordinates Transport Calculations
  - a) Adjoint calculations benchmarked to a reference-forward calculation, or stand-alone forward calculations.
  - b) Various geometrical models utilized with suitable mesh spacing in order to accurately represent the spatial distribution of the material compositions and source.
  - c) In performing discrete ordinates calculations, RG 1.190 also suggests that a P_3 angular decomposition of the scattering cross-sections be used, as a minimum.
  - d) RG 1.190 also recommends that discrete ordinates calculations utilize S_8 angular quadrature, as a minimum.
  - e) RG 1.190 indicates that the latest version of the Evaluated Nuclear Data File, or ENDF/B, should be used for determining the nuclear cross-sections; however, cross-sections based on earlier or equivalent nuclear data sets that have been thoroughly benchmarked are also acceptable.
- Monte Carlo Transport Calculations

A complete description of the Westinghouse pressure vessel neutron fluence methodology, which is based on the discrete ordinates transport calculations, is provided in Reference 2. The Westinghouse methodology adheres to the guidelines set forth in Regulatory Guide RG-1.190.

Plant-Specific Calculations

The most recent fast (E > 1.0 MeV) neutron fluence evaluation for the Seabrook reactor pressure vessel was based on a 2D/1D synthesis of neutron fluxes that were obtained from a series of plant- and cycle-specific forward discrete ordinates transport calculations run in R-θ, R-Z, and R geometric models. The set of calculations, which assessed dosimetry as part of the reactor vessel surveillance program and pressure vessel neutron fluences, were conducted in accordance with the guidelines that are specified in Regulatory Guide RG –1.190.

5.3.1.6.1 Results of Evaluation of Irradiated Capsules

The first surveillance capsule U(58.5°) was removed from the Seabrook Unit 1 reactor vessel in August 1991 after 0.913 effective full power years (EFPYs) of reactor operation.

The second surveillance capsule Y(241°) was removed in May 1997 from the Seabrook Unit 1 reactor vessel after 5.572 effective full power years of reactor operation. The results of the Capsules U and Y analyses are contained in References 4 and 5 respectively.

The third surveillance capsule V(61°) was removed in April 2005 from the Seabrook Unit 1 reactor vessel after 12.39 EFPYs of reactor operation. The results of the Capsule V analysis are contained in WCAP-16526-NP (Reference 6) which was submitted to the NRC as required by 10 CFR 50 Appendix H. This report summarizes results from all three capsules (U, Y, and V) and the initial unirradiated mechanical tests for comparison.

The reactor vessel lower shell plate material (R1808-3) and beltline weld (Heat No. 4P6052) were included in the surveillance capsules as the limiting beltline materials in all three capsules. The radiation induced transition temperature shifts (delta RTNDT) for the limiting plate and weld materials, from all three capsules, were within the standard two deviations of Regulatory Guide 1.99, Revision 2 predictions. The irradiated upper shelf energy values for the vessel weld metal and base materials samples were well in excess of the 50 ft.-lb. lower limit for continued safe operation and are expected to be maintained above 50 ft.-lbs. throughout vessel life as required by 10 CFR 50 Appendix G.

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure studs, nuts, and washers are designed and fabricated in accordance with the requirements of the ASME Code, Section III. The closure studs are fabricated of SA-540, Class 3, Grade B24. The closure stud material meets the fracture toughness requirements of the ASME Code, Section III and 10 CFR 50, Appendix G. Compliance with Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," is discussed in Section 1.8. Nondestructive examinations are performed in accordance with the ASME Code, Section III. Fracture toughness data for the Seabrook Unit 1 reactor vessel bolting materials is presented in Table 5.3-4.

Seabrook refueling procedures require the studs, nuts and washers to be removed from the reactor cavity, therefore the reactor closure studs are never exposed to the borated refueling cavity water. Threaded portions of the studs have been treated with an anti galling coating.

The stud holes in the reactor flange are sealed with special plugs before removing the reactor closure thus preventing leakage of the borated refueling water into the stud holes.

5.3.2 Pressure-Temperature Limits

5.3.2.1 Limit Curves

Startup and shutdown operating limitations will be based on the properties of the core region materials of the reactor pressure vessel. Actual material property test data will be used. The methods outlined in Appendix G to Section III of the ASME Code will be employed for the shell regions in the analysis of protection against nonductile failure. The initial operating curves are calculated assuming a period of reactor operation such that the beltline material will be limiting. The heatup and cooldown curves are given in the Technical Specifications. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil-ductility temperature shift (ΔRT_{NDT}).

Predicted RT_{NDT} values are derived using the maximum fluence at ¼T (thickness) and ¾T location (tips of the code reference flaw when flaw is assumed at inside diameter and outside diameter locations, respectively) curve. This curve is presented in the Technical Specifications. For a selected time of operation, this shift is assigned a sufficient magnitude so that no unirradiated ferritic materials in other components of the Reactor Coolant System (RCS) will be limiting in the analysis.

The operating curves including pressure-temperature limitations are calculated in accordance with 10 CFR 50, Appendix G and ASME Code, Section III, Appendix G, requirements. Changes in fracture toughness of the core region plates or forgings, weldments and associated heat-affected zones due to radiation damage will be monitored by a surveillance program which conforms with ASTM E185-79,

10 CFR 50, Appendix H. The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and ½T compact tension specimens. The post-irradiation testing will be carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core midheight and removable from the vessel at specified intervals.

The results of the radiation surveillance program will be used to verify that the RT_{NDT} predicted from the effects of the fluence and copper content curve is appropriate and to make any changes necessary to correct the fluence and copper curves if RT_{NDT} determined from the surveillance program is greater than the predicted RT_{NDT}. Temperature limits for preservice hydrotests and in-service leak and hydrotests will be calculated in accordance with 10 CFR 50, Appendix G.

Compliance with Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," is discussed in Section 1.8.

5.3.2.2 Operating Procedures

The transient conditions that are considered in the design of the reactor vessel are presented in Subsection 3.9(N).1.1. These transients are representative of the operating conditions that should prudently be considered to occur during plant operation. The transients selected form a conservative basis for evaluation of the RCS to insure the integrity of the RCS equipment.

Those transients listed as upset condition transients are given in Table 3.9(N)-1. None of these transients will result in pressure-temperature changes which exceed the heatup and cooldown limitations as described in Subsection 5.3.3.1 and in the Technical Specifications.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Design

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable, bolted, flanged and gasketed, hemispherical upper head. The reactor vessel flange and head are sealed by two hollow metallic 0-rings. Seal leakage is detected by means of two leakoff connections: one between the inner and outer ring and one outside the outer 0-ring. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains head adaptors. These head adaptors are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adaptors contain acme threads for the assembly of control rod drive mechanisms or instrumentation adaptors. The seal arrangement at the upper end of these adaptors consists of a welded flexible canopy seal. The threaded connection forms the pressure boundary. The canopy seals are not pressure retaining and are designed to provide a means to control leaks through the threads. If leakage develops through a canopy seal weld, either a weld repair or a canopy seal clamp assembly can be used to repair or prevent the leak. Inlet and outlet nozzles are located symmetrically around the vessel. Outlet nozzles are arranged on the vessel to facilitate optimum layout of the RCS equipment. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear incore instrumentation. Each nozzle consists of a tubular member made of either an Inconel or an Inconel-stainless steel composite tube. Each tube is attached to the inside of the bottom head by a partial penetration weld.

Internal surfaces of the vessel which are in contact with primary coolant are weld overlay with 0.125 inch minimum of stainless steel or Inconel. The exterior of the reactor vessel is insulated with canned stainless steel reflective sheets. The insulation is a minimum of 3 and a maximum of 4½ inches thick and contoured to enclose the top, sides and bottom of the vessel. Top, bottom and nozzle sections of the insulation modules are removable. Access to vessel side insulation is limited by the surrounding concrete.

The reactor vessel is designed and fabricated in accordance with the requirements of the ASME Code, Section III.

Principal design parameters of the reactor vessel are given in Table 5.3-1. The reactor vessel is shown in Figure 5.3-1.

There are no special design features which would prohibit the in situ annealing of the vessel. If the unlikely need for an annealing operation was required to restore the properties of the vessel material opposite the reactor core because of neutron irradiation damage, a metal temperature greater than 650°F for a period of 168 hours maximum would be applied. Various modes of heating may be used depending on the temperature.

The reactor vessel materials surveillance program is adequate to accommodate the annealing of the reactor vessel. Sufficient specimens are available to evaluate the effects of the annealing treatment.

Cyclic loads are introduced by normal power changes, reactor trip startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the projected plant life. Vessel analysis results in a usage factor that is less than 1.

The design specifications require analysis to prove that the vessel is in compliance with the fatigue and stress limits of the ASME Code, Section III. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. Per the reactor vessel design specification and subsequently documented in the qualification report, the vessel is capable of tolerating a heatup **and** cooldown rate of less **than** or equal to 100°F/hr during normal plant conditions. The heatup and cooldown rates may be further operationally limited by the governing pressure-temperature limit curves of Technical Specification 3.4.9.1

5.3.3.2 Materials of Construction

The materials used in the fabrication of the reactor vessel are discussed in Subsection 5.2.3.

5.3.3.3 Fabrication Methods

The Seabrook reactor vessel manufacturer is Combustion Engineering Corporation.

The fabrication methods used in the construction of the reactor vessel are discussed in Subsection 5.3.1.2.

5.3.3.4 Inspection Requirements

The nondestructive examinations performed on the reactor vessel are described in Subsection 5.3.1.3.

5.3.3.5 Shipment and Installation

The reactor vessel was shipped in a horizontal position on a shipping sled with a vessel lifting truss assembly. All vessel openings were sealed to prevent the entrance of moisture and an adequate quantity of desiccant bags was placed inside the vessel. These were usually placed in a wire mesh basket attached to the vessel cover. All carbon steel surfaces were painted with a heat-resistant paint before shipment except for the vessel support surfaces and the top surface of the external seal ring.

The closure head was also shipped with a shipping cover and skid. An enclosure attached to the ventilation shroud support ring protected the control rod mechanism housings. All head openings were sealed to prevent the entrance of moisture and an adequate quantity of desiccant bags was placed inside the head. These were placed in a wire mesh basket attached to the head cover. All carbon steel surfaces were painted with heat-resistant paint before shipment. A lifting frame was provided for handling the vessel head.

For a discussion of the degree of conformance with Regulatory Guides 1.37, 1.38 and 1.39, see Section 1.8.

5.3.3.6 Operating Conditions

Operating limitations for the reactor vessel are presented in Subsection 5.3.2, as well as in the Technical Specifications.

In addition to the analysis of primary components discussed in Subsection 3.9(N).1.4, the reactor vessel is further qualified to ensure against unstable crack growth under faulted conditions. Actuation of the Emergency Core Cooling System (ECCS) following a loss-of-coolant accident produces relatively high thermal stresses in regions of the reactor vessel, which come into contact with ECCS water. Primary consideration is given to these areas, including the reactor vessel beltline region and the reactor vessel primary coolant nozzle, to ensure the integrity of the reactor vessel under this severe postulated transient.

The principles and procedures of linear elastic fracture mechanics (LEFM) are used to evaluate thermal effects in the regions of interest. The LEFM approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption employed in LEFM is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the size of defect necessary to cause failure.

The elastic stress field at the crack tip in any cracked body can be described by a single parameter designated as the stress intensity factor, K. The magnitude of the stress intensity factor, K, is a function of the geometry of the body containing the crack, the size and location of the crack, and the magnitude and distribution of the stress.

The criterion for failure in the presence of a crack is that failure will occur whenever the stress intensity factor exceeds some critical value. For the opening mode of loading (stresses perpendicular to the major plane of the crack) the stress intensity factor is designated as K_I and the critical stress intensity factor is designated K_{IC}. Commonly called the fracture toughness, K_{IC} is an inherent material property which is a function of temperature and strain rate. Any combination of applied load, structural configuration, crack geometry and size which yields a stress intensity factor, K_{IC}, for the material will result in crack instability.

The criterion of the applicability of LEFM is based on plasticity considerations at the postulated crack tip. Strict applicability (as defined by ASTM) of LEFM to large structures where plane strain conditions prevail requires that the plastic zone developed at the tip of the crack does not exceed 2.25 percent of the crack depth. In the present analysis, the plastic zone at the tip of the postulated crack can reach 20 percent of the crack depth. However, LEFM has been successfully used to provide conservative brittle fracture prevention evaluations, even in cases where strict applicability of the theory is not permitted due to excessive plasticity. Recently, experimental results from Heavy Section Steel Technology (HSST) Program intermediate pressure vessel tests, have shown that LEFM can be applied conservatively as long as the pressure component of the stress does not exceed the yield strength of the material. The addition of the thermal stresses, calculated elastically, which results in total stresses in excess of the yield strength does not affect the conservatism of the results, provided that these thermal stresses are included in the evaluation of the stress intensity factors. Therefore, for faulted condition analyses, LEFM is considered applicable for the evaluation of the vessel inlet nozzle and beltline region.

In addition, it has been well established that the crack propagation of existing flaws in a structure subjected to cyclic loading can be defined in terms of fracture mechanics parameters. Thus, the principles of LEFM are also applicable to fatigue growth of a postulated flaw at the vessel inlet nozzle and beltline region.

An example of a faulted condition evaluation carried out according to the procedure discussed above is given in Reference 3. This report discussed the evaluation procedure in detail as applied to a severe faulted condition (a postulated loss-of-coolant accident), and concludes that the integrity of the reactor coolant pressure boundary would be maintained in the event of such an accident.

5.3.3.7 In-Service Surveillance

a. The reactor vessel will be examined during refueling per the requirements of ASME Section XI by means of a special automated inspection fixture that allows ultrasonic inspection of the vessel welds, the nozzle-to-shell welds and the nozzle inside surface areas.

b. The vessel lower head will also be ultrasonically examined per the requirements of ASME Section XI from the outside surface, since the lower head insulation has been designed to be easily removed.

c. With the bottom head insulation removed, the penetrations for the incore instrumentation will be examined visually.

d. The vessel cladding will be visually examined during refueling. The core barrel has been designed to be removed easily to permit examination.

e. The vessel closure head will be examined on both inner and outer surfaces during refueling per ASME Section XI requirements, since it is supported on a special fixture on the operating floor. The cladding, gasket seating surfaces and control rod drive mechanism housings will be examined by optical devices. The closure head cladding will be examined by volumetric or surface methods while on the support fixture.

f. The closure head flange to knuckle transition piece weld and transition piece to dome weld will be examined on the outer surface by volumetric examination methods per ASME Section XI requirements.

g. Clearances have been provided for volumetric and surface examination of the vessel nozzle to safe-end welds per ASME Section XI requirements, and the insulation has been designed for easy removal for this purpose.

h. The vessel closure studs, nuts and washers will be examined in situ or when removed, in accordance with the requirements of IWB-2500 and IWB-2600 of Section XI of the ASME Code.

i. The reactor vessel presents access problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for the periodic nondestructive tests which are required by the ASME in-service code. These are:

1. Shop ultrasonic examinations were performed on all internally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond and to allow later ultrasonic testing of the base metal from the inside surface. The size of cladding bond defect allowed was ¼ inch by ¾ inch, with the greater dimension parallel to the weld in the region bounded by 2T (T = wall thickness) on both sides of each full penetration pressure boundary weld. Unbounded areas exceeding 0.442 square inches (¾ inch diameter) in all other regions were rejected.

2. The design of the reactor vessel shell is an uncluttered cylindrical surface to permit future positioning of the test equipment without obstruction.

3. The weld-deposited clad surfaces on both sides of the welds to be inspected were specifically prepared to assure meaningful ultrasonic examinations.

4. During fabrication, all full penetration ferritic pressure boundary welds were ultrasonically examined in addition to code examinations.

5. After the shop hydrostatic testing, all full penetration ferritic pressure boundary welds, as well as the nozzle to safe-end welds, were ultrasonically examined in addition to ASME Code, Section III, requirements (see Subsection 5.3.1.3).

j. The in-service inspection program defined in Chapter 16 and discussed in Subsection 5.2.4 meets all the requirements of Section XI of the ASME Code with no exceptions. This comprehensive program of in-service inspection, when combined with the augmented material surveillance described above, should provide for early detection and timely monitoring of any flaws in either the base material, the cladding or the welds in the reactor vessel which could even remotely be considered to threaten reactor vessel integrity. When the conservative design of the reactor vessel is considered (see Subsection 5.3.3.1), these material surveillance and in-service inspection programs should guarantee that reactor pressure vessel boundary integrity will be maintained for the life of the plant.

5.3.4 References

1. Regulatory Guide RG-1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

2. WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology," S.L. Anderson, August 2000.

3. Buchalet, C. and Mager, T. R., "A Summary Analysis of the April 30 Incident at the San Onofre Nuclear Generating Station Unit 1," WCAP-8099, April 1973.

4. YAEC-1853, "Analysis of Seabrook Station Unit 1 Reactor Vessel Surveillance Capsule U", June 26, 1992 (FP 57955).

5. DES-NFQA-98-01, "Analysis of Seabrook Station Unit 1 Reactor Vessel Surveillance Capsules U and Y", May 1998 (FP 58503).

6. WCAP-16526-NP, "Analysis of Capsule V from the Florida Power and Light Energy Seabrook Unit 1 Reactor Vessel Radiation Surveillance Program", Rev. 0, March 2006 (FP 25626).

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Reactor Coolant Pump Assembly

5.4.1.1 Design Bases

The reactor coolant pump assembly ensures an adequate core cooling flow rate for sufficient heat transfer to maintain a Departure from Nucleate Boiling Ratio (DNBR) greater than or equal to the safety analysis limit value. The required net positive suction head is by conservative pump design always less than that available by system design and operation.

Sufficient pumping rotation inertia is provided by a flywheel, in conjunction with the pump and motor rotor assembly, to provide adequate flow during coastdown. This forces flow following an assumed loss of pump power and the subsequent natural circulation effect provides the core with adequate cooling.

The reactor coolant pump motor is tested, without mechanical damage, at overspeeds up to and including 125 percent of normal speed. The integrity of the flywheel during a loss-of-coolant accident (LOCA) is demonstrated in Reference 1, which is undergoing generic review by the NRC.

The reactor coolant pump is shown in Figure 5.4-1. The reactor coolant pump design parameters are given in Table 5.4-1.

Code and material requirements are provided in Section 5.2.

5.4.1.2 Pump Assembly Description

a. Design Description

The reactor coolant pump is a vertical, single stage, controlled leakage, centrifugal pump designed to pump large volumes of water at high temperatures and pressures.

The pump assembly consists of three major areas. They are the hydraulics, the seals, and the motor.

1. The hydraulic section consists of the casing, impeller, turning vane diffuser, and diffuser adapter.

2. The seal section consists of three seals arranged in series. These seals are contained within the seal housing. The first is a controlled leakage, film-riding seal; the second and third are rubbing face seals. The seal system provides a pressure reduction from the Reactor Coolant System (RCS) pressure to ambient conditions.

3. The motor is a drip-proof, water/air cooled, squirrel cage induction motor, with a vertical solid shaft, an oil-lubricated double-acting Kingsbury type thrust bearing, upper and lower oil-lubricated radial guide bearings, and a flywheel.

Additional components of the pump are the shaft, pump radial bearing, thermal barrier heat exchanger, coupling, spool piece, and motor stand.

b. Description of Operation

The reactor coolant enters the suction nozzle, is pumped through the turning vane diffuser, and exits through the discharge nozzle. The diffuser adapter limits the leakage of reactor coolant back to the suction.

Seal injection flow, under slightly higher pressure than the reactor coolant discharge pressure, enters the pump through a connection on the thermal barrier flange, and is directed into the plenum between the thermal barrier housing and the shaft. The flow splits with a portion flowing down the shaft through the radial bearing and into the RCS; the remainder flows up the shaft through the seals.

Component cooling water is provided to the thermal barrier heat exchanger. During normal operation, seal injection flow provides cooling to the radial bearing and to the seals.

The reactor coolant pump motor bearings are of conventional design. The radial bearings are the segmented pad type, and the thrust bearing is a double-acting Kingsbury type. All are oil-lubricated. Component cooling water is supplied to the external upper bearing oil cooler and to the integral lower bearing oil cooler.

The motor is a water/air cooled, Class B thermalastic epoxy insulated, squirrel cage induction motor. The rotor and stator are of standard construction and are cooled by air. Six resistance temperature detectors are imbedded in the stator windings to sense stator temperature. The top of the motor consists of a flywheel and an anti-reverse rotation device.

The internal parts of the motor are cooled by air. Integral vanes on each end of the rotor draw air in through cooling slots in the motor frame. This air passes through the motor with particular emphasis on the stator end turns. It is then routed to the external water/air heat exchangers, which are supplied with component cooling water. Each motor has two such coolers, mounted diametrically opposed to each other. In passing through the coolers, the air is cooled and then directed back to the motor air inlets through external ducts on the motor so that no air is discharged into the Containment from the motors.

Each of the reactor coolant pump assemblies is equipped for continuous monitoring of reactor coolant pump shaft and frame vibration levels. Shaft vibration is measured by two relative shaft probes mounted on top of the pump seal housing; the probes are located 90 degrees apart in the same horizontal plane and mounted near the pump shaft. Frame vibration is measured by two velocity seismoprobes located 90 degrees apart in the same horizontal plane and mounted at the top of the motor support stand. Proximeters and converters linearize the probe output which is displayed and alarmed in the Control Room.

A removable shaft segment, the spool piece, is located between the motor coupling flange and the pump coupling flange; the spool piece allows removal of the pump seals with the motor in place. The pump internals, motor, and motor stand can be removed from the casing without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

Each reactor coolant pump is equipped with an interlock which prevents pump start under conditions of (1) inadequate reactor coolant pump oil lift system pressure, or (2) inadequate reactor coolant pump number 1 seal differential pressure. This interlock prevents pump start under conditions which may result in pump motor bearing or seal system damage.

All parts of the pump in contact with the reactor coolant are austenitic stainless steel except for seals, bearings and special parts.

5.4.1.3 Design Evaluation

a. Pump Performance

The reactor coolant pumps are sized to deliver flow at rates which equal or exceed the required flow rates. Initial RCS tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial plant operation.

The estimated performance characteristic is shown in Figure 5.4-2. The "knee" at about 45 percent design flow introduces no operational restrictions, since the pumps operate at full flow.

The Reactor Trip System ensures that pump operation is within the assumptions used for loss-of-coolant flow analyses, which also assures that adequate core cooling is provided to permit an orderly reduction in power if flow from a reactor coolant pump is lost during operation.

An extensive test program has been conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long-term tests are conducted on less than full-scale prototype seals as well as on full-size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled leakage shaft seal pump design.

The support of the stationary member of the Number 1 seal ("seal ring") allows large deflections, both axial and tilting, while still maintaining its controlled gap relative to the seal runner. Even if all the graphite were removed from the pump bearings, the shaft could not deflect far enough to cause opening of the controlled leakage gap. The "spring-rate" of the hydraulic forces associated with the maintenance of the gap is high enough to ensure that the ring follows the runner under very rapid shaft deflections.

Testing of pumps with the Number 1 seal entirely bypassed (full system pressure on the Number 2 seal) shows that relatively small leakage rates would be maintained for a period of time which is sufficient to secure the pump even if the Number 1 seal fails entirely during normal operation; the Number 2 seal would maintain these small leakage rates if the proper action is taken by the operator. The plant operator is warned of Number 1 seal damage by the increase in Number 1 seal leakoff rate. Should an excessive seal leakage condition develop, the operator will secure the RCP in accordance with station procedures. Gross leakage from the pump does not occur if the proper operator action is taken subsequent to warning of excessive seal conditions.

The effect of loss of offsite power on the pump itself is to cause a temporary stoppage in the supply of injection flow to the pump seals and also of the cooling water for seal and bearing cooling. The emergency diesel generators are started automatically due to loss of offsite power so that component cooling flow is automatically restored; seal injection flow is subsequently restored.

An evaluation was performed to determine the effects of loss of seal injection to the RCPs as a result of operator actions to mitigate an inadvertent ECCS initiation at power event. The evaluation concluded that seal temperature will be maintained below the RCP shutdown limit of 220°F at least one hour for RCP No. 1 seal leak off flow rates between 1 gpm and 6 gpm during normal operation. There is negligible change in seal flow rate due to isolation of the seal leak-off line, and subsequent lifting of the relief valve following an inadvertent ECCS initiation at power event.

An evaluation of an extended loss of seal injection has determined that the RCP may have to be secured because the seal and bearing temperatures may reach their maximum operating limits. This evaluation assumed an extended loss of seal injection, an unlikely event, with low No. 1 seal leak off flow, and the Thermal Barrier heat exchanger cooling water at maximum design temperature and at minimum flow. The operator is warned of the loss of seal injection flow by alarm. He can then monitor bearing and seal temperatures. Should RCP bearing and/or seal temperatures approach operating limits, the operator will secure the RCP in accordance with station procedures.

b. Coastdown Capability

It is important to reactor protection that the reactor coolant continues to flow for a short time after reactor trip. In order to provide this flow in a station blackout condition, each reactor coolant pump is provided with a flywheel. Thus, the rotating inertia of the pump, motor and flywheel is employed during the coastdown period to continue the reactor coolant flow. The coastdown flow transients are provided in the figures in Section 15.3. The pump assembly is designed for the Safe Shutdown Earthquake at the site. Hence, it is concluded that the coastdown capability of the pumps is maintained even under the most adverse case of a blackout coincident with the Safe Shutdown Earthquake. Core flow transients and figures are provided in Section 15.3.

c. Bearing Integrity

The design requirements for the reactor coolant pump bearings are primarily aimed at ensuring a long life with negligible wear, provide accurate alignment and smooth operation over long periods of time. The surface bearing stresses are held at a very low value, and even under the most severe seismic transients do not begin to approach loads which cannot be adequately carried for short periods of time.

Because there are no established criteria for short time stress-related failures in such bearings, it is not possible to make a meaningful quantification of such parameters as margins to failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gives assurance of the adequacy of the bearing to operate without failure.

Low oil levels in the lube oil sumps signal alarms in the control room and require shutting down of the pump. Each motor bearing contains embedded temperature detectors, and so initiation of failure, separate from loss of oil, is indicated and alarmed in the control room as a high bearing temperature. This, again, requires pump shutdown. If these indications are ignored, and the bearing proceeded to failure, the low melting point of Babbitt metal on the pad surfaces ensures that sudden seizure of the shaft will not occur. In this event the motor continues to operate, as it has sufficient reserve capacity to drive the pump under such conditions. However, the high torque required to drive the pump will require high current which will lead to the motor being shutdown by the electrical protection systems.

d. Locked Rotor

In may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft fails in torsion just below the coupling to the motor, disengaging the flywheel and motor from the shaft. This constitutes a loss-of-coolant flow in the loop. Following such a postulated seizure, the motor continues to run without any overspeed, and the flywheel maintains its integrity, as it is still supported on a shaft with two bearings. Flow transients are provided in the figures in Subsection 15.3.3 for the assumed locked rotor.

There are no other credible sources of shaft seizure other than impeller rubs. A sudden seizure of the pump bearing is precluded by graphite in the bearing. Any seizure in the seals results in a shearing of the anti-rotation pin in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Indications of the pump malfunction in these conditions are initially by high temperature signals from the bearing water temperature detector, and excessive Number 1 seal leakoff indications, respectively. Following these signals, pump vibration levels are checked. Excessive vibration indicates mechanical trouble and the pump is shutdown for investigations.

e. Critical Speed

The reactor coolant pump shaft is designed so that its operating speed is below its first critical speed. This shaft design, even under the most severe postulated transient, gives low values of actual stress.

f. Missile Generation

Precautionary measures taken to preclude missile formation from primary coolant pump components assure that the pumps will not produce missiles under any anticipated accident condition. Appropriate components of the primary pump motors have been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller, because the small fragments that might be ejected would be contained by the heavy casing. Further discussion and analysis of missile generation is contained in Reference 1.

g. Pump Cavitation

The minimum net position suction head required by the reactor coolant pump at running speed is approximately at 192 feet of head (approximately 85 psi). Operating instructions prohibit pump operation when minimum differential pressure across the number 1 seal is less than 200 psi. This corresponds to a primary loop pressure at which the minimum net positive suction head is exceeded and no limitation on pump operation occurs from this source.

h. Pump Overspeed Considerations

For turbine trips actuated by either the Reactor Trip System or the Turbine Protection System, the generator breaker disconnects the generator, permitting the reactor coolant pumps to be maintained connected to the external network to prevent any pump overspeed condition. Further discussion of pump overspeed considerations is contained in Reference 1.

i. Anti-Reverse Rotation Device

Each of the reactor coolant pumps is provided with an anti-reverse rotation device in the motor. This anti-reverse mechanism consists of pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and shock absorbers.

At an approximate forward speed of 70 rpm, the pawls drop and bounce across the ratchet plate; as the motor continues to slow, the pawls drag across the ratchet plate. After the motor has slowed and come to a stop, the dropped pawls engage the ratchet plate and, as the motor tends to rotate in the opposite direction, the ratchet plate also rotates until it is stopped by the shock absorbers. The rotor remains in this position until the motor is energized again. When the motor is started, the ratchet plate is returned to its original position by the spring return.

As the motor begins to rotate, the pawls drag over the ratchet plate. When the motor reaches sufficient speed, the pawls are bounced into an elevated position, and are held in that position by centrifugal forces acting upon the pawls. While the motor is running at rated speed, there is no contact between the pawls and ratchet plate.

Considerable plant experience with the design of the anti-reverse rotation device has shown high reliability of operation.

j. Shaft Seal Leakage

Leakage along the reactor coolant pump shaft is controlled by three shaft seals arranged in series so that reactor coolant leakage to the Containment is essentially zero. Seal water injection flow is directed to each reactor coolant pump. Seal water prevents leakage of high temperature reactor coolant along the pump shaft. It enters the pumps through a connection on the thermal barrier flange and is directed to a plenum between the thermal barrier housing and the shaft. Here the flow splits: a portion flows down the shaft to cool the bearing and enters the RCS; the remainder flows up the shaft through the Number 1 seal, a controlled leakage seal. After passing through the seal, most of the flow leaves the pump via the Number 1 seal leakoff line. Minor flow passes through the Number 2 seal and leakoff line. A back flush injection from a standpipe flows into the Number 3 seal between its "double dam" seal area. At this point the flow divides with half flushing through one side of the seal and out the Number 2 seal leakoff, while the remaining half flushed through the other side and out the Number 3 seal leakoff. This arrangement assures essentially zero leakage of reactor coolant or trapped gases from the pump.

k. Seal Discharge Piping

The Number 1 seal drops the system pressure to that of the volume control tank. Water from each pump Number 1 seal is piped to a common manifold, and through the seal water return filter and through the seal water heat exchanger where the temperature is reduced to that of the volume control tank. The Number 2 and Number 3 leakoff lines dump Number 2 and 3 seal leakage to the reactor coolant drain tank and the containment sump, respectively.

5.4.1.4 Tests and Inspections

The reactor coolant pumps can be inspected in accordance with the ASME Code, Section XI, for in-service inspection of nuclear reactor coolant systems.

The pump casing is cast in one piece, eliminating welds in the casing. Support feet are cast integral with the casing to eliminate a weld region.

The design enables disassembly and removal of the pump internals for visual access to the internal surface of the pump casing.

The reactor coolant pump quality assurance program is given in Table 5.4-2.

5.4.1.5 Pump Flywheels

The integrity of the reactor coolant pump flywheel is assured on the basis of the following design and quality assurance procedures:

a. Design Basis

The calculated stresses at operating speed are based on stresses due to centrifugal forces. The stress resulting from the interference fit of the flywheel on the shaft is less than 2000 psi at zero speed, but this stress becomes zero at approximately 600 rpm because of radial expansion of the hub. The primary coolant pumps run at approximately 1190 rpm and may operate briefly at overspeeds up to 109 percent (1295 rpm) during loss of outside load. For conservatism, however, 125 percent of operating speed was selected as the design speed for the primary coolant pumps. The flywheels are given a test of 125 percent of the maximum synchronous speed of the motor.

b. Fabrication and Inspection

The flywheel consists of two thick plates bolted together. The flywheel material is produced by a process that minimizes flaws in the material and improves its fracture toughness properties, such as vacuum degassing, vacuum melting, or electroslag remelting. Each plate is fabricated from SA-533, Grade B, Class 1 steel. Supplier certification reports are available for all plates and demonstrate the acceptability of the flywheel material on the basis of the requirements of Regulatory Guide 1.14.

Flywheel blanks are flame-cut from the SA-533, Grade B, Class 1, steel plates with at least ½ inch of stock left on the outer and bore radii for machining to final dimensions. The finished machined bores, keyways, and drilled holes are subjected to magnetic particle or liquid penetrant examinations in accordance with the requirements of Section III of the ASME Code. The finished flywheels, as well as the flywheel material (rolled plate), are subjected to 100 percent volumetric ultrasonic inspection using procedures and acceptance standards specified in Section III of the ASME Code.

The reactor coolant pump motors are designed such that, by removing the cover to provide access, the flywheel is available to allow an in-service inspection program in accordance with the recommendations of Regulatory Guide 1.14, which references Section XI of the ASME Code.

c. Material Acceptance Criteria

The reactor coolant pump motor flywheel conforms to the following material acceptance criteria:

1. The nil-ductility transition temperature (NDTT) of the flywheel material is obtained by two Drop Weight Tests (DWT) which exhibit "no-break" performance at 20°F in accordance with ASTM E208. The above drop weight tests demonstrate that the NDTT of the flywheel material is no higher than 10°F.

2. A minimum of three Charpy V-notch impact specimens from each plate shall be tested at ambient (70°F) temperature in accordance with the specification ASME SA-370. The Charpy V-notch (C_v) energy in both the parallel and normal orientation with respect to the rolling direction of the flywheel materials is at least 50 foot pounds at 70°F and therefore an RT_{NDT} of 10°F can be assumed. An evaluation of flywheel overspeed has been performed which concludes that flywheel integrity will be maintained (Reference 1).

Thus, it is concluded that flywheel plate materials are suitable for use, and can meet Regulatory Guide 1.14 acceptance criteria on the bases of suppliers' certification data. The degree of compliance with Regulatory Guide 1.14 is further discussed in Section 1.8.

5.4.2 Steam Generators

5.4.2.1 Design Basis

Steam generator design data are given in Table 5.4-3. Code classifications of the steam generator components are given in Section 3.2. Although the ASME classification for the secondary side is specified to be Class 2, all pressure retaining parts of the steam generator, and thus both the primary and secondary pressure boundaries, are designed to satisfy the criteria specified in Section III of the ASME Code for Class 1 components. The design stress limits, transient conditions and combined loading conditions applicable to the steam generator are discussed in Subsection 3.9.1. Estimates of radioactivity levels anticipated in the secondary side of the steam generators during normal operation, and the bases for the estimates, are given in Chapter 11. The accident analysis of a steam generator tube rupture is discussed in Chapter 15.

A design objective of the internal moisture separation equipment is that moisture carryover should not exceed 0.25 percent by weight under the following conditions:

a. Steady state operation up to 100 percent of full load steam flow, with water at the normal operating level.

b. Loading or unloading at a rate of 5 percent of full power steam flow per minute in the range of 15 to 100 percent of full load steam flow.

c. A step load change of 10 percent of full power in the range from 15 to 100 percent full load steam flow.

The water chemistry on the reactor side, selected to provide the necessary boron content for reactivity control should minimize corrosion of RCS surfaces. The effectiveness of the water chemistry of the steam side in maintaining corrosion control are discussed in Chapter 10. Compatibility of steam generator tubing with both primary and secondary coolants is discussed further in Subsection 5.4.2.4c.

The steam generator is designed to minimize unacceptable damage from mechanical or flow induced vibration. Tube support adequacy is discussed in Subsection 5.4.2.3c. The tubes and tube sheet are analyzed and confirmed to withstand the maximum accident loading conditions as they are defined in Subsection 3.9.1. Further consideration is given in Subsection 5.4.2.3d to the effect of tube wall thinning on accident stresses.

5.4.2.2 Design Description

The steam generator is a Model F, vertical shell and U-tube evaporator, with integral moisture separating equipment. Figure 5.4-4 shows the model, indicating several of its improved design features which are described in the following paragraphs.

On the primary side, the reactor coolant flows through the inverted U-tubes, entering and leaving through nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical divider plate extending from the apex of the head to the tube sheet.

Steam is generated on the shell side, flows upward and exits through the outlet nozzle at the top of the vessel. Feedwater enters the steam generator at an elevation above the top of the U-tubes, through a feedwater nozzle. The water is distributed circumferentially around the steam generator by a feedwater ring and then flows through an annulus between the tube wrapper and shell. The feedwater enters the ring via a welded thermal sleeve connection and leaves it through inverted "J" tubes located at the flow holes which are at the top of the ring. The "J" tubes are arranged to distribute the bulk of the colder feedwater to the hot leg side of the tube bundle. The feed ring is designed to minimize conditions which can result in water hammer occurrences in the feedwater piping. At the bottom of the wrapper, the water is directed toward the center of the tube bundle by a flow distribution baffle. This baffle arrangement serves to minimize the tendency in the relatively low velocity fluid for sludge deposition. Flow blocking devices discourage the water from flowing up the bypass lane, as it enters the tube bundle where it is converted to a steam-water mixture. Subsequently, the steam-water mixture from the tube bundle rises into the steam drum section, where 16 individual centrifugal moisture separators remove most of the entrained water from the steam. The steam continues to the secondary separators for further moisture removal, increasing its quality to a designed minimum of 99.75 percent. The moisture separators direct the separated water, which is combined with entering feedwater to flow back down the annulus between the wrapper and shell for recirculation through the steam generator. The dry steam exits from the steam generator through the outlet nozzle which is provided with a steam flow restrictor, described in Subsection 5.4.4.

5.4.2.3 Design Evaluation

a. Forced Convection

The effective heat transfer coefficient is determined by the physical characteristics of the Model F steam generator and the fluid conditions in the primary and secondary systems for the "nominal" 100 percent design case. It includes a conservative allowance for fouling and uncertainty. A designed heat transfer area is provided to permit the achievement of full design heat removal rate.

b. Natural Circulation Flow

The driving head created by the change in coolant density as it is heated in the core and rises to the outlet nozzle initiates convection circulation. This circulation is enhanced by the fact that the steam generators, which provide a heat sink, are at a higher elevation than the reactor core which is the heat source. Thus, natural circulation is provided for the removal of decay heat during hot shutdown in the unlikely event of loss of forced circulation.

c. Mechanical and Flow Induced Vibration Under Normal Operating Conditions

In the design of the steam generators, the possibility of degradation of tubes due to either mechanical or flow induced excitation is thoroughly evaluated. This evaluation includes detailed analysis of the tube support systems as well as an extensive research program with tube vibration model tests and instrumentation of an operating Model F steam generator.

In evaluating tube degradation due to vibration, consideration is given to sources of excitation such as those generated by primary fluid flowing within the tubes, mechanically induced vibration and secondary fluid flow on the outside of the tubes. During normal operation, the effects of primary fluid flow within the tubes and mechanically induced vibration are considered to be negligible and should cause little concern. Thus, the primary source of tube vibrations is the hydrodynamic excitation by the secondary fluid impinging on the outside surface of the tubes. Three vibration mechanisms have been identified in tube bundle arrays:

1. Vortex shedding
2. Fluidelastic excitation
3. Turbulence.

Vortex shedding is not expected to produce a mechanism for vibration in close-arrayed tube bundles. There are several reasons why this happens:

1. Flow turbulence in the downcomer and tube bundle inlet region may inhibit the formation of Von Karman's vortex train.
2. The spatial variations of cross flow velocities along the tube preclude vortex shedding at a single frequency.
3. The axial flow velocity components existing on the tubes is expected to reduce the potential for the formation of Von Karman vortex sheets.

Fluidelastic vibration analytical model analysis computed fluidelastic stability ratios to be less than 1.0 indicating that the tubes are stable relative to fluidelastic excitation.

Fluidelastic excitation was not observed during prototypical and operational flow testing. Therefore, fluidelastic excitation is excluded from consideration as a factor in steam generator tube bundle vibrations.

Flow vibrations due to fluid turbulence induce stresses in the tubes that are two orders of magnitude below the endurance limit (30,000 psi) of the tube material. Therefore, the vibration contribution to fatigue degradation from flow-induced vibration is not anticipated during normal operation.

Summarizing the results of analysis and tests of the Model F steam generator tubes for vibration, it can be stated that a check of all modes of tube vibration mechanisms has been completed. Conclusions can be drawn that the primary source of tube vibration is cross-flow turbulence, that the amplitude of the tube vibration is small, and that fatigue degradation due to flow-induced vibration is not anticipated.

The impact of operation of Seabrook Station at a power level of 3678 MWt has been evaluated and it is concluded that significant levels of vibration will not occur from the fluidelastic, vortex shedding, or turbulent mechanisms. The projected level of tube wear as a result of vibration will remain small and will not result in unacceptable tube wear.

d. Allowable Tube Wall Thinning Under All Plant Conditions

An analysis has been performed to define the structural limits for an assumed uniform thinning mode of degradation in both the axial and circumferential directions at a power level of 3678 MWt for Seabrook Unit 1. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. The allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating into the structural limit a growth allowance for continued operation until the next scheduled inspection, as well as an allowance for eddy current measurement uncertainty.

Calculations have been performed to establish the tube structural limit for the straight leg (free span) region of the tube for degradation over an unlimited axial length, and for degradation over limited axial extent at the tube support plate (TSP), the flow distribution baffle (FDB), and anti-vibration bar intersections for the 3678 MWT power level conditions at Seabrook Station.

The minimum structural limit is calculated to be 57.5% allowable tube wall loss for a straight length location at the low T_{avg} condition at a power level of 3678 MWt. A steam generator tube with wall thinning to this extent can withstand normal operating and accident condition loadings and meet the stress limits as defined by the ASME Code.

The results of a study made on "D series" (0.75 inch nominal diameter, 0.043 inch nominal wall thickness) tubes under accident loadings are discussed in Reference 3. These results demonstrate that a minimum wall thickness of 0.026 inches would have a maximum faulted condition stress (i.e., due to combined LOCA and safe shutdown earthquake loads) that is less than the allowable limit. This thickness is 0.010 inches less than the minimum "D series" tube wall thickness of 0.039 inches which is reduced to 0.036 inches by the assumed general corrosion and erosion rate. Thus, an adequate safety margin is exhibited. The corrosion rate is based on a conservative weight loss rate of Inconel tubing in flowing 650°F primary side reactor coolant fluid. The weight loss, when equated to a thinning rate and projected over a 40-year design operating objective, with appropriate reduction after initial hours, is equivalent to 0.083 mils thinning. The assumed corrosion rate of 3 mils leaves a conservative 2.917 mils for general corrosion thinning on the secondary side.

The Model F steam generator has to be analyzed using similar assumptions of general corrosion and erosion rates. The nominal dimensions of tubing used in the Model F design are 0.688 inch 0D by 0.040 inch wall thickness; the minimum wall thickness of such tubing is 0.036 inches. For Seabrook, the minimum allowable tube wall thickness to withstand postulated accident condition loadings has been calculated to be 0.014 inch. Reducing the minimum wall thickness of 0.036 inch to 0.033 inch by an assumed general corrosion and erosion rate over a 40-year period of 0.003 inch, results in a margin of 0.019 inch.

5.4.2.4 Steam Generator Materials

a. Selection and Fabrication of Materials

All pressure boundary materials used in the steam generator are selected and fabricated in accordance with the requirements of Section III of the ASME Code. A general discussion of materials specifications is given in Subsection 5.2.3, with types of materials listed in Table 5.2-2 and Table 5.2-3. Fabrication of reactor coolant pressure boundary materials is also discussed in Subsection 5.2.3, particularly in Subsections 5.2.3.3 and 5.2.3.4.

Testing has justified the selection of corrosion resistant Inconel-600, a nickel-chromium-iron alloy (ASME SB-163), for the steam generator tubes. The channel head divider plate is Inconel (ASME SB-168). The interior surfaces of the reactor coolant channel head, nozzles, and manways are clad with austenitic stainless steel. The primary side of the tube sheet is weld clad with Inconel (ASME SFA-5.14). The tubes are then seal welded to the tube sheet cladding. These fusion welds are performed in compliance with Sections III and IX of the ASME Code, and are dye penetrant inspected and leak proof tested before each tube is hydraulically expanded the full depth of the tube sheet bore.

Code cases used in material selection are discussed in Subsection 5.2.1. The extent of conformance with Regulatory Guides 1.84, "Design and Fabrication Code Case Acceptability ASME Section III Division 1," and 1.85 "Materials Code Case Acceptability ASME Section III Division 1," is discussed in Section 1.8.

During manufacture, cleaning is performed on the primary and secondary sides of the steam generator in accordance with written procedures which follow the guidance of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants." Onsite cleaning and cleanliness control also follow the guidance of Regulatory Guide 1.37, as discussed in Section 1.8. Cleaning process specifications are discussed in Subsection 5.2.3.4.

The fracture toughness of the materials is discussed in Subsection 5.2.3.3. Adequate fracture toughness of ferritic materials in the reactor coolant pressure boundary is provided by compliance with 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and with paragraph NB-2300 of Section III of the ASME Code.

b. Steam Generator Design Effects on Materials

Several features have been introduced into the Model F steam generator to minimize the deposition of contaminants from the secondary side flow. Such deposits could otherwise produce a local environment in which a condition could develop and result in material corrosion. The support plates are made of corrosion resistant stainless steel 405 alloy and incorporate a four-lobe-shaped tube hole design that provides greater flow area adjacent to the tube outer surface and eliminates the need for interstitial flow holes. The resulting increase in flow provides higher sweeping velocities at the tube/tube support plate intersections. Figure 5.4-3 is an illustration of the "quatrefoil" broached holes. This modification in the support plate design is a major factor contributing to the increased circulation ratio. The increased circulation results in increased flow in the interior of the bundle, as well as increased horizontal velocity across the tube sheet, which reduces the tendency for sludge deposition. The effect of the increased circulation on the vibrational stability of the tube bundle has been analyzed with consideration given to flow induced excitation frequencies. The unsupported span length of tubing in the U-bend region and the corresponding optimum number of anti-vibration bars has been determined. The anti-vibration bars are fabricated from square Inconel barstock which is then chromium plated to improve frictional characteristics. Also, due to the increased circulation ratio, the moisture separating equipment has been modified to maintain an adequate margin with respect to moisture carryover. To provide added strength as well as resistance to vibration, the quatrefoil tube support plate thickness has been increased.

In addition, 13 peripheral supports also provide stability to the plates so that tube fretting or wear due to flow induced plate vibrations at the tube support contact regions is reduced.

Assurance against significant flow induced tube vibration has been obtained by a combination of analysis and testing.

Combining both vortex shedding and turbulence effects in a conservative manner, the maximum predicted local tube wear depth over a 40-year operating design objective remains less than 0.006 inches with the operation of Seabrook Unit 1 at a power level of 3678 MWt.

This value is considerably below the plugging limit for a Model F steam generator tube.

c. Compatibility of Steam Generator Tubing with Primary and Secondary Coolants

As mentioned in Subsection 5.4.2.4a, corrosion tests, which subjected the steam generator tubing material, Inconel-600 (ASME SB-163), to simulated steam generator water chemistry, have indicated that the loss due to general corrosion over the 40-year design operating objective is insignificant compared to the tube wall thickness. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel-600 has resisted general and pitting-type corrosion in severe operating water conditions. Many reactor years of successful operation have shown the same low general corrosion rates as indicated by the laboratory tests.

Recent operating experience, however, has revealed areas on secondary surfaces where localized corrosion rates were significantly greater than the low general corrosion rates. Both intergranular stress corrosion and tube wall thinning were experienced in localized areas, although not simultaneously at the same location or under the same environmental conditions (water chemistry, sludge composition).

The adoption of the All Volatile Treatment (AVT) control program minimizes the possibility for recurrence of the tube wall thinning phenomenon. Successful AVT operation requires maintaining low concentrations of impurities in the steam generator water, thus reducing the potential for formation of highly concentrated solutions in low flow zones, which is the precursor of corrosion. By restriction of the total alkalinity in the steam generator and prohibition of extended operation with free alkalinity, the AVT program should minimize the possibility for recurrence of intergranular corrosion in localized areas due to excessive levels of free caustic.

Laboratory testing has shown that the Inconel-600 tubing is compatible with the AVT environment. Isothermal corrosion testing in high purity water has shown that commercially produced Inconel-600 exhibiting normal microstructures tested at normal engineering stress levels does not suffer intergranular stress corrosion cracking in extended exposure to high temperature water. These tests also showed that no general type of corrosion occurred. A series of autoclave tests in reference secondary water with planned excursions have produced no corrosion attack after 1,938 days of testing on any as-produced Inconel-600 tube samples.

Model boiler tests have been used to evaluate the AVT chemistry guidelines adopted in 1974. The guidelines appear to be adequate to preserve tube integrity, with one significant alteration: operation with contaminant ingress must be limited.

Additional extensive operating data are presently being accumulated with the conversion to AVT chemistry. A comprehensive program of steam generator inspections, including the recommendations of Regulatory Guide 1.83, "In-Service Inspection of Pressurized Water Reactor Steam Generator Tubes," with the exceptions as stated in Section 1.8, should provide for detection of any degradation that might occur in the steam generator tubing.

Increased margin against stress corrosion cracking has been obtained by the use of thermally treated Inconel-600 tubing. Thermal treatment of Inconel tubes has been shown to be particularly effective in resisting caustic corrosion. Tubing used in the Model F is thermally treated in accordance with a laboratory derived treatment process.

The tube support plates used in the Model F are ferritic stainless steel, which has been shown in laboratory tests to be resistant to corrosion in the AVT environment. If corrosion of ferritic stainless steel were to occur, due to concentration of contaminants, the volume of the corrosion products is essentially equivalent to the volume of the parent material consumed. This would be expected to preclude denting. The support plates are also designed with quatrefoil tube holes rather than cylindrical holes. The quatrefoil tube hole design promotes high velocity flow along the tube, and is expected to minimize the accumulation of impurities at the support plate locations.

Additional measures are incorporated in the Model F design to prevent areas of dryout in the steam generator and accumulations of sludge in low velocity areas. Modifications to the wrapper have increased water velocities across the tube sheet. A flow distribution baffle is provided which forces the low flow area to the center of the bundle. Increased capacity blowdown pipes have been added to enable continuous blowdown of the steam generators at a high volume. The intakes of these blowdown pipes are located below the center cutout section of the flow distribution baffle in the low velocity region where sludge may be expected to accumulate. Continuous blowdown should provide protection against inleakage of impurities from the condenser.

The impact of operating Seabrook Unit 1 at a power level of 3678 MWt on steam generator water chemistry has been considered. The occurrence of stress corrosion cracking and other forms of degradation that might occur at current and enhanced rates will be found using the non-destructive examination techniques specified in the degradation assessment that must be completed for subsequent plant outages. The degradation assessment is the key document in planning for the SG tube inspection, where inspection plans and related actions are determined, documented, and communicated prior to the outage. The degradation assessment provides assurance that non-destructive examination detection and sizing performance is known and can be accounted for in the condition monitoring and operational assessments required by NEI 97-06, Rev. 1, "Steam Generator Program Guidelines."

d. Cleanup of Secondary Side Materials

Several methods are employed to clean operating steam generators of corrosion causing secondary side deposits. Sludge lancing, a procedure in which a hydraulic jet inserted through an access opening (handhole) loosens deposits which are removed by means of a suction pump, can be performed when the need is indicted by the results of steam generator tube inspection. Six 6-inch access ports are provided for sludge lancing and inspection. Three of these are located above the tube sheet and three above the flow distribution baffle. Continuous blowdown is performed to monitor water chemistry. The location of the blowdown piping suction, adjacent to the tube sheet and in a region of relatively low flow velocity, facilities the removal of particulate impurities to minimize the accumulation on the tube sheet.

5.4.2.5 Steam Generator In-Service Inspection

a. Design Provisions for Inspection

The Seabrook Station steam generators have been designed to facilitate the inspection and repair of all Code Class 1 and 2 components, including the individual steam generator tubes.

1. Access to the vessel welds is provided by means of reusable fiberglass blanket with stainless steel jacket insulation panels which can be quickly and easily removed to expose the welds and the adjacent areas.

2. Access for inspection of the steam generator internals is provided by means of four manways: two providing access to the two chambers of the reactor coolant channel-head and two providing access to the steam drum moisture separators. Six-inch hand holes have been provided, three just above the tube sheet and three just above the distribution baffle. Access to the U-bends is provided through each of the three deck plates.

b. Inspection Plans

In general, the in-service inspection program for the steam generators will be conducted in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Subarticles IWB-2500, IWC-2500 and Regulatory Guide 1.83, Rev. 1.

Prior to startup, a baseline inspection will be performed on all Class 1 and 2 components. Permanent records of baseline and in-service inspections will be maintained. Detailed plans for baseline and in-service inspection of all Code Class 1 and 2 components of the steam generators, including criteria for tube plugging, are presented in Technical Specification 3/4.4.5, Steam Generators.

Volumetric, surface and visual examinations of the following components will be performed:

1. Circumferential and meridional head welds (primary side), circumferential and longitudinal shell welds, tube-sheet-to-head and tube-sheet-to-shell welds, nozzle to vessel welds (primary side), nozzle inside radius sections (primary side), steam generator tubing, class 1 bolts and studs over 2 inch diameter, in place, and class 2 bolts and studs over 2 inch diameter will be examined by volumetric methods.

2. Nozzle-to-safe-end welds, class 1 bolts and studs over 2 inch diameter when removed, and nozzle-to-vessel welds on the secondary side over ½ inch nominal vessel wall thickness will be examined by volumetric and surface methods.

3. Secondary side nozzle-to-vessel welds less than ½ inch nominal vessel wall thickness will be examined using surface methods.

4. Surfaces of pressure-retaining bolting, and nonpressure-retaining bolts and studs 2 inch diameter and smaller will be visually examined using VT-1 methods.

5. Pressure retaining components will be visually examined during leak tests or hydrotest using VT-2 methods.

6. Component supports will be examined using VT-3 methods or VT-4 methods for mechanical or hydraulic supports.

c. Steam Generator Tubing Inspection

Steam generator tubing in-service inspection will be performed in accordance with the requirements of Regulatory Guide 1.83, Rev. 1. The details of the in-service inspection program, such as a description of the equipment, procedures, sensitivity of the examination and recording methods, criteria used to select tubes for examination, inspection intervals and actions to be taken if defects are found (including criteria for plugging defective tubes), are presented in Technical Specification 3/4.4.5, Steam Generators.

The tubing examination equipment and procedures will be capable of detecting and locating defects with a penetration of 20 percent or more of wall thickness. The recommendations of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" will be followed for the resolution of problems regarding degraded tubes.

The allowable Steam Generator Tube Plugging (SGTP) limit is 10% of the total tubes in all four steam generators. This limit is uniform. That is, no individual steam generator may exceed 10% tube plugging. This limit is an input assumption to the Seabrook Station safety analyses. Additional discussion of the SGTP limit is found in UFSAR section s15.0.3.5.2 and 15.6.5.2.

5.4.2.6 Quality Assurance

The steam generator quality assurance program is given in Table 5.4-4.

Radiographic inspection and acceptance standard will be in accordance with the requirements of Section III of the ASME Code.

Liquid penetrant inspection is performed on weld deposited tube sheet cladding, channel head cladding, divider plate to tube sheet and to channel head weldments, tube-to-tube sheet weldments, and weld deposit cladding. Liquid penetrant inspection and acceptance standards are in accordance with the requirements of Section III of the ASME Code.

Magnetic particle inspection is performed on the tube sheet forging, channel head casting, nozzle forgings, and the following weldments:

a. Nozzle to shell

b. Support brackets

c. Instrument connection (secondary)

d. Temporary attachments after removal

e. All accessible pressure retaining welds after hydrostatic test.

Magnetic particle inspection and acceptance standards are in accordance with requirements of Section III of the ASME Code.

Ultrasonic tests are performed on the tube sheet forging, tube sheet cladding, secondary shell and head plate, and nozzle forgings.

The heat transfer tubing is subjected to eddy current testing and ultrasonic examination.

Hydrostatic tests are performed in accordance with Section III of the ASME Code.

5.4.3 Reactor Coolant Piping

5.4.3.1 Design Bases

The RCS piping is designed and fabricated to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions. Stresses are maintained within the limits of Section III of the ASME Boiler and Pressure Vessel Code. Code and material requirements are provided in Section 5.2.

Materials of construction are specified to minimize corrosion/erosion and to ensure compatibility with the operating environment.

The piping in the RCS is Safety Class 1, and is designed and fabricated in accordance with ASME Code, Section III, Class 1 requirements.

Stainless steel pipe conforms to ANSI B36.19 for sizes ½ inch through 12 inches and wall thickness Schedules 40S through 80S. Stainless steel pipe outside the scope of ANSI B36.19 conforms to ANSI B36.10.

The minimum wall thickness of the loop pipe and fittings are not less than that calculated using the ASME Code, Section III, Class 1 formula of paragraph NB-3641.1(3) with an allowable stress value of 17,550 psi. The pipe wall thickness for the pressurizer surge line is Schedule 160. The minimum pipe bend radius is 5 nominal pipe diameters, and ovality does not exceed 6 percent.

All butt welds, branch connection nozzle welds, and boss welds are of a full penetration design.

Processing and minimization of sensitization are discussed in Subsection 5.2.3.

Flanges conform to ANSI B16.5.

Socket weld fittings and socket joints conform to ANSI B16.11.

In-service inspection is discussed in Subsection 5.2.4.

5.4.3.2 Design Description

Principal design data for the reactor coolant piping are given in Table 5.4-5.

Reactor coolant pipes are forged. Reactor coolant fittings are cast. Both pipe and fittings are seamless without longitudinal or electroslag welds, and comply with the requirements of the ASME Code, Section II, (Parts A and C), Section III, and Section IX.

The RCS piping is specified in the smallest sizes consistent with system requirements. This design philosophy results in the reactor inlet and outlet piping diameters given in Table 5.4-5. The line between the steam generator and the pump suction is larger, to reduce pressure drop and to improve flow conditions to the pump suction.

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. There will be no electroslag welding on these components. All smaller piping which comprise part of the RCS such as the pressurizer surge line, spray and relief line, loop drains and connecting lines to other systems are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. All joints and connections are welded, except for the pressurizer code safety valves, where flanged joints are used. Thermal sleeves are installed on the pressurizer spray line nozzle and on the pressurizer surge nozzle located on the bottom of the pressurizer vessel.

All piping connections from auxiliary systems are made above the horizontal centerline of the reactor coolant piping, with the exception of:

a. Residual heat removal pump suction lines, which are 45 degrees down from the horizontal centerline. This enables the water level in the RCS to be lowered in the reactor coolant pipe while continuing to operate the Residual Heat Removal System, should this be required for maintenance.

b. Loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.

c. The differential pressure taps for flow measurement, which are downstream of the steam generators on the first 90 degree elbow.

d. The pressurizer surge line, which is attached at the horizontal centerline.

e. The 3-inch charging line connections, the 1-inch excess letdown connection and the 4-inch pressurizer spray line nozzles with scoops which are all located on the horizontal centerline.

Penetrations into the coolant flow path are limited to the following:

a. The pressurizer spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force.

b. The reactor coolant sample system taps protrude into the main stream to obtain a representative sample of the reactor coolant.

c. The narrow range RCS resistance temperature detectors (RTDs) are mounted in thermowells in the hot and cold legs. With one exception on each hot leg, the original bypass scoops are machined and used to house the thermowells. One scoop on each hot leg, however, has been capped and the thermowell has been relocated to an independent boss. The independent boss is located on the same cross-sectional plane as the scoops on loops A, B, and D. On loop C, the boss has been relocated to a position approximately 12 inches upstream of the scoops location at approximately 105°.

d. The wide range temperature detectors are located in resistance temperature detector wells that extend into both the hot and cold legs of the reactor coolant pipes.

Separate thermowell-mounted RTDs for each reactor coolant loop hot and cold leg are provided so that individual temperature signals may be developed for use in the reactor control and protection system. A representative hot leg temperature is obtained by averaging three thermowell-mounted RTDs.

Signals from the thermowell-mounted RTDs are used to compute the reactor coolant ΔT (temperature of the hot leg, T_{hot}, minus the temperature of the cold leg, T_{cold}) and an average reactor coolant temperature (T_{avg}). The T_{avg} for each loop is indicated on the main control board.

The RCS piping includes those sections of piping interconnecting the reactor vessel, steam generator, and reactor coolant pump. It also includes the following:

a. Charging line and alternate charging line from the system isolation valve up to the branch connections on the reactor coolant loop.

b. Letdown line and excess letdown line from the branch connections on the reactor coolant loop to the system isolation valve.

c. Pressurizer spray lines from the reactor coolant cold legs to the spray nozzle on the pressurizer vessel.

d. Residual heat removal lines to or from the reactor coolant loops up to the designated check valve or isolation valve.

e. Safety injection lines from the designated check valve to the reactor coolant loops.

f. Accumulator lines from the designated check valve to the reactor coolant loops.

g. Loop drain, sample*, and instrument* lines to or from the designated isolation valve to or from the reactor coolant loops.

h. Pressurizer surge line from one reactor coolant loop hot leg to the pressurizer vessel inlet nozzle.

---

* Lines with a $^3/_8$ inch or less flow restricting orifice qualify as Safety Class 2; in the event of a break in one of these Safety Class 2 lines, the normal makeup system is capable of providing makeup flow while maintaining pressurizer water level.

i. Resistance temperature detector scoop element, pressurizer spray scoop, sample connection* with scoop, reactor coolant temperature element installation boss, and the temperature element well itself.

j. All branch connection nozzles attached to reactor coolant loops.

k. Pressure relief lines from nozzles on top of the pressurizer vessel up to and through the power-operated pressurizer relief valves and pressurizer safety valves.

l. Seal injection water lines to the reactor coolant pump to the designated check valve (injection line).

m. Auxiliary spray line from the isolation valve to the pressurizer spray line header.

n. Sample lines* from pressurizer to the isolation valve.

Details of the materials of construction and codes used in the fabrication of reactor coolant piping and fittings are discussed in Section 5.2.

5.4.3.3 Design Evaluation

Piping load and stress evaluation for normal operating loads, seismic loads, blowdown loads, and combined normal, blowdown and seismic loads is discussed in Section 3.9(N).

a. Material Corrosion/Erosion Evaluation

The water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications.

The design and construction are in compliance with the ASME Code, Section XI. Pursuant to this, all pressure containing welds out to the second valve that delineate the RCS boundary are available for examination with removable insulation.

* Lines with a 3/8 inch or less flow restricting orifice qualify as Safety Class 2; in the event of a break in one of these Safety Class 2 lines, the normal makeup system is capable of providing makeup flow while maintaining pressurizer water level.

Components constructed with stainless steel will operate satisfactorily under normal plant chemistry conditions in pressurized water reactor systems, because chlorides, fluorides, and particularly oxygen, are controlled to very low levels using the guidance in EPRI PWR Primary Water Chemistry Guidelines and implemented in the Chemistry Manual.

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to desired reactor coolant water quality. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and sampling system which are described in Chapter 9.

b. Sensitized Stainless Steel

Sensitized stainless steel is discussed in Subsection 5.2.3.

c. Contaminant Control

Contamination of stainless steel and Inconel by copper, low melting temperature alloys, mercury and lead is prohibited. Colloidal graphite is the only permissible thread lubricant.

Except for those times when thermal insulation is removed for the installation of new thermal insulation (due to mechanical damage or age), the austenitic stainless steel surfaces are cleaned and analyzed to a halogen limit specified by the NSSS vendor.

5.4.3.4 Tests and Inspections

The RCS piping quality assurance program is given in Table 5.4-6.

Volumetric examination is performed throughout 100 percent of the wall volume of each pipe and fitting, in accordance with the applicable requirements of Section III of the ASME Code for all pipe 27½ inches and larger. All unacceptable defects are eliminated in accordance with the requirements of the same section of the code.

A liquid penetrant examination is performed on both the entire outside and inside surfaces of each finished fitting in accordance with the criteria of the ASME Code, Section III. Acceptance standards are in accordance with the applicable requirements of the ASME Code, Section III.

The pressurizer surge line conforms to SA-376, Grade 304, 304N, or 316 with supplementary requirements S2 (transverse tension tests), and S6 (ultrasonic test). The S2 requirement applies to each length of pipe. The S6 requirement applies to 100 percent of the piping wall volume.

The end of pipe sections, branch ends and fittings are machined back to provide a smooth weld transition adjacent to the weld path.

5.4.4 Main Steam Line Flow Restrictor

5.4.4.1 Design Basis

The outlet nozzle of the steam generator is provided with a flow restrictor designed to limit steam flow in the unlikely event of a break in the main steam line. A large increase in steam flow will create a backpressure which limits further increase in flow. Several protective advantages are thereby provided: rapid rise in containment pressure is prevented, the rate of heat removal from the reactor coolant is kept within acceptable limits, thrust forces on the main steam line piping are reduced, and stresses on internal steam generator components, particularly the tube sheet and tubes, are limited. The resistor is also designed to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.4.2 Design Description

The flow restrictor consists of seven Inconel (ASME SB-163) venturi inserts which are inserted into the holes in an integral steam outlet nozzle forging. The inserts are arranged with one venturi at the centerline of the outlet nozzle and the other six equally spaced around it. After insertion into the nozzle forging holes, the Inconel venturi inserts are welded to the Inconel cladding on the inner surface of the forgings.

5.4.4.3 Design Evaluation

The flow restrictor design has been sufficiently analyzed to assure its structural adequacy. The equivalent throat diameter of the steam generator outlet is 16 inches, and the resultant pressure drop through the restrictor at 100 percent steam flow is approximately 3.28 psi. This is based on a design flow rate of 4.135×10^6 lb/hr. Materials of construction and manufacturing of the flow restrictor are in accordance with Section III of the ASME Code.

5.4.4.4 Tests and Inspections

Since the restrictor is not a part of the steam system boundary, no tests and inspection beyond those during fabrication are anticipated.

5.4.5 Main Steam Isolation System

The Main Steam Isolation System is composed of the main steam isolation valves, actuators, logic cabinets and portions of the Engineered Safety Features Actuation System. The MSIVs are located in the mainsteam and feedwater pipe chases adjacent to the Containment.

5.4.5.1 Design Bases

a. Isolate the steam generators and main steam lines in the event of a main steam or feedwater line rupture to prevent the uncontrolled blowdown of more than one steam generator.

b. Isolate the Containment from the outside environment in the event of a design basis accident, upon receipt of containment isolation signals as described in Subsection 6.2.4.

5.4.5.2 Description

The main steam isolation system components are described in Section 10.3 and Subsection 6.2.4.

5.4.5.3 System Operation

The isolation system operation is described in Subsection 6.2.4. The isolation valve operation is described in Subsection 10.3.2.

5.4.5.4 Design Evaluation

The main steam isolation valves will close in 3 to 5 seconds after receipt of a signal from the Engineered Safety Features Actuation System, and are capable of stopping flow from either the forward or reverse directions.

Immediately after a pipe break and until isolation occurs, all steam generators will be partially blown down. After isolation occurs, only the inventory portions of the main steam piping will be blown down, if the break is downstream of an isolation valve. If the break occurs upstream of an isolation valve, either inside or outside Containment, one steam generator will be blown down.

5.4.5.5 Tests and Inspections

Each MSIV is given an operability test every three months. This involves partially stroking the valve. See Subsection 10.3.2 for additional discussion of MSIV testing.

5.4.6 Reactor Core Isolation Cooling System

Not applicable to Seabrook.

5.4.7 Residual Heat Removal System

The Residual Heat Removal System (RHRS) transfers heat from the Reactor Coolant System (RCS) to the Component Cooling System (CCS) to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal plant cooldown, and maintains this temperature until the plant is started up again.

Parts of the RHRS also serve as parts of the ECCS during the injection and recirculation phases of a LOCA (see Section 6.3).

The RHRS also is used to transfer refueling water between the refueling cavity and the refueling water storage tank at the beginning and end of the refueling operations.

Nuclear plants employing the same RHRS design as the Seabrook Station are given in Section 1.3.

5.4.7.1 Design Bases

RHRS design parameters are listed in Table 5.4-7.

The RHRS is placed in operation approximately four hours after reactor shutdown, when the temperature and pressure of the RCS are approximately 350°F and 365 psig, respectively. Assuming that two heat exchangers and two pumps are in service and that each heat exchanger is supplied with component cooling water at 3000 gpm, the RHRS is designed to reduce the temperature of the reactor coolant from 350°F to 125°F in 23.9 hours. The time required under these conditions to reduce the reactor coolant temperature from 350°F to 212°F is less than 7 hours. Maximum anticipated component cooling water temperature is 102°F during initial RHRS operation; however, it will approach 85°F at 24 hours after reactor shutdown. The heat load handled by the RHRS during the cooldown transient includes residual and decay heat from the core, coolant sensible heat, and reactor coolant pump heat. The design heat load is based on the decay heat fraction that exists at 24 hours following reactor shutdown from an extended run at full power. The normal cooldown curve is shown in Figure 5.4-5.

Assuming that only one heat exchanger and pump are in service and that the heat exchanger is supplied with component cooling water at 3000 gpm, the RHRS is capable of reducing the temperature of the reactor coolant from 350°F to 200°F within 33.2 hours. The time required, under these conditions, to reduce reactor coolant temperature from 350°F to 212°F is less than 25 hours. Maximum anticipated component cooling water temperature is 102°F during initial RHRS operation; however, it will approach 85°F at 120 hours after reactor shutdown. The single train RHR cooldown curve is shown on Figure 5.4-6.

Assuming 3000 gpm water at 85°F with 2 pumps and 2 heat exchangers in operation, the system is cooled from 350°F to 125°F so that 125°F is achieved within 24 hours after shutdown. As the cooldown proceeds, the CCW temperature may rise as high as 102°F but will drop off again to 85°F.

The RHRS is designed to be isolated from the RCS whenever the RCS pressure exceeds the RHRS design pressure. The RHRS is isolated from the RCS on the suction side by two motor-operated valves in series on each suction line. Each motor-operated valve is interlocked to prevent its opening if RCS pressure is greater than that which could result in the RHR system design pressure being exceeded. (This includes the effects of instrument uncertainty and bistable deadband.) Refer to the Station Technical Specifications. (See Figure 5.4-7.) The RHRS is isolated from the RCS on the discharge side by two check valves in each return line. Also provided on the discharge side is a normally open motor-operated valve downstream of each RHRS heat exchanger. (These check valves and motor-operated valves are not considered part of the RHRS; they are shown as part of the ECCS, see Figure 6.3-1, Figure 6.3-2, Figure 6.3-3, Figure 6.3-4 and Figure 6.3-5.)

Those valves used for hot leg injection are in the residual heat removal pump and safety injection pump discharge and do not qualify as high-to-low pressure isolation barriers since they are backed up by normally closed motor-operated gate valves which form an additional high-to-low pressure barrier. Those valves used for charging/safety injection are in the high pressure discharge piping of the charging pumps and do not qualify as high-to-low pressure isolation barriers. Additionally, these check valves are backed up by normally closed motor-operated gate valves.

Although they are backed up by normally closed motor-operated gate valves, periodic leakage testing is currently performed for the following hot leg and hi-head check valves as stated in Table 5.2-7 and Section 6.3.4.2:

RH-V-0050,	RH-V-0051,	RH-V-0052,	RH-V-0053
SI-V-0081,	SI-V-0082,	SI-V-0086,	SI-V-0087
SI-V-0106,	SI-V-0110,		
SI-V-0140,	SI-V-0144,	SI-V-0148,	SI-V-0152 and S-IV-0156

The check valves in the cold leg injection paths do form high-to-low pressure isolation barriers.

Appropriate connections for leak testing these valves are shown in Figure 5.4-9, Figure 5.4-10, Figure 5.4-11, and Figure 6.3-1, Figure 6.3-2, Figure 6.3-3, Figure 6.3-4, Figure 6.3-5.

Each inlet line to the RHRS is equipped with a pressure relief valve designed to relieve the combined flow of all the charging pumps at the relief valve set pressure. These relief valves also protect the system from inadvertent overpressurization during plant cooldown or startup.

Relief valves have been installed to provide thermal relief protection for water trapped between the redundant RHR suction isolation valves (RC-V22, -V23 in line 13-1-2501-12, and RC-V87, -V88 in line 58-1-2501-12). These lines and valving will relieve excessive pressure to the pressurizer relief tank.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve designed to relieve the maximum possible back leakage through the valves isolating the RHRS from the RCS.

The RHRS is designed to be operable from the control room with limited action outside the control room for normal operation. Manual operations required of the operator are: unlocking and closing the circuit breakers for the RHR suction isolation valves, opening the suction isolation valves, positioning the flow control valves downstream of the RHRS heat exchangers, and starting the RHRS pumps. Because of its redundant two train design, the RHRS is designed to accept all major component single failures, with the only effect being an extension in the required cooldown time. For two low probability electrical system single failures i.e., failure in the suction isolation valve interlock circuitry, or diesel generator failure in conjunction with loss of offsite power; limited operator action outside the control room is required to open the suction isolation valves. Manual actions are discussed in further detail in Subsections 5.4.7.2f and 5.4.7.2g. The RHRS motor-operated isolation valves located inside Containment are not susceptible to flooding following a steam line break or a loss-of-coolant accident. Although Westinghouse considers it to be of low probability, spurious operation of a single motor-operated valve can be accepted without loss of function, as a result of the redundant two train design.

In its normal function, the RHRS transfers heat from the Reactor Coolant System to the Component Cooling Water System during normal plant cooldown (two trains operating) and during plant maintenance/refueling (one train operating). For this function, connections between the two trains of residual heat removal are isolated to prevent a passive failure from affecting more than one loop.

The RHRS also serves as the low pressure pump suction of the Emergency Core Cooling System (ECCS). As discussed in Subsection 6.3.2.5, the system has been designed and proven by analysis as having the ability to withstand any single credible active failure during the injection phase (post-LOCA), or any active or passive failure during the recirculation phase. This design approach is consistent with ANSI/ANS 58.9-1981 "Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems."

Additional information about ECCS design protection for a passive failure is contained in Subsection 6.3.2.5b and a single passive failure analysis is presented in Table 6.3-6.

Missile protection, protection against dynamic effects associated with the postulated rupture of piping, and seismic design are discussed in Sections 3.5, 3.6(N), respectively.

5.4.7.2 System Design

a. Schematic Piping and Instrumentation Diagrams

The RHRS, as shown in Figure 5.4-9, Figure 5.4-10, Figure 5.4-11 and Figure 5.4-12, consists of two residual heat exchangers, two residual heat removal pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet lines to the RHRS are connected to the hot legs of two reactor coolant loops, while the return lines are connected to the cold legs of each of the reactor coolant loops. These return lines are also the ECCS low head injection lines (see Figure 6.3-1, Figure 6.3-2, Figure 6.3-3, Figure 6.3-4 and Figure 6.3-5).

The RHRS suction lines are isolated from the RCS by two motor-operated valves in series located inside the Containment. Each discharge line is isolated from the RCS by two check valves located inside the Containment and by a normally-open motor-operated valve located outside the Containment. (The check valves and the motor-operated valve on each discharge line are not part of the RHRS; these valves are shown as part of the ECCS, see Figure 6.3-1.)

During RHRS operation, reactor coolant flows from the RCS to the residual heat removal pumps, through the tube side of the residual heat exchangers, and back to the RCS. The heat is transferred to the component cooling water circulating through the shell side of the residual heat exchanger.

Coincident with operation of the RHRS, a portion of the reactor coolant flow may be diverted from downstream of the residual heat exchangers to the Chemical and Volume Control System (CVCS) low pressure letdown line for cleanup and/or pressure control. By regulating the diverted flowrate and the charging flow, the RCS pressure may be controlled. Pressure regulation is necessary to maintain the pressure range dictated by the fracture prevention criteria requirements of the reactor vessel and by the Number 1 seal differential pressure and net positive suction head requirements of the reactor coolant pumps.

The RCS cooldown rate is manually controlled by regulating the reactor coolant flow through the tube side of the residual heat exchangers. The flow control valve in the bypass line around each residual heat exchanger automatically maintains a constant return flow to the RCS. Instrumentation is provided to monitor system pressure, temperature and total flow.

The RHRS is also used for filling the refueling cavity before refueling. Also, after refueling operations, the RHRS is used to pump the water back to the refueling water storage tank until the water level is brought down to the flange of the reactor vessel. The remainder of the water is removed via a drain connection at the bottom of the refueling canal.

When the RHRS is in operation, the water chemistry is the same as that of the reactor coolant. Provision is made for the Process Sampling System to extract samples from the flow of reactor coolant downstream of the residual heat exchangers. A local sampling point is also provided on each residual heat removal train between the pump and heat exchanger.

The RHRS also functions, in conjunction with the high head portion of the ECCS, to provide injection of borated water from the refueling water storage tank into the RCS cold legs during the injection phase following a loss-of-coolant accident.

In its capacity as the low head portion of the ECCS, the RHRS provides long-term recirculation capability for core cooling following the injection phase of the loss-of-coolant accident. This function is accomplished by aligning the RHRS to take fluid from the containment sump, cool it by circulation through the residual heat exchangers, and supply it to the core directly as well as via the centrifugal charging pumps and safety injection pumps.

The use of the RHRS as part of the ECCS is more completely described in Section 6.3.

The RHR pumps, in order to perform their ECCS function, are interlocked to start automatically on receipt of a safety injection signal (see Section 6.3).

Each RHR loop is isolated from the Reactor Coolant System by two redundant motor-operated valves. One valve in each loop is located inside the missile barrier and is powered from Train B. The redundant valve is located outside the missile barrier and is powered from Train A. This ensures that isolation can be maintained between the RHR and the RCS in the event of a single failure of the power supply.

The RHR suction isolation valves in each inlet line from the RCS are separately interlocked to prevent being opened when RCS pressure is greater than that which could result in the RHR system design pressure being exceeded. (This includes the effects of instrument uncertainty and bistable deadband.) Refer to the Station Technical Specifications. These interlocks are described in more detail in Subsections 5.4.7.2d and 7.6.2 (see Figure 5.4-7).

The RHR suction isolation valves are also interlocked to prevent their being opened unless the isolation valves in the recirculation lines from the residual heat exchanger outlets to the suctions of the safety injection pumps and centrifugal charging pumps are closed (see Figure 5.4-7).

Failure of an interlock will result in failure of a valve to open.

Failures of interlocks are inherently addressed in the failure modes and effects analyses by addressing the valves that are controlled by the interlocks.

The motor-operated valves in the RHR miniflow bypass lines are interlocked to open when the RHR pump discharge flow is less than 750 gpm and close when the flow exceeds approximately 1400 gpm (see Figure 5.4-7, sh. 2).

b. Equipment and Component Descriptions

The materials used to fabricate RHRS components are in accordance with the applicable code requirements. All parts of components in contact with borated water are fabricated or clad with austenitic stainless steel or equivalent corrosion resistant material. Component parameters are given in Table 5.4-8.

1. Residual Heat Removal Pumps

Two pumps are installed in the RHRS. The pumps are sized to deliver reactor coolant flow through the residual heat removal heat exchangers to meet the plant cooldown requirements. The use of two separate residual heat removal trains assures that cooling capacity is only partially lost should one pump become inoperative.

The residual heat removal pumps are protected from overheating and loss of suction flow by miniflow bypass lines that assure flow to the pump suction. A valve located in each miniflow line is controlled by a signal from the flow switches located in each pump discharge header. The control valves open when the residual heat removal pump discharge flow is less than approximately 750 gpm and close when the flow exceeds approximately 1400 gpm.

A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high pressure alarm is also provided in the control room.

The two pumps are vertical, centrifugal units with mechanical seals on the shafts. All pump surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

The residual heat removal pumps also function as the low head safety injection pumps in the ECCS (see Section 6.3 for further information and for the residual heat removal pump performance curves).

2. Residual Heat Exchangers

Two residual heat exchangers are installed in the system. The heat exchanger design is based on heat load and temperature differences between reactor coolant and component cooling water existing 24 hours after reactor shutdown when the temperature difference between the two systems is small.

The installation of two heat exchangers in separate and independent residual heat removal trains assures that the heat removal capacity of the system is only partially lost if one train becomes inoperative.

The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

The residual heat exchangers also function as part of the ECCS (see Section 6.3).

3. Residual Heat Removal System Valves

Valves that perform a modulating function are equipped with two sets of packings and an intermediate leakoff connection that discharges to the drain header.

Manual and motor-operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open.

Leakage connections are provided where required by valve size and fluid conditions.

c. System Operation

1. Plant Startup

Generally, while at cold shutdown condition, decay heat from the reactor core is being removed by the RHRS. The number of pumps and heat exchangers in service depends upon the heat load at the time.

At initiation of plant heatup, the RHRS remains aligned to the RCS in order to maintain a low pressure letdown path to the CVCS. This flowpath may provide RCS pressure control while the pressurizer heaters are forming the steam bubble and heating the pressurizer. As the reactor coolant pumps are started, their thermal input heats the reactor coolant inventory. Once the pressurizer steam bubble formation is complete, the RHRS is isolated from the RCS and aligned for operation as part of the ECCS.

2. Power Generation and Hot Standby Operation

During power generation and hot standby operation, the RHRS is not in service but is aligned for operation as part of the ECCS.

3. Plant Cooldown

Plant cooldown is defined as the operation which brings the reactor from no-load temperature and pressure to cold conditions.

The initial phase of plant cooldown is accomplished by transferring heat from the RCS to the steam and power conversion system through the use of the steam generators.

When the reactor coolant temperature and pressure are reduced to approximately 350°F and 362 psig, approximately four hours after reactor shutdown, the second phase of cooldown starts with the RHRS being placed in operation.

Startup of the RHRS includes a warmup period during which reactor coolant flow through the heat exchangers is limited to minimize thermal shock. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through the residual heat exchangers. By adjusting these control valves downstream of the residual heat exchangers, the mixed mean temperature of the return flows is controlled. Coincident with the manual adjustment of flow through the heat exchangers, each heat exchanger bypass valve is automatically regulated to give the required total flow.

The controls for the heat exchanger flow control valves and bypass valves are located on the main control board. During normal operation, when the system is aligned for safety injection, the flow control valve must be in full-open position and the bypass valve must be in full-closed position. A bypassed/inoperable status alarm is provided in the control room in case these valves are not aligned for safety injection.

During reduced inventory operations, RHR pump flow is administratively controlled to limit flow such that **vortexing or air entrainment** do not occur.

The reactor cooldown rate is limited by RCS equipment cooling rates based on allowable stress limits, as well as the operating temperature limits of the Component Cooling Water System. As the reactor coolant temperature decreases, the reactor coolant flow through the residual heat exchangers is increased by adjusting the control valve in each heat exchanger's tube side outlet line.

The maximum available cooldown rate would not exceed 200°F/hr following initiation of residual heat cooling (350°F).

Additionally, maintaining such a cooldown rate is difficult if not impossible. A cooldown rate of 50° per hour becomes difficult, if not impossible, to maintain at temperatures below 250°F.

Analysis has shown that, from a stress standpoint, a cooldown rate greater than 200°F/hr is acceptable for such a hypothetical cooldown from 350°F to 250°F even though, as discussed above, a rate of cooldown of 50°F/hr may not be achievable below 250°F.

As cooldown continues, the pressurizer is filled with water and the RCS is operated in the water solid condition.

At this stage, pressure control is accomplished by regulating the charging flow rate and the rate of letdown from the RHRS to the CVCS.

After the reactor coolant pressure is reduced and the temperature is 125°F or lower, the RCS may be opened for refueling or maintenance.

4. Refueling

After the reactor vessel head is lifted and the control rod drive shafts are verified disengaged, the reactor vessel head is removed to the reactor vessel storage stand.

The refueling water is then introduced into the reactor vessel through the RHRS and into the refueling cavity through the open reactor vessel. After the water level reaches the normal refueling level, the refueling water storage tank supply valves are closed, and residual heat removal is resumed.

During refueling, the RHRS is maintained in service with the number of pumps and heat exchangers in operation as required by the heat load.

Following refueling, one of the residual heat removal pumps is used to drain the refueling cavity to the top of the reactor vessel flange by pumping water from the RCS to the refueling water storage tank. The vessel head is then replaced and the normal RHRS flowpath re-established. The remainder of the water is removed from the refueling canal via a drain connection in the bottom of the canal.

d. Control

Each inlet line to the RHRS from the RCS is equipped with a pressure relief valve sized to relieve the combined flow of all the charging pumps at the relief valves set pressure. These relief valves also protect the system from inadvertent overpressurization during plant cooldown or startup. Each valve has a relief flow capacity of 900 gpm at a set pressure of 450 psig. An analysis has been conducted to confirm the capability of the RHRS relief valve to prevent overpressurization in the RHRS. All credible events were examined for their potential to overpressurize the RHRS. These events included normal operating conditions, infrequent transients, and abnormal occurrences.

The most severe credible overpressure transient is the mass input transient resulting from one centrifugal charging pump operating in an unthrottled condition with flow to the Reactor Coolant System while letdown flow is isolated. The capacity of a single RHRS inlet relief valve is sufficient to satisfy RHRS overpressure requirements for this transient during the hot shutdown and cold shutdown operational modes. Procedures and administrative controls ensure that more severe RHRS overpressure transients do not occur during RHRS operations.

Two situations were analyzed to confirm the adequacy of the RHRS relief valve to prevent overpressurization of the RHRS. The first considers the RHRS in the initial phase of RCS cooldown. RCS temperature and pressure are 350°F and 400 psig respectively, and one centrifugal charging pump is in operation. The operator initiates RHRS operation by opening one inlet line and starting the corresponding RHR pump. At this time, a complete loss of plant air occurs, the charging line flow control valve fails open, and the low pressure letdown flow control valve fails closed. The maximum charging pump injection rate is approximately 400 gpm at the relief valve set pressure of 450 psig.

The second transient consists of the RCS in the final stages of cooldown. RCS temperature and pressure are less than 200°F and 450 psig, respectively. The additional conservatism of a second centrifugal charging pump is assumed. The combined flow from both charging pumps is less than 600 gpm.

As each RHRS inlet relief valve has a capacity of 900 gpm at a set pressure of 450 psig, sufficient margin is present to bound liquid and two-phase relief rate uncertainties.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the valves separating the RHRS from the RCS. Each valve has a relief flow capacity of 20 gpm at a set pressure of 600 psig. These relief valves are located in the ECCS (see Figure 6.3-1).

The fluid discharge by the suction side relief valves is collected in the pressurizer relief tank. The fluid discharged by the discharge side relief valves is collected in the primary drain tank of the equipment and floor drain system.

The design of the RHRS includes two motor-operated gate isolation valves in series on each inlet line between the high pressure RCS and the lower pressure RHRS. They are closed, with power locked out, during normal operation and are only opened for residual heat removal during a plant cooldown after the RCS pressure is reduced to 362 psig or lower and RCS temperature is reduced to approximately 350°F. During a plant startup, the inlet isolation valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above 400 psig. These isolation valves are provided with a "prevent-open" interlock that is designed to prevent possible exposure of the RHRS to normal RCS operating pressure. The two inlet isolation valves in each residual heat removal subsystem are separately and independently interlocked with pressure signals to prevent their being opened whenever the RCS pressure is greater than approximately that which could result in the RHR system design pressure being exceeded. (This includes the effects of instrument uncertainty and bistable deadband.) Refer to the Station Technical Specifications. Power to the suction line isolation valve motor operators is locked out to prevent spurious opening. An alarm is actuated if a suction isolation valve is open when RCS pressure is above 362 psig as a backup to administrative control to ensure that both valves, rather than just one, are closed at normal RCS operating pressure, thereby ensuring compliance with reactor coolant pressure boundary isolation criteria.

The use of two independently powered motor-operated valves in each of the two inlet lines, along with two independent pressure interlock signals assures a design which meets applicable single failure criteria. Not only more than one single failure but also different failure mechanisms must be postulated to defeat the function of preventing possible exposure of the RHR system to normal RCS operating pressure. These protective interlock and alarm designs, in combination with plant operating procedures, provide diverse means of accomplishing the protective function. For further information on the instrumentation and control features, see Subsection 7.6.2.

Valve position indication is provided on the main control board for both the open and closed positions for the RHR suction line isolation valves and RHR recirculation valves. On loss of suction, the RHR system would go into a recirculation mode, thus protecting the operating RHR pump. Alarms are provided that will actuate, if either suction valve for an operating RHR pump is not fully open and if the flow through the RHR pump is below the minimum expected. Indication of reduced flow and pump amperage is also provided in the control room for the operator to observe during normal system surveillance.

Isolation of the low pressure RHRS from the high pressure RCS is provided on the discharge side by two check valves in series. These check valves are located in the ECCS and their testing is described in Subsection 6.3.4.2.

e. Applicable Codes and Classifications

The entire RHRS is designed as Nuclear Safety Class 2, with the exception of the suction isolation valves which are Safety Class 1. Component codes and classifications are given in Section 3.2.

f. System Reliability Considerations

General Design Criterion 34 requires that a system to remove residual heat be provided. The safety function of this system is to transfer fission product decay heat and other residual heat from the core at a rate sufficient to prevent fuel or pressure boundary design limits from being exceeded. Safety grade systems are provided in the plant design, both NSSS scope and BOP scope, to perform this function. The safety grade systems which perform this function for all plant conditions except a LOCA are: the RCS and steam generators, which operate in conjunction with the Emergency Feedwater System; the steam generator safety and power-operated relief valves; and the RHRS which operates in conjunction with the Component Cooling Water System and the Service Water System. For LOCA conditions, the safety grade system which performs the function of removing residual heat from the reactor core is the ECCS, which operates in conjunction with the Component Cooling Water System and the Service Water System.

The Emergency Feedwater System, along with the steam generator safety and power-operated relief valves, provides a completely separate, independent, and diverse means of performing the safety function of removing residual heat, which is normally performed by the RHR system when RCS temperature is less than 350°F. The Emergency Feedwater System is capable of performing this function for an extended period of time following plant shutdown.

The RHR system is provided with two residual heat removal pumps, and two heat exchangers arranged in two separate, independent flow paths. To assure reliability, each residual heat removal pump is connected to a different vital bus. Each residual heat removal train is isolated from the RCS on the suction side by two motor-operated valves in series with each valve receiving power via a separate motor control center and from a different vital bus. Each suction isolation valve is also interlocked to prevent exposure of the RHR system to the normal operating pressure of the RCS (see Subsection 5.4.7.2d).

RHR system operation for normal conditions and for major failures is accomplished from the control room. Limited action outside the control room is required to prepare the RHRS for operation as discussed in Subsection 5.4.7.2f. The redundancy in the RHR system design provides the system with the capability to maintain its cooling function even with major single failures, such as failure of an RHR pump, valve, or heat exchanger, since the redundant train can be used for continued heat removal.

Although such major system failures are within the system design basis, there are other less significant failures which can prevent opening of the RHR suction isolation valves from the control room. Since these failures are of a minor nature, improbable to occur, and easily corrected outside the control room, with ample time to do so, they have been realistically excluded from the engineering design basis. Such failures are not likely to occur during the limited time period in which they can have any effect (i.e., when opening the suction isolation valves to initiate RHR operation); however, even if they should occur, they have no adverse safety impact and can be readily corrected. In such a situation, the Emergency Feedwater System and steam generator power-operated relief valves can be used to perform the safety function of removing residual heat. In fact, they can be used to continue the plant cooldown below 350°F, until the RHR system is made available.

One failure of this type is a failure in the interlock circuitry which is designed to prevent exposure of the RHR system to the normal operating pressure of the RCS (see Subsection 5.4.7.2d). In the event of such a failure, RHR system operation can be initiated by defeating the failed interlock through corrective action at the solid-state protection system cabinet or by taking local control at the remote safe shutdown control panels.

The other type of failure which can prevent opening the RHR suction isolation valves from the control room is a failure of an electrical power train. Such a failure is extremely unlikely to occur during the few minutes out of a year's operating time during which it can have any consequence.

To ensure operation of at least one 100 percent RHR train in the event of loss of a single power supply, a temporary power connection will be provided to the MCC compartment of the RHR valve whose power supply has failed. This temporary power connection will be a cable from a designated compartment of the unaffected redundant MCC to the compartment of the affected valve. The MCCs involved are MCC E521 and E621. See Figure 8.3-27 for location of MCCs. Terminals are provided in the MCC compartments to facilitate a temporary connection to the line side of the breaker-starter combination of valve. Control of the valve is still from the MCB. The individual MCC compartments are capable of being withdrawn from the bus bar connection and secured in the disconnected position, which precludes a faulted MCC from affecting the operation of the valve having the temporary connection.

The only impact of either of the above types of failures is some delay in initiating residual heat removal operation, while action is taken to open the residual heat removal suction isolation valves. This delay has no adverse safety impact because of the capability of the Emergency Feedwater System and steam generator power-operated relief valves to continue to remove residual heat, and in fact to continue plant cooldown.

A failure mode and effects analysis of the Residual Heat Removal System is provided in Table 5.4-17.

g. Manual Actions

The RHRS is designed to be operable from the control room for normal operation. Manual operations required of the operator are: energizing the valve controls, opening the suction isolation valves, positioning the flow control valves downstream of the RHR heat exchangers, and starting the residual heat removal pumps.

The RHR suction line isolation valves, which are normally de-energized with the circuit breaker locked open, will require an operator to energize these valves just before establishing RHR. There is sufficient time during the plant cooldown to send an operator to the MCC to energize these valves and then operate them from the control room.

Manual actions required outside the control room, under conditions of single failure, are discussed in Subsection 5.4.7.2f.

h. Passive Failures Evaluation

While in the shutdown cooling mode, only the RHR and CC systems are utilized to maintain reactor core cooling. Because the RHR system operates as a "high energy" system for less than 2 percent of the time which it operates, it is considered a "moderate energy" system in accordance with BTP ASB 3-1, Appendix A - Definitions.

1. The maximum discharge rate due to a moderate energy line break of the above two systems is as follows:

   (a) For RHR lines RC-13-2-601-12" and RC-58-2-601-12" (see note below), the maximum discharge rate is 121.2 cfm.

   (b) For CC lines C-775-5-152-20" and CC-829-2-152-20", the maximum discharge rate is 85.4 cfm.

   NOTE: Cracks in the 14" and 16" sections of the RHR suction lines are not postulated because maximum stresses in these lines are less than those specified in BTP MEB 3-1, Subsection B.2.c.(1).

2. In determining the time available for recovery, a conservative estimate was made as to how much RCS inventory would have to be lost before the RHR system would loose its ability to perform core cooling. Although the RHR system will continue to perform its function with the RCS water level as low as the centerline of the vessel loop nozzles, only the inventory associated with 1) the pressurizer (at 25 percent level), (2) the surge line, (3) the primary side of the steam generators and (4) reactor vessel above the loop nozzles was considered. This inventory conservatively amounts to more than 5840 cu. ft.

   Assuming that a loss of 5840 cu. ft would begin to degrade the RHR system's ability to maintain core cooling, and neglecting any mass input associated with the charging system or operator action associated with manually initiating safety injection, a leak rate of 121.2 cfm would take over 48 minutes to reduce the RCS inventory by 5840 cu. ft.

   The effect on core cooling during this time would be a reduction to the RCS cooldown rate.

   A leak in the component cooling lines supplying the RHR heat exchangers of the magnitude presented in 1(b) above, would empty the CC surge tank in about 1 minute 43 seconds. Although air introduction into the CC system and cavitation of the CC pump would commence at this point, because of the large volume of water remaining in the CC return lines and still supplying the CC pump suction, a complete loss of CC flow to the RHR heat exchanger is not expected at this time.

   The effect upon core cooling for this postulated event would be a reduction in the cooldown rate as the plant continues its cooldown on the remaining unaffected RHR/CC systems.

3. Assuming the leak rate postulated in the response to 1(a) above, a low pressurizer level alarm would be generated in about 41 seconds. A second lo-lo pressurizer level alarm would be generated less than 25 seconds after the first alarm. Additionally, a high sump level alarm in the affected vault would be actuated approximately 1 minute from the commencement of the leak.

For the postulated leak in the component cooling lines identified in 1(b) above, a component cooling surge tank low level alarm would activate in less than 1 minute. Within an additional 25 seconds, a lo-lo surge tank level alarm would also annunciate. Within 1½ minutes from the onset of the leak, a high sump level alarm in the affected vault would also be actuated.

For either of the above postulated leaks, the operator response would be to isolate the affected system and continue the RCS cooldown using the redundant, nonaffected RHR and Component Cooling System.

i. Functional Requirements

The safe shutdown design basis for Seabrook is hot standby. The cold shutdown capability of the plant has been evaluated in order to demonstrate how the plant can be brought to a cold shutdown condition using only safety-grade equipment following a safe shutdown earthquake, loss of offsite power and the most limiting single failure. Under such conditions, the plant is capable of achieving RHR system operating conditions (approximately 350°F and 400 psig) in approximately nine to ten hours, which includes remaining in hot standby for up to four hours.

The selected method of achieving the cold shutdown condition for Seabrook is natural circulation without RCS letdown. Core decay heat and cooldown energy is removed by a combination of steam generator atmospheric venting with secondary coolant makeup from the condensate storage tank via the Emergency Feedwater System. Reactivity control is achieved by making up to the RCS from borated water sources, taking advantage of the reactor coolant's specific volume decrease as the RCS is cooled to ≤ 350°F. At this point, the RHR system is placed in service to complete the cooldown to cold shutdown conditions subsequent to RCS depressurization to approximately 400 psig.

This scenario is detailed more fully as follows:

1. System Energy Removal

To maintain the RCS in hot standby (constant average temperature), core decay heat energy must be removed at a rate equivalent to fission product energy production. To cool down the RCS, the energy contained in the reactor coolant and all system components must also be removed.

The combination of decay heat energy and mass energy of the coolant and system components is initially removed by heat transfer to the steam generators. This heat transfer is made possible by natural convection with the reactor core as the heat source and the steam generators as the heat sink. Since the steam generators are located at a higher elevation than the reactor core, a natural thermosiphon is created. The resulting natural flow created for Seabrook is in the order of three to four percent of normal forced flow (reactor coolant pumps operating) and is more than adequate to transfer decay energy and mass energy to the steam generators. Simple analysis indicates that RCS loop ΔT values are in the order of 15 - 30°F which are typical of power operation with forced convection. These values of natural circulation flow and loop ΔTs correspond well with actual natural circulation tests conducted at similar PWRs.

To ensure this natural circulation flow in the RCS, the steam generators must be maintained as a heat sink. To achieve this condition, the safety-grade steam generator power-operated atmospheric relief valves are used to vent vaporized secondary coolant. The rate of venting is adjusted by the operator to set the RCS cooldown rate to a value $\leq$ 50°F/hour. Secondary coolant makeup is provided via the Emergency Feedwater System from the seismic Category I condensate storage tank. The minimum volume which is available for this scenario is the 196,000 gallons which is dedicated for EFW use. The heat removal capability of this secondary coolant volume is determined by heating this coolant to saturation in the steam generators and removing the latent heat of vaporization by venting the steam generator. The total worth of the 200,000 gallons in the condensate storage tank in terms of RCS heat removal is, on a Btu basis, sufficient to accommodate a four-hour period at hot standby plus a cooldown to $\leq$ 350°F.

Since it is recognized that continued mass addition to the RCS is desirable for reactor coolant pump seal injection and that the letdown system would be isolated, the RCS cooldown would normally commence without a four hour period at hot standby. This means that the portion of the condensate storage tank volume allotted by design for a four-hour period at hot standby is not actually required for this scenario. This adds additional conservatism to the secondary coolant volume of 196,000 gallons provided for this cooldown.

Seabrook is classified at a T_{cold} plant by the NSSS vendor. This means that with normal forced convection (reactor coolant pumps running) the temperature of the coolant under the vessel head remains close to T_{cold} because a small portion of the vessel inlet flow is diverted into this region. Under natural circulation conditions, the coolant under the vessel head does not remain at T_{cold}. Vendor data shows that with a 50°F/hour RCS cooldown rate, the cooldown rate of the coolant under the head would be approximately 34°F/hour without the aid of the nonsafety grade control rod drive mechanism fans. For a 50°F/hour cooldown rate, the RCS could be depressurized to 400 psig for RHR operation in about five hours. Although bulk coolant temperature under the head would be at a higher temperature, no steam voiding would occur based on a 34°F/hour cooldown rate.

When the steam generators are being used as the reactor heat sink during the cooldown to 350°F, a single failure of any active component does not render all steam generators ineffective as a heat sink. Either of the two emergency feedwater pumps has sufficient capacity to provide for all steam generator makeup requirements. The steam generator power-operated relief valves have manual loading stations should remote operation be lost. The Emergency Feedwater System and the steam generator power-operated relief valves are seismic Category I subsystems.

The second stage of the cooldown is from 350°F to cold shutdown. During this stage, the RHR system is brought into operation. Circulation of the reactor coolant is provided by the RHR pumps, and the heat exchangers in the RHR system act as the means of heat removal from the RCS. In the RHR heat exchangers, the residual heat is transferred to the Component Cooling Water System which ultimately transfers the heat to the Service Water System.

The RHR system is a fully redundant system. Each RHR subsystem includes one RHR pump and one RHR heat exchanger. Each RHR pump is powered from different emergency power trains and each RHR heat exchanger is cooled by a different component cooling water system loop. The Component Cooling Water and Service Water Systems are both designed to seismic Category I.

If any component in one of the RHR subsystems were rendered inoperable as the result of a single failure, cooldown of the plant could still be continued.

At Seabrook, a single RHR cooling loop can be cut in under full flow conditions with all air-operated temperature control valves in their failed (maximum cooling) positions. The resulting maximum RCS cooldown rate would not exceed 50°F/hour; therefore, special control functions are not necessary.

2. Reactivity and Inventory Control

Core reactivity is controlled during the cooldown by adding borated water to the RCS in conjunction with the cooldown. As the cooldown progresses, the specific volume of the reactor coolant decreases. The resulting coolant contraction allows the addition of borated water to the RCS to maintain a constant pressurizer level during cooldown.

Boration is accomplished using portions of the Chemical and Volume Control Systems (CVCS). At the beginning of the cooldown, the operators align one of the two boric acid tanks to the suction side of the centrifugal charging pumps. One of the two centrifugal charging pumps would inject borated water to the RCS through the reactor coolant pump seal injection flow path and/or the boron injection portion of the safety injection system. The capacity of one boric acid tank is sufficient to make up for reactor coolant contraction down to and beyond the point of cutting in RHR. The concentration of boron in the boric acid tank is maintained between 7000-7700 ppm H_3BO_3. At the minimum concentration of 7000 ppm, gross shutdown margin in the order of 2-5 percent $\Delta K/K$ is maintained during the cooldown considering most limiting conditions (end-of-life, most reactive rod stuck out).

Makeup in excess of that required for boration can be provided from the refueling water storage tank (RWST) using the centrifugal charging pumps and the same injection flow paths as described for boration. Two independent motor-operated valves, each powered from different emergency diesels, transfer the suction of the charging pumps to the RWST.

The two boric acid tanks, two centrifugal charging pumps and the associated piping are of seismic Category I design and are independently train associated.

Under natural circulation conditions, the RCS loop transport time is approximately five minutes and the coolant Reynold's number is in the order of 25,000. Since either boration path distributes RCS makeup to all four loops, adequate mixing and distribution of boron can be assumed.

Provisions are made to obtain RCS coolant samples to determine boron concentration during the cooldown. This can be done considering single failure and without the need for a containment entry. The onshift chemistry technician would aid the operators in following boron concentration.

3. RCS Depressurization

For this scenario, RCS depressurization is accomplished by opening one of the two safety-grade pressurizer power-operated relief valves. The discharge is directed to the pressurizer relief tank where it is condensed and cooled.

The depressurization process is integrated with the cooldown process to maintain the RCS within normal pressure-temperature limits. Just before cutting in an RHR cooling loop at 350°F, the RCS is depressurized to ≤ 400 psig.

Analysis shows that the pressurizer relief tank can accommodate the RCS depressurization to the RHR cut in pressure of 400 psig without opening the rupture discs. Operation of the Pressurizer Relief Tank Cooling System is not required.

Single failure of one pressurizer power-operated relief valve does not prevent RCS depressurization. Isolation valves are provided for each power-operated relief valve should it fail to properly reseat after depressurization.

4. Instrumentation

Class 1E instrumentation is available in the control room to monitor the key functions associated with achieving cold shutdown and includes the following:

(a) RCS wide-range temperature, T_H and T_C

(b) RCS wide range pressure

(c) Pressurizer water level

(d) Steam generator water level (per steam generator, narrow and wide range)

(e) Steam line pressure (per steam line)

(f) Condensate storage tank level

(g) Boric acid tank level (per boric acid tank)

(h) Emergency feedwater flow.

This instrumentation is sufficient to monitor the key functions associated with cold shutdown and to maintain the RCS within the design pressure, temperature, and inventory relationships.

5. Summary of Manual Actions

This scenario can be easily accomplished with the normal onshift complement and on-call personnel.

Depending on the nature of any single failure that may or may not be present, the following manual actions could be required outside the main control room:

(a) Manual valve alignment to feed boric acid tank contents to the suction of the operating centrifugal charging pump prior to cooldown. This would only be necessary if the normal feed path was inoperable or if direct gravity feed from the boric acid tank to the centrifugal charging pump was desired. This task is simple and would take less than ten minutes. The backup borated water source (RWST) can be lined up from the control room until this task is complete.

(b) Prior to cutting in an RHR cooling loop when RCS temperature and pressure are reduced to ≤ 350°F and 400 psig, the operator must manually close the power supply breakers for the RHR suction valves (RC-V22, -23, -87, and -88). As stated in the response to BTP RSB 5-1, this task would take 2-4 minutes to complete, although there is no real need to do this quickly. The RHR suction valve power supply breakers are locked open during normal plant operation to reduce concerns relating to "interfacing LOCAs" during a postulated fire event.

Should either Train A or Train B power sources be unavailable, the associated RHR suction valve in each RHR system would be inoperable. In this case, provisions are made to power the affected valve from the opposite train power sources. Here again, the time to complete this task does not seriously hamper transition to RHR system cooling. The on-call maintenance electrician would aid operators in completing this task.

5.4.7.3 Performance Evaluation

The performance of the RHRS in reducing reactor coolant temperature is evaluated through the use of heat balance calculations on the RCS, and the Component Cooling Water System at stepwise intervals following the initiation of residual heat removal operation. Heat removal through the residual heat removal and component cooling water heat exchangers is calculated at each interval by use of standard water-to-water heat exchanger performance correlations. The resultant fluid temperatures for the Residual Heat Removal and Component Cooling Water Systems are calculated and used as input to the next interval's heat balance calculation.

Assumptions used in the series of heat balance calculations describing a normal plant RHR cooldown are as follows:

a. Residual heat removal operation is initiated four (4) hours after reactor shutdown.

b. Residual heat removal operation begins at a reactor coolant temperature of 350°F.

c. Thermal equilibrium is maintained throughout the RCS during the cooldown.

d. The component cooling water flow through a RHR heat exchanger is 3000 gpm instead of the design value of 5000 gpm.

e. Component cooling water temperature during cooldown is limited to a maximum of 102°F.

f. Cooldown rates of 50°F per hour are not exceeded.

Cooldown curves calculated using this method are provided for the case of all RHR components operable (Figure 5.4-5) and for the case of a single train RHR cooldown (Figure 5.4-6).

5.4.7.4 Preoperational Testing

The RHR system is tested prior to initial core loading for integrity, performance and operability. The preoperational tests of the system are described in Chapter 14.

5.4.8 Reactor Water Cleanup System (BWR)

Not applicable to Seabrook.

5.4.9 Main Steam Line and Feedwater Piping

The Main Steam and Feedwater Systems are secondary cooling systems, and are isolated from the Reactor Coolant System by the steam generators. The main steam piping is described in Section 10.3; the feedwater piping is described in Subsection 10.4.7. Piping material specifications and test methods for both systems are described in Subsection 10.3.6.

5.4.10 Pressurizer

5.4.10.1 Design Basis

The general configuration of the pressurizer is shown in Figure 5.4-13. The design data of the pressurizer are given in Table 5.4-9. Codes and material requirements are provided in Section 5.2.

The pressurizer provides a point in the RCS where liquid and vapor can be maintained in equilibrium under saturated conditions to maintain pressure during steady state operations and limits pressure changes during transients.

a. Pressurizer Surge Line

The surge line is sized to minimize the pressure drop between the RCS and the pressurizer/safety valves with maximum allowable discharge flow from the safety valves, as necessary.

The surge line and the thermal sleeves are designed to withstand the thermal stresses which occur during operation.

The pressurizer surge line nozzle diameter is given in Table 5.4-9 and the pressurizer surge line dimensions are shown in Figure 5.1-3, sh.4.

b. Pressurizer

The volume of the pressurizer is equal to, or greater than, the minimum volume of steam, water, or total of the two which satisfies all of the following requirements:

1. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

2. The water volume is sufficient to prevent the heaters from being uncovered during a step load increase of 10 percent.

3. The steam volume is large enough to accommodate the surge resulting from 50 percent reduction of full load with automatic reactor control and 40 percent steam dump without the water level reaching the high level reactor trip point.

4. The steam volume is large enough to prevent water relief through the safety valves following a loss of load with the high water level initiating a reactor trip, without reactor control or steam dump.

5. The pressurizer will not empty following reactor trip and turbine trip.

6. The emergency core cooling signal is not activated during reactor trip and turbine trip.

5.4.10.2 Design Description

a. Pressurizer Surge Line

The pressurizer surge line connects the pressurizer to one reactor hot leg enabling continuous coolant volume pressure adjustments between the RCS and the pressurizer.

b. Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of low alloy steel, with austenitic stainless steel cladding on all internal surfaces exposed to the reactor coolant. A stainless steel liner or tube may be used in lieu of cladding in some nozzles.

The surge line nozzle connects the bottom of the pressurizer to an RCS hot leg. The electric heaters are located in the lower portion of the vessel, where they maintain the steam and water contents at equilibrium conditions. The heaters are removable for maintenance or replacement.

A thermal sleeve is provided to minimize thermal stresses in the surge line nozzle. A retaining screen is located above the nozzle to prevent any foreign matter from entering the RCS. Baffles in the lower section of the pressurizer prevent an insurge of cold water from flowing directly to the steam/water interface and assist in mixing.

Spray line nozzles, relief and safety valve connections are located in the top head of the vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves also can be operated manually by a switch in the control room.

A small continuous spray flow is provided through a manual bypass valve around the power-operated spray valves to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excessive cooling of the spray piping.

During an outsurge from the pressurizer, flashing of water to steam and generating of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. During an insurge from the RCS, the spray system, which is fed from two cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the setpoint of the power-operated relief valves for normal design transients.

Heaters are energized on high water level during insurge to heat the subcooled surge water that enters the pressurizer from the reactor coolant loop.

Material specifications are provided in Table 5.2-2 for the pressurizer, pressurizer relief tank, and the surge line. Design transients for the components of the RCS are discussed in Subsection 3.9(N).1. Additional details on the pressurizer design cycle analysis are given in Subsection 5.4.10.3e.

1. Pressurizer Instrumentation

   Refer to Chapter 7 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

2. Spray Line Temperatures

   Temperatures in the spray lines from the cold legs of two loops are measured and indicated. Alarms from these signals are actuated to warn the operator of low spray water temperature or indicate insufficient flow in the spray lines.

3. Safety and Relief Valve Discharge Temperatures

   Temperatures in the pressurizer safety valve discharge lines and relief valve manifold are measured and indicated. An increase in a discharge line temperature is an indication of leakage or relief through the associated valve.

5.4.10.3 Design Evaluation

a. System Pressure

Whenever a steam bubble is present within the pressurizer, the RCS pressure will be maintained by the pressurizer. Analyses indicate that proper control of pressure is maintained for the normal operating conditions.

A safety limit has been set to ensure that the RCS pressure does not exceed the maximum transient value allowed under the ASME Code, Section III, and thereby assure continued integrity of the RCS components. Evaluation of plant conditions of operations, which follow, indicate that this safety limit is not reached.

During startup and shutdown, the rate of temperature change in the RCS is controlled by the operator. Heatup rate is controlled by pump energy and by the pressurizer electrical heating capacity. This heatup rate takes into account the continuous spray flow provided to the pressurizer. When the reactor core is in cold shutdown (except during initial pressurizer fill), the heaters are de-energized.

When the pressurizer is filled with water, e.g., during initial system heatup, and near the end of the second phase of plant cooldown, RCS pressure is maintained by the letdown flow rate via the RHRS.

b. Pressurizer Performance

The normal operating water volume at full load conditions is a percentage of the internal vessel volume. Under part load conditions, the water volume in the vessel is reduced for proportional reductions in plant load to accommodate the accompanying thermal contraction of the reactor coolant. The various plant operating transients are analyzed and the design pressure is not exceeded with the pressurizer design parameters as given in Table 5.4-9.

A comparison of the design basis operating conditions determined that the conditions the pressurizer was originally qualified for envelop the operation of Seabrook Unit 1 at a power level of 3678 MWt.

c. Pressure Setpoints

The RCS design and operating pressure together with the safety, power-operated relief and pressurizer spray valves setpoints, and the protection system pressure setpoints are listed in Table 5.4-10. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics.

d. Pressurizer Spray

Two separate, automatically controlled spray valves with remote manual overrides are used to initiate pressurizer spray. In parallel with each spray valve is a manual throttle valve which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open, and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping routed to the pressurizer forms a water seal which prevents the steam buildup back to the control valves. The spray rate is selected to prevent the pressurizer pressure from reaching the operating setpoint of the power-operated relief valves during a step reduction in power level of 15 percent of full load.

The pressurizer spray lines and valves are large enough to provide adequate spray using as the driving force the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop to utilize the velocity head of the reactor coolant loop flow to add to the spray driving force. The spray valves and spray line connections are arranged so that the spray will operate when one reactor coolant pump is not operating. The line may also be used to assist in equalizing the boron concentration between the reactor coolant loops and the pressurizer.

A flow path from the Chemical/Volume Control System (CVCS) to the pressurizer spray line is also provided. This additional facility provides auxiliary spray to the vapor space of the pressurizer during cooldown when the reactor coolant pumps are not operating.

The thermal sleeves on the pressurizer spray connection and the spray piping are designed to withstand the thermal stresses resulting from the introduction of cold spray water.

e. Pressurizer Design Analysis

The occurrences for pressurizer design cycle analysis are defined as follows:

1. The temperature in the pressurizer vessel is always, for design purposes, assumed equal to the saturation temperature for the existing RCS pressure, except in the pressurizer steam space subsequent to a pressure increase. In this case the temperature of the steam space will exceed the saturation temperature since an isentropic compression of the steam is assumed.

   The only exception of the above occurs when the pressurizer is filled water solid, such as, during plant startup and cooldown.

2. The temperature shock on the spray nozzle is assumed to equal the temperature of the nozzle minus the minimum spray line temperature and the temperature shock on the surge nozzle is assumed to equal the pressurizer water space temperature minus the hot leg temperature.

3. Pressurizer spray is assumed to be initiated instantaneously to its design value as soon as the RCS pressure increases 40 psi above the nominal operating pressure. Spray is assumed to be terminated as soon as the RCS pressure falls below the operating pressure plus 40 psi unless otherwise noted.

4. Unless otherwise noted, pressurizer spray is assumed to be initiated once per occurrence of each transient condition. The pressurizer surge nozzle is also assumed to be subject to one temperature transient per transient condition, unless otherwise noted.

5. Each upset condition transient results in a reactor trip. At the end of each transient, except the faulted conditions, the RCS is assumed to return to no-load conditions consistent with the pressure and temperature changes.

6. For design purposes, the following assumptions are made with respect to Condition III (Emergency Conditions) and Condition IV (Faulted Conditions) transients.

   (a) The plant eventually reaches cold shutdown conditions after all Condition IV transients and after the following Condition III transients: small loss-of-coolant accident and small steam line break.

   (b) For the other Condition III transients, the plant goes to hot shutdown until the condition of the plant is determined. It is then brought either to no-load conditions or to cold shutdown conditions, with pressure and temperature changes controlled within allowable limits.

7. Temperature changes occurring as a result of pressurizer spray are assumed to be instantaneous. Temperature changes occurring on the surge nozzle are also assumed to be instantaneous.

8. Whenever spray is initiated in the pressurizer, the pressurizer water level is assumed to be at the no-load level.

5.4.10.4 Tests and Inspections

The pressurizer is designed and constructed in accordance with the ASME Code, Section III.

To implement the requirements of the ASME Code, Section XI, the following welds are designed and constructed to present a smooth transition surface between the parent metal and the weld metal. The path is ground smooth for ultrasonic inspection.

a. Support skirt to the pressurizer lower head

b. Surge nozzle to the lower head

c. Nozzle to safe end attachment welds

d. All girth and longitudinal full penetration welds

e. Manway attachment welds.

The liner within the safe end nozzle region extends beyond the weld region to maintain a uniform geometry for ultrasonic inspection.

Peripheral support rings are furnished for the removable insulation modules.

The pressurizer quality assurance program is given in Table 5.4-11.

5.4.11 Pressurizer Relief Discharge System

The pressurizer relief discharge system collects, cools and directs for processing the steam and water discharged from the various safety and relief valves in the Containment. The system consists of the pressurizer relief tank, pressurizer relief tank pump, pressurizer relief tank heat exchanger, the safety and relief valve discharge piping, the relief tank internal spray header and associated piping, and the tank nitrogen supply, the vent to Containment and the drain to the equipment and floor drain system.

5.4.11.1 Design Bases

Codes and materials of the pressurizer relief tank and associated piping are given in Section 5.2. Design data for the tank are given in Table 5.4-12.

The system design is based on the requirement to condense and cool a discharge of pressurizer steam equal to 110 percent of the volume above the full power pressurizer water level setpoint. The system is not designed to accept a continuous discharge from the pressurizer. The volume of water in the relief tank is capable of absorbing the heat from the assumed discharge, with an initial temperature of 120°F. If the temperature in the tank rises during plant operation, the tank is cooled by spraying in cool water and draining out the warm mixture to the equipment and floor drain system. Alternatively, the tank may be cooled by circulating its contents through the pressurizer relief tank heat exchanger. The volume and pressure of nitrogen gas in the tank are selected to limit the maximum pressure following a design discharge to less than the minimum rupture disc relief pressure with conservative margin.

The vessel saddle supports and anchor bolt arrangement are designed to withstand the loadings resulting from a combination of nozzle loadings acting simultaneously with the vessel seismic and static loadings.

5.4.11.2 System Description

The piping and instrumentation diagram for the Pressurizer Relief Discharge System is given in Figure 5.1-4.

The steam and water discharged from the various safety and relief valves inside the Containment is routed to the pressurizer relief tank, if the discharged fluid is of reactor grade quality. Table 5.4-13 provides an itemized list of valves discharging to the tank, together with references to the corresponding piping and instrumentation diagrams.

The tank normally contains water and a predominantly nitrogen atmosphere. In order to obtain effective condensing and cooling of the discharged steam, the tank is installed horizontally, with the steam discharged through a sparger pipe located near the tank bottom and under the water level. The sparger holes are designed to insure a resultant steam velocity close to sonic.

The tank is also equipped with an internal spray and a drain which are used to cool the water following a discharge. Cooling is effected by cold water drawn from the Reactor Makeup Water System, or by circulation of the contents of the tank through the pressurizer relief tank heat exchanger and back into the spray header.

The nitrogen gas blanket is used to control the atmosphere in the tank, and to allow room for the expansion of the original water plus the condensed steam discharge. Provision is made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and/or oxygen.

The contents of the vessel can be drained to the primary drain tank in the equipment and floor drain system via the reactor coolant drain tank pumps in the equipment and floor drain system. When the level in the tank is above the high level setpoint, and the liquid temperature is less than 120°F, the three-way valve is changed to the discharge position, and the pressurizer relief tank pump transfers the liquid to the suction side of the reactor coolant drain tank pumps, which then transfers the liquid to the primary drain tank.

The general configuration of the pressurizer relief tank is shown in Figure 5.4-14. The tank is a horizontal, cylindrical vessel with elliptical dished heads. The vessel is constructed of austenitic stainless steel and is overpressure protected in accordance with ASME Code, Section VIII, Division 1, by means of two safety heads with stainless steel rupture discs.

A flanged nozzle is provided on the tank for the pressurizer discharge line connection to the sparger pipe. The tank is also equipped with an internal spray connected to a cold water inlet and with a bottom drain, which are used to cool the tank following a discharge.

5.4.11.3 Safety Evaluation

The Pressurizer Relief Discharge System does not constitute part of the reactor coolant pressure boundary per 10 CFR 50, Section 50.2, since all of its components are downstream of the RCS safety and relief valves. Thus, General Design Criteria 14 and 15 are not applicable. Furthermore, complete failure of the auxiliary systems serving the pressurizer relief tank will not impair the capability for safe plant shutdown.

The design of the system piping layout and piping restraints is consistent with Regulatory Guide 1.46, with the safety and relief valve discharge piping restrained so that integrity and operability of the valves are maintained in the event of a rupture. Regulatory Guide 1.67 is not applicable since the system is not an open discharge system.

The Pressurizer Relief Discharge System is capable of handling the design discharge of steam without exceeding the design pressure and temperature of the pressurizer relief tank.

The volume of water in the pressurizer relief tank is capable of absorbing the heat from the assumed discharge. If a discharge exceeding the design basis should occur, the rupture discs on the tank would pass the discharge through the tank to the Containment.

The rupture discs on the relief tank have a relief capacity equal to or greater than the combined capacity of the pressurizer safety valves. The minimum rupture disc relief pressure is significantly above the calculated pressure resulting from the design basis safety valve discharge described in Subsection 5.4.11.1. The tank and rupture disc holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20 percent of the setpoint pressure at full flow.

5.4.11.4 Instrumentation Requirements

The pressurizer relief tank pressure transmitter provides an indication of tank pressure at the main control board. An alarm is provided to indicate high tank pressure.

The pressurizer relief tank level transmitter supplies a signal for an indicator with high and low level alarms at the main control board.

The temperature of the water in the pressurizer relief tank is indicated at the main control board, and an alarm actuated by high temperature informs the operator that cooling of the tank contents is required. A temperature in excess of a predetermined value automatically starts the pressurizer relief tank pump. This pump may also be operated manually from the control room.

5.4.11.5 Inspection and Testing Requirements

The Pressurizer Relief Discharge System is tested prior to initial core loading for integrity, performance and operability. The preoperational tests of the system are described in Chapter 14.

5.4.12 Valves

5.4.12.1 Design Bases

As noted in Section 5.2, all valves in the Reactor Coolant System (RCS), out to and including the second valve normally closed or capable of automatic or remote closure and larger than ¾ inch, are ANS Safety Class 1, and ASME Boiler and Pressure Vessel Code, Section III, Code Class 1, valves. All ¾ inch or smaller valves in lines connected to the RCS are Class 2, since the interface with the Class 1 piping is provided with suitable orificing for such valves. Design data for the RCS valves are given in Table 5.4-14.

For a check valve to qualify as part of the RCS, it must be located inside the containment system. When the second of two normally open check valves is considered part of the RCS (as defined in Section 5.1), means are provided to periodically assess back-flow leakage of the first valve when closed.

To ensure that the valves will meet the design objectives, the materials of construction minimize corrosion/erosion and ensure compatibility with the environment. Leakage is minimized to the extent practicable by design.

5.4.12.2 Design Description

Initially, all manually and motor-operated valves of the RCS which are 3 inches and larger were provided with double-packed stuffing boxes and intermediate lantern ring leakoff connections. Exceptions to this criterion are gate valves that have been determined to be susceptible to pressure locking, which have been modified to utilize the valve stem leakoff connection as a vent path for the bonnet cavity. Packing configurations have evolved so that the preferred packing configuration is a single packing set. The industry has moved away from double packed stuffing boxes. These changes in packing configuration have been approved for use at Seabrook Station. Accordingly, either packing design configuration is acceptable for use at Seabrook. These valves use only a single packing set. Leakage to the atmosphere is essentially zero for these valves. Gate valves at the engineered safety features interface are wedge design and are essentially straight through. The wedges are flex-wedge or solid. All gate valves have backseats. Globe valves are "T" and "Y" style. Check valves are swing type for sizes 3 inches and larger. All check valves which contain radioactive fluid are stainless steel and do not have body penetrations other than the inlet, outlet and bonnet. The check hinge is serviced through the bonnet. All operating parts are contained within the body. The disc has limited rotation to provide a change of seating surface and alignment after each valve opening.

5.4.12.3 Design Evaluation

The design requirements for Class 1 valves, as discussed in Subsection 3.9.1, limit stresses to levels which ensure structural integrity of the valves. In addition, the testing programs described in Section 3.9(N) demonstrate the ability of the valves to operate as required during anticipated and postulated plant conditions.

Reactor coolant chemistry parameters are specified in the design specifications to assure the compatibility of valve construction materials with the reactor coolant. To ensure that the reactor coolant continues to meet these parameters, the chemical composition of the coolant will be analyzed periodically.

The above requirements and procedures, coupled with the previously described design features for minimizing leakage, ensure that the valves will perform their intended functions as required during plant operation.

5.4.12.4 Tests and Inspections

The tests and inspections discussed in Section 3.9(N) are performed to ensure the operability of active valves.

There are no full-penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection to the extent practical. Plant layout configurations determine the degree of inspectability. The valve nondestructive examination program is given in Table 5.4-15. In-service inspection is discussed in Subsection 5.2.4.

5.4.13 Safety and Relief Valves

5.4.13.1 Design Bases

The combined capacity of the pressurizer safety valves is designed to accommodate the maximum surge resulting from a complete loss of load. This objective is met without reactor trip or any operator action by the opening of the steam generator safety valves when steam pressure reaches the steam side safety setting.

The pressurizer power-operated relief valves are designed to fail to the closed position on loss of electrical power. Subsection 5.2.2.11 discusses pressurizer power-operated relief valve operation during RCS control during low temperature operations.

5.4.13.2 Design Description

The pressurizer safety valves are of the pop type. The valves are spring loaded, open by direct fluid pressure action, and have backpressure compensation features.

The pressurizer power-operated relief valves are solenoid actuated valves which respond to a signal from a pressure sensing system or to manual control. Remotely-operated stop valves are provided to isolate the power-operated relief valves if excessive leakage develops.

Temperatures in the pressurizer safety valves discharge lines and relief valve discharge manifold are measured and indicated, with high temperature annunciated in the control room. An increase in a discharge line temperature is an indication of leakage or relief through the associated valve.

Design parameters for the pressurizer spray control, safety, and power relief valves are given in Table 5.4-16.

5.4.13.3 Design Evaluation

The pressurizer safety valves prevent RCS pressure from exceeding 110 percent of system design pressure, in compliance with the ASME Boiler and Pressure Vessel Code, Section III.

The pressurizer power-operated relief valves limit undesirable opening of the spring-loaded safety valves.

5.4.13.4 Tests and Inspections

All safety and relief valves are subjected to hydrostatic tests, seat leakage tests, operational tests, and inspections, as required. For safety valves that are required to function during a faulted condition, additional tests are performed. These tests are described in Section 3.9(N).

There are no full penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection.

5.4.14 Component Supports

5.4.14.1 Design Bases

Component supports allow virtually unrestrained lateral thermal movement of the RCS loop during plant operation and provide restraint to the loops and components during accident and seismic conditions. The loading combinations and design stress limits are discussed in Subsection 3.9(N).1.4. (Reference 4 provides the original criteria for postulating breaks in the RC loop. The basis for eliminating eight of these postulated large pipe breaks in the RC loop is provided in References 5 and 6. However, the design of the RCS component supports remains unchanged.) Support design is in accordance with the ASME Code, Section III, Subsection NF, except as noted in Subsection 3.8.3. The design maintains the integrity of the RCS boundary for normal, seismic, and accident conditions and satisfies the requirements of the piping code. Results of piping and supports stress evaluation are presented in Section 3.9(N).

5.4.14.2 Description

The support structures are welded structural steel sections. Linear type structures (tension and compression struts, columns, and beams) are used in all cases, except for the reactor vessel supports, which are plate type structures. Attachments to the supported equipment are nonintegral type that are bolted to or bear against the components. The supports-to-concrete attachments are either anchor bolts or embedded fabricated assemblies.

The supports permit virtually unrestrained thermal growth of the supported systems but restrain vertical, lateral, and rotational movement resulting from seismic and pipe break loadings. This is accomplished using spherical bushings in the columns for vertical support and girders, bumper pedestals, hydraulic snubbers, and tie-rods for lateral support.

Because of manufacturing and construction tolerances, ample adjustment in the support structures is provided to ensure proper erection alignment and fit-up. This is accomplished by shimming or grouting at the supports-to-concrete interface and by shimming at the supports-to-equipment interface.

The supports for the various components are described in the following paragraphs.

a. Reactor Pressure Vessel

The reactor pressure vessel is supported laterally and vertically by two distinct support systems as shown in Figure 5.4-15.

1. Lateral Support

Four individual curved girders provide lateral support for the reactor vessel. They are located on a ledge on the primary shield wall and are in contact with the vertical supports. A more complete description is contained in Subsection 3.8.3.

2. Vertical Support

Vertical supports for the reactor vessel (Figure 5.4-15) are individual air-cooled rectangular box structures located beneath the vessel nozzles and bolted to the primary shield wall concrete. Each box structure consists of a horizontal top plate that receives loads from the reactor vessel shoe, a horizontal bottom plate which transfers the loads to the primary shield wall concrete, and connecting vertical plates which bear against an embedded support. The supports are air cooled to maintain the supporting concrete temperature within acceptable levels.

b. Steam Generator

As shown in Figure 5.4-16, the steam generator supports consist of the following elements:

1. Vertical Support

Four individual columns provide vertical support for each steam generator. These are bolted at the top to the steam generator and at the bottom to the concrete structure. Spherical ball bushings at the top and bottom of each column allow unrestrained lateral movement of the steam generator during heatup and cooldown. The column base design permits both horizontal and vertical adjustment of the steam generator for erection and adjustment of the system.

2. Lower Lateral Support

Lateral support is provided at the generator tube sheet by fabricated steel girders and struts. These are bolted to the compartment walls and include bumpers that bear against the steam generator, but permit unrestrained movement of the steam generator during changes in system temperature. Stresses in the beams caused by wall displacements during compartment pressurization are considered in the design.

3. Upper Lateral Support

The upper lateral support of the steam generator is provided by a ring band at the operating deck. Two-way acting snubbers restrain sudden seismic or blowdown-induced motion, but permit the normal thermal movement of the steam generator. Movement perpendicular to the thermal growth direction of the steam generator is prevented by struts.

c. Reactor Coolant Pump

Three individual columns, similar to those used for the steam generator, provide the vertical support for each pump. Lateral support for seismic and blowdown loading is provided by three lateral tension tie bars. The pump supports are shown in Figure 5.4-17.

d. Pressurizer

The supports for the pressurizer, as shown in Figure 5.4-18 and Figure 5.4-19, consist of:

1. A steel ring plate between the pressurizer skirt and the supporting concrete slab. The ring serves as a leveling and adjustment member for the pressurizer, and may also be used as a template for positioning the concrete anchor bolts.

2. The upper lateral support consists of a frame that bears against the "seismic lugs" provided on the pressurizer, and is attached to the compartment walls.

5.4.14.3 Evaluation

Detailed evaluation ensures the design adequacy and structural integrity of the reactor coolant loop and the primary equipment supports system. This detailed evaluation is made by comparing the analytical results with established criteria for acceptability. Structural analyses are performed to demonstrate design adequacy for safety and reliability of the plant in case of a large or small seismic disturbance and/or LOCA conditions. Loads which the system is expected to encounter often during its lifetime (thermal, weight, pressure) are applied and stresses are compared to allowable values as described in Subsection 3.9(N).1.4.

5.4.14.4 Tests and Inspections

Nondestructive examinations are performed in accordance with the procedures of the ASME Code, Section V, except as modified by the ASME Code, Section III, Subsection NF or as noted in Subsection 3.8.3.

5.4.15 References

1. "Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September 1973.

2. Deleted

3. De Rosa, P., et al., "Evaluation of Steam Generator Tube, Tube Sheet and Divider Plate Under Combined LOCA Plus SSE Conditions," WCAP-7832-A, April 1978.

4. "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," WCAP-8082-P-A (Proprietary) and WCAP-8172-A (Nonproprietary), January 1975.

5. "Technical Bases for Eliminating Large Primary Loop Rupture as the Structural Design Basis for Seabrook Units 1 and 2," WCAP-10567, June 1984 (Proprietary) and WCAP-10566 (Nonproprietary), June 1984.

6. Final Rule Modifying 10 CFR 50 Appendix A, GDC-4 dated October 27, 1987 [52 FR 41288]

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 5
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

TABLES

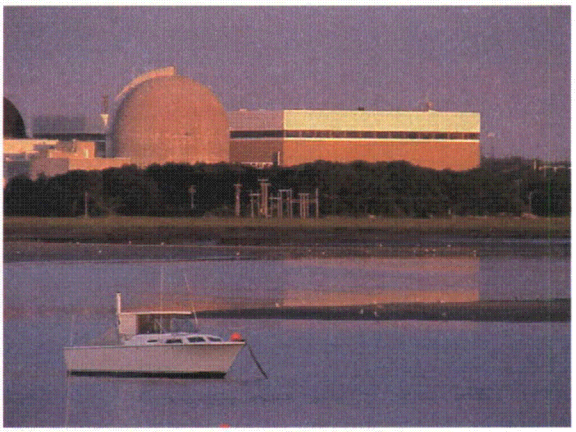


TABLE 5.1-1 SYSTEM DESIGN AND OPERATING PARAMETERS

Plant Design Life, years			40
Nominal Operating Pressure, psig			2235
Total System Volume Including Pressurizer and Surge Line, ft^3			12,265
System Liquid Volume, Including Pressurizer Water at Maximum Guaranteed Power, ft^3			11,524
Pressurizer Spray Rate, Maximum, gpm			900
Pressurizer Heater Capacity, kW			1753.8
System Thermal and Hydraulic Data	Tavg*		4 Pumps Running
NSSS Power, MWt	589.1	571.0	3678**
Reactor Power, MWt			3659**
Thermal Design Flows, gpm			
Active Loop			93,600
Idle Loop			-
Reactor			374,400
Total Thermal Design Flow, 10^6 lb./hr	139.4	143.0	-
Temperatures, °F			
Reactor Vessel Outlet	621.4	604.3	-
Reactor Vessel Inlet	556.8	537.7	-
Steam Generator Outlet	556.4	537.4	-
Steam Generator Steam	540.0	520.3	-
Feedwater	452.4	452.4	-
Steam Pressure, psia	962	815	-

* 0% SGTP

** Includes Core Power Uncertainty

Total Steam Flow, 10^6 lb./hr	16.52	16.42	-

TABLE 5.1-1 SYSTEM DESIGN AND OPERATING PARAMETERS

Best Estimate Flows, gpm***	
Active Loop	100,500+
Idle Loop	-
Reactor	402,000
Mechanical Design Flows, gpm	
Active Loop	104,200
Idle Loop	-
Reactor	416,800
System Pressure Drops**+	
Reactor Vessel ΔP, psi	47.6
Steam Generator ΔP, psi	40.1
Hot Leg Piping ΔP, psi	1.3
Pump Suction Piping ΔP, psi	3.3
Cold Leg Piping ΔP, psi	3.4*
Pump Head, ft	290

*** Initial Core
** At best estimate flow of 100,500 gpm.
* Includes pump weir ΔP of 2.0 psi
+ 0% SGTP

TABLE 5.2-1 APPLICABLE CODE ADDENDA FOR REACTOR COOLANT SYSTEM COMPONENTS

Reactor Vessel	ASME III, 1971 Edition, through Winter 72
Steam Generator	ASME III, 1971 Edition, through Summer 73
Pressurizer	ASME III, 1971 Edition, through Summer 73
CRDM Housing Full Length	ASME III, 1974 Edition, through Summer 74
CRDM Head Adapter	ASME III, 1971 Edition, through Winter 72
Reactor Coolant Pump	ASME III, 1971 Edition, through Summer 73
Reactor Coolant Pipe	ASME III, 1974 Edition, through Summer 75
Class 1 Interconnecting	ASME III, 1971 Edition, through Winter 72
Piping to the RCS* Surge Line	ASME III, 1974 Edition, through Summer 75
Valves	
Pressurizer Safety	ASME III, 1971 Edition, through Winter 72
Motor Operated	ASME III, 1974 Edition, through Summer 74
Manual	(3" and larger)
ASME III, 1974 Edition, through Summer 74 (2" and smaller)	ASME III, 1974 Edition, through Summer 75
Control	ASME III, 1974 Edition, through Summer 75
Pressurizer Spray	ASME III, 1971 Edition, through Summer 72

* Piping provided by A/E.

TABLE 5.2-2 CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS

Reactor Vessel Components	
Shell and head plates (other than core region)	SA-533, Gr. B, Class 1 (vacuum treated)
Shell plates (core region)	SA-533, Gr. B, Class 1 (vacuum treated)
Shell, flange and nozzle forgings nozzle safe ends	SA-508, Class 2; SA-182, Type F316
CRDM and/or ECCS appurtenances, upper head	SB-167; SA-182, Type F304
Instrumentation tube appurtenances, lower head	SB-166
Closure studs, nuts, washers	SA-540, Class 3, Gr. B24 threaded portions of the closure studs have been treated with an anti galling coating
Core support pads	SB-166 with carbon less than 0.10%
Monitor tubes	SA-312, Type 316; SB-166
Vent pipe	SB-166; SB-167
Vessel supports, seal ledge and head lifting lugs	SA-516, Gr. 70 quenched and tempered; SA-533, Gr. B, Class 1 (vessel supports may be of weld metal build up of strength equivalent to nozzle material)
Cladding and buttering	Stainless steel weld metal analysis A-7* and Ni-Cr-Fe weld metal F-Number 43
Steam Generator Components	
Pressure plates	SA-533, Gr. A, B or C, Class 1 or 2
Pressure forgings (including nozzles and tubesheet)	SA-508, Class 2, 2a or 3
Nozzle safe ends	Stainless steel weld metal analysis A-7*
Channel heads	SA-533, Gr. A, B or C, Class 1 or 2 or SA-216, Gr. WCC
Tubes	SB-163, Ni-Cr-Fe annealed
Cladding and buttering	Stainless steel weld metal analysis A-7* and Ni-Cr-Fe weld metal F-Number 43
Closure bolting	SA-193, Gr. B7 threaded portions may be treated with an anti galling coating

* Designated A-8 in the 1974 edition of the ASME Code

Pressurizer Components	
Pressure plates	SA-533, Gr. A, Class 2
Pressure forgings	SA-508, Class 2a
Nozzle safe ends	SA-182, Type 316L and Ni-Cr-Fe weld metal F-Number 43**
Cladding and buttering	Stainless steel weld metal analysis A-8 and Ni-Cr-Fe weld metal F-Number 43
Closure bolting	SA-193, Gr. B7
Reactor Coolant Pump	
Pressure forgings	SA-182, Type F304, F316, F347 or F348
Pressure casting	SA-351, Gr. CF8, CF8A or CF8M
Tube and pipe	SA-213, 376 or 312, seamless Type 304 or 316
Pressure plates	SA-240, Type 304 or 316
Bar material	SA-479, Type 304 or 316
Closure bolting	SA-193, 320, 540 or 453, Gr. 660
Flywheel	SA-533, Gr. B, Class 1
Reactor Coolant Piping	
Reactor coolant pipe	SA-376, Gr. 304N or SA-351, Gr. CF8A centrifugal casting
Reactor coolant fittings	SA-351, Gr. CF8A
Branch nozzles	SA-182, Gr. F316N
Surge line	SA-376, Gr. 304, 316 or F304N
Auxiliary piping ½ through 12 inch and wall schedules 40S through 80S (ahead of second isolation valve)	ANSI B36.19
All other auxiliary piping (ahead of second isolation valve)	ANSI B36.10
Socket weld fittings	ANSI B16.11
Piping flanges	ANSI B16.5

** Alloy 52M (ERNiCrFe-7A) Weld Overlay Installed

Control Rod Drive Mechanism

Latch housing	SA-182, Gr. F304; SA-351, Gr. CF8
Rod travel housing	SA-182, Gr. F304; SA-336, Gr. F8; SA-312, Gr. Type 304
Cap	SA-479, Type 304
Welding materials	Stainless steel weld metal analysis A-8

TABLE 5.2-3 CLASS 1 AND 2 AUXILIARY COMPONENTS MATERIAL SPECIFICATIONS

Valves	
Bodies	SA-182, Type F316; SA-351, Gr. CF8 or CF8M
Bonnets	SA-182, Type F316; SA-351, Gr. CF8 or CF8M
Discs	SA-182, Type F316; SA-564, Gr. 630
Pressure retaining bolting	SA-453, Gr. 660
Pressure retaining nuts	SA-453, Gr. 660; SA-194, Gr. 6
Auxiliary Heat Exchangers	
Heads	SA-240, Type 304
Nozzle necks	SA-182, Gr. F304; SA-312, Type 304
Tubes	SA-213, Type 304; SA-249, Type 304
Tubesheets	SA-182, Gr. F304; SA-240, Type 304; SA-516, Gr. 70 with stainless steel analysis A-8 cladding
Shells	SA-240 and 312, Type 304
Auxiliary Pressure Vessels, Tanks, Filters, etc.	
Shells and heads	SA-351, Gr. CF8A; SA-240, Type 304; SA-264 clad plate consisting of SA-537, Class 1 with SA-240, Type 304 cladding and stainless steel weld overlay analysis A-8
Flanges and nozzles	SA-182, Gr. F304; SA-350, Gr. LF2 with SA-240, Type 304 and stainless steel weld overlay analysis A-8
Piping	SA-312 and SA-240, Type 304 or 316 seamless
Pipe fittings	SA-403, Gr. WP304 seamless
Closure bolting and nuts	SA-193, Gr. B7; SA-194, Gr. 2H
Auxiliary Pumps	
Pump casing and heads	SA-351, Gr. CF8 or CF8M; SA-182, Gr. F304 or F316
Flanges and nozzles	SA-182, Gr. F304 or F316; SA-403, Gr. WP316L seamless
Piping	SA-312, Type 304 or 316 seamless
Stuffing or packing box cover	SA-351, Gr. CF8 or CF8M; SA-240, Type 304 or 316; SA-182, Gr. F304 or F316
Pipe fittings	SA-403, Gr. WP316L seamless; SA-213, Gr. TP304, TP304L, TP316, or TP316L
Closure bolting and nuts	SA-193, Gr. B6, B7 or B8M and SA-194, Gr. 2H or 8M, SA-453, Gr. 660, and nuts, SA-194, Gr. 2H, 8M, 6, and 7

TABLE 5.2-4 REACTOR VESSELS INTERNALS FOR EMERGENCY CORE COOLING

Forgings	SA-182, Type F304
Plates	SA-240, Type 304
Tubes	SA-213, Type 304
Bars	SA-479, Type 304
Bolting	SA-193, Gr. B8M (65 MYS/90MTS) Code Class 1618, Inconel X-750 SA-673, Gr. 688
Nuts	SA-194, Gr. B8, SA-479, Type 304
Locking Devices	SA-479, Type 304L and Type 304
Hold Down Springs	Type 403 Modified Per WPS 80280NL
Clevis Inserts	SB 166
Clevis Insert Bolts	Inconel X-750, SA 637, Gr. 688, Type II

TABLE 5.2-5 Deleted

TABLE 5.2-6 REACTOR COOLANT PRESSURE BOUNDARY PIPING LINE NUMBERS

1-1	12-1	75-1	155-5	251-7	328-6
1-2	13-1	76-1	158-5	256-4	328-7
2-1	21-1	80-1	160-5	258-2	329-4
3-1	30-1	80-2	160-6	259-3	329-5
4-1	30-2	80-6	162-2	260-2	330-4
4-2	30-4	90-5	163-2	261-2	330-5
5-1	30-4	91-1	180-2	261-3	33-14
6-1	30-6	93-1	180-3	261-4	331-5
7-1	30-1	94-1	201-2	270-2	343-1
7-2	48-1	96-1	202-2	272-4	365-2
8-1	48-2	97-1	203-2	272-5	366-2
9-1	48-3	97-3	204-2	273-1	368-1
10-1	49-1	97-4	250-5	274-1	
10-2	58-1	97-5	251-5	275-1	
11-1	74-1	98-1	251-6	275-4	

TABLE 5.2-7 REACTOR COOLANT PRESSURE BOUNDARY VALVE NUMBERS

CSV-0002	RCV-0115	SIV-0050*
CSV-0018	RCV-0116	SIV-0051*
CSV-0034	RCV-0117	SIV-0081*
CSV-0050	RCV-0122	SIV-0082*
CSV-0175	RCV-0124	SIV-0086*
CSV-0176	RCV-0475*	SIV-0087*
CSV-0178	RCV-0479*	SIV-0106*
CSV-0179	RHV-0015*	SIV-0110*
CSV-0181	RHV-0029*	SIV-0118*
CSV-0182	RHV-0030*	SIV-0122*
CSV-0185	RHV-0031*	SIV-0126*
CSV-0186	RHV-0050*	SIV-0130*
CSV-0471	RHV-0051*	SIV-0140*
CSV-0472	RHV-0052*	SIV-0143
CSV-0473	RHV-0053*	SIV-0144*
CSV-0474	RHV-0059	SIV-0147
CSV-0752	RHV-0061	SIV-0148*
RCV-0001	RHV-0063	SIV-0151
RCV-0017	RHV-0065	SIV-0152*
RCV-0022*	SIV-0003	SIV-0155
RCV-0023*	SIV-0005*	SIV-0156*
RCV-0050	SIV-0006*	RC-PCV-456A
RCV-0079	SIV-0017	RC-PCV-456B
RCV-0080	SIV-0020*	RC-LCV-459
RCV-0081	SIV-0021*	RC-LCV-460
RCV-0087*	SIV-0032	RC-PCV-455A
RCV-0088*	SIV-0035*	RC-PCV-455B
RCV-0109	SIV-0036*	
RCV-0110	SIV-0047	

* Reactor coolant pressure isolation valves which require leakage testing in accordance with the Technical Specifications.

TABLE 5.2-8 TYPICAL PLANT THERMAL-HYDRAULIC PARAMETERS

	Units	2-Loop	3-Loop	4-Loop	Seabrook
Heat Output, Core	MWt	1,780	2,652	3,411	3,411
System Pressure	psia	2,250	2,250	2,250	2,250
Coolant Flow	gpm	178,000	265,500	354,000	382,800
Average Core Mass Velocity	10^6lb./hr-ft^2	2.42	2.33	2.50	2.62
Inlet Temperature	°F	545	544	552.5	558.8
Core Average T_{mod}	°F	581	580	588	591.8
Core Length	Ft	12	12	12	12
Average Power Density	kW/l	102	100	104	104
Maximum Fuel Temperature	°F	4,100	4,200	4,200	4,200
Pressurizer Volume	Ft^3	1,000	1,400	1,800	1,800
Pressurizer Volume Ratioed to Primary System Volume		0.157	0.148	0.148	0.146
Peak Surge Rate for Pressurizer Safety Valve Sizing Transient	Ft^3/sec	21.8	33.2	41.0	36.733
Pressurizer Safety Valve Flow at 2500 psia - +3% Accumulation	Ft^3/sec	26.1	36.1	43.3	43.2
Ratio of Safety Valve Flow to Peak Surge Rate		1.197	1.087	1.056	1.179
Full Power Steam Flow per Loop	lb./sec	1,078	1,076	1,038	1,051
Nominal Shell-Side Steam Generator Water Mass per Loop	lb.	100,300	106,000	106,000	107,000

TABLE 5.3-1 REACTOR VESSEL DESIGN PARAMETERS

Design/operating Pressure, psig	2485/2317
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft-in (Bottom Head Outside Diameter to top of Control Rod Mechanism Adapter)	43-10
Thickness of Insulation, in.	3 to 4½
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head/Studs, in. (minimum shank)	6 13/16
Inside Diameter of Flange, in.	167
Outside Diameter of Flange, in.	205
Inside Diameter at Vessel Beltline Shell, in.	173
Inlet Nozzle Inside Diameter, in.	27½
Outlet Nozzle Inside Diameter, in.	29
Clad Thickness, minimum, in.	1/8
Lower Head Thickness, minimum, in.	5 3/8
Vessel Beltline Thickness, minimum, in.	8 5/8
Closure Head Thickness, minimum, in.	7

TABLE 5.3-2 REACTOR VESSEL QUALITY ASSURANCE PROGRAM

	NDT METHOD*			
Forgings	RT	UT	PT	MT
1. Flanges		yes		yes
2. Studs, nuts		yes		yes
3. CRD Head Adapter Flange		yes	yes	
4. CRD Head Adapter Tube		yes	yes	
5. Instrumentation Tube		yes	yes	
6. Main Nozzles		yes		yes
7. Nozzle Safe Ends		yes	yes	
Plates	yes		yes	
Weldments				
1. Main Seam	yes	yes		yes
2. CRD Head Adapter to Closure Head Connection			yes	
3. Instrumentation Tube to Bottom Head Connection			yes	
4. Main Nozzle	yes	yes		yes
5. Cladding		yes	yes	
6. Nozzle Safe Ends	yes	yes	yes	

* RT – Radiographic
UT – Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle

TABLE 5.3-2 REACTOR VESSEL QUALITY ASSURANCE PROGRAM

	NDT METHOD*			
Weldments	RT	UT	PT	MT
7. CRD Head Adaptor Flange to CRD Adaptor Tube	yes		yes	
8. All Full Penetration Ferritic Pressure Boundary Welds Accessible After Hydrotest		yes		yes
9. All Full Penetration Nonferritic Pressure Boundary Welds Accessible After Hydrotest			yes	
10. Seal Ledge				yes
11. Head Lift Lugs				yes
12. Core Pad Welds			yes	

* RT – Radiographic
UT – Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle

TABLE 5.3-3 SEABROOK UNIT NO. 1 REACTOR VESSEL MATERIALS PROPERTIES

Component	Code No	Material Spec. No.	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	AVG. SHELF ENERGY* NMWD (Ft-Lb)	AVG. SHELF ENERGY* MWD (Ft-Lb)
Closure Head Dome	R1809-1	A533B,CL.1	0.15		0.011	-40	10	80.5	-
Closure Head Torus	R1810-1	A533B,CL.1	0.08		0.012	-50	0	104	-
Closure Head Flange	R1802-1	A508,CL.2	-		0.013	10	10	105.5	-
Vessel Flange	R1801-1	A508,CL.2	-		0.012	20	30	91	-
Inlet Nozzle	R1804-1	A508,CL.2	0.10		0.011	0	0	125	-
Inlet Nozzle	R1804-2	A508,CL.2	0.09		0.010	-20	-20	125	-
Inlet Nozzle	R1804-3	A508,CL.2	0.08		0.010	-20	-20	131	-
Inlet Nozzle	R1804-4	A508,CL.2	0.10		0.013	-20	-20	128	-
Outlet Nozzle	R1805-1	A508,CL.2	-		0.003	-20	-10	115	-
Outlet Nozzle	R1805-2	A508,CL.2	-		0.004	-20	-20	132	-
Outlet Nozzle	R1805-3	A508,CL.2	-		0.009	-10	-10	128	-
Outlet Nozzle	R1805-4	A508,CL.2	-		0.005	-10	-10	117	-
Nozzle Shell	R1807-1	A533B,CL.1	0.08		0.011	-30	30	66	-
Nozzle Shell	R1807-2	A533B,CL.1	0.09		0.012	-40	30	66.5	-
Nozzle Shell	R1807-3	A533B,CL.1	0.06		0.010	-20	10	107	-

* NMWD - Normal to Major Working Direction
MWD - Major Working Direction

Component	Code No	Material Spec. No.	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	AVG. SHELF ENERGY* NMWD (Ft-Lb)	AVG. SHELF ENERGY* MWD (Ft-Lb)
Inter. Shell	R1806-1	A533B,CL.1	0.045	0.61	0.012	-30	40	82	139.5
Inter. Shell	R1806-2	A533B,CL.1	0.06	0.64	0.007	-30	0	102	143.5
Inter. Shell	R1806-3	A533B,CL.1	0.075	0.63	0.007	-40	10	115	138
Lower Shell	R1808-1	A533B,CL.1	0.06	0.58	0.005	-30	40	78	120.7
Lower Shell	R1808-2	A533B,CL.1	0.06	0.58	0.007	-20	10	77	127
Lower Shell	R1808-3	A533B,CL.1	0.07	0.59	0.007	-20	40	78	130.7
Bottom Head Torus	R1811-1	A533B,CL.1	0.15		0.010	-50	0	94.5	-
Bottom Head Dome	R1812-1	A533B,CL.1	0.09		0.009	-30	0	97.5	-
Inter. & Lower Shell Long Weld Seams[1,2]	G1.72[3]	Sub Arc Weld	0.047	0.049	0.008	-50	-50	200	-
Inter. & Lower Shell Girth Weld Seam[1,2]	G1.72[3]	Sub Arc Weld	0.047	0.049	0.008	-50	-50	200	-
Surveillance Weld[1,2]		Sub Arc Weld	0.047	0.049		-60	-60	160	

1. The welds were fabricated with Wire Heat No. 4P6052, Linde 0091 Flux, Lot No. 0145 per WCAP-10110 (FP 56056)
2. Weld Cu and Ni Values from WCAP-16526 (FP 25626) which are best estimates.
3. Combustion Engineering weld qualification in response to NRC Bulletin No. 78-12, June 8, 1979.

TABLE 5.3-4 SEABROOK UNIT NO. 1 REACTOR VESSEL BOLTING MATERIAL PROPERTIES CLOSURE HEAD STUDS

Heat No.	Material Spec. No.	Bar No.	0.2% Yield Strength KSI	Ultimate Tensile Strength KSI	Elong. %	Reduction in Area (RA) %	Energy Energy Ft-lb	LateraL Expansion Mils	BHN
84176	SA540, B24	392	145.0	160.0	16.5	51.4	49, 50, 52	29, 30, 30	341
84176	SA540, B24	392-1	144.0	159.0	16.0	50.0	52, 54, 53	30, 33, 31	331
84176	SA540, B24	397	145.5	159.0	16.0	50.0	49, 48, 48	26, 26, 28	321
84176	SA540, B24	397-1	146.5	160.0	16.0	51.9	51, 51, 53	30, 30, 36	331
84176	SA540, B24	401	144.5	159.5	17.0	52.5	49, 51, 50	25, 28, 28	331
84176	SA540, B24	401-1	146.2	162.0	17.0	53.3	50, 50, 50	31, 31, 30	341
63182	SA540, B24	403	139.0	154.5	16.5	50.3	57, 54, 55	32, 33, 33	331
63182	SA540, B24	403-1	144.5	158.0	16.0	51.9	51, 50, 51	28, 30, 30	331
63182	SA540, B24	407	141.0	156.0	17.0	53.0	54, 55, 52	33, 33, 31	331
63182	SA540, B24	407-1	147.0	161.0	17.0	54.7	50, 51, 50	26, 28, 25	341
43320	SA540, B24	490	137.2	153.0	16.0	48.9	55, 55, 56	37, 36, 36	321
43320	SA540, B24	490-1	140.0	154.5	17.0	51.0	54, 52, 54	36, 31, 35	331
43320	SA540, B24	491	144.2	159.0	16.5	50.0	48, 47, 47	29, 29, 27	-
43320	SA540, B24	491-1	134.5	150.0	16.5	48.6	50, 54, 52	31, 36, 33	-

TABLE 5.3-4 SEABROOK UNIT NO. 1 REACTOR VESSEL BOLTING MATERIAL PROPERTIES CLOSURE HEAD STUDS

Heat No.	Material Spec. No.	Bar No.	0.2% Yield Strength KSI	Ultimate Tensile Strength KSI	Elong. %	Reduction in Area (RA) %	Energy Energy Ft-lb	LateraL Expansion Mils	BHN
Closure Head Nuts and Washers									
63182	SA540, B24	135	147.5	161.0	17.0	53.0	49, 49, 51	28, 29, 30	321
63182	SA540, B24	135-1	143.2	157.0	17.5	55.2	55, 54, 52	33, 32, 31	321
63182	SA540, B24	137	145.0	159.0	16.5	54.8	54, 54, 53	33, 33, 29	331
63182	SA540, B24	137-1	147.0	160.0	17.0	55.7	54, 55, 54	34, 36, 33	321
63182	SA540, B24	143	145.0	159.0	18.0	58.1	55, 54, 54	33, 32, 32	331
63182	SA540, B24	143-1	147.0	160.0	17.0	57.3	54, 50, 52	33, 29, 30	321
63182	SA540, B24	145	145.0	159.0	17.0	56.0	54, 54, 55	34, 35, 34	321
63182	SA540, B24	145-1	146.2	159.8	17.0	57.0	56, 55, 54	36, 35, 36	331
63182	SA540, B24	148	144.0	157.5	17.5	56.5	56, 55, 55	33, 34, 34	331
63182	SA540, B24	148-1	148.5	162.0	17.0	55.6	52, 51, 52	33, 28, 30	321
63182	SA540, B24	150	144.8	158.0	17.5	55.7	55, 55, 54	33, 30, 31	331
63182	SA540, B24	150-1	145.8	160.0	17.0	56.5	53, 50, 52	33, 30, 31	331

TABLE 5.3-5 SEABROOK UNIT 1 REACTOR VESSEL BELTLINE REGION SHELL PLATE TOUGHNESS (TRANSVERSE DIRECTION)[a]

Inter. Shell Plate R1806-1				Inter. Shell Plate R1806-2				Inter. Shell Plate R1806-3			
Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)
-20	12	0	8	-20	10	0	6	-20	9	0	4
-20	10	0	7	-20	14	0	9	-20	11	0	7
-20	12	0	9	-20	14	0	9	-20	10	0	5
30	26	10	18	30	43	20	28	20	30	10	19
30	22	10	15	30	40	15	26	20	37	15	26
30	33	15	22	30	23	10	16	20	40	15	28
60	44	20	28	50	48	20	34	60	40	15	27
60	40	20	25	50	47	20	35	60	78	30	52
60	41	20	27	50	47	20	34	60	67	25	46
90	49	25	38	60	51	20	36	70	64	30	48
90	55	25	42	60	50	20	37	70	50	20	36
90	52	25	38	60	50	20	37	70	65	30	45

(a) From Lukens Steel CMTRs

TABLE 5.3-5 SEABROOK UNIT 1 REACTOR VESSEL BELTLINE REGION SHELL PLATE TOUGHNESS (TRANSVERSE DIRECTION)[a]

Inter. Shell Plate R1806-1				Inter. Shell Plate R1806-2				Inter. Shell Plate R1806-3			
Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)
100	53	30	41	100	69	40	53	100	85	60	62
100	50	25	40	100	71	40	52	100	82	50	57
100	61	35	48	100	71	40	53	100	86	60	64
160	78	90	61	160	94	90	67	160	105	95	71
160	84	95	66	160	93	90	66	160	102	90	69
160	76	90	58	160	93	90	64	160	107	95	75
212	85	100	67	212	102	100	76	212	119	100	77
212	86	100	62	212	100	100	71	212	118	100	76
212	75	100	56	212	105	100	74	212	108	100	68

(a) From Lukens Steel CMTRs

TABLE 5.3-5 SEABROOK UNIT 1 REACTOR VESSEL BELTLINE REGION SHELL PLATE TOUGHNESS (TRANSVERSE DIRECTION)[a]

Lower Shell Plate R1808-1				Lower Shell Plate R1808-2				Lower Shell Plate R1808-3			
Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)
-40	8	0	4	-40	9	0	6	-40	9	0	5
-40	12	0	8	-40	9	0	7	-40	8	0	5
-40	10	0	6	-40	13	0	8	-40	7	0	4
0	21	5	16	0	24	10	19	0	18	5	10
0	17	0	12	0	19	5	14	0	18	5	12
0	17	0	13	0	28	15	22	0	17	5	10
30	26	10	21	40	34	15	26	40	23	5	17
30	28	10	22	40	40	20	32	40	34	15	23
30	32	15	24	40	39	20	30	40	26	10	19
60	37	15	29	60	56	30	43	60	36	15	28
60	43	15	34	60	41	25	31	60	32	15	26
60	38	15	32	60	41	25	32	60	38	15	28

(a) From Lukens Steel CMTRs

TABLE 5.3-5 SEABROOK UNIT 1 REACTOR VESSEL BELTLINE REGION SHELL PLATE TOUGHNESS (TRANSVERSE DIRECTION)[a]

Lower Shell Plate R1808-1				Lower Shell Plate R1808-2				Lower Shell Plate R1808-3			
Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)
90	49	25	38	70	51	25	37	90	50	25	42
90	54	35	40	70	54	25	40	90	45	20	34
90	55	40	42	70	51	25	38	90	51	25	40
100	74	70	55	100	67	90	51	100	54	20	42
100	60	65	50	100	63	80	48	100	52	20	39
100	58	65	47	100	61	80	46	100	50	20	38
160	72	100	60	160	75	100	60	160	67	90	54
160	82	100	62	160	69	100	58	160	75	90	59
160	74	100	58	160	77	100	61	160	76	90	61
212	76	100	50	212	77	100	60	212	76	100	56
212	79	100	63	212	79	100	61	212	78	100	56
212	78	100	64	212	75	100	60	212	80	100	59

(a) From Lukens Steel CMTRs

TABLE 5.3-6 SEABROOK UNIT 1 REACTOR VESSEL BELTLINE REGION WELD METAL AND HAZ MATERIAL TOUGHNESS

Inter & Lower Shell Long & Girth Welds							
Weld Qualification No. G1.72 For Wire Heat No. 4P6052				Weld HAZ Material			
Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)
-70	7	0	2	-110	22	5	10
-70	6	0	2	-110	28	5	13
-70	7	0	2	-110	19	0	7
-30	19	10	13	-80	38	15	19
-30	20	10	15	-80	30	10	13
-30	24	10	18	-80	49	20	25
-10	58	30	42	-40	70	35	39
-10	128	70	77	-40	103	60	55
-10	122	70	72	-40	101	60	53
10	139	80	81	-10	117	70	58
10	58	30	38	-10	119	70	61
10	122	70	70	-10	81	40	43
60	188	100	72	60	121	100	66
60	205	100	70	60	135	100	68
60	208	100	68	60	152	100	66
100	173	100	81	100	156	100	67
100	227	100	76	100	146	100	70
100	180	100	75	100	132	100	65

TABLE 5.3-6 SEABROOK UNIT 1 REACTOR VESSEL BELTLINE REGION WELD METAL AND HAZ MATERIAL TOUGHNESS

Inter & Lower Shell Long & Girth Welds							
Weld Qualification No. G1.72 For Wire Heat No. 4P6052				Weld HAZ Material			
Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)	Test Temp. (°F)	Energy (ft-lb.)	Shear (%)	Lat. Exp. (mils)
				160	133	100	66
				160	150	100	68
				160	144	100	67

Note: There is also a complete set of charpy data for the surveillance weld that is reported in WCAP.10110 (FP 56056). It is the value used for PTS and USE.

TABLE 5.4-1 REACTOR COOLANT PUMP DESIGN PARAMETERS

Unit design pressure (psig)	2485
Unit design temperature (°F)	650[a]
Unit overall height (ft)	27
Seal water injection (gpm)	8
Seal water return (gpm)	3
Cooling water flow (gpm)	596
Maximum continuous cooling water inlet temperature (°F)	105
<u>Pump</u>	
Type	Vertical, centrifugal, single stage
Capacity (gpm)	100,200[c]
Developed head (ft)	290[c]
NPSH required (ft)	Figure 5.4-2
Suction temperature (°F)	558
Pump discharge nozzle, inside diameter (in.)	27½
Pump suction nozzle, inside diameter (in.)	31
Speed (rpm)	1190[c]
Water volume (ft^3)	80[b]
Weight, dry (lb.)	203,960

[a] Design temperature of pressure retaining parts of the pump assembly exposed to the reactor coolant and injection water on the high pressure side of the controlled leakage seal shall be that temperature determined for the parts for a primary loop temperature of 650°F.

[b] Composed of reactor coolant in the casing and of injection and cooling water in the thermal barrier.

[c] These values represent the pump hydraulic design point. Actual heads, flows, temperatures, and currents are dependent upon system parameters such as fuel, modifications to the reactor internals, percentage of steam generator tube plugging and operating $T_{avg.}$ Values for use in analyses should reflect current conditions.

Motor	
Type	Drip proof, squirrel cage induction, water/air cooled
Power (hp)	7000
Voltage (volts)	13,200
Phase	3
Frequency (Hz)	60
Insulation class	Class B, thermalastic epoxy insulation
Starting current	1720 amp @ 13,200 volts
Input, hot reactor coolant	267 ± 6 amp[c]
Input, cold reactor coolant	334 ± 7 amp[c]

Pump assembly moment of inertia, maximum (lb/ft^2)	
Flywheel	70,000
Motor	22,500
Shaft	520
Impeller	1,980
Total	95,000

[c] These values represent the pump hydraulic design point. Actual heads, flows, temperatures, and currents are dependent upon system parameters such as fuel, modifications to the reactor internals, percentage of steam generator tube plugging and operating $T_{avg.}$ Values for use in analyses should reflect current conditions.

TABLE 5.4-2 REACTOR COOLANT PUMP QUALITY ASSURANCE PROGRAM

	NDT Method*			
	RT	UT	PT	MT
Castings	yes		yes	
Forging				
Main shaft		yes		yes
Main studs		yes		yes
Flywheel (rolled plate)		yes		
Weldments				
Circumferential	yes		yes	
Instrument connections			yes	

* RT - Radiographic
UT - Ultrasonic
PT - Dye penetrant
MT - Magnetic particle

TABLE 5.4-3 STEAM GENERATOR DESIGN DATA

Design Pressure, reactor coolant side, psig	2485
Design Pressure, steam side, psig	1185
Design Pressure, primary to secondary, psi	1600
Design Temperature, reactor coolant side, °F	650
Design Temperature, steam side, °F	600
Design Temperature, primary to secondary, °F	650
Total Heat Transfer Surface Area, ft^2	55,000
Maximum Moisture Carryover, wt percent	0.25
Overall Height, ft-in.	67-8
Number of U-Tubes	5626
U-Tube Nominal Diameter, in.	0.688
Tube Wall Nominal Thickness, in.	0.040
Number of Manways	4
Inside Diameter of Manways, in.	16
Number of Handholes	6
Design Fouling Factor, ft^2-hr-°F/Btu	0.00006
Steam Flow, lbs/hr	4.106×10^6 - 3.769×10^6

TABLE 5.4-4 STEAM GENERATOR QUALITY ASSURANCE PROGRAM

	NDT Method*				
	RT	UT	PT	MT	ET
Tubesheet					
Forging		yes		yes	
Cladding		yes(+)	yes		
Channel-Head (if fabricated)					
Fabrication	yes(++)	yes(+++)		yes	
Cladding			yes		
Secondary Shell and Head					
Plates			yes		
Tubes			yes		
Nozzles.-(Forgings)		yes		yes	yes
Weldments					
Shell, longitudinal	yes			yes	
Shell, circumferential	yes			yes	
Cladding (channel head- tubesheet joint cladding restoration)			yes		
Primary nozzles to fab head		yes			yes
Manways to fab head	yes			yes	
Steam and Feedwater nozzle to shell	yes			yes	
Support brackets				yes	
Tube to tubesheet			yes		
Instrument connections (primary and secondary)				yes	
Temporary attachments after removal				yes	
After hydrostatic test (all major pressure boundary welds and complete cast channel head - where accessible)				yes	
Nozzle safe ends (if weld deposit)	yes		yes		

* RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current

(+) Flat Surface Only
(++) Weld Deposit
(+++) Base Material Only

TABLE 5.4-5 REACTOR COOLANT PIPING DESIGN PARAMETERS

Reactor inlet piping, inside diameter (in.)	27½
Reactor inlet piping, nominal wall thickness (in.)	2.32
Reactor outlet piping, inside diameter (in.)	29
Reactor outlet piping, nominal wall thickness (in.)	2.45
Coolant pump suction piping, inside diameter (in.)	31
Coolant pump suction piping, nominal wall thickness (in.)	2.60
Pressurizer surge line piping, nominal pipe size (in.)	14
Pressurizer surge line piping, nominal wall thickness (in.)	1.406
Reactor Coolant Loop Piping	
Design/operating pressure (psig)	2485/2235
Design temperature (°F)	650
Pressurizer Surge Line	
Design pressure (psig)	2485
Design temperature (°F)	680
Pressurizer Safety Valve inlet Line	
Design pressure (psig)	2485
Design temperature (°F)	680
Pressurizer (Power Operated) Relief Valve Inlet Line	
Design pressure (psig)	2485
Design temperature (°F)	680

TABLE 5.4-6 REACTOR COOLANT PIPING QUALITY ASSURANCE PROGRAM

	NDT Method*		
	RT	UT	PT
Castings	yes		yes (after finishing)
Forgings		yes	yes (after finishing)
Weldments			
Circumferential	yes		yes
Nozzle to runpipe (except no RT for nozzles less than 6 inches)	yes		yes
Instrument connections			yes

* RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant

TABLE 5.4-7 DESIGN BASES FOR RESIDUAL HEAT REMOVAL SYSTEM OPERATION

Residual Heat Removal System startup	Approximately 4 hours after reactor shutdown
Reactor Coolant System initial pressure (psig)	Approximately 425
Reactor Coolant System initial temperature (°F)	Approximately 350
Component cooling water design temperature (°F)	102
Cooldown time (hours after initiation of Residual Heat Removal System operation)	Approximately 20
Reactor Coolant System temperature at end of cooldown (°F)	125
Decay heat generation at 24 hours after reactor shutdown (Btu/hr)	76.6×10^6

TABLE 5.4-8 RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pump		
Number	2	
Design pressure (psig)	600	
Design temperature (°F)	400	
Design flow (gpm)	3000	
Design head (ft)	375	
Power (hp)	400	
Residual Heat Exchanger		
Number	2	
Design heat removal capacity (Btu/hr)	35.1×10^6	
Estimated UA (Btu/hr-°F)	2.0×10^6	
	Tube side	Shell side
Design pressure (psig)	600	150
Design temperature (°F)	400	200
Design flow (lb/hr)	1.48×10^6	2.475×10^6
(gpm)	3000	5000
Inlet temperature (°F)	125	85
Outlet temperature (°F)	102.7	98.3
Material	Carbon steel/Austenitic stainless steel	Carbon Steel
Fluid	Reactor coolant	Component cooling water

TABLE 5.4-9 PRESSURIZER DESIGN DATA

Design pressure (psig)	2485
Design temperature (°F)	680
Surge line nozzle diameter (in.)	14
Heatup rate of pressurizer using heaters only (°F/hr)	55
Internal volume (ft^3)	1800

TABLE 5.4-10 REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	psig
Hydrostatic test pressure	3107
Design Pressure	2485
Operating pressure	2235
Safety valves (begin to open)	2485
High pressure reactor trip	2385
High pressure alarm	2310
Power-operated relief valves	2385*
Pressurizer spray valves (full open)	2310
Pressurizer spray valves (begin to open)	2260
Proportional heaters (begin to operate)	2250
Proportional heaters (full operation)	2220
Backup heaters on	2210
Low pressure alarm	2210
Pressurizer power-operated relief valve interlock	2335
Low pressure reactor trip (typical, but variable)	1945

* At 2385 psig, a pressure signal initiates actuation (opening) of these valves. Remote manual control is also provided.

TABLE 5.4-11 PRESSURIZER QUALITY ASSURANCE PROGRAM

	NDT Method[a]			
	RT	UT	PT	MT
Heads				
Plates		yes		
Cladding			yes	
Shell				
Plates		yes		
Cladding			yes	
Heaters				
Tubing[b]		yes	yes	
Centering of element	yes			
Nozzle (Forgings)		yes	yes[c]	yes[c]
Weldments				
Shell, longitudinal	yes			yes
Shell, circumferential	yes			yes
Cladding			yes	
Nozzle safe end	yes		yes	
Instrument connection			yes	
Support skirt, longi tudinal seam	yes			yes
Support skirt to lower head		yes		yes
Temporary attachments(after removal)				yes
All external pressure boundary welds after shop hydrostatic test				yes

(a) RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle

(b) Or a UT and ET

(c) MT or PT

TABLE 5.4-12 PRESSURIZER RELIEF TANK DESIGN DATA

Design Pressure, psig		100
Normal Operating Pressure, psig		2-6
Rupture Disc Release Pressure, psig	Nominal Range	91 86-100
Normal Water Volume, ft^3		1350
Normal Gas Volume, ft^3		450
Design Temperature, °F		340
Normal Operating Water Temperature, °F		<120
Total Rupture Disc Relief Capacity, Saturated Steam at100 psig, lb/hr		$1.6x10^6$

TABLE 5.4-13 SAFETY/RELIEF VALVES DISCHARGING TO THE PRESSURIZER RELIEF TANK

Reactor Coolant System		
3	Pressurizer safety valves	Figure 5.1-4
2	Pressurizer power-operated relief valves	Figure 5.1-4
2	Residual heat removal pump suction line from the Reactor Coolant System hot legs	Figure 5.1-3, Sheets 1 & 4
2	Between RHR Loop Isolation Valves	Figure 5.1-3, Sheets 1 & 4
Chemical and Volume Control System		
2	Seal water return line	Figures 9.3-26 through 9.3-29
1	Letdown line	Figures 9.3-26 through 9.3-29

TABLE 5.4-14 REACTOR COOLANT SYSTEM VALVE DESIGN PARAMETERS

Design/normal operating pressure (psig)	2485/2235
Preoperational plant hydrotest (psig)	3107
Design temperature (°F)	650

TABLE 5.4-15 REACTOR COOLANT SYSTEM VALVES NONDESTRUCTIVE EXAMINATION PROGRAM

	NDT Method[a]		
	RT	UT	PT
Castings (larger than 4 inches)	yes		yes
(2 inches to 4 inches)	yes[b]		yes
Forgings (larger than 4 inches)	yes[b]		yes
(2 inches to 4 inches)	[c]	[c]	yes

(a) RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
(b) Welds ends only
(c) Either RT or UT

TABLE 5.4-16 PRESSURIZER VALVES DESIGN PARAMETERS

Pressurizer Spray Control Valves	
Number	2
Design pressure, psig	2485
Design temperature, °F	650
Design flow for valves full open, each, gpm	450
Pressurizer Safety Valves	
Number	3
Design pressure, psig	2485
Design temperature, °F	680
Maximum relieving capacity, ASME rated flow, lb/hr (per valve)	420,000
Set pressure, psig	2485
Fluid Saturated steam	
Backpressure:	
Expected during discharge, psig	500
Pressurizer Power-operated Relief Valves	
Number 2	
Design pressure, psig 2485	
Design temperature, °F 680	
Relieving capacity at 2385 psig, lb/hr (per valve)	210,000
Fluid Saturated steam	

TABLE 5.4-17 FAILURE MODE AND EFFECTS ANALYSIS - RESIDUAL HEAT REMOVAL SYSTEM - PLANT COOLDOWN OPERATION

Component	Failure Mode	*Effect on System Operation	**Failure Detection Method	Remarks
1. Motor-operated gate valve RC-V23 (RC-V88 Analogous)	a. Fails to open on demand (open manual mode CB switch selection)	a. Failure blocks reactor coolant flow from hot leg of RC loop #1 through Train A of RHRS. Fault reduces redundancy of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop #4 flowing through Train B of RHRS; however, time required to reduce RCS temperature will be extended	a. Valve position indication (closed to open position change) at CB; RC Pressure indication (RC-PI-405) at CB; RHR Train A discharge flow indication RH-FI-618) at CB; and RHR pump discharge pressure indication (RH-PI-614) at CB.	1. Valve is electrically interlocked with RHR to charging pump suction line isolation valve (RH-V35) and with a "prevent-open" RC pressure interlock (PB-405). The valve cannot be opened from the CB if the indicated isolation valve is open or if RC pressure exceeds the interlock condition.
	b. Once open, fails closed (open manual mode CB switch selection)	b. Same effect on system operation as that stated above for failure mode "Fails to open on demand."	b. Same method of detection as those stated above for failure mode "Fails to open on demand."	2. If both trains of RHRS system are unavailable for plant cooldown due to multiple component failures, the Emergency Feedwater System and S.G. power operated relief valves can be used to perform the safety function of removing residual heat.
2. Motor-operated gate valve RC-V22 (RC-V87 analogous)	a. Same failure modes as those stated for item #1.	a. Same effect on system operation as that stated for item #1.	a. Same methods of detection as those stated for item #1, except for RC pressure indication (RC-PI-403) at CB.	1. Same remarks as those stated for item #1, except for pressure interlock (PB-403) control.
3. Residual heat removal pump RH-P- 8A, (pump RH-P-8B analogous)	a. Fails to deliver working fluid.	a. Failure results in loss of reactor coolant flow from hot leg of RC loop #1 through Train A of RHRS. Fault reduces redundance of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop #4 flowing through Train B of RHRS; however, time required to reduce RCS temperature will be extended.	a. Open pump switchgear circuit breaker indication at CB; circuit breaker close position monitor light for group monitoring of components at CB; common breaker trip alarm at CB; RC pressure indication (RC-PI-405) at CB; RHR Train A discharge flow indication (RH-FI-618) (see item #11) at CB; and pump discharge pressure indication (RH-PI-614) at CB.	1. The RHRS shares components with the ECCS. Pumps are tested as part of the ECCS testing program (see Subsection 6.3.4). Pump failure may also be detected during ECCS testing.

TABLE 5.4-17 FAILURE MODE AND EFFECTS ANALYSIS - RESIDUAL HEAT REMOVAL SYSTEM - PLANT COOLDOWN OPERATION

Component	Failure Mode	*Effect on System Operation	**Failure Detection Method	Remarks
4. Motor-operated globe valve RH-FCV-610 (RH-FCV-611 analogous)	a. Fails to open on demand (open manual mode CB switch selection).	a. Failure blocks miniflow line to suction of RHR pump A during cooldown operation. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop#4 flowing through Train B of RHRS; however, time required to reduce RCS temperature will be extended.	a. Valve position indication (closed to open position change) at CB.	1. Valve is automatically controlled to open when pump discharge is less than ~750 gpm and close when the discharge exceeds ~1400 gpm. The valve protects the pump from dead-heading during ECCS operation. CB switch set to "Auto" position for automatic control of valve positioning.
	b. Fails to close on demand ("Auto" mode CB switch selection).	b. Failure allows for a portion of RHR heat exchanger A discharge flow to be bypassed to a suction of RHR pump A. RHRS Train A is degraded for the regulation of coolant temperature by RHR heat exchanger A. No effect on safety for system operation. Cooldown of RCS within established specification cooldown rate may be accomplished through operator action of throttling flow control valve RH-HCV-606 and controlling cooldown with redundant RHRS Train B	b. Valve position indication (open to closed positioning change) and RHRS Train A discharge flow indication(RH-FI-618) at CB.	
5. Air diaphragm operated butterfly Valve RH-FCV-618 (RH-FCV-619 analogous)	a. Fails to open on demand ("Auto" mode CB switch selection).	a. Failure prevents coolant Discharged from RHR pump A from bypassing RHR heat Exchanger A resulting in Mixed mean temperature of coolant flow to RCS being low. RHRS Train A is degraded for the regulation of controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished through operator action of throttling flow control valve RH-HCV-606 and controlling cooldown with redundant RHRS Train B.	a. RHR pump A discharge flow temperature and RHRS Train A discharge to RCS cold leg flow temperature recording (RH-TR-612) at CB; and RHRS Train A discharge to RCS cold leg flow indication (RH-FI-618) at CB.	1. Valve is designed to Fail "closed" and is electrically wired so that electrical solenoid of the air diaphragm operator is energized to open the valve. Valve is normally "closed" to align RHRS for ECCS operation during plant power operation and load follow.
	b. Fails to close on demand ("Auto mode CB switch selection).	b. Failure allows coolant discharged from RHR pump A to bypass RHR heat exchanger A resulting in mixed mean temperature of coolant flow to RCS being high. RHRS Train A is degraded for the regulation of controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS within established specification rate will be accomplished with redundant RHRS Train B; however, cooldown time will be extended.	b. Same method of detection as those stated above.	

TABLE 5.4-17 FAILURE MODE AND EFFECTS ANALYSIS - RESIDUAL HEAT REMOVAL SYSTEM - PLANT COOLDOWN OPERATION

Component	Failure Mode	*Effect on System Operation	**Failure Detection Method	Remarks
6. Air diaphragm operated butterfly valve RH-HCV-606 (RH-HCV-607 analogous)	a. Fails to close on demand for flow reduction.	a. Failure prevents control of coolant discharge flow from RHR heat exchanger A resulting in loss of mixed mean temperature coolant flow adjustment to RCS. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished by operator action of controlling cooldown with redundant RHRS Train B.	a. Same methods of detection as those stated for item #5. In addition, monitor light and alarm (valve closed) for group monitoring of components at CB. operation and load follow.	1. Valve is designed to fail "open." The valve is normally"open" to align RHRS for ECCS operation during plant power
	b. Fails to open on demand for increased flow.	b. See 5b.	b. Same methods as those stated above for failure mode "Fails to close on demand for flow reduction."	
7. Manual globe valve RH-V18 (RH-V19 analogous)	a. Fails closed	a. Failure blocks flow from Train A of RHRS to CVCS letdown heat exchanger. Fault prevents (during the initial phase of plant cooldown) the adjustment of boron concentration level of coolant in lines of RHRS Train A so that it equals the concentration level in the RCS using the RHR cleanup line to CVCS. No effect on safety for system operation. Operator can balance boron concentration levels by cracking open flow control valve RH-HCV-606 to permit flow to cold leg of loop #1 of RCS in order to balance levels using normal CVCS letdown flow.	a. CVCS letdown flow indication (CS-FI-132) at CB	1. Valve is normally "closed" to align the RHRS for ECCS operation during plant power operation and load follow.
8. Air diaphragm operated globe valve CS-HCV-128.	a. Fails to open on demand.	a. Failure blocks flow from Trains A and B of RHRS to CVCS letdown heat exchanger. Fault prevents use of RHR cleanup line to CVCS for balancing boron concentration levels of RHR Trains A and B with RCS during initial cooldown operation and later in plant cooldown for letdown flow. No effect on safety for system operation. Operator can balance boron concentration levels with similar actions, using pertinent flow control valve RH-HCV-607, as stated above for item #7. Normal CVCS let- down flow can be used for purification if RHRS cleanup line is not available.	a. Valve position indication (degree of opening) at CB and CVCS letdown flow indication (CS-FI-132) at CB.	1. Same remark as that stated above for item #7. 2. Valve is a component of the CVCS that performs an RHR function during plant cooldown operation.
9. Motor-operated gate valve CBS-V2 (CBS- V5 analogous)5	a. Fails to close on demand.	a. Failure reduces the redundancy of isolation valves provided to flow isolate RHRS Train A from RWST. No effect on safety for system operation. Check valve CBS-V55 in series with MO-valve provides the primary isolation against the bypass of RCS coolant flow from the suction of RHR pump A to RWST.	a. Valve position indication (open to closed position change) at CB and valve (closed) monitor light and alarm at CB.	1. Valve is a component of the ECCS that performs an RHR function during plant cooldown. Valve is normally "open" to align the RHRS for ECCS operation during plant operation.

TABLE 5.4-17 FAILURE MODE AND EFFECTS ANALYSIS - RESIDUAL HEAT REMOVAL SYSTEM - PLANT COOLDOWN OPERATION

List of acronyms and abbreviations

Auto - Automatic

CB - Control Board

CVCS - Chemical and Volume Control System

ECCS - Emergency Core Cooling System

MO - Motor-Operated

RC - Reactor Coolant

RCS - Reactor Coolant System

RHR - Residual Heat Removal

RHRS - Residual Heat Removal System

RWST - Refueling Water Storage Tank

SG - Steam Generator

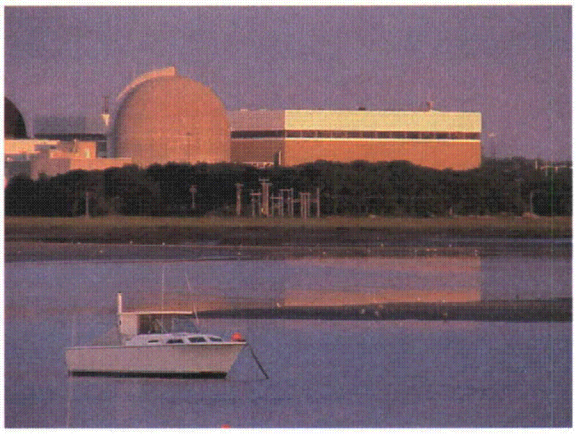
* See list at end of table for definition of acronyms and abbreviations used.

** As part of plant operation, periodic tests, surveillance inspection and instrument calibrations are made to monitor equipment and performance. Failures may be detected during such monitoring of equipment in addition to detection methods noted.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 5
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

FIGURES



See PID-1-RC-B20840

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant System Overview	
		Figure 5.1-1

See PID-1-RC-B20845

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant System Reactor Vessel P&ID	
		Figure 5.1-2

See PID-1-RC-B20841

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant System, Loop No. 1 through Loop 4 [4 Sheets]	
		Figure 5.1-3 Sh. 1 of 4

See PID-1-RC-B20842

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant System, Loop No. 1 through Loop 4 [4 Sheets]	
		Figure 5.1-3 Sh. 2 of 4

See PID-1-RC-B20843

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant System, Loop No. 1 through Loop 4 [4 Sheets]	
		Figure 5.1-3 Sh. 3 of 4

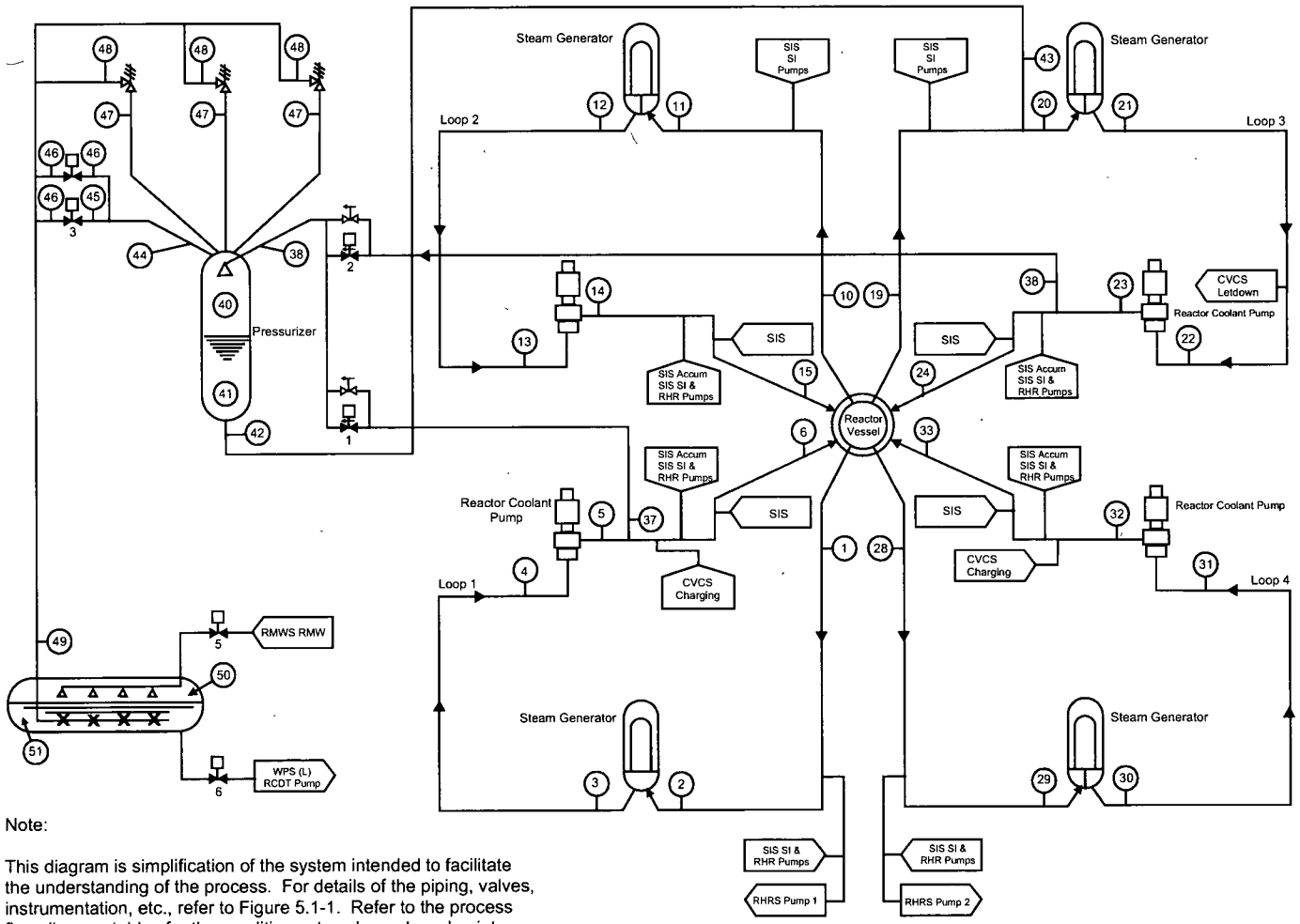
See PID-1-RC-B20844

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant System, Loop No. 1 through Loop 4 [4 Sheets]	
		Figure 5.1-3 Sh. 4 of 4

See PID-1-RC-B20846

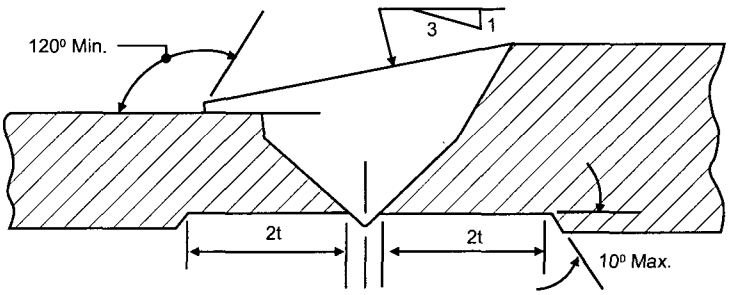
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant System Pressurizer	
		Figure 5.1-4

Reactor Coolant System Process Flow Diagram

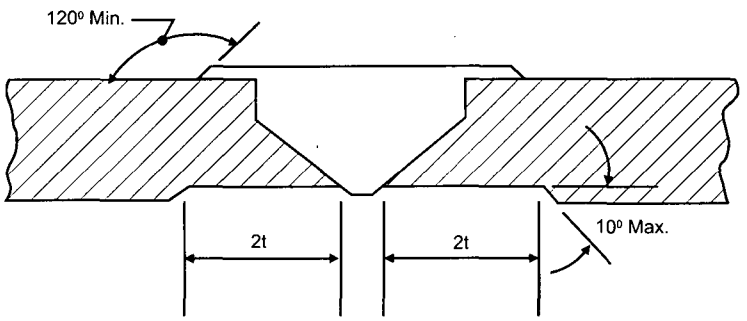


Note:

This diagram is simplification of the system intended to facilitate the understanding of the process. For details of the piping, valves, instrumentation, etc., refer to Figure 5.1-1. Refer to the process flow diagram tables for the conditions at each numbered point.



For Weld Ends of Unequal Thickness



For Weld Ends of Equal Thickness

Notes:

1. Machine or Grind Shop or Field Welds To Obtain Surface Finish of 250μ-Inches RMS Maximum.
2. No Undercutting is Permitted.
3. Remove Sharp Edges.
4. The 2t Dimension Does Not Apply To Fittings.
5. For Details of Weld End Preparation See Drawing 5000-F-1382.

G:\Word\Images_P\UFSAR\521.ds4

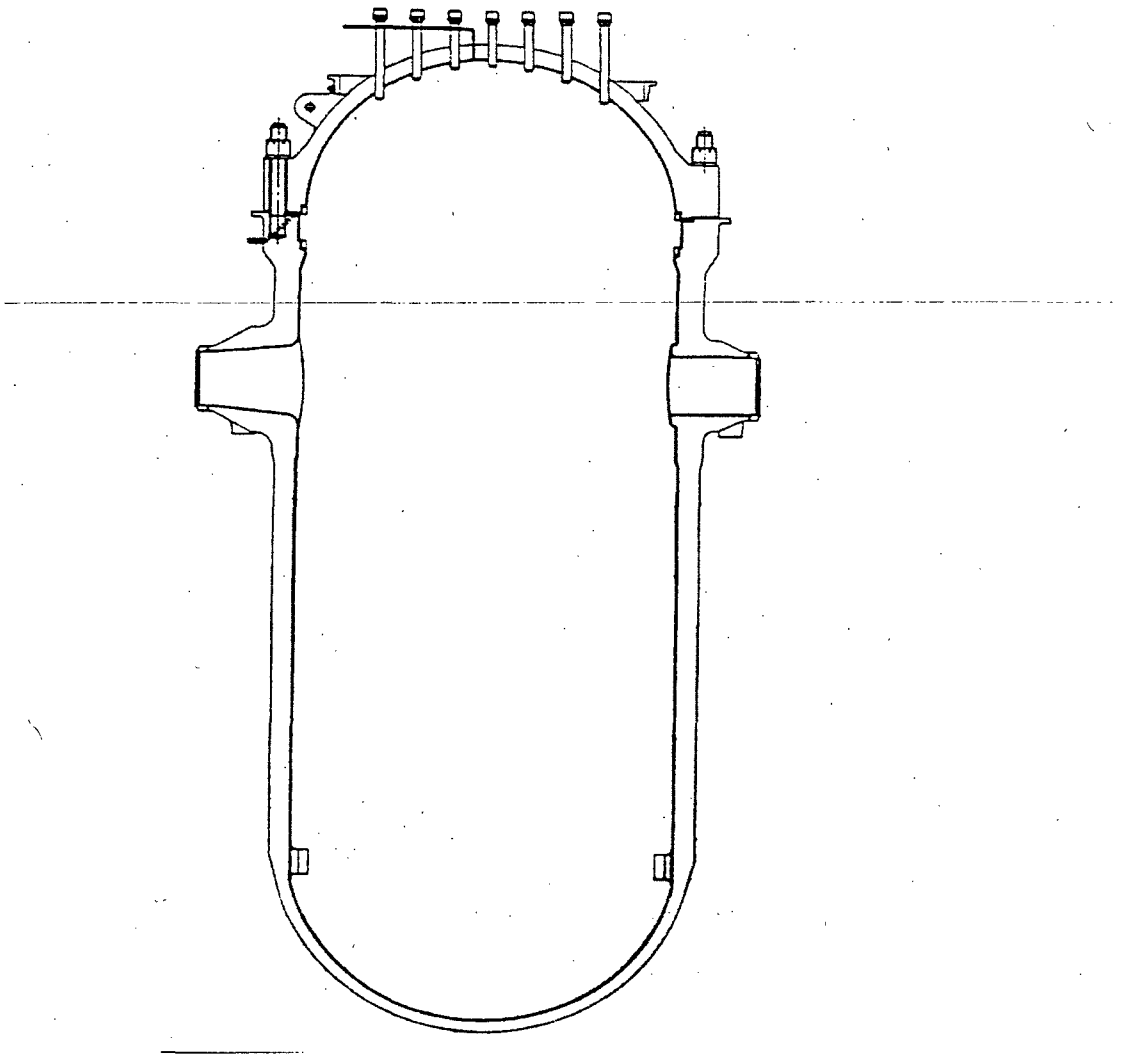
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Weld Surface and I.D. Preparation for Welds Requiring In-Service Inspection	
		Figure 5.2-1

See 1-NHY-500037-1

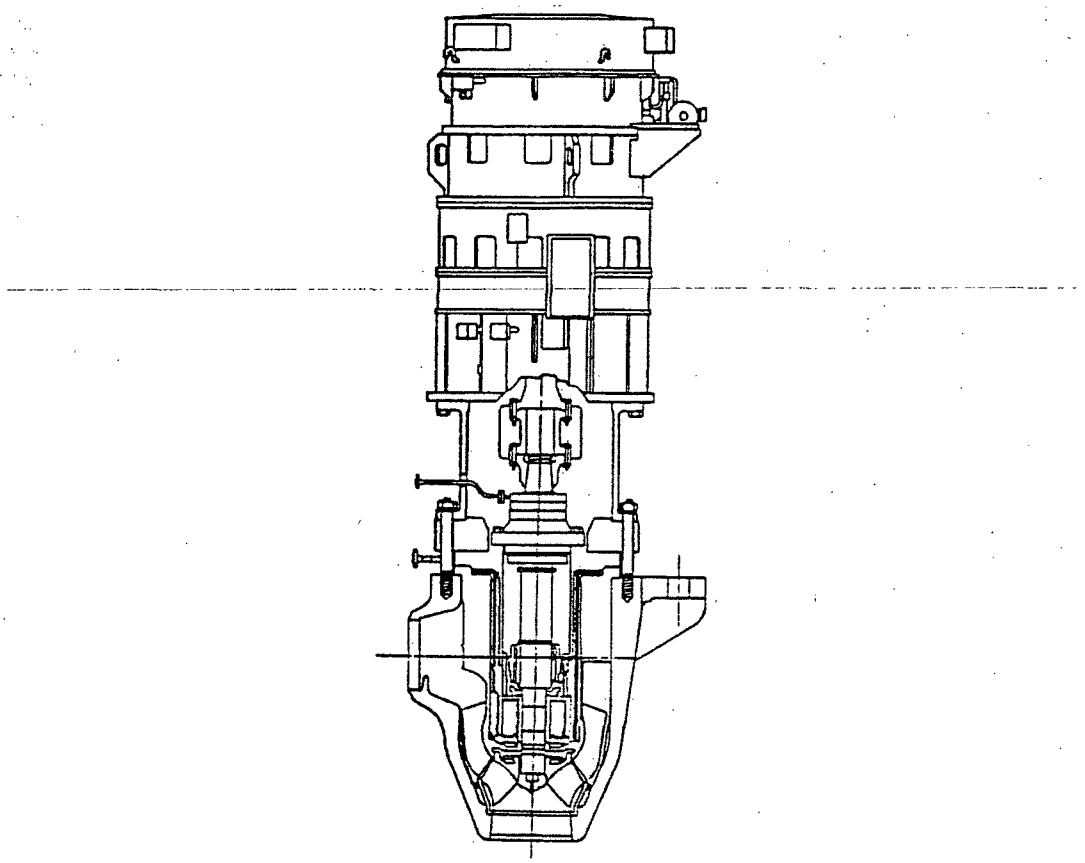
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Leak Detection System Instrumentation Engineering Diagram [2 Sheets]	
		Figure 5.2-2 Sh. 1 of 2

See 1-NHY-500037-2

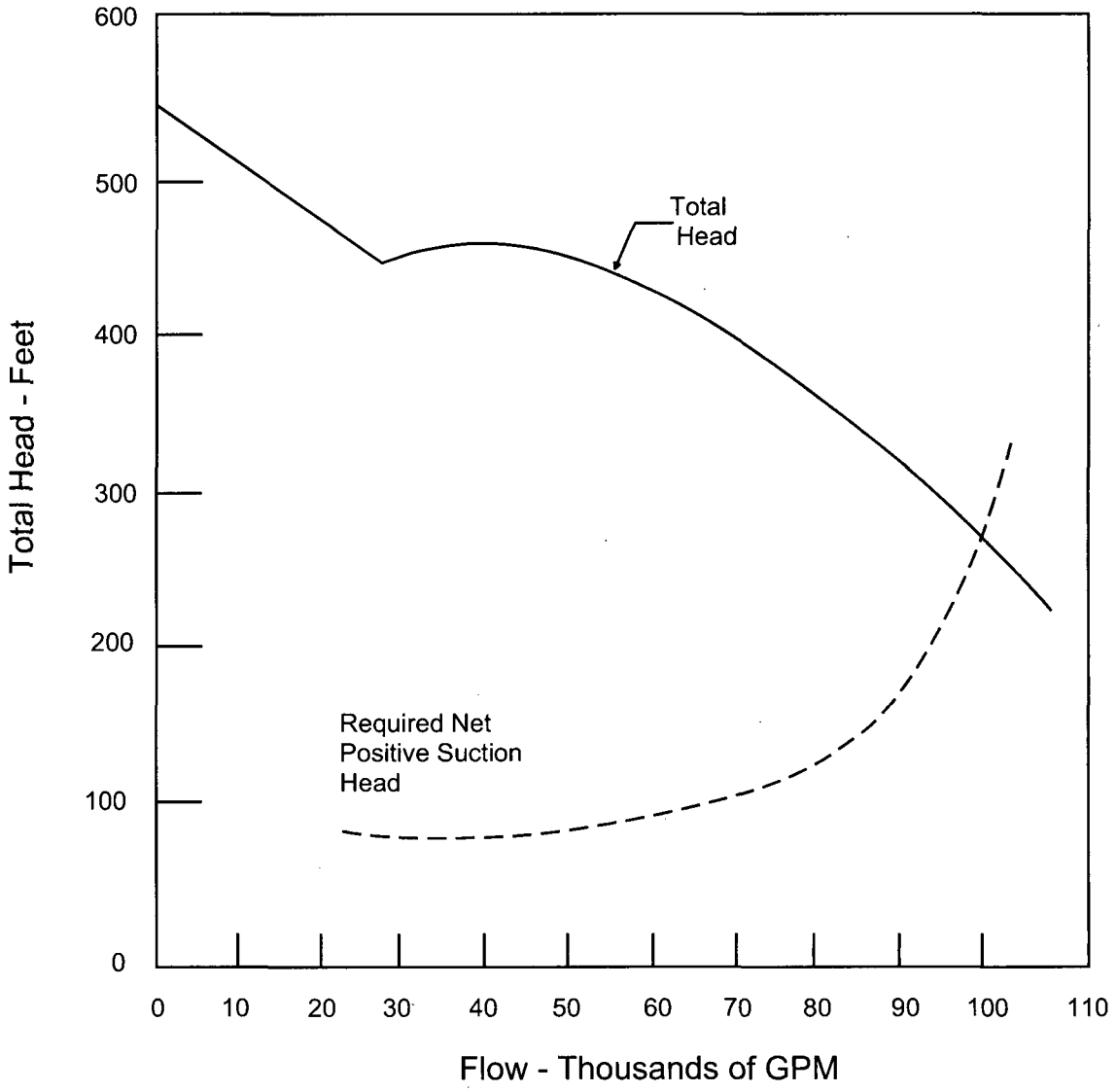
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Leak Detection System Instrumentation Engineering Diagram [2 Sheets]	
		Figure 5.2-2 Sh. 2 of 2



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Vessel	
		Figure 5.3-1

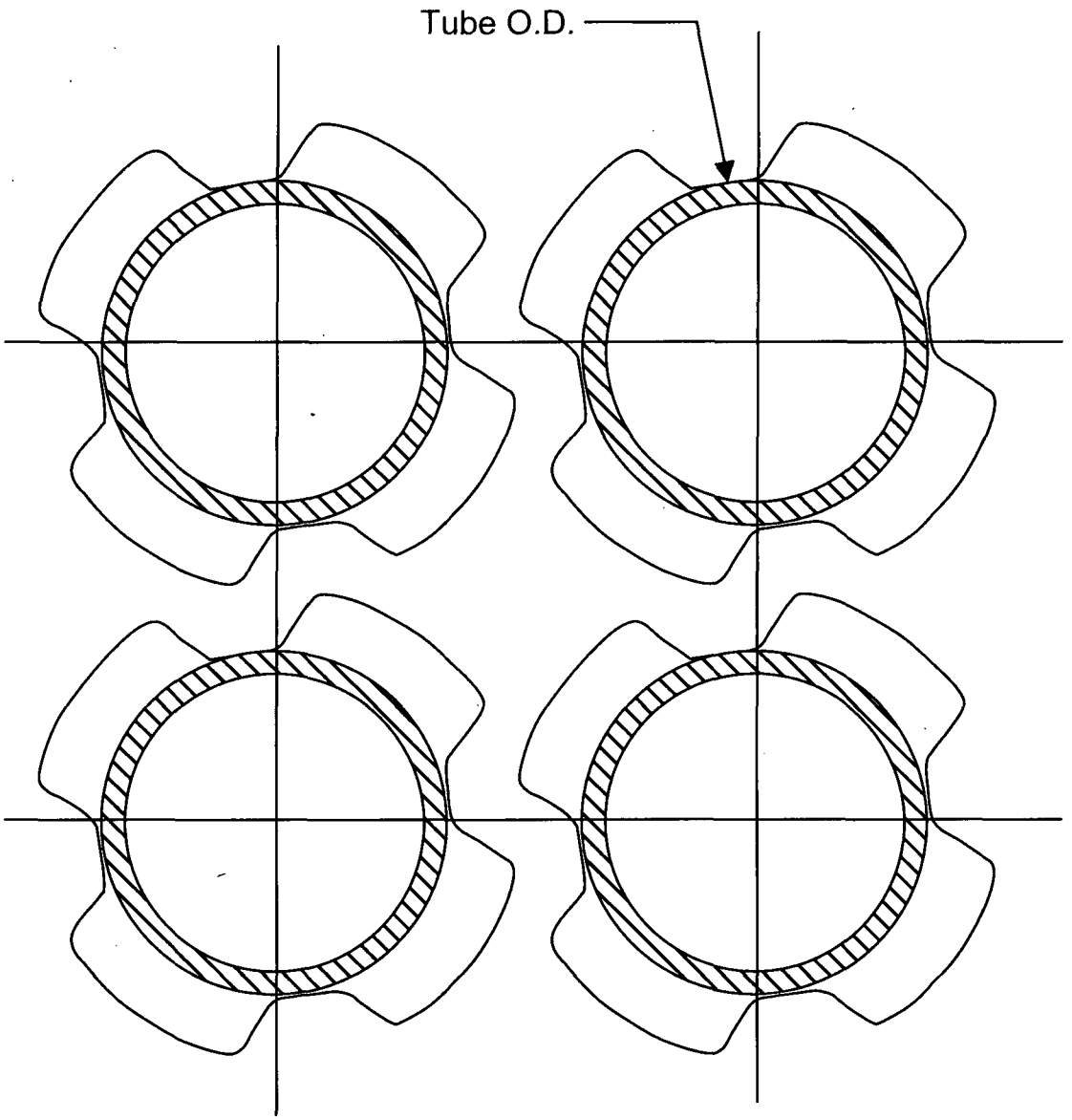


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant Controlled Leakage Pump	
		Figure 5.4-1



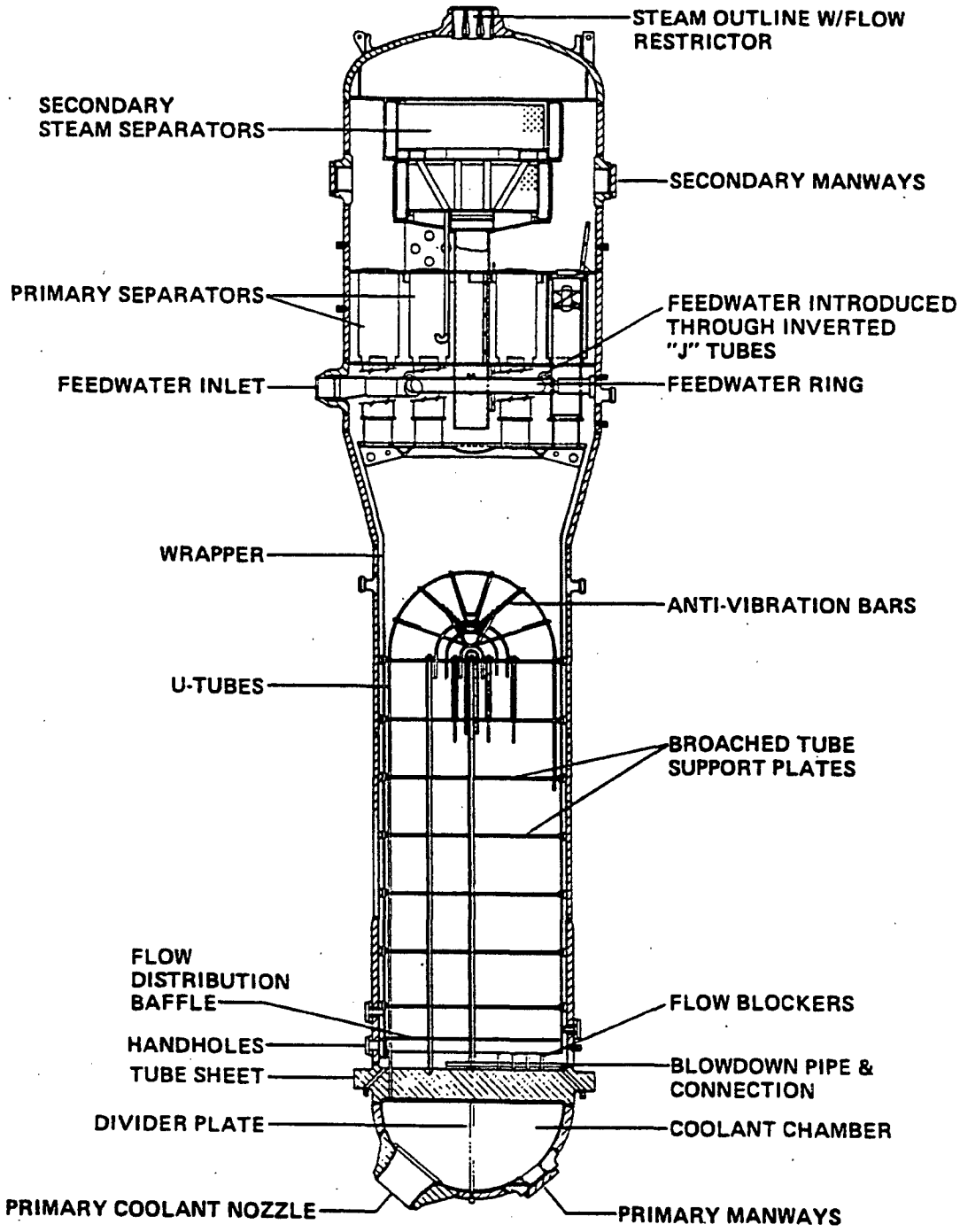
G:\Word\Images_P\UFSAR\542.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant Pump Estimated Performance Characteristic	
		Figure 5.4-2

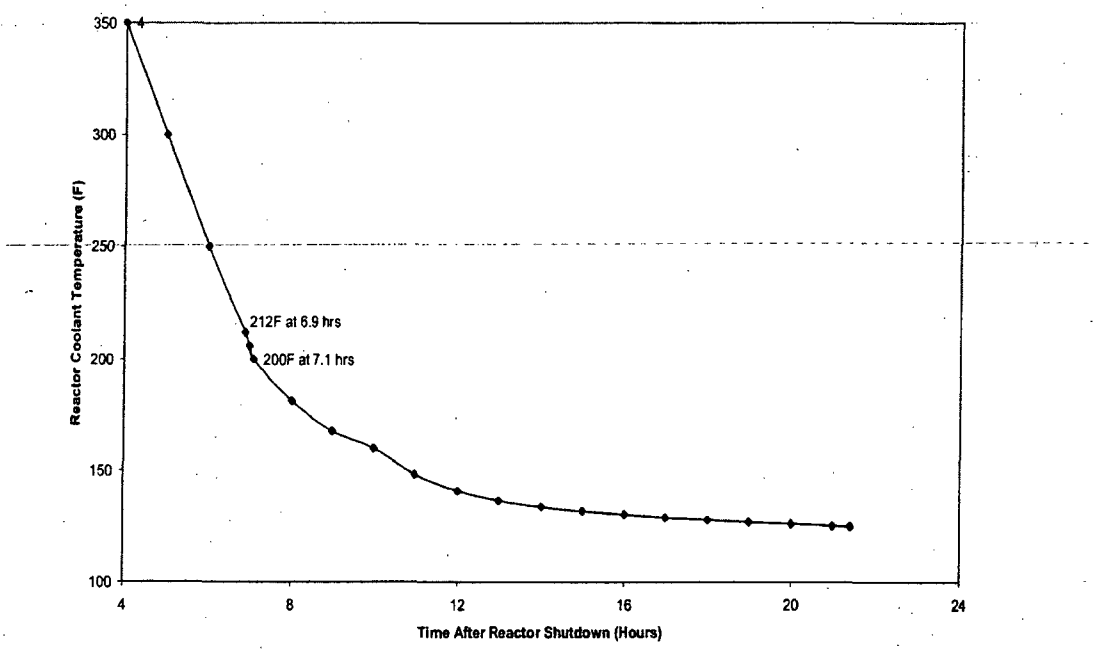


G:\Word\Images_P\UFSAR\543.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	The Quatrefoil Broached Holes	
		Figure 5.4-3

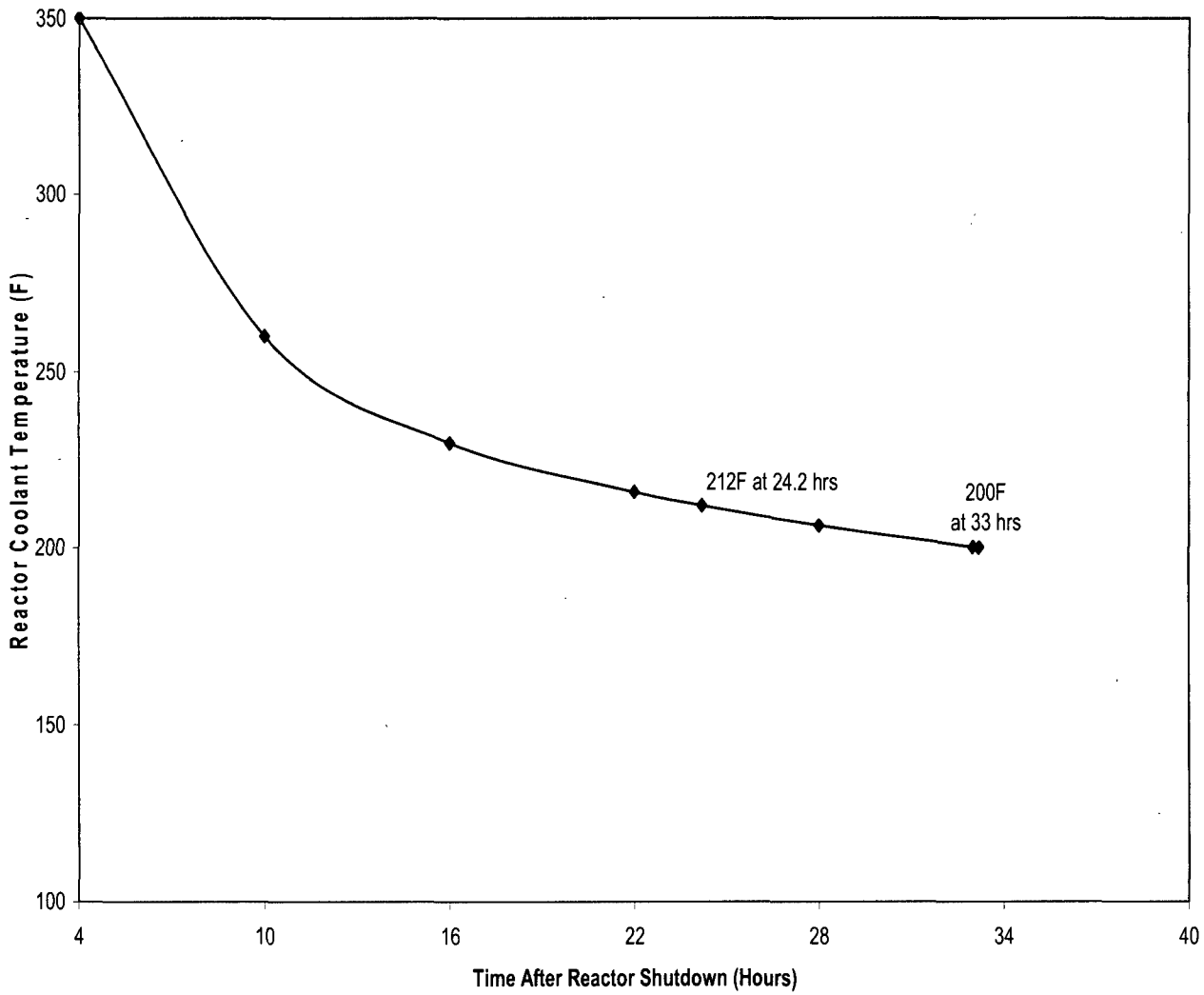


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator	
		Figure 5.4-4



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normal RHR Cooldown	
		Figure 5.4-5

Single Train RHR Cooldown - UFSAR Figure 5.4-6



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Single Train RHR Cooldown	
		Figure 5.4-6

See 1-NHY-503747

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RC - Isolation Valves Logic Diagram [2 Sheets]	
		Figure 5.4-7 Sh. 1 of 2

See 1-NHY-503748

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RC - Isolation Valves Logic Diagram [2 Sheets]	
		Figure 5.4-7 Sh. 2 of 2

See 1-NHY-503764

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RH Pumps Low Flow Recirc. Valves Logic Diagram	
		Figure 5.4-8

See PID-1-RH-B20660

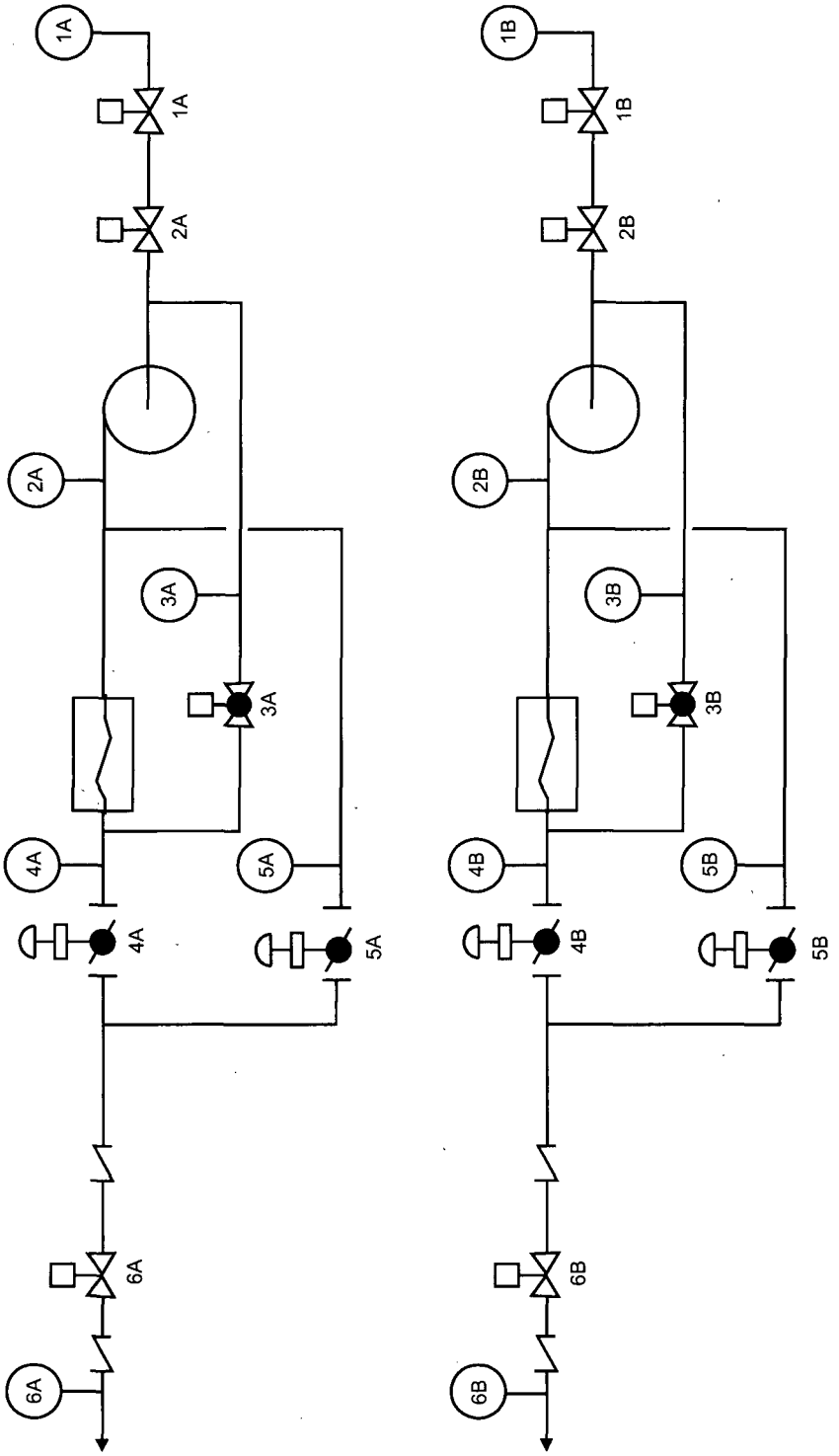
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Residual Heat Removal System Overview	
		Figure 5.4-9

See PID-1-RH-B20662

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Residual Heat Removal System Train A Detail	
		Figure 5.4-10

See PID-1-RH-B20663

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Residual Heat Removal System Train B Cross-Tie Detail	
		Figure 5.4-11



G:\Word\Images_PUFSAR\5412.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Residual Heat Removal System Process Flow Diagram	
	Sheet 1 of 3	Figure 5.4-12

NOTES TO FIGURE 5.4-12
(Sheet 1 of 2)

MODES OF OPERATION [1]

1. MODE A, INITIATION OF RHR OPERATION

When the reactor coolant temperature and pressure are reduced to 350°F and 400 psig, approximately four hours after reactor shutdown, the second phase of plant cooldown starts with the RHRS being placed in operation. Before starting the pumps, the inlet isolation valves are opened, the heat exchanger flow control valves are set at minimum flow, and the outlet valves are verified open. The automatic miniflow valves are open and remain so until the pump flow exceeds ~1400 gpm at which time they trip closed. Should the pump flow drop below ~750 gpm, the miniflow valves open automatically.

Startup of the RHRS includes a warn-up period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock on the RCS. The rate of heat removal from the reactor coolant is controlled manually by regulating the reactor coolant flow through the residual heat exchangers. The total flow is regulated automatically by control valves in the heat exchanger bypass line to maintain a constant total flow. The cooldown rate is limited to 50°F/hr based on equipment stress limits and a 120°F maximum component cooling water temperature.

2. MODE B, END CONDITIONS OF A NORMAL COOLDOWN

This situation characterizes most of the RHRS operation. As the reactor coolant temperature decreases, the flow through the residual heat exchanger is increased until all of the flow is directed through the heat exchanger to obtain maximum cooling.

VALVE ALIGNMENT CHART

Valve	Operational Mode	
	A	B
1AB	O	O
2AB	O	O
3AB	C	C
4AB	P	O
5AB	P	C
6AB	O	O

O = Open C = Closed P = Partial

(1) For the safeguards functions perfumed by the Residual Heat Removal System, refer to Section 6.3, Emergency Core Cooling System.

NOTES TO FIGURE 5.4-12
(Sheet 2 of 2)

MODE A – BEGINNING OF COOLDOWN

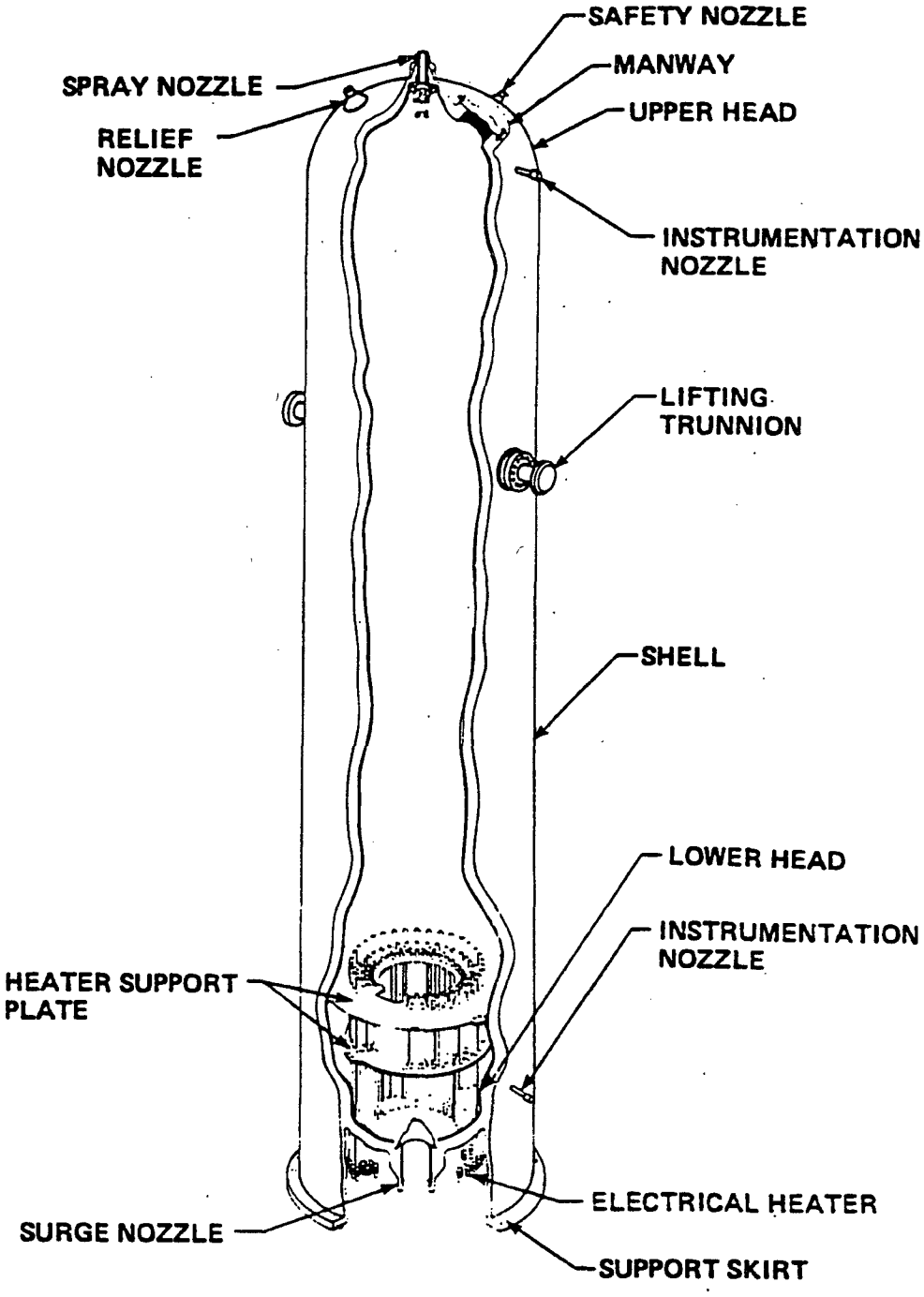
Location	Pressure (psig)	Temperature (°F)	Flow (gpm)	Flow (lb/hr x 10^5)
1A, B	400	350	3000	1.48
2AB	574.5	350	3000	1.48
3A, B	420	350	0	0
4AB	555	147	1320	0.66
5AB	559	350	1780	0.82
6AB	400	260	3000	1.48

MODE B – END OF COOLDOWN

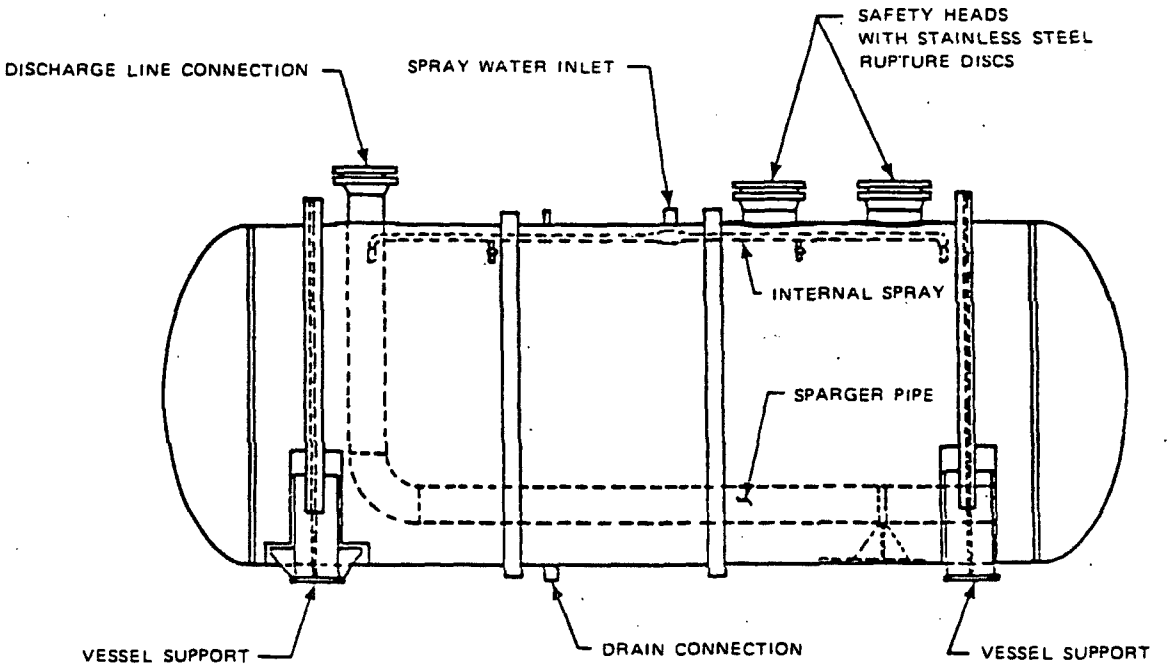
Location	Pressure (psig)	Temperature (°F)	Flow (gpm)	Flow (lb/hr x 10^5)
1AB	0	125	2960	1.48
2AB	161	125	2960	1.48
3AB	20	125	2960	1.48
4AB	117	100	2960	1.48
5AB	146	125	0	0
6AB	0	100	2960	1.48

NOTE:

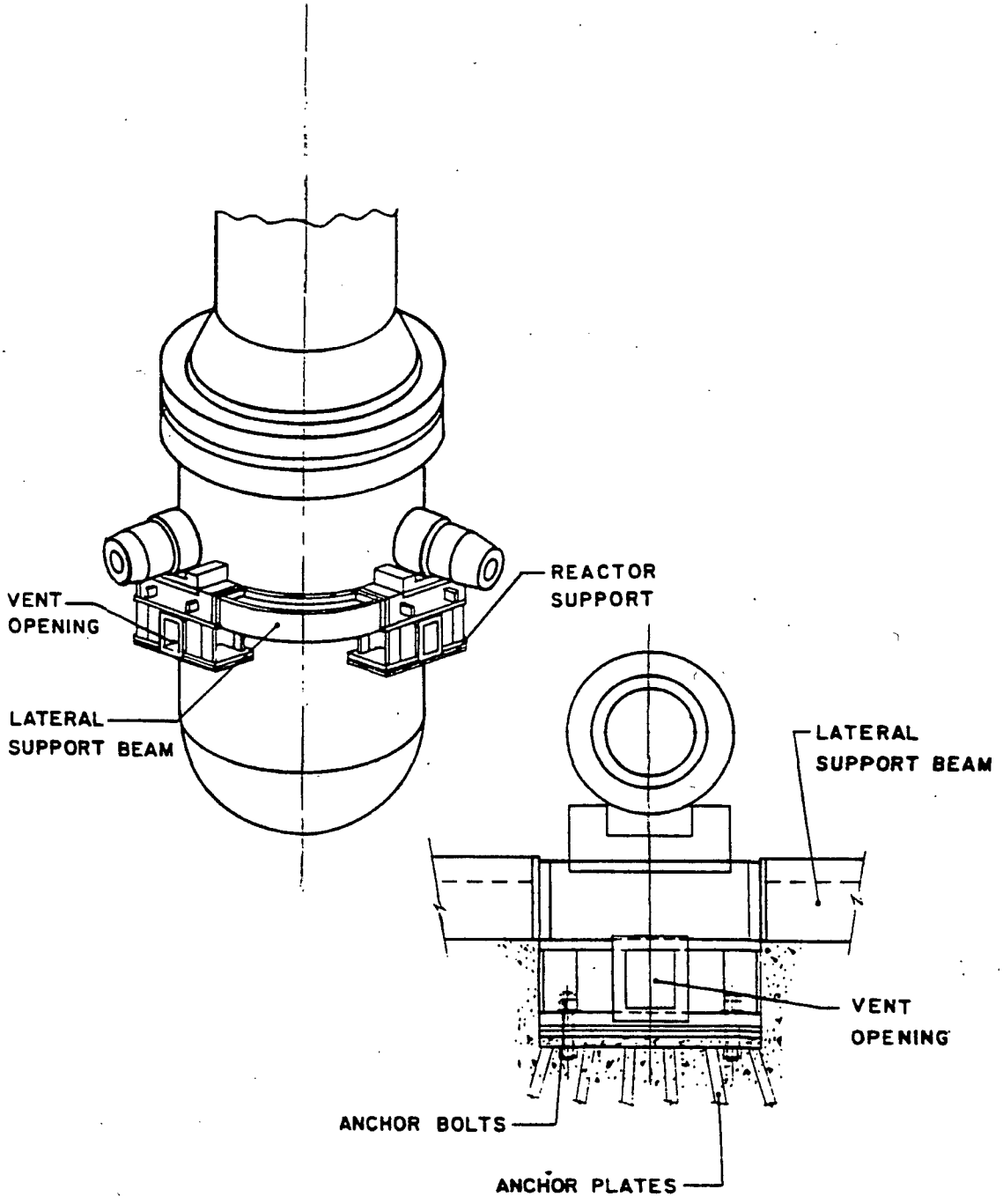
This table is illustrative of the RHR system performance in the cooldown mode when system flow is maintained at 3000 GPM. Current practice at Seabrook Station is to operate the RHR system at 3500 GPM. System flow rate operating pressure and component differential pressure may vary. Overall system performance in unaffected.



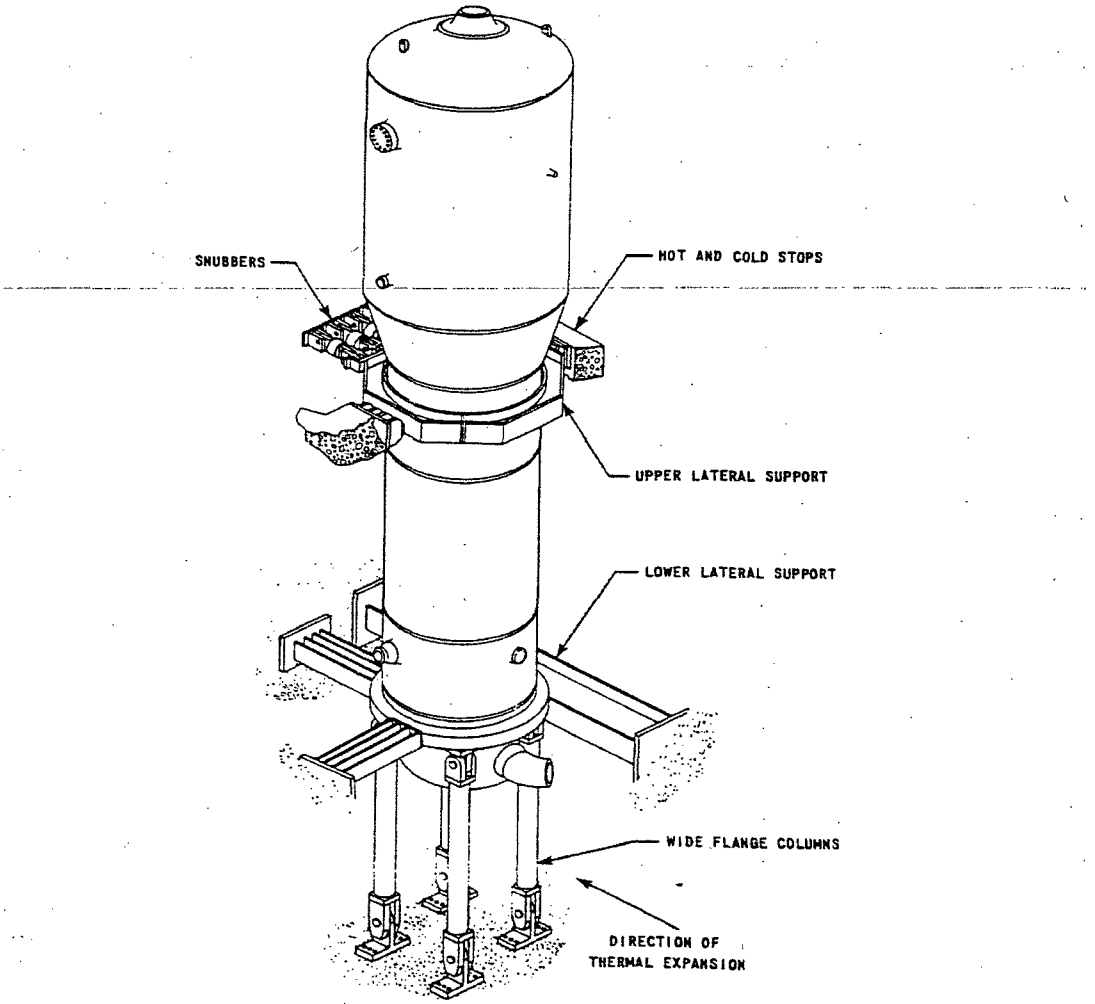
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer	
		Figure 5.4-13



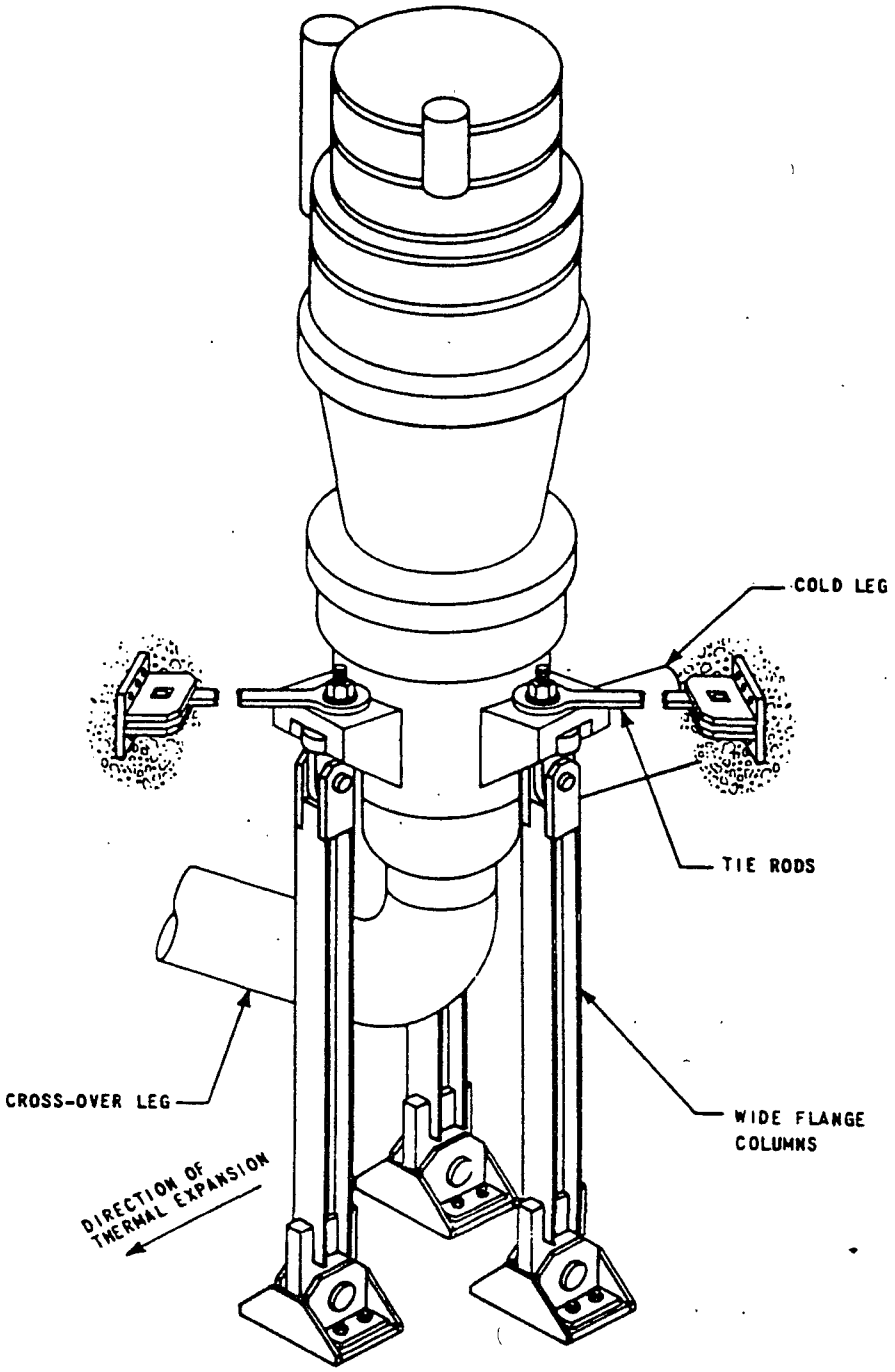
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Relief Tank	
		Figure 5.4-14



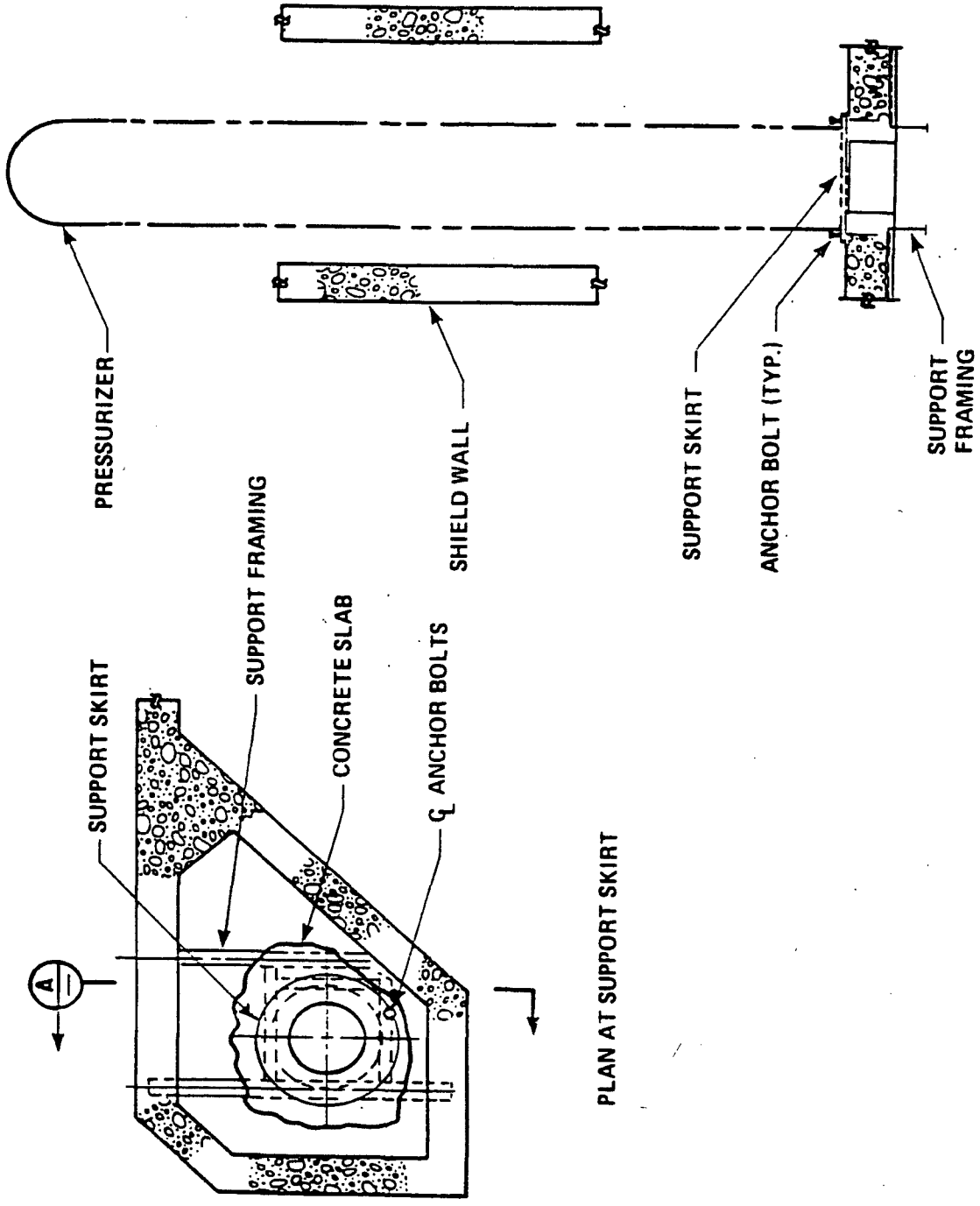
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Vessel Supports	
		Figure 5.4-15



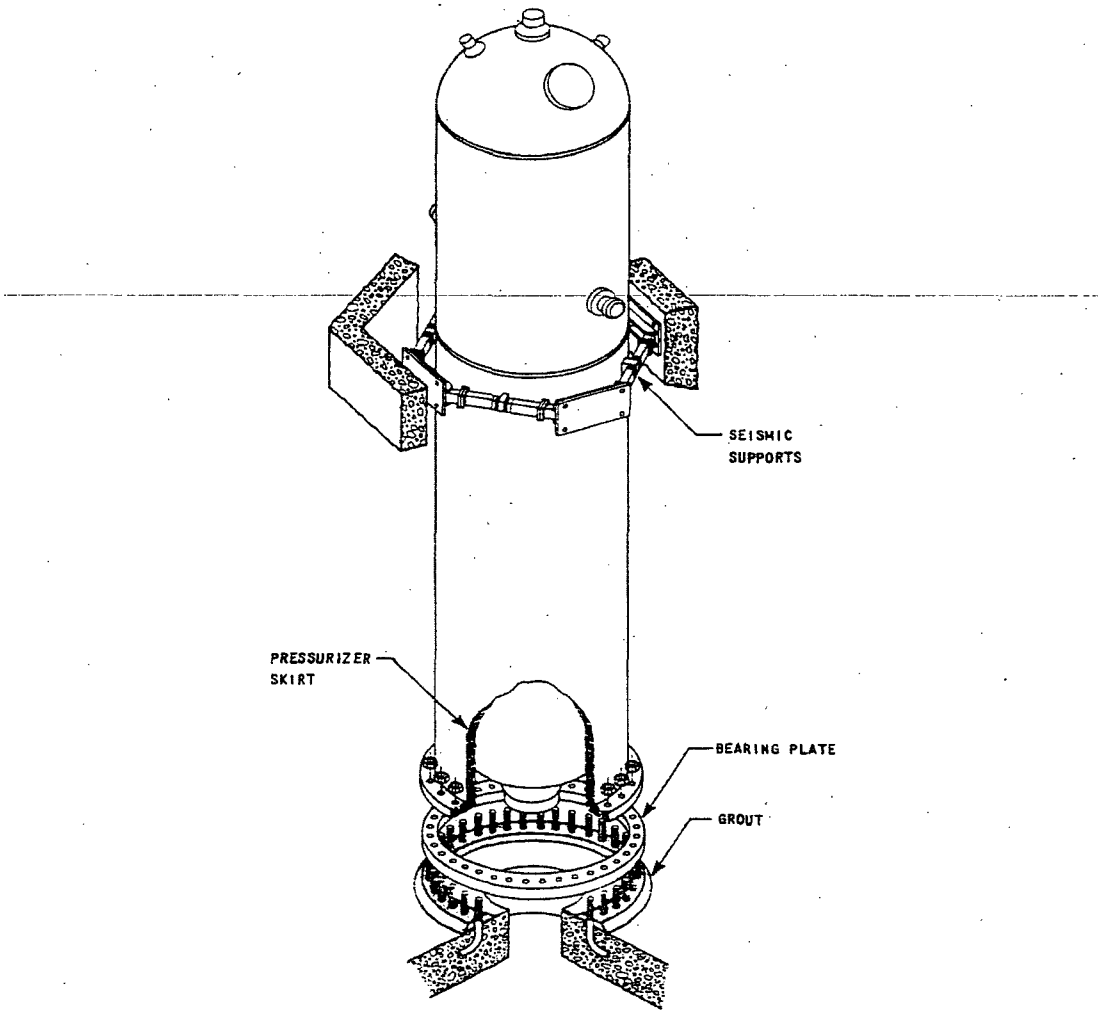
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Supports	
		Figure 5.4-16



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant Pump Supports	
		Figure 5.4-17



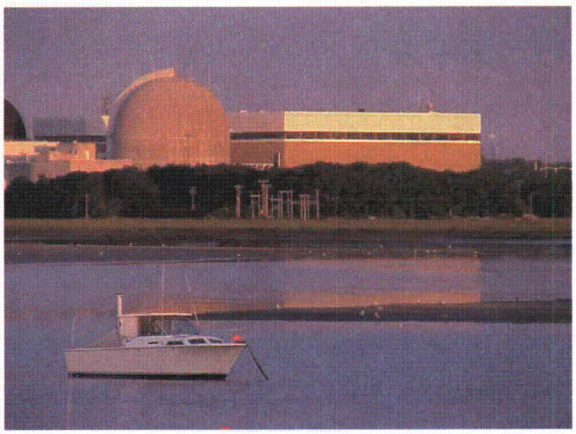
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Building Internals Pressurizer Supports	
		Figure 5.4-18



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Supports	
		Figure 5.4-19

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 6
ENGINEERED SAFETY FEATURES



6.1(B) ENGINEERED SAFETY FEATURE MATERIALS

This section provides a discussion of the materials used in Engineered Safety Feature (ESF) components and the material interactions that potentially could impair operation of ESF.

6.1(B).1 Metallic Materials

6.1(B).1.1 Materials Selection and Fabrication

Typical material specifications applicable to components in the ESF not covered by Subsection 6.1(N).1 are listed in Table 6.1(B)-1. In some cases this list of materials may not be totally inclusive; however, the listed specifications are representative of those materials used. Materials utilized in ESF have been selected for their compatibility with core and containment spray solution, and conform with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Articles NC-2160 and NC-3120, plus all applicable addenda and code cases.

Typical specifications for principal pressure-retaining ferritic materials, austenitic stainless steels and nonferrous materials, including bolting and weld rod materials used in pressure-boundary welds in the Engineered Safety Features, are listed in Table 6.1(B)-1. These materials are qualified to the requirements of the ASME Code, Section III and Section IX, and are used in procedures which have been qualified to these same rules.

The following controls are placed on fabrication and assembly of austenitic stainless steel materials used in the ESF components to insure a high degree of quality and reliability. These controls assure compliance with Regulatory Guides 1.31, 1.37, and 1.44:

a. Significant sensitization of austenitic stainless steel is avoided by imposing the following controls:

   1. Use of low-heat input welding procedures and processes, as well as maximum interpass temperature control
   2. Use of fully annealed material heat-treated in accordance with specific parameters (e.g., water quenching)
   3. Prohibition of stress relief after welding
   4. Engineering review of welding and heat-treatment procedures.

b. Specific controls are imposed during fabrication and installation to preclude contamination of stainless steel by chlorides and low melting point constituents, particularly during welding and heat treatment. These controls are monitored by checking chemical analysis certifications of materials that contact stainless steel (such as tapes, marking crayons and cleaning solutions), and engineering review of final cleaning procedures.

c. Cold working of stainless steel is prohibited after solution annealing except in mild environments where residual stresses from bending or forming are minimal. Piping for the containment spray system spray headers undergoes moderate bending during fabrication; however, this will have no deleterious affects on system performance since internal pressure during system operation is low and the chemical environment mild. In no case is cold-worked stainless steel with a yield strength of 90,000 psi or greater used in ESF constituents.

d. Each heat or lot of filler material is required to be checked to assure the presence of 5 to 20 percent ferrite as calculated from the chemical composition and/or by a magnetic measuring check of a weld pad made with the subject filler material. Maximum interpass temperature control is also imposed during welding to minimize hot cracking.

The thermal-insulation used on ESF piping and equipment inside containment is fiberglass blanket insulation of the type commercially known as Nukon, manufactured by Owens-Corning Fiberglass, with a stainless steel jacket over the outside surface of the insulation. Nukon is consistent with the recommendations of Regulatory Guide 1.36. Owens-Corning Fiberglass submitted Topical Reports OCF-1 on Nukon to the NRC for review in August 1977. The thermal insulation used on ESF piping and equipment outside containment shall be either fiberglass or calcium silicate molded sections with an aluminum jacket over the outside surface of the insulation.

6.1(B).1.2 Composition Compatibility and Stability of Containment Core Coolants

The pH of the coolants for the ESF during a loss of coolant is dependent upon the boron concentration of the Reactor Coolant System, the accumulators and the refueling water storage tank, and the concentration of sodium hydroxide in the spray additive tank. The concentrations of boron in the accumulators, and refueling water storage tank, and the concentration of sodium hydroxide in the spray additive tank are fixed and are verified periodically by analyzing samples of the solution in each tank and adjusting the chemical compositions if necessary. A recirculation/sample system is installed for the Spray Additive Tank to verify its concentration. The boron concentration in the Reactor Coolant System can vary from 0 to 4000 ppm depending upon the requirements for reactivity control. Depending on the various initial chemical compositions of the injected coolant and the Reactor Coolant System, the pH of the coolant can range from 8.8 to 9.5 at the beginning of the recirculation phase of ESF operation, and remains constant thereafter. The Ph of the spray solution during the injection phase will average between 9.1 and 10.3. The pH of the containment spray system sump water (and therefore the long-term ESF coolant) following a LOCA is monitored by withdrawing samples downstream of the RHR heat exchanger. Two sample points exist: the normal connection to the sample sink and a local sample point. Sodium hydroxide can be added, if necessary for pH adjustment, using the chemical and volume control system tanks and pumps. This assures the capability of maintaining a sump pH greater than 7.0 as recommended in Branch Technical Position MTEB 6-1. The solution would be prepared in the chemical mixing tanks and supplied to the suction of the charging pumps. The charging pump suction is fed from a cross-connect by the RHR system.

Information concerning hydrogen release by the corrosion of containment metals and the control of the hydrogen and combustible gas concentrations within the containment following a loss-of-coolant accident is contained in Subsection 6.2.5.

Water for containment spray and emergency core cooling is stored in the refueling water storage tank and the spray additive tank. Both tanks are constructed of stainless steel which has been demonstrated by test and experience to be compatible with solutions of borated water and sodium hydroxide.

6.1(B).2 Organic Materials

Significant quantities of coated surfaces inside containment that would be exposed to the post-LOCA environment are listed in Table 6.1(B)-2. The coating systems for these surfaces, except PCCW piping, are epoxy-based Keeler & Long coating systems designed for a 40-year life and are in compliance with the applicable ANSI standards for coating systems inside containment (ANSI N45.2, ANSI N101.2, ANSI N101.4 and ANSI N512). Thus the coating systems meet Regulatory Guide 1.54.

Other significant quantities of organic materials inside containment are listed in Table 6.1(B)-4.

6.1(N) ENGINEERED SAFETY FEATURE MATERIALS

6.1(N).1 Metallic Materials

6.1(N).1.1 Materials Selection and Fabrication

Typical materials specifications used for components in the Engineered Safety Features are listed in Table 6.1(N)-1. In some cases, this list of materials may not be totally inclusive. However, the listed specifications are representative of those materials used. Materials used conform with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, plus applicable and appropriate addenda and code cases.

The welding materials used for joining the ferritic base materials of the Engineered Safety Features conform to or are equivalent to ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18, and 5.20. The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Material Specifications SFA 5.11 and 5.14. The welding materials used for joining the austenitic stainless steel base materials conform to ASME Material Specifications SFA 5.4 and 5.9. These materials are qualified to the requirements of the ASME Code, Section III and Section IX, and are used in procedures which have been qualified to these same rules. The methods utilized to control delta ferrite content in austenitic stainless steel weldments are discussed in Subsection 5.2.3.

All parts of components in contact with borated water are fabricated of or clad with austenitic stainless steel or equivalent corrosion-resistant material. The integrity of the safety-related components of the Engineered Safety Features is maintained during all stages of component manufacture. Austenitic stainless steel is utilized in the final heat-treated condition as required by the respective ASME Code, Section II, material specification. Furthermore, it is required that austenitic stainless steel materials used in the engineered safety features components be handled, protected, stored, and cleaned according to recognized and accepted methods which are designed to minimize contamination that could lead to stress corrosion cracking. These controls are stipulated in Westinghouse specifications which are discussed in Subsection 5.2.3. Additional information concerning austenitic stainless steel, including the avoidance of sensitization and the prevention of intergranular attack, can be found in Subsection 5.2.3. No cold-worked austenitic stainless steels having yield strengths greater than 90,000 psi are used for components of the Engineered Safety Features within Westinghouse scope of supply.

Westinghouse-supplied engineered safety features components within the containment that would be exposed to core cooling water and containment sprays in the event of a loss-of-coolant accident used materials listed in Table 6.1(N)-1. These components are manufactured primarily of stainless steel or other corrosion-resistant material. The integrity of the materials of construction for engineered safety features equipment when exposed to post-design basis accident conditions has been evaluated. Post-design basis accident conditions were conservatively represented by test conditions. The test program (Reference 1) performed by Westinghouse considered spray and core-cooling solutions of the design chemical compositions, as well as the design chemical compositions contaminated with corrosion and deterioration products which may be transferred to the solution during recirculation. The effects of sodium (free caustic), chlorine (chloride), and fluorine (fluoride) on austenitic stainless steels were considered. Based on the results of this investigation, as well as testing by ORNL and others, the behavior of austenitic stainless steels in the post-design basis accident environment will be acceptable. No cracking is anticipated on any equipment even in the presence of postulated levels of contaminants, provided the core cooling and spray solution pH is maintained at an adequate level. The inhibitive properties of alkalinity (hydroxyl ion) against chloride cracking and the inhibitive characteristic of boric acid on fluoride cracking have been demonstrated.

Information concerning compliance with Regulatory Guides 1.31, 1.37, and 1.44 can be found in Section 1.8.

6.1(N).1.2 Composition, Compatibility, and Stability of Containment and Core Spray Coolants

Westinghouse supplied the accumulator vessels used for storing ESF coolants. The accumulators are carbon steel clad with austenitic stainless steel. Because of the corrosion resistance of these materials, significant corrosive attack on the accumulator vessels is not expected.

The accumulator vessels are filled with borated water and are pressurized with nitrogen gas. The boron concentration, as boric acid, is 2600-2900 parts per million (ppm). Samples of the solution in the accumulators are taken periodically for checks of boron concentration. Principal design parameters of the accumulators are listed in Table 6.3-1.

The method of establishing containment spray and recirculation sump pH following a loss-of-coolant accident is discussed in Subsection 6.2.2. Information concerning hydrogen release by the corrosion of containment metals and the control of the hydrogen and combustible gas concentrations within the containment following a loss-of-coolant accident is discussed in Subsection 6.2.5.

6.1(N).2 Organic Materials

Quantification of significant amounts of protective coatings on Westinghouse-supplied components located inside the Containment Building is given in Table 6.1(N)-2; the painted surfaces of Westinghouse-supplied equipment comprise a small percentage of the total painted surfaces inside containment.

For large equipment requiring protective coatings (specifically itemized in Table 6.1(N)-2, Westinghouse specifies or approves the type of coating systems utilized; requirements with which the coating system must comply are stipulated in Westinghouse process specifications, which supplement the equipment specifications. For these components, the generic types of coatings used are zinc-rich silicate or epoxy-based primer with or without chemically cured epoxy or epoxy-modified phenolic top coat.

The remaining equipment requires protective coatings on much smaller surface areas and is procured from numerous vendors; for this equipment, Westinghouse specifications require that high quality coatings be applied using good commercial practices. Table 6.1(N)-2 includes identification of this equipment and total quantities of protective coatings on such equipment.

Protective coatings for use in the reactor containment have been evaluated as to their suitability in post-design basis accident conditions. Tests have shown that certain epoxy and modified phenolic systems are satisfactory for in-containment use. This evaluation (Reference 2) considered resistance to high temperature and chemical conditions anticipated during a loss-of-coolant accident, as well as high radiation resistance.

Information regarding assurance requirements for protective coatings is addressed in the discussion on conformance to Regulatory Guide 1.54 in Section 1.8. Further compliance information has been submitted to the NRC for review (via letter NS-CE-1352 dated February 1, 1977 to C. J. Heltemes, Jr., Quality Assurance Branch, NRC, from C. Eicheldinger, Westinghouse PWRSD, Nuclear Safety Dept.) and accepted (via letter dated April 27, 1977, to C. Eicheldinger from C. J. Heltemes, Jr.).

6.1(N).3 References

1. "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss-of-Coolant Accident Environment," WCAP-7803, December 1971.
2. "Evaluation of Protective Coatings for Use in Reactor Containments," WCAP-7825, December 1971.

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

a. Design Bases

The containment design bases are established by the requirement that the system safely withstand the consequences of postulated accidents in conjunction with simultaneous occurrences of adverse environmental conditions. The containment structure and the containment enclosure, together with the exhaust system, are designed so that the offsite doses from radioactivity released under accident conditions are less than the limits set forth in 10 CFR 100.

The GOTHIC computer program (Reference 28) was used to develop a model of the Seabrook Station containment and associated safety systems. The model was employed to determine the containment response to various LOCAs and main steam line breaks at an analyzed core power level of 3659 MWt. The GOTHIC model is similar to the CONTRAST-S-MOD1 model that was originally used to determine the Seabrook Station containment response.

The peak containment pressure and temperature predicted by GOTHIC for an analyzed core power level of 3659 MWt is bounded by the results of the original containment analysis described in the following subsections. Additionally, an evaluation of the short-term LOCA mass and energy releases presented in Table 6.2-30 determined that they are bounding for an analyzed core power level of 3659 MWt, without the need to be adjusted.

Therefore, the containment design bases, evaluation, and results presented in the following subsections remain bounding and applicable for an analyzed core power level of 3659 MWt and have not been revised.

1. Postulated Accident Conditions for Containment Design

Accidents postulated to determine the containment internal design pressure and the containment design temperature include ruptures of the primary and secondary coolant system piping concurrent with a variety of single failures. The simultaneous loss of offsite power (LOOP) has also been assumed whenever it results in more restrictive design conditions.

The detailed accident conditions for primary system pipe rupture are given in Subsection 6.2.1.3; those for secondary system pipe ruptures in Subsection 6.2.1.4.

The single failures postulated for the primary system pipe ruptures include failure of a containment spray train and failure of a diesel generator. Those postulated for the main steam line breaks include failure of main feedwater pump to trip, a feedwater isolation or control valve, a main steam isolation valve, an emergency feedwater pump run out control, and a containment spray train.

The calculated maximum internal containment pressure is 49.6 psig, resulting from a (full) double-ended guillotine rupture of the primary coolant system pipe at the pump suction, with one of the two containment spray pumps failed at time of containment spray actuation, maximum initial containment pressure of 1.5 psig and minimum flow rate of 2808 gpm for the Containment Building Spray System. The system design flow rate is 2930 gpm. This is the containment design basis (DB) accident. In accordance with General Design Criterion 50 of 10 CFR 50, this value was increased to 52.0 psig, thus providing a 4.8 percent margin between the design and maximum calculated values.

Use of containment temperature responses following the main steam line breaks (MSLBs), which are more severe than those for loss-of-coolant-accident (LOCAs), to obtain an envelope for equipment qualification, is discussed in Section 3.11.

2. Postulated Accident Conditions for Subcompartment Design

Ruptures of appropriate high-energy lines at various locations within a subcompartment, concurrent with the SSE, have been postulated to determine the design requirements for the subcompartment structure. The maximum calculated pressure is not affected by LOOP or any postulated single failure because of the rapid occurrence of the peak pressure.

The accidents postulated for each subcompartment are described in Subsection 6.2.1.2. This subsection also contains the maximum calculated pressures and the design pressures associated with each subcompartment.

3. Mass and Energy Releases

Accidents involving ruptures in the primary or secondary coolant system pipes can result in the release of a significant amount of mass and energy into the containment atmosphere.

(a) Loss-of-Coolant Accident

The assumptions and details of calculational methods concerning the mass and energy releases into the containment following a rupture in the Primary Coolant System are treated in Subsection 6.2.1.3.

The sources of water mass available for release include the initial reactor coolant, borated water from the accumulators and the refueling water storage tank (RWST), and the aqueous solution in the spray additive tank (SAT). The amount of water in each source is shown in Table 6.2-1.

During the blowdown phase, the reactor coolant energy is the principal source of energy released to the containment. The high-enthalpy, high-pressure water is rapidly discharged from the break at a critical-flow rate that depends upon the conditions at the break location. Some portion of it flashes into steam due to the comparatively lower pressure and lower temperature of the containment atmosphere. The discharge rate soon drops as the reactor coolant pressure is relieved. The end of blowdown is defined as the time when the flow at the break reaches a minimum.

Following the blowdown phase, additional heat is transferred via the coolant to the containment. This is comprised of decay heat, core internal energy, reactor vessel metal energy, steam generator energy, metal-water heat of reaction, and the energy of the coolant itself.

The post-blowdown phase is characterized by a long, slow transient in which mass and energy discharge rates depend upon the flooding rate of the core. The flooding rate and the performance of the Emergency Core Cooling System (ECCS) are discussed in Section 6.3.

(b) Secondary System Pipe Rupture

The details concerning assumptions and calculational methods dealing with mass and energy releases into the containment following a rupture in the Secondary Coolant System are discussed in Subsection 6.2.1.4.

For ruptures in the main steam line, the sources of mass available for release include the steam generator initial inventory, the main and emergency feedwater pumped into the steam generator before isolation, the feedwater in the unisolated piping which eventually flashes into the steam generator, and the steam in the unisolated main steam piping. The first item varies from 124,000 lbm to 168,000 lm, depending on the initial operating power level. The rest depends on the time of isolation and the single failure postulated. The sources of energy for release into the containment include the energy of the initial steam generator inventory and the heat transferred from the primary system to the secondary system during the transient period.

The high-enthalpy and high-pressure steam is discharged from the break at the critical-flow rate upon rupture of a main steam pipe at the exit of a steam generator flow restrictor. The discharge rate decreases as the affected steam generator is depressurized. Depending on the break type, an isolation signal is generated by either the Reactor Protection System or by the instrumentation system monitoring the containment pressure. The signal causes isolation of the main feedwater lines and the main steam. The blowdown drops considerably following the isolation, but the flow from the affected steam generator continues until dryout time. Thereafter, the blowdown flow rate drops to a value equal to the emergency feedwater flow rate.

The effects of a postulated feedwater line rupture are not as severe as the main steam line break because the break effluent of a feedwater line rupture is at a lower specific enthalpy. Therefore, feedwater line break mass and energy releases to the containment are not addressed since they are bounded by steam line break releases.

4. Effects of Engineered Safety Features on Energy Removal

The energy released as a result of a LOCA or a secondary system pipe rupture is partially removed from the containment by the Containment Building Spray (CBS) System and the Residual Heat Removal (RHR) System through the Station Service Water (SSW) System.

Shortly following a coolant system pipe rupture, ECCS automatically starts. When the containment pressure reaches a high-pressure setpoint, borated water from the RWST mixed with spray additives is pumped through the spray nozzles to the containment atmosphere by the CBS system. When the RWST reaches a low-level setpoint, the source of water for the CBS system and the RHR portion of the ECCS is switched to the containment sump. In this recirculation mode the containment sump water is cooled by the CBS and RHR heat exchangers, and either pumped into the reactor vessel or sprayed into the containment atmosphere. The CBS and RHR heat exchangers transfer heat to the service water via the intermediate closed-loop primary component cooling water.

All Engineered Safety Features which are available for containment heat removal are described in Subsection 6.2.2. Relevant system parameters are summarized in Table 6.2-2.

5. Capability of ESF for Post-Accident Pressure Reduction and Energy Removal

All Engineered Safety Features are separated into two independent subsystems of equal capability to meet the single failure criteria. One hundred percent redundancy is also provided in the associated electrical actuation systems. Emergency power is supplied from redundant onsite power sources.

The containment and its heat removal systems are designed so that operation of only one of the two CBS trains, in conjunction with the ECCS at any point in its range of capability, is sufficient to reduce the pressure of the containment atmosphere to within half of its calculated peak value in less than 24 hours following containment design basis LOCA.

6. Containment Leakage Rate Bases

The containment is isolated from the outside environment following major accidents by the Containment Isolation System. The presence of the containment enclosure and the use of exhaust fans to produce a slightly sub-atmospheric pressure in the space between the containment enclosure and the containment structure reduce the direct leakage from the structure to the environment to zero. The containment design is such that the maximum rate of leakage from the containment structure to the containment enclosure following a coolant pipe rupture is 0.15 percent of the containment air mass per day. The containment heat removal systems are capable of reducing the containment pressure, within 24 hours following the accident, to such a value that the volumetric leakage rate is less than one-half of the maximum value. The use of HEPA and charcoal filters in the exhaust line from the containment enclosure reduces the discharge of radioactive iodine into the environment to the extent that offsite doses following an accident are within the guidelines of 10 CFR 100. The exhaust system is discussed in Subsection 6.2.3.

The periodic testing and surveillance program to assure the above containment leakage rate is discussed in Subsection 6.2.1.6 and in the Technical Specifications.

7. Bases for Minimum Containment Pressure Analysis

Assumptions in the minimum pressure analysis for ECCS confirmatory studies are based upon maximizing the ESF heat removal capability and other heat removal mechanisms. They are discussed in Subsection 6.2.1.5.

b. Design Features

1. Containment Structure

The containment structure is a reinforced concrete cylinder with a hemispherical dome and a reinforced concrete foundation, keyed into the rock by the depression for the reactor cavity pit and by continuous bearing around the periphery of the foundation mat. A welded steel liner plate is anchored to the inside face of the concrete as a leak-tight membrane. The liner plate on top of the foundation slab is protected by a 4-foot thick concrete slab which serves to carry internal equipment loads and forms the floor of the containment. A detailed description of the containment structure is given in Subsection 3.8.1. Figures showing typical sections through the containment can be found in Section 1.2.

2. Containment Enclosure

The containment enclosure is a reinforced concrete cylindrical structure with a hemispherical dome. Detailed descriptions of the structure are presented in Subsection 3.8.4. Figures showing sections and elevations of the containment enclosure can be found in Section 1.2.

Fans maintain the pressure in the space between the containment structure and the containment enclosure at a value slightly below the atmospheric pressure following a LOCA. All joints and penetrations are welded or sealed to ensure air tightness.

3. Protection Against Dynamic Effects

Provision is made for protecting the containment structure, internal compartment and ESF systems against loss of function from effects that could occur following postulated accidents. These provisions include physical barriers designed to minimize the dynamic effects of missiles and pipe whip, pipe whip restraints to limit damage from ruptured lines, physical separation by distance and redundancy of components and/or safety trains, as appropriate. A detailed discussion of provisions for protection against dynamic effects inside the containment is presented in Section 3.6 and Subsection 3.5.1.2.

4. Codes, Standards and Guides

Codes, standards and guides applied to the design of the containment structure and internal structures are identified in Subsections 3.8.1 and 3.8.3.

5. Protection Against External Pressure

The containment structure, including its steel components, is designed to withstand a maximum external pressure of 3.5 psi (differential pressure). The most limiting event for establishing the required external pressure is the inadvertent actuation of the Containment Spray System, which results in a negative pressure differential of 2.6 psi. The analysis follows:

The containment is normally maintained at a slight positive pressure by the containment online purge subsystem (see Updated FSAR Subsection 9.4.5.2c.2). With this system in operation, the containment is maintained at a nominal positive pressure of 0.5 psig, with high and low pressure alarms at 0.65 psig and 0.35 psig, respectively. Accordingly, if this system is in operation, the initial pressure inside containment is always positive as an initial starting condition.

However, because the above system is not redundant, the analysis for this event is based upon an initial containment ambient pressure of 14.6 psia. Initial temperatures and relative humidity are 120°F and 90 percent. The analysis utilized the containment free volume, geometrical data and passive heat sink data as provided in Updated FSAR Table 6.2-1, Table 6.2-3 and Table 6.2-4.

Both spray trains are assumed to operate with a maximum flow rate of 3500 gpm per train. The analysis assumes the total capacity of the refueling water storage tank of 475,000 gallons is available for spray at a minimum temperature of 50°F (see Updated FSAR Subsection 6.2.2.3). Using the computer code CONTRAST-S-MOD-1, the minimum resultant containment pressure of 12.0 psia results from the inadvertent actuation of sprays.

The above event produces a negative pressure differential of 2.7 psig. This is to be compared with the containment structure design differential pressure of negative 3.5 psig (see Updated FSAR Subsection 6.2.1.1b.5). This provides a nominal negative pressure margin of 0.8 psig.

6. Potential Water Traps

Principal areas where water could be trapped and prevented from being circulated by the ECCS and CBS system during a major accident are the reactor cavity and refueling canal. Figures showing plan and sectional views of these areas are given in Section 1.2. Other minor volumes not addressed here have been taken into account in calculating the NPSH available to the ECCS and CBS system pumps, as discussed in Subsection 6.2.2.

In the event of the LOCA, the reactor cavity, incore instrument sump, ECCS sump, and other minor volumes will be filled with water up to elevation (-)26 feet during the operation of the ECCS and CBS systems. A total of 17,070 cubic feet of water is required to fill these two volumes up to the (-)26 feet elevation, resulting in a reduction of water height of approximately 17 inches which would otherwise contribute to NPSH available to the ESF pumps. This has been factored into the NPSH calculations for the pumps.

Three four-inch diameter drain lines connected to a common header permit a flow path between the reactor cavity, refueling canal, and the rest of the containment, thereby preventing water from being trapped in the reactor cavity and refueling canal to elevations above the water level in the rest of the containment. In the event that all the drain lines were blocked or all the normally open drain valves were left closed (see Subsection 6.2.2.3a.3 for further details) the loss of 5760 cubic feet of water above the (-)26 feet elevation would occur. This would result in a further reduction of water height of approximately 6 inches. This has been factored into the NPSH calculation for pumps also.

7. Containment Cooling and Ventilation System

During normal operation, fan coolers maintain the containment atmosphere below 120°F. It uses the cooling water from the Primary Component Cooling Water System, while its humidity is permitted to vary. Five of six fan coolers operate continuously, with the sixth fan cooler serving as an installed spare.

Cooled air from the containment fan coolers is also directed into the reactor cavity at various locations to maintain equipment and concrete at or below design temperatures.

Three induced draft fans, two of which are normally operating, draw air past the control rod drive mechanism through a cooling shroud to maintain the equipment at or below its design operating temperature. The third fan serves as an installed spare.

An Online Purge System is provided to periodically purge the containment air to control airborne radioactivity. A separate system provides pre-entry purging of the containment and purging during refueling operations. The Containment Structure Heating and Cooling System and the Containment Online and Pre-entry Purge Systems are discussed in detail in Subsection 9.4.5.

c. Design Evaluations

1. Containment Pressure-Temperature Response Following A LOCA or a Secondary Coolant System Pipe Rupture

Containment pressure and temperature responses following a variety of postulated ruptures in the primary and secondary coolant system pipes have been calculated by means of the computer program CONTRAST-S-MOD1 which is described in detail in Reference 1.

(a) System Parameters and Initial Conditions

The system parameters and the initial conditions used in the pressure-temperature response analysis are presented in Table 6.2-1 and Table 6.2-2. The parameters and the initial conditions are chosen to maximize the containment pressure and temperature responses unless their effects are insignificant.

(b) Actuation of Containment Sprays

The Containment Spray System is initiated by a containment spray actuation signal which is generated by the containment Hi-3 ("P" signal). The analysis limit for this setpoint is 19.8 psig. The maximum delay in signal processing and the response time of the protection system instrumentation is one second. The stroking time of the spray system valves is 20 seconds to become fully open with the exception of CBS-V38 and CBS-V43 which have a maximum stroke time of 25 seconds. The maximum delay time to bring the pumps to full speed and to fill the feed lines and headers is 38 seconds following the receipt of this signal. The valve opening and fill-up of the line takes place concurrently. Therefore, the maximum delay of 39 seconds after generation of the "P" signal consists of 1 second plus 38 seconds for fill-up.

In the case of loss of offsite power concurrent with a coolant system pipe rupture, the emergency electric power from the onsite diesel generators will be available in 12 seconds. Receipt of the "P" signal by the actuation sequencer, which is discussed in detail in Subsection 8.3.1, will cause the spray valves to start opening immediately, or as soon as the emergency power is available. If the "P" signal is received within 27 seconds or between 27 and 52 seconds following an accident, the spray pumps will be started at 27 seconds or 52 seconds respectively. If the signal is received after 52 seconds, the pumps will be started immediately. Thus, for any "P" signal received before 27 seconds following an accident, which is true for all LOCA cases analyzed, the spray time is always at 65 seconds after the accident. For MSLB cases analyzed, the time to generate the "P" signal varies over a wide range. However, for all MSLB cases, a constant conservative spray delay time of 65 seconds after receipt of the signal has been assumed.

(c) Containment Passive Heat Sinks

The structure and equipment within the containment which have been modeled as passive heat sinks are listed in Table 6.2-3. Table 6.2-4 gives the thermophysical properties of the materials for the heat sinks. In Table 6.2-3, some of the heat sinks of similar structure have been lumped into various thickness groups. For internal heat sinks exposed to the containment on one side only, it is conservatively assumed that the other side is insulated. For those exposed on both sides and having a plane of symmetry, only half of the heat sink is modeled but the effective surface area is doubled. For the steel-lined structures, the steel-concrete interface resistance has been modeled as an equivalent air gap of inch. This is conservative when compared with experimental data in References 2 and 3. The interface conductance therein varies from 100,000 Btu/hr-ft^2-°F for a very good contact to 10 Btu/hr-ft^2-°F for a very poor contact. The lower limit corresponds to the equivalent conductance of an air gap approximately 20 mils thick.

The heat transfer between the containment and the passive heat sinks is calculated in the CONTRAST-S code by combining the contribution from the condensation and convection. An effective heat transfer coefficient, h_{eff}, based on the temperature difference between the containment atmosphere and the heat sink surface can be defined as follows:

$$h_{eff} = f(h_{cond}-h_{conv})(T_{sat}-T_{wall})/(T_{con}-T_{wall}) + h_{conv}$$

where,

h_{cond} = condensing heat transfer coefficient

h_{conv} = convective heat transfer coefficient

T_{sat} = containment atmosphere dew point

T_{wall} = surface temperature of heat sink

T_{con} = containment atmosphere temperature

f = 1, if $T_{sat} > T_{wall}$, 0 otherwise.

In the estimation of the condensing heat transfer coefficient, the modified Tagami correlation (Reference 4) is used for the LOCA analysis while the Uchida correlation (Reference 5) is used for the MSLB analysis. The convective heat transfer coefficient has been assumed to have a constant value of 2 Btu/hr-ft^2-°F in all accident cases analyzed. This corresponds to the minimum value given in the Tagami correlation. The heat transfer through a passive heat sink is computed in the CONTRAST-S code by solving the partial differential equation for unsteady, one-dimensional heat conduction using a fully implicit finite-difference scheme which is unconditionally stable. The accuracy of the solution can be improved by decreasing the grid spacing as well as the time increment. A sensitivity study (Reference 6) has been performed to establish the upper limit of the grid spacing in concrete for a reasonable degree of accuracy. It was found that grid spacings of 0.05 inch for concrete a few inches thick is adequate. Considerably larger grid spacings have been found adequate for steel, due to the relatively smaller temperature gradients.

The selection of the grid spacings has also been guided by the following criteria, suggested in Reference 7, to avoid large, meaningless fluctuations in the solution:

$$\frac{(\Delta x)}{\propto(\Delta t)} \geq 1 + \frac{h(\Delta x)}{k}$$

where,

Δx = grid spacing

Δt = time increment

$\propto$ = thermal diffusivity of the material

h = heat transfer coefficient at the heat sink surface

k = thermal conductivity of the material

For a typical heat transfer coefficient of 80 Btu/hr-ft^2-°F during the containment pressure-temperature transient, this criterion suggests a grid spacing greater than 0.038 inch for concrete (k = 0.083 Btu/hr-ft-°F and $\propto$ = 0.028 ft^2/hr) and greater than 0.138 inch for carbon steel (k = 27.0 Btu/hr-ft-°F and $\propto$ = 0.459 ft^2/hr), when a time increment of 1 second is used.

(d) Containment Pressure and Temperature Responses

(1) Loss-of-Coolant Accident

A spectrum of postulated reactor coolant system pipe ruptures has been considered. This includes three break locations listed in Table 6.2-5 and three break types and sizes given in Table 6.2-6.

The ESF systems relied upon to mitigate the consequences of a LOCA are the ECCS and CBS systems operating in conjunction with the Primary Component Cooling Water System and the SSW system. Failure of one of the two CBS trains obviously would result in more severe containment conditions. The failure may be caused by failure of a pump, failure of a valve, or, assuming loss of offsite power concurrent with the LOCA, failure of a diesel generator. As to the ECCS, any of its various components may fail, leading to partial loss of its cooling effect. In the extreme case, one of the two trains may fail entirely, due to failure of a diesel generator to start, assuming a concurrent loss of offsite power. The above two limiting single active failures (SAFs) are delineated in Table 6.2-7.

The cooling water for ultimate heat disposal is available from two sources, namely, the circulating water tunnels and the cooling tower basin. Section 9.2 presents a detailed description of these sources. The tunnel water temperature is not expected to exceed 65°F. During the summer months, extended hot weather combined with ocean current changes can result in minor ocean temperature excursions above the 65°F design temperature threshold. System analysis has been performed to permit continued plant operation up to a maximum ocean temperature of 68.5°F. Concerning containment heat removal, the use of tunnel water following a LOCA is more restrictive than that of the cooling tower basin water. All LOCA analyses presented have been performed using the ocean as the ultimate heat sink.

The effects of the break type/size and the SAF have been fully investigated in the case of a rupture at the pump suction (Location No. 1). In the case of Locations No. 2 (cold leg) and No. 3 (hot leg), the results are expected to be less severe than those for Location No. 1 for the most limiting break type/size and SAF combination, namely Break No. 1 and SAF No. 2. Overall, a total of six cases have been analyzed. The details of the calculation of the mass and energy releases for the six cases analyzed are given in Subsection 6.2.1.3. Those results have been based on a temperature of 120°F for the safety injection water. For Seabrook Station Unit 1, the maximum temperature of the injection water is 100°F.

The time for switchover from the injection mode to the recirculation mode depends upon the injection flow (charging pumps, high pressure safety injection pumps, and low pressure safety injection pumps), spray flow and the quantity of water available in the RWST. Since one of the two spray trains has been assumed to fail in all cases analyzed, the recirculation times are calculated for the maximum safety injection (two injection trains) and the minimum safety injection (one injection train) cases. With the injection and spray flow rates given in Table 6.2-2 and the available quantity of water in the RWST provided in Table 6.2-1, the recirculation times for the maximum safety injection and the minimum safety injection cases are calculated to be 1688 seconds and 2755 seconds, respectively.

The transient responses of the containment pressure, temperature and sump water temperature for the six cases analyzed are shown in Figure 6.2-1, Figure 6.2-2, Figure 6.2-3, Figure 6.2-4, Figure 6.2-5, Figure 6.2-6, Figure 6.2-7, Figure 6.2-8, Figure 6.2-9, Figure 6.2-10, Figure 6.2-11, Figure 6.2-12, Figure 6.2-13, Figure 6.2-14, Figure 6.2-15, Figure 6.2-16, Figure 6.2-17 and Figure 6.2-18. The transients show that, following blowdown, and prior to refill, the containment pressure and containment temperature drop because the mass and energy released through the breaks ceases completely at the end of the blowdown period. However, the reflood and post-reflood mass and energy released from the break increase the containment pressure and temperature again. The containment pressure and temperature eventually drop due to decreases in the mass and energy release rates, and due to energy removal by containment spray and passive heat sinks. After the switchover from the injection mode to the recirculation mode, the containment spray water is taken from the containment sump through the containment spray heat exchanger, and is at a higher temperature than that of the RWST. The spray heat removal rate thus drops.

The containment pressure and temperature, therefore, increase after the start of recirculation. The mass and energy release data presented in Subsection 6.2.1.3 is based on the conservative assumption that the remainder of the energy (Reactor Coolant System, core-stored, primary and secondary metal, etc.) exits through the break within 3600 seconds. The reduced heat removal capacity by the spray due to increased temperature, together with conservative energy release to the containment atmosphere, results in a recirculation pressure peak which is observed to be the maximum peak for all pump suction breaks. The energy release through the break drops at 3600 seconds, and only the decay heat is released thereafter. The containment pressure and temperature continue to drop monotonically from then on.

The calculated peak containment pressure and peak containment temperature, along with the energy released to the containment up to the end of blowdown, are summarized in Table 6.2-8 for each of the six cases analyzed. The maximum peak containment pressure is seen to be 49.6 psig (see also discussion presented in Subsection 6.2.1.1a.1). It occurs in Case 1.1.2, namely, a full double-ended guillotine rupture at the pump suction with the single active failure of one spray train, which corresponds to maximum safety injection and minimum spray cooling. Case 1.1.2 is therefore taken as the containment DB LOCA. The transient responses have been calculated for up to 10^5 seconds after the accidents, except for Case 3.1.1, (Hot-Leg Break). As can be seen from the plots presented, the periods covered include the most important aspects of the transient and show the general trend of responses.

The accident chronology is given in Table 6.2-9, one for each break location. It includes the time when the ESF system begins operation and time of occurrence of other important events. The distribution of energy inventories prior to the accident, at the end of the blowdown phase and at the end of the core reflood phase, and also the steam generator energy releases during post-reflood phase, are provided in Subsection 6.2.1.3.

The long-term recirculation operation causes reduction of the containment pressure to within a few psi above the atmospheric pressure in one day, which is well below one-half of the calculated peak pressure in all cases analyzed.

For the DB LOCA, the effective heat transfer coefficient based on the temperature difference between the containment atmosphere and the heat sink surface is plotted as a function of time in Figure 6.2-19.

(2) Secondary System Pipe Ruptures

Containment temperature and pressure responses have been evaluated following a spectrum of breaks in the main steam line occurring at various plant operating power levels, each with a single failure in the safety systems postulated to concur with the accident. The details of calculating the mass and energy releases into the containment after a main steam line break (MSLB) are given in Subsection 6.2.1.4.

The initial plant operating power levels and the spectrum of break types and sizes analyzed are summarized in Table 6.2-10 and Table 6.2-11, respectively.

The blowdown data for double-ended breaks has been developed for breaks located downstream of the steam generator flow restrictor and upstream of the main steam isolation valve (MSIV). The postulation of a break downstream of the flow restrictor is conservative for large double-ended breaks since a break upstream of the flow restrictor would result in a smaller energy release because of severe water entrainment in the forward flow and the flow-limiting effect of the restrictor on the reverse flow. For small double-ended breaks, there is no difference between the two locations. Since a break immediately downstream of the flow restrictor allows the steam in this section to blow down completely shortly after isolation, blowdown based on this break location is considered conservative.

For the split rupture, the blowdown data developed are valid for any break location in the steam piping and the header.

As presented in Table 6.2-10 and Table 6.2-11, a total of 17 operating power break type/size combinations for a MSLB have been considered. The concurrence of a single failure in the safety systems can result in more severe conditions in the containment. In this analysis, a loss of offsite power has been assumed to concur with the MSLB whenever it results in more severe containment conditions, in addition to a single failure in the safety systems.

However, for the conservative estimation of mass and energy releases, offsite power is assumed to be available.

The safety-related equipment relied upon to mitigate the consequences of an MSLB include those required for isolation of the main feedwater lines and the main steam lines, delivery of emergency feedwater into the steam generators, and delivery of spray water into the containment. The Containment Spray System is the only active heat removal system for which credit has been taken in mitigating the consequences of a MSLB. The signal to isolate the main feedwater lines and the main steam lines and to start the Emergency Feedwater System is generated by the Reactor Protection System for all double-ended ruptures listed in Table 6.2-11. For the split ruptures, this isolation signal is conservatively assumed to be generated when the containment pressure reaches the Hi-1 setpoint of 6.8 psig for isolation of feedwater lines and Hi-2 setpoint of 7.4 psig for isolation of main steam lines, although the nominal values for the Hi-1 and Hi-2 setpoints are 4.3 psig. After the isolation setpoint is reached, a delay of 1.0 second for instrument response and signal processing has been allowed. The containment spray actuation signal ("P" signal) is generated when the containment pressure reaches the Hi-3 setpoint, as discussed in Subsection 6.2.1.1c.1(b).

The isolation of the main feedwater lines is achieved by tripping the main feedwater pumps and closing the feedwater control valves (FCVs) and the feedwater isolation valves (FWIVs). The pump trip is immediate. The FCVs are capable of closing completely within 10 seconds (5-second delay and 5-second stroke) after receipt of the isolation signal, and the FWIVs within 10 seconds after receipt of the isolation signal. Failure to trip a main feedwater pump would result in more main feedwater being pumped into the steam generators and an increase in the blowdown. Failure of the broken-loop FWIV would allow additional feedwater, namely that in the piping between the FWIV and FCV to flash into the steam generator. Loss of offsite power, however, would result in less feedwater being fed to the steam generators because of the coastdown of the various pumps in the Condensate and Feedwater System.

The isolation of the main steam lines is achieved by closing the MSIVs, the turbine control valves (TCVs) and the turbine stop valves (TSVs). The MSIVs are capable of closing completely within 5 seconds after receipt of the isolation signal and the TCVs and the TSVs within 0.2 second. Failure of the broken-loop MSIV to close would allow additional steam, namely that remaining in the steam piping bounded by the MSIVs, the TSVs, the condenser, and the moisture separators/reheaters, to blow down after isolation. Failure of a TCV or a TSV, however, would not cause more blowdown since closure of either one would effectively isolate that line. Loss of offsite power has no effect on the isolation of the main steam lines.

The emergency feedwater that enters the affected steam generator would eventually blow down. For conservatism, it is assumed that the emergency feedwater is pumped into the affected steam generator immediately after the MSLB. The flow isolation valves in each line are pre-set in the open position to provide a flow of at least 235 gpm to the intact steam generator, and are closed by a high flow signal in the event of a broken loop. In case of failure of a flow isolation valve in the open position, the flow is limited by the flow restricting venturi to at most 750 gpm. In either case, the flow of emergency feedwater to the affected steam generator is assumed to be terminated manually 30 minutes after the MSLB.

For the Containment Spray System, failure of one of the two trains would reduce its heat removal capacity by half and result in more severe containment conditions. The effect of loss of the offsite power on the actuation time of the spray system has been taken into account, as previously discussed by assuming a later spray time.

The single failures which have been considered, as discussed above, are listed in Table 6.2-12. The first 5 failures would increase mass-energy release to the containment, while the last failure would result in a reduction in the heat-removing capacity of the Containment Spray System.

Table 6.2-10, Table 6.2-11 and Table 6.2-12 show that there is a total of 102 combinations of plant operating powers, break type/sizes, and single failures. Mass and energy releases to the containment for these 102 cases have been calculated. The effect of single failures No. 3 and No. 4 of Table 6.2-12 on the blowdown for any type of ruptures, and single failure No. 2 for split ruptures, is to increase the steam generator dryout time only. This is essentially true for single failure No. 5 for all ruptures since a somewhat larger blowdown rate after the dryout time has been observed not to be sufficient to cause a second peak in the containment temperature or the containment pressure. Thus, the above single failure combinations can be eliminated. However, to clearly illustrate the effect of the single failures on the containment pressure-temperature transient, all 6 single failures have been analyzed for the 102 percent power level and for each of the three break types, No. 1, No. 2 and No. 3. This amounts to a total of 56 cases. After carefully examining the containment pressure-temperature responses for these cases, the following general inferences are drawn:

The containment pressure and temperature transients show that the peak containment pressure occurs either at the time the spray water enters the containment or at the steam generator dryout time. The latter occurs when the blowdown is severe enough to cause increases in the containment pressure even with the containment spray system operating, which is true for all double-ended ruptures. The peak containment temperature always occurs at the spray time.

When considering accidents at the same power level and with the same single failure, the peak containment pressure increases with the size of the break in the case of double-ended ruptures. The peaks for the split ruptures are always higher than those for the small double-ended ruptures, but lower than those for the full double-ended ruptures. The effect of power level, with the inherent break size change in the case of small double-ended breaks, is not always monotonic. The peak containment pressure varies monotonically with power level only in small double-ended breaks with a concurrent single failure of a MSIV or a containment spray train, and in full double-ended breaks with a concurrent single failure of a containment spray train. in the former, the peak pressure increases monotonically as the power level increases, while in the latter, it decreases monotonically.

The computed peak containment pressure and peak containment temperature, and the times of their occurrence are summarized in Table 6.2-13, Table 6.2-14 and Table 6.2-15 for full double-ended ruptures, small double-ended ruptures, and split ruptures, respectively. Only those cases which effectively envelope containment pressure and temperature responses following MSLBs are shown. Also included in the tables is the total energy released to the containment in each case. The transients for Cases 5.1.6 and 1.1.3 are shown in Figure 6.2-20, Figure 6.2-21, Figure 6.2-22 and Figure 6.2-23.

The maximum peak containment pressure, as can be seen in these tables, is 36.1 psig and occurs in Case 5.1.6, namely a full double-ended guillotine rupture at hot shutdown with a concurrent failure of one containment spray train. This pressure is significantly lower than that for the DB LOCA. For this maximum peak containment pressure case, the mass-energy release is given in Subsection 6.2.1.4. The effective heat transfer coefficient, based on the temperature difference between the containment atmosphere and the heat sink surface, is plotted in Figure 6.2-24 as a function of time for four representative heat sinks, along with the Uchida correlation (Reference 5) which was used to calculate the condensing heat transfer coefficient. The heat sinks are the containment liner and typical thin-steel, thick-steel and concrete structures. All four effective heat transfer coefficients are seen to be much lower than those predicted by the Uchida correlation (taking into account the 0.4 factor for concrete surfaces) when the containment atmosphere is superheated. When the containment atmosphere becomes saturated, the effective heat transfer coefficients become identical to that of the Uchida correlation, as long as the heat sink surface temperature is lower than the containment atmosphere (which is approximately equal to the saturation temperature corresponding to the partial pressure of steam in the containment). If the heat sink surface temperature exceeds the containment atmosphere temperature, condensation ceases and convective heat transfer from the heat sink to the containment atmosphere takes place.

The maximum peak containment temperature is 364°F and occurs in Case 1.1.2, namely a full double-ended guillotine rupture at 102 percent power with a concurrent failure in the broken-loop MSIV. The mass-energy release for this case is presented in Subsection 6.2.1.4.

A plot of the heat transfer coefficients similar to that presented for the maximum peak containment pressure is shown in Figure 6.2-25 for Case 1.1.2. Again, a similar behavior is observed. The effective heat transfer coefficients for the containment liner, the thin steel structure, the thick steel structure, and the concrete structure are considerably lower than those predicted by the Uchida correlation when the containment atmosphere is superheated. They become identical only when the containment atmosphere becomes saturated and the heat sink surface temperature is lower than the containment atmosphere saturation temperature.

(e) Post-Accident Containment Temperature/Pressure Monitoring

Instrumentation provided in the containment can be used to monitor and record containment pressure in the event of an accident. This system is discussed in detail in Section 7.5. Containment sump temperature is not monitored since it is not consequential to verification of proper operation of the Post-Accident Heat Removal Systems.

6.2.1.2 Containment Subcompartments

a. Design Bases

The major subcompartments within the containment are the reactor cavity, the steam generator compartments, the pressurizer compartment and the pressurizer skirt cavity. These subcompartments are designed to withstand the differential pressures and jet impingement forces resulting from a postulated pipe break. Reactor cavity and steam generator compartment overpressurization has been deleted from the design basis in accordance with the final NRC Rulemaking with respect to GDC-4, dated October 27, 1987. Sufficient openings for venting these subcompartments are provided to keep differential pressures on the subcompartment walls and forces imposed on equipment supports within their structural limits. In addition, restraints and supports on the various equipment contained within these subcompartments are designed so that pipe whip and forces transmitted through component supports do not threaten the structural integrity of these subcompartments or the containment structure.

The pipe breaks considered in all subcompartments are full double-ended ruptures.

Of all the postulated break locations, the ones considered severe for the subcompartment pressurization analyses are listed in Table 6.2-16. The loads on the subcompartment walls and on the equipment supports for a given subcompartment are determined for various break locations.

The subcompartment walls and the equipment supports are designed so that the maximum calculated load does not exceed the design load.

b. Design Features

1. Reactor Cavity

The reactor cavity is a cylindrical annulus around the reactor vessel. The surrounding structure, termed the primary shield wall, is a heavily reinforced concrete structure which provides support for the reactor vessel and its associated coolant system piping. The lower portion of the cavity, where the core lies, has an outside diameter of 17.08 ft. The inner diameter of the annulus, formed by the outer diameter of the reactor vessel, is 16.76 ft. The upper region of the cavity where the hot and cold leg nozzles emanate from the vessel has a diameter of 25.5 ft. Contained within this upper region is the ring girder which gives added rigidity to the vessel support. The reactor coolant loop pipe whip restraints were eliminated from the design bases by ECA 25/113665, Rev. A, in accordance with the final NRC Rulemaking with respect to GDC-4, dated October 27, 1987.

A neutron shield consisting of borated concrete, and which is integral to the permanent reactor cavity seal ring, is installed around the reactor vessel refueling flange to reduce neutron streaming and dose rates on the containment operating floor during power operation. The permanent seal ring is equipped with removable hatch covers and neutron shield plugs to allow for the required ventilation air flow rate and access to the reactor cavity annular space, respectively.

Reactor cavity overpressurization has been deleted from the design basis in accordance with the final NRC Rulemaking with respect to GDC-4, dated October 27, 1987.

A vertical section through the reactor cavity is shown in Figure 6.2-26. Figure 6.2-27 shows the plan view at the elevation of the nozzles. The total free volume of the reactor cavity plus the reactor pool region above is 48,978 cu. ft. The total vent area from the reactor cavity to the containment is 1,757 sq. ft, primarily through the reactor pool region. The free volume and the vent area are conservatively calculated with the insulation in place.

2. Steam Generator Compartment

A steam generator compartment is a reinforced concrete structure which encloses the steam generator, a reactor coolant pump and its associated primary and secondary coolant system piping. Figure 6.2-29 and Figure 6.2-30 show sections of the steam generator compartment. Horizontal sections at various elevations are provided in Figure 6.2-31, Figure 6.2-32, Figure 6.2-33 and Figure 6.2-34. Large vent paths from the steam generator compartment to the containment are available via the reactor coolant pump area, and the adjacent steam generator compartment. The top of the steam generator compartment is also open to the containment. The steam generator stands on four supports anchored to the floor at El. (-)26'-0." The total free volume of a steam generator compartment is 23,040 cu. ft. There are no blowout panels or other pressure-dependent areas considered in the analysis. Free volumes and vent areas have been calculated with the insulation in place. Steam generator compartment overpressurization has been deleted from the design basis in accordance with the final NRC Rulemaking with respect to GDC-4, dated October 27, 1987.

3. Pressurizer Compartment

The pressurizer compartment is a reinforced concrete structure extending from El. 0'-0" to El. 63'-0" which encloses the pressurizer and its associated piping. The pressurizer skirt, which is a cylindrical support extending from the bottom of the pressurizer, anchors the pressurizer to the compartment floor. A ring support at El. 23'-6¾" provides lateral support for the pressurizer. Section and plan drawings of the pressurizer compartment are shown in Figure 6.2-35, Figure 6.2-36, Figure 6.2-37, Figure 6.2-38, Figure 6.2-39 and Figure 6.2-40. Free volumes and vent areas have been calculated assuming the insulation remains intact during the transient. The HVAC ducting and sheet metal panels at elevation 16'-6" are designed to blow out in the event of a pressure buildup of 0.25 psig in the compartment to provide additional vent area. The total free volume of the compartment used in the analysis is 6638 cu. ft and the total vent area to the containment is 400 sq. ft.

4. Pressurizer Skirt Cavity

The pressurizer skirt cavity is formed by the bottom of the pressurizer and its supporting skirt. A 14" surge line which connects the reactor coolant system with the pressurizer passes through a 5½ ft diameter opening in the pressurizer compartment floor. Figure 6.2-41 and Figure 6.2-42 show the plan and elevation drawings of the pressurizer skirt cavity. The volume below the pressurizer skirt has a large vent opening to the containment. The total free volume of the skirt cavity is 1860 cu. ft and the total vent area to the containment is 238 sq. ft. The insulation on the pressurizer and on the surge line is assumed intact in calculating the free volume and vent openings.

c. Design Evaluation

1. Mass and Energy Release Data

The mass and energy release data for all the breaks considered for the subcompartment analyses has been generated by Westinghouse. Discussions of the blowdown model are provided in Reference 8.

2. Computer Code for Subcompartment Pressurization Analysis

The subcompartment pressure transients were calculated using COMPRESS - a digital computer program. A detailed description of the analytical method can be found in Appendix 15C and Reference 9. Some important aspects of the method are outlined below:

(a) Mathematical Model

The COMPRESS computer program calculates the pressure and temperature transient responses in a set of inter-connected volumes following a high energy line rupture. The subcompartment initial conditions, free volumes, vent path areas, inertias and loss coefficients, as well as the mass and energy release from the break, are input to the code. For vent paths covered by blowout panels or hinged doors, the flow area, inertia and loss coefficient are to be supplied as functions of the position of the vent cover. Information is also required on the physical properties of the vent cover related to the dynamics.

Computations for the transient pressure and temperature are made by performing mass and energy balances in each volume during small time steps. The flow between inter-connected volumes is calculated assuming quasi-steady thermodynamic conditions in the volumes. It is assumed that the air-steam-water mixture in each volume is homogeneous and in thermodynamic equilibrium during the time step. The entrainment of the water from condensed vapor is conservatively assumed to be 100 percent. The flow is based on the vent path characteristics at the previous time step. The position of a blowout panel or hinged door, which determines the vent path characteristics, is obtained by solving the equation of motion at each time step, utilizing the calculated nodal pressures.

(b) Vent Flow Calculations

The COMPRESS code offers various options for calculating the flow between connected volumes. The flow can be sonic or subsonic depending upon the ratio of upstream and downstream pressures. Subsonic flow is calculated using the incompressible flow equation or the ideal nozzle equation. Sonic flow is calculated using the ideal nozzle equation or Moody's two-phase flow correlation. However, in the present analyses, the vent flow calculations were made using the incompressible flow equation and the ideal nozzle equation, and the smaller of the two calculated flows is used as the actual vent flow for the conservative prediction of the subcompartment pressure. The complete entrainment of water with the air-steam mixture, the homogeneity of air-steam-water mixture and the thermodynamic equilibrium between the gas and liquid phases are assumed throughout the vent path.

(c) Vent Path Loss Coefficient

The resistance to flow in a vent path is, in general, due to change in the flow area (expansion and contraction), friction, change in the flow direction and flow obstructions. Various components of the loss coefficient are calculated following the procedures outlined in Reference 10. All contractions and expansions are considered to be sudden changes in the flow area. All the components are appropriately lumped together and used for a given vent path.

3. Pressure Transient Analyses

(a) Subcompartment Modeling

Each subcompartment is subdivided into a number of nodes to determine a realistic subcompartment pressure response. Boundaries between the nodes are placed at locations where large pressure gradients can occur either due to physical flow restrictors or due to the fluid inertia. A particular nodalization is developed for a given subcompartment through a nodalization sensitivity study, to determine the minimum number of nodes required to adequately predict the pressure profiles in the subcompartment. This involves several nodalizations for each subcompartment with various numbers of nodes, especially in regions where large pressure gradients exist. By comparing the results of various nodalization schemes for a given subcompartment, that scheme which results in a converged solution so that the subcompartment pressure profiles are not appreciably changed by further nodal refinement is selected for the pressurization analysis.

(b) Nodalization Sensitivity Study

The transient pressure responses in various subcompartments are analyzed by subdividing into a set of inter-connecting nodes. The pressure at any particular location is usually sensitive to the nodalization scheme employed. As the nodalization is made more refined, the calculated subcompartment pressure responses are closer to the real situation. To ensure that a particular nodalization scheme is adequate to predict the subcompartment pressure responses, a nodalization sensitivity study is usually performed by employing various nodalization schemes. When it is established that by further refinement of the nodalization the subcompartment pressure responses or the equipment loads within the subcompartment do not change significantly, the nodalization scheme is considered adequate.

(1) Reactor Cavity

Reactor cavity overpressurization has been deleted from the design basis in accordance with the final NRC Rulemaking with respect to GDC-4, dated October 27, 1987.

(2) Steam Generator Compartment

Steam generator compartment overpressurization has been deleted from the design basis in accordance with the final NRC Rulemaking with respect to GDC-4, dated October 27, 1987.

(3) Pressurizer Compartment

The pressurizer compartment pressure transient analysis for the nodalization sensitivity study has been performed following a double-ended rupture in the spray line at El. 34'-0". The three nodal schemes employed for the sensitivity study are given in Figure 6.2-54. In the 15-node scheme, the break region is divided into 2 nodes and the whole compartment is divided into 6 vertical regions. The 23-node scheme has 6 nodes in the break region and has 6 vertical regions. In the 29-node scheme, the whole compartment is divided into 7 vertical regions.

The horizontal pressure profiles around the pressurizer at the break elevation are shown in Figure 6.2-55, while Figure 6.2-56 gives the vertical profile. For the three nodalizations, Figure 6.2-57 depicts the sidewise forces and Figure 6.2-58 depicts the moments that act on the pressurizer. The horizontal and vertical pressure profiles, as well as the forces and moments on the pressurizer, demonstrate that the 23-node scheme adequately predicts the compartment pressure response.

(4) Pressurizer Skirt Cavity

The pressure transients in the pressurizer skirt cavity are calculated following a double-ended guillotine rupture in the surge line. The two nodal schemes used for the sensitivity study are shown in Figure 6.2-59 and Figure 6.2-60. In the 5-node scheme the volume underneath the pressurizer down to El. (-)5'-11" is taken as the break node. In the 6-node scheme the break node consists of the volume down to El. 0'-0" only. The peak pressure in the two cases is within 3 percent, even though the break node volume in the 6-node model is about half of that in the 5-node model. Further refinement of the nodalization below El. (-)5'-11" will not affect the pressure in the skirt cavity because a choked flow condition is established from the break node to the adjacent volume in the skirt cavity. The 6-node scheme, therefore, is adequate for the skirt cavity pressurization analysis.

(c) Results

(1) Reactor Cavity

Reactor cavity overpressurization has been deleted from the design basis in accordance with the final NRC Rulemaking with respect to GDC-4, dated October 27, 1987.

(2) Steam Generator Compartment

Steam generator compartment overpressurization has been deleted from the design basis in accordance with the final NRC Rulemaking with respect to GDC-4, dated October 27, 1987.

(3) Pressurizer Compartment

All postulated double-ended guillotine ruptures in the spray line at various discontinuities have been considered to determine the design loads on the pressurizer supports and also on the compartment structures.

The sensitivity study performed for a full double-ended rupture of the spray line (5.187 in I.D.) demonstrated the adequacy of the 23-nodal model. Using the model, the compartment pressurization was analyzed for a break at El. 34'-0". Figure 6.2-69 graphically shows the transient pressures in each node in the pressurizer compartment. The differential pressures for various compartment walls are given in Figure 6.2-70. The mass and energy releases following a double-ended guillotine rupture used in the analysis are provided in Table 6.2-30. Table 6.2-31 presents the vent path characteristics used in the analysis. The nodal data, the pressurizer compartment initial conditions, calculated peak differential pressures, design peak differential pressures and the design margins for a spray line rupture are presented in Table 6.2-32. The calculated peak differential pressures listed in Table 6.2-32 are the maximum differential pressures considering all the postulated break locations listed in Table 6.2-16.

(4) Pressurizer Skirt Cavity

A double-ended guillotine rupture of the surge line (11.188 in. I.D.) at the pressurizer nozzle was analyzed using a 6-node model. The pressure response of the nodes is shown in Figure 6.2-71. The mass and energy release data used for this break is given in Table 6.2-33. The vent path characteristics are presented in Table 6.2-34.

Table 6.2-35 presents the nodal data, skirt cavity initial conditions, calculated peak differential pressures, design peak differential pressures and the design margins for this rupture.

6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

This analysis presents the mass and energy releases to the containment subsequent to a hypothetical loss-of-coolant accident (LOCA). The release rates are calculated for pipe failure at three distinct locations:

- Hot leg (between vessel and steam generator)
- Pump suction (between steam generator and pump)
- Cold leg (between pump and vessel).

During the reflood phase, these breaks have the following different characteristics. For a break in the pump suction or cold leg piping, a portion of the accumulator or safety injection flow in the intact loop can bypass the downcomer and flow directly to the break. For a cold leg pipe break, all of the fluid which leaves the core must vent through a steam generator and be vaporized by heat addition to the primary from the secondary. However, relative to breaks at the other locations, the core flooding rate (and therefore the rate of fluid leaving the core) is low, because all the core vent paths include the resistance of the reactor coolant pump. For a hot leg pipe break, the vent path resistance is relatively low which results in a high core flooding rate but the majority of the fluid which exits the core bypasses the steam generators in venting to the containment. The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and steam generator heat addition as in the cold leg break. As a result, the pump suction breaks yield the highest energy flow rates during the post-blowdown period.

The spectrum of breaks analyzed includes the largest cold and hot leg breaks, reactor inlet and outlet respectively, and a range of pump suction breaks from the largest to a 3.0 ft^2 break. Table 6.2-36 presents the specific cases analyzed and a list of tables which contain the results for each case. Because of the phenomena of reflood as discussed above, the pump suction break location is the worst case. For this reason a spectrum of break sizes has been used in this analysis for the pump suction location. Other break locations result in less severe containment pressure transients than the pump suction location. Smaller break sizes at these locations result in less severe transients than full double-ended guillotine breaks. Therefore the hot leg and cold leg locations have only been analyzed with an assumed double-ended guillotine break.

The LOCA transient is typically divided into four phases:

- Blowdown - includes the period from accident occurrence (when the reactor is at steady-state operation) to the time when the total break flow stops.
- Refill - the period of time when the lower plenum is being filled by accumulator and safety injection water. (This phase is conservatively neglected in computing mass and energy releases for containment evaluations.)
- Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.
- Post-Reflood - describes the period following the reflood transient. For the pump suction and cold leg breaks a two-phase mixture exits the core, passes through the hot legs and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

a. Mass and Energy Release Data

1. Blowdown Mass and Energy Release Data

Table 6.2-37, Table 6.2-38, Table 6.2-39, Table 6.2-40, Table 6.2-41 and Table 6.2-42 present the calculated mass and energy releases for the blowdown phase of the various breaks analyzed with the corresponding break size.

2. Reflood Mass and Energy Release Data

Table 6.2-43, Table 6.2-44, Table 6.2-45, Table 6.2-46 and Table 6.2-47 present the calculated mass and energy releases for the reflood phase of the various breaks analyzed, along with the corresponding safeguards assumption (maximum or minimum). The reflood results have been omitted for the hot leg break since the blowdown releases are sufficient to determine the peak containment pressure for this break location.

3. Dry Steam Post-Reflood Mass and Energy Release Data

The calculated mass and energy releases for the post-reflood phase with dry steam are provided in the reflood mass and energy release tables (Table 6.2-45, Table 6.2-46, Table 6.2-47) after end of 10-foot entrainment occurs.

4. Two Phase Post-Reflood Mass and Energy Release Data

Table 6.2-48 and Table 6.2-49 present the two phase (froth) mass and energy release data for a double-ended pump suction break using minimum and maximum safeguards assumptions, respectively. Table 6.2-50 presents the results for a 0.6 ft^2 double-ended pump suction break using minimum safeguards.

The double-ended pump suction minimum safeguards case is normally limiting. The two phase results are provided for other cases to prove that an upper bound calculation has been performed. This information is not provided for the three-foot squared pump suction split or the double-ended cold leg or hot leg cases. The peak containment pressures for these cases will occur during the blowdown phase of the transient.

5. Equilibration and Depressurization Energy Release Data

The equilibration and depressurization energy release has been incorporated in the post-reflood mass and energy data. This eliminates the need to determine additional releases due to the cooling of steam generator secondaries and primary metal.

b. Mass and Energy Sources

The sources of mass considered in the LOCA mass and energy release analysis are given in the mass balance tables (Table 6.2-51, Table 6.2-52, Table 6.2-53, Table 6.2-54, Table 6.2-55 and Table 6.2-56). These sources are:

1. Reactor Coolant System
2. Accumulators
3. Pumped injection.

Likewise the sources of energy considered in the LOCA mass and energy release analysis are given in the energy balance tables (Table 6.2-57, Table 6.2-58, Table 6.2-59, Table 6.2-60, Table 6.2-61 and Table 6.2-62). These sources include:

1. Reactor Coolant System
2. Accumulator
3. Pumped injection
4. Decay heat
5. Core stored energy
6. Primary metal energy
7. Secondary metal energy
8. Steam generator secondary energy
9. Secondary transfer of energy (feedwater into the steam out of the steam generator secondary).

The balances are presented at the following times:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time (The only difference that will be noted from the values at the end of blowdown is that some accumulator water will be transferred to the reactor coolant. Thus, the low plenum will be full at the beginning of the reflood transient.)
4. End of reflood time
5. The time when the broken loop steam generator reaches thermal equilibrium (for froth cases only)

6. The time when the intact loop steam generator reaches thermal equilibrium (for froth cases only)
7. Time of full depressurization (for froth cases only).

The methods and assumptions used to release the various energy sources are given in Reference 11.

The following items ensure that the core energy release is conservatively analyzed for maximum containment pressure.

1. Maximum expected operating temperature
2. Allowance in temperature of instrument error and dead band (+4°F)
3. Margin in volume (1.4 percent)
4. Allowance in volume for thermal expansion (1.6 percent)
5. Margin in core power associated with use of engineered safeguards design rating (ESDR)
6. Allowance for calorimetric error (2 percent of ESDR)
7. Conservatively modified coefficients of heat transfer
8. Allowance in core-stored energy for effect of fuel densification
9. Margin in core stored energy (+15 percent).

c. Blowdown Model Description

The model used for the blowdown transient (SATAN-VI) is the same as that used for the ECCS calculation. This model is described in Reference 12 and 13. Reference 11 provides the method by which this model is used.

d. Refill Model Description

At the end of blowdown, a large amount of water remains in the cold legs, downcomer and lower plenum. To conservatively model the refill period for the purpose of containment mass and energy releases, this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. Thus, the time required for refill is conservatively neglected.

e. Reflood Model Description

The model used for the reflood transient (WREFLOOD) is a slightly modified version of that used in the ECCS calculation. This model is described in Reference 12 and 14. Reference 11 describes the method by which this model is used, and the modifications. Transients of the principal parameters during reflood are given in Table 6.2-63 and Table 6.2-64 for the double-ended pump suction break with minimum and maximum safeguards.

f. Post-Reflood Model Description (FROTH)

The transient model (FROTH) along with its method of use is described in Reference 8.

g. Single Failure Analysis

The effect of single failures of various ECCS components on the mass and energy releases is included in these data. Two analyses bound this effect.

No single failure is assumed in determining the mass and energy releases for the maximum safeguards case. For this case a failure must be assumed in the Containment Cooling Systems. Normally the limiting case is the loss of one spray pump. For the minimum safeguards case, the single failure assumed is the loss of one emergency diesel. This failure results in the loss of one pumped safety injection train. The analysis of both maximum and minimum safeguards cases assures that the effect of all credible single failure is bounded.

h. Metal-Water Reaction

In the mass and energy release data presented here, no Zr-H_20 reaction heat was considered because the clad temperature did not rise high enough for the rate of the Zr-H_20 reaction to be of any significance.

i. Energy Inventories

Energy inventories for primary and secondary systems are tabulated for hot leg, cold leg, and pump suction breaks in Table 6.2-57, Table 6.2-58, Table 6.2-59, Table 6.2-60, Table 6.2-61 and Table 6.2-62.

j. Additional Information Required for Confirmatory Analysis

System parameters needed to perform confirmatory analysis are provided in Table 6.2-65.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures inside Containment

a. Mass and Energy Release Data

The mass and energy releases into the containment following a postulated main steam line break (MSLB) have been calculated by using the model described in Subsection 6.2.1.4d and incorporating the balance-of-plant parameters for Seabrook Station via the procedure described in Reference 15.

The effects of a postulated feedwater line rupture are not as severe as the main steam line break because the break effluent of a feedwater line rupture is at a lower specific enthalpy. Therefore, feedwater line break mass and energy releases to the containment are not addressed here since they are bounded by steam line break releases.

1. Break Type/Size and Operating Power

The plant operating power levels at the time of the MSLB and the spectrum of break types and sizes analyzed have been presented in Table 6.2-10 and Table 6.2-11, respectively. Full double-ended rupture (DER) area is determined by the integral flow restrictor area. This break represents the largest possible break. A small double-ended rupture has been considered for each power level. These break sizes have been chosen to be large enough to generate a steam line isolation signal from the Primary Protection System. For any ruptures smaller than these small double-ended ruptures, an isolation signal is generated by containment pressure. Two such cases have been analyzed with approximately half the corresponding size of the small double-ended rupture. These breaks are expected to cover adequately the full spectrum of double-ended break sizes. For the split ruptures, the break sizes selected are the largest sizes which will not generate a steam line isolation signal from the Primary Protection System. An isolation signal is generated on containment pressure. Larger split ruptures will generate primary protection signals and are expected to be bounded by the double-ended ruptures. The breaks are assumed to be at the exit of a steam generator flow restrictor for double-ended ruptures, and at any point on the piping between a steam generator and the first main steam pipe whip restraint inside the containment for split ruptures.

For full-size double-ended guillotine ruptures, the model allows credit for the flow-limiting effect of the MSIV (one for each loop) on the reverse flow blowdown. The reverse flow consists of (1) an initial blowdown of the steam in the piping between the break (immediately downstream of a steam generator flow restrictor) and the nearest MSIV, which is controlled by the piping cross-sectional area (4.18 ft^2); (2) a subsequent flow from the intact steam generators, which is controlled by the MSIV seat area (1.97 ft^2); and (3) a post-isolation piping steam blowdown. The initial piping blowdown is assumed at a constant choked flow rate corresponding to the initial pressure in the line. The flow from the intact steam generators, which is controlled by MSIV seat area, is conservatively calculated assuming a pressure decay curve for a break area smaller than the MSIV seat area for a given power level. Following isolation, the blowdown of the steam remaining in the piping (from the break up to the MSIVs of the intact steam generators for a single failure of a MSIV and up to the nearest MSIV for all other single failures) is calculated assuming the flow rate drops to zero linearly.

The post-isolation blowdown was calculated by following the procedure given in Reference 15 for the small double-ended breaks.

The reverse flow consists of (1) a reverse flow from the intact steam generators, and (2) a post-isolation piping steam blowdown, both controlled by the break size.

For split ruptures, isolation of the main feedwater and main steam lines is initiated when the containment pressure reaches 6.8 and 7.4 psig, which are the upper bounds of the isolation setpoints for the main feedwater and main steam. The time when this pressure is reached was found by computing the containment pressure-temperature response using blowdown data obtained by assuming no isolation. The analytical method and initial conditions for this calculation have been presented in Subsection 6.2.1.1. The correct blowdown in the period following isolation was then calculated by applying the calculated isolation time.

2. Fluid Inventory for Release

The total inventory of fluid available for release is characterized by the steam generator dryout time, which is defined as the time when the blowdown rate of the affected steam generator is equal to the rate at which the feedwater is entering. The dryout time depends on four steam/water sources: (1) initial steam generator mass, (2) mass added by feedwater flashing, (3) mass added by the Main Feedwater System, and (4) mass added by the Emergency Feedwater System. Item (1) is discussed in Subsection 6.2.1.4c. For Item (2), the actual water volume in the main feedwater piping between the affected steam generator and the valve effecting the isolation was used. Item (3) is obtained by integrating the transient flow rate over an appropriate time period. The flow rate was first obtained as a function of the pressure in the affected steam generator, considering the system resistance and characteristics of the various pumps. The pressure in the intact steam generators was assumed to remain unchanged during the transient. The main feedwater pumps were assumed to be operating at the maximum speed until they were tripped and all feedwater control valves (FCV) were assumed to be at their initial position, except the one in the broken loop which was assumed fully open, until their isolation is completed. It was assumed that the main feedwater pumps are immediately tripped upon receipt of the isolation signal, while all FCVs close instantaneously after a further delay equal to the maximum valve stroking time. The maximum stroking time of the feedwater isolation valves (FWIVs) and FCVs is discussed in Subsection 6.2.1.1. The pressure-flow relationship thus calculated was then converted to give the main feedwater flow rate as a function of time using the calculated pressure transient for the affected steam generator. Finally, Item (4) was obtained by multiplying the constant flow rate by the dryout time, or 30 minutes (time of isolation of the emergency feedwater line), as appropriate.

3. Mass-Energy Release Data for Most Severe Ruptures

The mass and energy releases have been calculated for the 102 accident-single failure combinations, as can be deduced from Table 6.2-10, Table 6.2-11 and Table 6.2-12. The mass flow rate and the enthalpy for the forward flow and the reverse flow for the case of maximum containment peak pressure (a full DE rupture at hot shutdown with one containment spray train failed) are given in Table 6.2-66 and Table 6.2-67, respectively. Similar blowdown information for the case of maximum containment peak temperature (a full DE rupture at 102 percent power with the broken loop MSIV failed) is given in Table 6.2-68 and Table 6.2-69.

b. Single Failure Analysis

The single failures postulated to concur with the MSLB have been presented in Table 6.2-12. Failure of a FWIV (Single Failue No. 1) increases the dryout time through Item (2) described in the previous subsection. The main feedwater available for flashing with and without this failure is 558.0 cu. ft and 160.5 cu. ft, respectively. Failure of a FCV (Single Failure No. 3) and failure of the main feedwater pump trip (Single Failure No. 4) increase the dryout time through Item (3) only, while failure of the emergency feedwater pump run out control (Single Failure No. 5) increases the dryout time through Item (4) only. Failure of the broken-loop MSIV (Single Failure No. 2) extends the steam volume available for the post-isolation piping blowdown to include the portion bounded by the MSIVs, the turbine stop valves (TSVs), the condensers, and the moisture separators/reheaters. The steam volume with and without this failure is 11907.1 cu. ft and 969.9 cu. ft, respectively. Failure of one of the two containment spray trains (Single Failure No. 6) results in a reduced containment heat removal rate, but has no effect on the blowdown.

c. Initial Conditions

A spectrum of power levels spanning the operating range, 102 percent, 75 percent, 50 percent, 25 percent, as well as the hot shutdown condition, has been considered. At each power level, plant initial conditions corresponding to the power level were assumed. Initial steam generator mass corresponding to the design mass limits was assumed.

Offsite power was assumed to be available. Specifically, this means no credit was taken for tripping of the reactor coolant pumps in determining the mass and energy releases. Tripping of the pumps reduces heat transfer capability from the primary plant into the steam generator, which reduces the effects of core power generation decay heat and thick metal energy and energy from intact steam generators on break releases. Further details of the initial conditions may be found in Reference 15.

d. Description of Blowdown Model

The steam generator blowdown model assumes dry steam. Details are available in Reference 15.

e. Energy Inventories

Mass and energy balances are provided in Table 6.2-69a and Table 6.2-69b for the most severe secondary pipe ruptures based on the highest peak calculated containment pressure and temperature.

f. Additional Information Required for Confirmatory Analysis

No additional information is deemed necessary for the performance of a confirmatory analysis.

6.2.1.5 Deleted

6.2.1.6 Testing and Inspection

Information concerning preoperational leakage testing and periodic in-service leakage surveillance of the containment to ensure functional capability of the containment and associated structures is provided in Subsection 6.2.6. Testing and inspection requirements for safety systems which support the functional capability of the containment and associated structures are discussed in the respective sections for the individual systems.

6.2.1.7 Instrumentation Requirements

Instrumentation is provided to monitor containment pressure, temperature humidity, hydrogen concentration, radiation levels and sump and flood water level to assist normal plant operations.

Instrumentation to monitor containment parameters for accident monitoring is discussed in detail in Section 7.5.

Containment post-LOCA radiation monitoring, area and airborne radioactivity monitoring instrumentation is discussed in detail in Subsection 12.3.4.

Containment post-LOCA hydrogen monitoring is discussed in Subsection 6.2.5.

6.2.1.8 Containment Analysis at an Analyze Core Power Level of 3659 MWt

As discussed in Section 6.2.1.1.a, the peak containment pressure and temperature predicted by the GOTHIC model for an analyzed core power level of 3659 MWt is bounded by the results of the original containment analysis performed using the CONTRAST-S-MOD1 program. This section provides a discussion of the results/data associated with containment response analyses of record, as shown on the applicable tables and figures in this section, as compared with the results/data associated with the containment response at the analyzed core power level of 3659 MWt.

a. Long-Term LOCA Containment Pressure and Temperature Response

The pressure and temperature time/histories shown in Tables 6.2-8, 6.2-9 and Figures 6.2-1 through 6.2-18 are based on the original and bounding containment analysis. A new GOTHIC code analysis was performed to baseline the original analysis and determine the impact of the analyzed core power level of 3659 MWt. The peak temperature and pressure values shown for the bounding EQ profile curves remain bounding at the 3659 MWt analyzed core power level. The actual calculated pressure and temperature values at the 3659 MWt analyzed power level vs. time differ slightly from the curves presented. The results show that the peak containment temperatures and pressures and the peak containment sump water temperature at an analyzed core power level of 3659 MWt are bounded by the peak values determined in the analysis of record.

b. Long-term LOCA Mass and Energy Release Analysis

The data presented in Tables 6.2-37 through 6.2-64 and Figures 6.2-87 through 6.2-90, applicable to the containment response analysis of record, remain bounding at the analyzed core power level of 3659 MWt for the following reasons: (1) The LOCA mass and energy release data used in the analysis of record are generic, whereas the mass and energy release data used in the analyzed 3659 MWt core power analysis were generated based on a Seabrook Station plant-specific model, (2) The decay heat model used in the generic data was based on American Nuclear Society (ANS) 1971 + 20%, whereas the 3659 MWt analyzed core power analysis was performed with ANS 1979 + 2σ, and (3) In the 3659 MWt analyzed core power analysis, credit was taken for steam-water interaction in the Reactor Coolant System loop piping, which was not available when the generic data was calculated.

c. MSLB Containment Pressure and Temperature Response

The pressure and temperature time/histories shown in Tables 6.2-13 through 6.2-15 and Figures 6.2-20 through 6.2-23 are based on the original and bounding MSLB containment analysis. For the 3678 MWt NSSS analyzed thermal power level, a new GOTHIC based analysis was performed to baseline original analysis and determine the impact of the analyzed core power level of 3659 MWt. The peak temperature and pressure values shown for the bounding EQ profile curves remain bounding at the 3659 MWt analyzed core power level. The actual calculated pressure and temperature values at the 3659 MWt analyzed core power level vs. time differ slightly from the curves presented.

The limiting peak temperature for MSLB under 3659 MWt analyzed core power conditions is 357.4°F for a double ended rupture at 100% power with a single train failure. This is bounded by the existing peak containment temperature during a MSLB of 364°F.

The existing containment peak pressure following a MSLB is 36.1 psig determined for a double-ended rupture of the main steam line at hot standby power with failure of one spray train. This peak containment pressure determined for MSLB conditions is well below the peak containment pressure determined under LOCA following a full double-ended rupture of the reactor coolant pump suction piping with two trains of safety injection in operation.

The highest containment peak pressure based on 3659 MWt analyzed core power level conditions following a MSLB is calculated to be 37.3 psig for a doubled ended rupture at near zero power with failure of one diesel generator (i.e., one train). Peak containment pressure of 30.8 psig is developed for MSLB under 3659 MWt analyzed core power conditions for the double-ended rupture at 100% power with failure of one train. The current analysis of record was performed using mass and energy release data for a generic Westinghouse plant, whereas the results under the 3659 MWt analyzed core power level conditions are specific to Seabrook Station. Although the peak containment pressure resulting from MSLB under 3659 MWt analyzed core power conditions is slightly higher than that developed in the analysis of record, this value remains bounded by the peak pressure determined in the loss of coolant accident analysis.

d. MSLB Mass and Energy Release Analysis

The containment MSLB response analysis of record was performed using mass and energy release data for a generic Westinghouse plant (Tables 6.2-66 through 6.2-69b), whereas the 3659 MWt core power analysis uses the RETRAN computer code for determining the MSLB mass and energy releases, as well as inputs that are plant-specific for Seabrook Station. As indicated above, results of the 3659 MWt core power analysis show that the containment MSLB response analysis of record remains bounding for peak containment temperature; the peak containment pressure determined in the 3659 MWt core power analysis is bounded by the peak pressure determined in the containment LOCA response analysis of record.

6.2.2 Containment Heat Removal System

The containment is maintained below design pressure following a primary or secondary system line rupture by the parallel action of the Emergency Core Cooling System (ECCS) and the Containment Building Spray (CBS) System as active heat removal systems and by the passive heat sinks such as structural components. The ECCS is discussed in Section 6.3; details of the CBS system and the inter-relationship between the ECCS and CBS systems for removing heat from the containment are discussed in this section. Passive heat sinks, such as the containment liner and other structures, are described in Subsection 6.2.1.

6.2.2.1 Design Bases

a. The ECCS and the CBS system are each comprised of two identical trains, each train independent of the other and fully redundant. Failure of a single active component will not cause the loss of more than half of either system's 200 percent heat removal capacity. Sufficient capacity to mitigate the consequences of an accident is thus assured with one CBS train and one ECCS train available.

b. The reactor unit has its own CBS system and ECCS.

c. The CBS system is designed to remove the energy discharged to the containment following a loss-of-coolant accident (LOCA) or main steam line break (MSLB) to prevent the containment pressure from exceeding design pressure and to reduce and maintain containment temperature and pressure within acceptable limits. The postulated accident conditions for which the CBS performance is evaluated are discussed in Subsection 6.2.1.1.

d. The sources and amounts of energy released to the containment during accident conditions that determine the required capacities of the containment heat removal systems are discussed in Subsections 6.2.1.3 and 6.2.1.4.

e. Only minimum containment spray capacity (one train) is required for design heat removal. The containment energy removal rate is sufficient to reduce containment pressure so that leakage is reduced to one-half of the design leakage of 0.15 percent of the containment air mass per day within 24 hours after the DBA.

f. Assuming a loss of offsite power, the most limiting single failure is the failure of an emergency diesel generator to function leading to the loss of one safety train at the time of actuation. A detailed discussion of single active failures is presented in Subsection 6.2.1.3.

g. Components in the CBS system that are required to function during and subsequent to an accident are designated seismic Category I, and are designed to withstand the SSE without loss of function.

h. The capability of mechanical, instrumentation and electrical components in the containment heat removal systems to withstand the post-accident containment environmental conditions is discussed in Section 3.11.

i. Design of the CBS system to withstand the effects of wind and tornado loading, floods, and missiles is discussed in Sections 3.3, 3.4, and 3.5, respectively.

j. Components in the CBS system that are required to function during and subsequent to an accident are protected against the dynamic effects of pipe rupture, as discussed in Section 3.6.

k. Design of the CBS system to withstand the effects of floods is discussed in Section 3.4.

The contents of the RWST and SAT are required for the safe shutdown of the reactor and cooling of the containment following the rupture of primary or secondary coolant system piping in the containment. The water in the RWST and SAT is not required for reactor cooldown if there is no pipe rupture. The RWST and SAT are not protected from tornado wind loads and missiles since the simultaneous occurrence of a pipe rupture in the containment and a tornado is considered incredible.

6.2.2.2 System Design

The P&I diagrams of the ECCS and CBS system are shown in Figure 6.3-1 and Figure 6.2-74, respectively. Figure 6.2-75 is a complete flow diagram of all the Engineered Safety Feature Systems. The valves in Figure 6.2-75 are shown in their open position so that the same diagram can be used to represent all modes of operation. Design parameters for the CBS system components are listed in Table 6.2-75.

a. Operation

The CBS system is actuated by a containment spray actuation signal (CSAS), which is initiated by high pressure in the containment. The CSAS is discussed in Section 7.3. The CBS system pumps water from the refueling water storage tank (RWST) to the spray nozzles located high in the Containment Building. The RWST contains a minimum of 450,000 gallons of borated water at a maximum temperature of 98°F, and provides cooling for a minimum of 26 minutes after an accident, based upon maximum pumps in operation at maximum flow rates. Upon a low-low level signal from the RWST (approximately 350,000 gallons removed) in conjunction with an "S" signal, the suctions of the residual heat removal (RHR) and CBS pumps automatically re-align to take suction from the containment recirculation sumps. The operator then manually re-aligns the centrifugal charging pumps to take suction from RHR pump P-8A discharge and the safety injection pumps to take suction from RHR pump P-8B discharge. All pumps continue to operate in the recirculation mode until no longer required. Heat tracing is not required for the piping in this system since no part of the system is exposed to temperatures below 40°F.

b. Component Description

The following are descriptions of the components in the CBS system. RHR pumps and heat exchangers are described in Section 6.3 and Subsection 5.4.7; ECCS component descriptions are found in Section 6.3.

1. Containment Spray Pumps

The CBS pumps are horizontal centrifugal pumps selected to supply the design spray flow rate at containment design pressure. The pumps are designed to take suction from the containment sump at the most limiting NPSH condition (atmospheric pressure and a temperature of 212°F) and pump it back into the containment through the spray nozzles. Design pump discharge pressure takes into account containment pressure, elevation head to the highest nozzles, and piping frictional losses.

Each CBS pump is designed to deliver 3010 gpm from the lowest level in the RHR equipment vault to the highest point in the Containment Building. The minimum calculated CBS system flow rate of 2808 gpm during injection has been shown to be adequate to maintain the containment pressure and temperature within the design envelope for worst case primary and secondary side ruptures.

2. Spray Additive Tank

The spray additive tank (SAT) is mounted adjacent to the RWST, and drains by gravity into the RWST mixing chamber through a six inch diameter pipe which has redundant valving. This line connects the bottoms of the SAT with the RWST mixing chamber. External heaters are provided to prevent freezing or chemical precipitation during cold weather. The mixing ratio of the spray additive tank volume to the RWST volume is such that the pH of the spray solution during the injection phase will average between 9.0 and 9.6 units. The tank is sized to provide the correct amount of sodium hydroxide solution to insure that the final containment recirculation sump pH after injection will be between 8.5 and 11.0 units for the various reactor coolant conditions. No provision is made in the design of the SAT to prevent the reaction of NaOH with atmospheric carbon dioxide during long-term storage.

Proper concentration of sodium hydroxide between 19 and 21% by weight will be verified periodically by chemical analysis.

3. Containment Spray Heat Exchangers

The containment spray heat exchangers are shell and tube-type heat exchangers with spray flow in the tube side and primary component cooling water (PCCW) on the shell side. They are sized such that one containment spray heat exchanger and one residual heat removal heat exchanger provide 100 percent of design heat removal capacity.

Heat exchanger parameters, including flow rates, were selected so that one RHR heat exchanger and one CBS heat exchanger satisfy containment cooling requirements. Table 6.2-76 contains the heat exchanger performance data used for the accident analyses.

4. Spray Headers and Nozzles

The spray headers are positioned in the containment dome to maximize coverage of the containment volume. Four separate headers are used to obtain the distribution of the flow, two for each train. Each train contains 198 nozzles with each nozzle providing a design flow of 15.2 gpm (see Figure 6.2-76 and Figure 6.2-77).

5. Refueling Water Storage Tank

The refueling water storage tank (RWST) is designed to store 475,000 gallons of borated water. This tank is designed to supply water both for refueling operations and to the Containment Spray System and the Emergency Core Cooling System during accident operations. The RWST capacity is based on accident requirements and will supply the safety injection, the charging, residual heat removal and containment spray pumps for at least 26 minutes during the injection phase of a design base accident.

Margin is provided to allow time for transfer of the systems to the recirculation mode and to account for instrument errors. Analysis is based on a minimum of 350,000 gallons of water being injected. An external steam heating supply system is provided to protect against freezing. Tank temperature is indicated locally and alarmed in the main control room.

c. Material Compatibility

The components of the CBS system, including the spray nozzles, are fabricated of materials listed in Table 6.2-75.

The pH of the sump water following an accident is monitored to ensure that the pH is maintained in the correct range (discussed in part f. of this section) as the hydroxide is consumed by chemical reaction with zinc and aluminum within the containment. Two sample points exist to withdraw samples downstream of the RHR heat exchangers: the normal connection to the sample sink, and a local sample point. Sodium hydroxide can be added for pH adjustment using the chemical and volume control system tanks and pumps. The solution is prepared in the chemical mixing tanks and supplied to the suction of the charging pumps. The charging pump suction is fed from the RHR system during recirculation.

Neither the containment spray pumps or motors nor other engineered safeguard pumps or motors are exposed to the containment atmosphere; accordingly, no adverse effects are considered due to the post-accident containment environment.

d. Redundancy

Minimum allowable cooling capacity is assured by utilizing the "double train" concept in both the Emergency Core Cooling and the Containment Spray Systems. These trains are independent of each other, with no interconnection, so that a single active component failure will not cause loss of function of the system. An analysis of a failure of each component is presented in Table 6.2-77.

e. Recirculation Piping

There are two penetrations from the containment sump to the Primary Auxiliary Building (PAB), with each pipe encased in a sleeve. In each line, immediately inside the PAB, is a motor-operated gate valve. After passing through the isolation valve, the flow in each line divides to supply one CBS and one RHR pump. Each isolation valve is enclosed within a housing designed to withstand containment design pressure to prevent any leakage to the PAB atmosphere.

f. Containment Spray System pH Values

The pH of the system is dependent on the ratio of boric acid to caustic. The interaction of these two compounds has been investigated (References 17 and 18) and this data has been used to correlate containment spray composition and pH.

The 21 percent by weight sodium hydroxide in the spray additive tank, when mixed with the borated water in the RWST, nominally produces a pH of 9.6 or less during the injection phase. For maximum initial tank level mismatch, due to instrument uncertainties, the spray pH could exceed 10.3 for about 6 minutes.

The calculations of the spray pH range included such considerations as the physical piping arrangement between the additive and water storage tanks, variation in relative liquid heights in the tanks due to instrument uncertainties and permissible variations in the concentrations of sodium hydroxide in the spray additive and boron in the refueling water storage tanks. Since the UE&C computer program (MIXCH) used in the above calculations has successfully simulated the test data gathered in the testing of a similar gravity feed chemical injection system (Arkansas Nuclear One), a full-scale pH test is not considered necessary for Seabrook. This analysis technique (MIXCH) is detailed in Reference 23.

The maximum pH of the containment sump after a LOCA depends on the concentration of boron in the reactor coolant, i.e., the sump pH is 9.4 at zero ppm boron in the reactor coolant and 8.8 at a 4000 ppm boron concentration. Corrosion products in the solution have a tendency to slightly suppress the pH, generally to an extent less than 0.1 unit. In particular, for the composition of the containment sump water, the reduction in pH is expected to be even less (about 0.02 unit).

g. Net Positive Suction Head Requirements

Adequate net positive suction head is assured by locating the RHR and CBS pumps at the lowest level in the Auxiliary Building. The RHR and CBS pump available net positive suction heads from the containment sump were determined by assuming the limiting conditions in accordance with NRC Regulatory Guide 1.1 (pressure equal to atmospheric and temperature equal to 212°F). The CBS pump available and required net positive suction head is shown in Table 6.2-75. The RHR pump available and required net positive suction head is shown in Table 6.3-1.

h. Heat Exchanger Surface Fouling

The materials used for the CBS heat exchanger are listed in Table 6.1(B)-1. The shell side of the heat exchanger is cooled by the PCCW system which contains a corrosion-inhibiting agent and operates as a closed system. The tube side, which is in contact with the emergency core coolant, is corrosion resistant. The effect of corrosion fouling on heat exchanger surfaces will, therefore, be minimal; however, fouling factors were included in the detailed design of the units to assure the required heat removal capability through conservative design.

i. Heat Exchanger Performance

The heat exchanger (residual heat removal, containment spray, primary component cooling water) temperatures have been selected based upon maximum service water (ultimate heat sink) temperatures and the amount of heat removal required. The flows, geometry, and surface area were studied in the evaluation of the containment pressure-temperature analysis. Those parameters selected and tabulated in Table 6.2-76 are those which meet the design basis requirement for containment cooling.

j. Containment Recirculation Sump and Strainer Design

The containment recirculation sump collects and strains the water available for supplying the residual heat removal, containment spray, safety injection and high head charging pumps during the recirculation mode of operation following an accident. The sump is designed to meet the intent of Regulatory Guide 1.82. Two completely independent sumps are located in the containment to maintain the "double train" concept as described in Subsection 6.2.2.2d.

One sump supplies water to Train A and the other sump supplies Train B. The arrangement of these sumps is shown in Figure 6.2-79. The minimum water level in containment during a loss-of-coolant accident is nominally Elevation (-)23.79 ft.

A series of debris interceptors are provided on the containment floor within the recirculation flow paths. The debris interceptors reduce the quantity of debris transported to the sumps by trapping debris and allowing the remaining debris more time to settle prior to reaching the sumps.

Heavy particles are prevented from reaching the sumps by sloping the surrounding floor away from the sumps. This facilitates settling of debris on the floor prior to reaching the sump area.

A strainer is installed in each sump. Each strainer consists of rows of vertically oriented strainer panels, consisting of a framework sandwiched between two sets of wire cloth attached to perforated plates. The maximum hole size of the perforated plates and maximum width of a gap between bolted structures is 0.068 inches. The strainer would therefore prevent debris particles 0.068 inches or greater in diameter which may be generated following a large break LOCA from passing through or bypassing the strainer and entering the ECCS system. The minimum physical restriction in the ECCS flow path consists of 0.073 inches, which is the effective opening of the fuel assembly debris filter bottom nozzle in combination with the P-grid. Therefore, the strainer will prevent recirculation of debris particles of sufficient size to impede cooling flow to the core.

The strainer panels are mounted on a plenum structure within the sump. The plenum is sealed to the sump floor and at the sump wall adjacent to the ECCS pipe inlet to ensure that all water entering the sump passes through the strainer panels. Water is drawn through the strainer panels and plenum and into the lower portion of the sump.

The strainer will also act as a vortex preventor to further preclude air intrusion into the ECCS piping.

The strainers have been designed to accommodate the debris generated and transported to the sump during the recirculation phase of a LOCA. The head loss due to debris on the strainer is less than the available NPSH margins for operating ECCS pumps, thereby ensuring that cavitation of the ECCS pumps will not occur. Therefore, the design meets the intent of Regulatory Guide 1.82.

The potential for clogging of the sump strainers by equipment and piping insulation or loose insulation in the containment is minimized by the type of insulation used.

The thermal insulation inside the containment for piping and equipment except the reactor pressure vessel is fiberglass blanket insulation of the type commercially known as Nukon, manufactured by Owen's-Corning Fiberglass. The outside surface of the insulation blankets is covered with a stainless-steel jacket or is encapsulated in stainless steel wire mesh. Nukon is consistent with the recommendations of Regulatory Guide 1.36. The reactor pressure vessel is insulated with stainless-steel reflective insulation.

Clogging of the strainers by nonsafety-related equipment is unlikely due to the remote location of the sumps relative to the NNS equipment and physical barriers separating the sumps from other areas in the containment. The supplementary neutron shielding around the reactor vessel which could be displaced by blowdown forces during an accident is designed to remain anchored and intact; hence, it is not a potential source of strainer blockage during an accident.

The design of the sump suction piping ensures that adequate flow and net positive suction head are available to all pumps under the most limiting containment conditions, as required by Regulatory Guide 1.1. The two sumps and the pumps they service are designed so that any single active or passive failure will not cause the loss of both A and B Train components.

The sumps are visually inspected on a periodic basis to assure that they are clean, free of debris and that all strainers are intact and in position. The containment sump line isolation valves are exercised periodically to assure operability within Technical Specification requirements.

k. Periodic Testing

The provisions for periodic testing and inspection of the containment spray system are discussed in Subsection 6.2.2.4.

l. Applicable Codes, Standards and Guides

The codes, standards and guides applicable to the containment spray system are summarized in Table 6.2-79.

m. Remote Manual Operation of the Containment Building Spray System

The CBS system is designed to function completely automatically under accident conditions, hence there are no operations which must be performed manually by the operator from the main control board to initiate the proper function of the system during an accident. After the suctions of the CBS pumps are automatically switched over from the RWST to the containment recirculation sumps, the isolation valves (CBS-V2, CBS-V5) in the discharge line from the RWST will be closed by the operator.

n. Plant Protection System Signals and Setpoints

Operation of the CBS system is initiated automatically upon receipt of a containment spray actuation signal (CSAS). A CSAS is generated when the containment pressure reaches 19.8 psig (see also Subsection 6.2.1.1c.1(b)). The analysis limit of 19.8 psig is established by the requirement to maintain the maximum containment pressure during an accident as low as practical while keeping the setpoint as high as practical to minimize the probability of spray actuation following a small high energy line break. The Engineered Safety Features Actuation System is further described in Section 7.3.

o. Equipment Qualification

Components in the CBS system which are required to function during the accident are qualified (vendor certification) to verify the ability of the components to perform their intended functions under the conditions specified in the purchase documents and/or by test. Environmental qualification of safety-related equipment is discussed in Section 3.11. Tests and inspections are discussed in Subsection 6.2.2.4. Seismic qualification is addressed in Section 3.10. Pump and valve operability assurance is discussed in Subsection 3.9.3.2.

p. Containment Spray System Response Time

Containment spray system response time is discussed in Subsection 6.2.1.1c.

6.2.2.3 Design Evaluation

The analyses of the post-accident containment pressure transients are discussed in Subsection 6.2.1. The double train concept insures that sufficient heat removal capacity will exist, even with a single active failure. Containment design pressure is not exceeded and containment pressure reduction reduces containment leakage to 50 percent of the design leak rate within 24 hours after the DBA.

The refueling water storage tank and the spray additive tank are located within an enclosure building in the yard area. The RWST and SAT are fully enclosed with insulated siding and roof, as well as by two heated buildings (PAB and WPB). Included within these enclosures is the associated piping, vent lines, and instrument tubing.

During cold weather conditions, both the SAT and RWST are heated by steam heating panels mounted on the exterior surface of the tanks. Calculations demonstrate that the RWST heating panel can maintain a minimum water tank temperature of 50°F, and concurrently provide sufficient heat into the enclosure area to maintain an enclosure temperature of 39°F. No credit was taken for heat contributions from the SAT heaters. The site environmental condition for this design evaluation assumed -17°F and 30 mph winds, and are more conservative than the minimum outdoor conditions listed in Updated FSAR Figure 3.11-1.

For the above environmental conditions, the heat loss from the Enclosure Building, including infiltration losses, is 158,000 Btu/hr. as compared to an RWST heating panel capacity of 674,000 Btu/hr. Accordingly, freeze protection is provided for all equipment. In addition, both tank low temperature and enclosure temperature alarms are provided in the main control room.

The maximum temperature of the water in the tanks is calculated to be 86°F, using plant specific meteorological data, assuming maximum solar heat gain and failure of the ventilation fans. The heat of reaction upon the mixing of the boric acid and the sodium hydroxide would raise the temperature of the tank contents approximately 2°F, thereby raising the RWST maximum supply temperature to 88°F. Following a refueling operation, the RWST water temperature may reach a higher value. The maximum temperature considered for all evaluations is 98°F. Neither tank is protected against tornado missiles, and a tornado and accident are not considered simultaneous events. In the event of tornado damage to either tank, the affected unit would be shut down.

The SAT is connected to the RWST by two parallel lines each with an automatic motor-operated valve. The valves are actuated and powered from separate sources to insure that the NaOH solution can be added to the containment spray even in the event of a single active failure.

The method of addition of 20 percent NaOH solution in required concentrations to the borated water drawn from the RWST immediately following a LOCA is primarily dependent on passive components, such as tanks, pipes and a baffled mixing chamber. The rate of addition is dependent on the drawdown rate of the RWST and is based on principles of hydrostatics and hydrodynamics. A description of the system is contained in the following paragraphs.

The system outlined here was chosen in preference to other available systems because it relies upon a minimum number of active components for adding and mixing the NaOH with the borated water and therefore is not dependent on the proper operation of active components (such as eductors and associated recirculation valves, etc.), to achieve the required mixing ratio. Addition of NaOH to the borated water is accomplished by gravity feed through a 6" pipe connecting the SAT and RWST which together remain in hydrostatic equilibrium throughout the period of ECCS injection into the core and the containment atmosphere.

A mixing chamber is provided inside the RWST to thoroughly mix the NaOH with the borated water supplied to the containment spray system by the RWST.

The redundant motor-operated isolation valves between the SAT and the RWST are normally closed and are automatically opened by the containment spray actuation signal. Once these isolation valves are open, the SAT draws down simultaneously with the RWST as the residual heat removal, safety injection, charging and containment spray pumps withdraw water from the RWST. The operating levels of the SAT and RWST are maintained during normal operation to ensure that the two tanks are in hydraulic equilibrium. Since the two tanks are in hydraulic conjunction with each other, they remain in hydrostatic equilibrium throughout the injection phase.

The Containment Spray Actuation System and power supplies are independent and redundant to ensure actuation and power in the event of a single failure. An analysis of possible failures is presented in Table 6.2-77.

a. Passage of Spray Water to the Recirculation Sumps

The containment spray water is distributed over the operating floor at elevation 25' and is then directed to one of three different flow paths to the containment sump. Water holdup is minimized by grated floors and adequate openings in compartment walls and shielding to allow passage of water to the recirculation sumps, as described below:

1. The operating floor has 3520 ft^2 of grating which will pass the spray directly to the lower floor at elevation 0'-0". Spray that impacts and collects on the solid floor at this elevation will freely drain to the grated floor areas. Elevation 0'-0" has 1270 ft^2 of grating which will allow the water draining from above to drain to elevation -26' where the spray will drain to the recirculation sump.

2. The steam generator shielding extends above the operating floor to elevation 32 ft. The top of these shielded cubicles is open to collect spray. However, this path is open to elevation -26 ft and the spray will drain directly to the sump.

3. There are openings in the reactor operating floor for the refueling canal, reactor internals lay down and access to the reactor head region. Eventually, all spray impacting this area drains to the annular region between the reactor vessel and the concrete primary shield wall. This area is not isolated during normal plant operation. Low elevation portions of the refueling canal (See Figure 1.2-6) will drain to the -26' elevation through three four-inch lines each having two valves in series, normally open, but both closed during refueling connected to a common drain path. Each drain path is isolated by two valves in series during refueling.

Adequate openings have been provided in the missile shield walls to allow free passage of water at elevation -26' to the recirculation sump.

The maximum total trapped volume of spray water is 22,830 ft^3 (170,780 gal). See Subsection 6.2.1.1b.6 for details. This is significantly less than the 46,788 ft^3 (350,000 gal) of water supplied from the RWST.

b. Changeover from Injection to Recirculation

The changeover from the injection mode to the recirculation mode during an accident is described in Subsection 6.3.2.8. The containment spray pumps function the same as the RHR pumps during the changeover.

Upon reaching a low-low level in the RWST, the recirculation mode of operation is automatically initiated. The two containment sump isolation valves are automatically opened when the low-low level signal in the RWST is indicated in conjunction with an "S" signal. This valve opening realigns the suction of the containment spray pumps to the sumps for the start of recirculation. The containment spray pumps continue to operate during this switchover. After the switchover is completed, the isolation valves in the discharge line from the RWST are closed by the operator.

Each of the containment spray subsystems is capable of satisfying the system function and either can be removed from service. A single failure does not prevent the transfer to the recirculation mode since each active component is duplicated by the dual train concept.

c. Spray Effectiveness

Each spray train contains 198 SPRAYCO 1713A hollow-cone ramp bottom nozzles. Sixty-five nozzles were randomly selected from a quantity of 325 to evaluate the performance of the nozzles and verify the required flow of 15.2 gpm at 40 psi differential pressure. The average mean droplet diameter for the nozzles tested was 660 microns. This compares to a conservative value of 1250 microns used in the containment iodine removal analysis. The average mean drop diameter was arrived at by numerical averaging based on an instantaneous sampling of spray at design conditions.

Table 6.2-80 lists the percentage of sprayed volume and Figure 6.2-80 and Figure 6.2-81 show the extent of overlapping of the sprays in plan for spray loops A and B, respectively. These figures show virtually 100 percent coverage of the containment at the operating floor level. Figure 6.2-82 and Figure 6.2-83 show the spray loops A and B coverage pattern in elevation views.

Figure 6.2-84, Figure 6.2-85 and Figure 6.2-86 summarize the operating characteristics of the spray nozzles at 40 psi. It should be noted that 99.99 percent of the drops are below a diameter of 1500 microns, have a terminal velocity less than 17.88 fps and contain over 99.99 percent of the total liquid volume. It is assumed that, following a LOCA, the sprays are initiated at a time when the containment atmospheric temperature is 266°F, so that the air steam ratio is 0.78 pounds air per pound steam and the initial spray temperature is 100°F. The drop is considered to be a rigid sphere of radius r_o initially at T_o in an air/steam atmosphere at T_{oo}, and T_{oo} does not change during the time the drop is falling. Steam in contact with the drop will condense, leaving a boundary layer of air around the drop. This is equivalent to having an extremely large air-steam mass ratio, essentially 100 percent air and 0 percent steam. It is also conservatively assumed that this boundary layer is created instantaneously. Heat is transferred to the drop through the boundary layer by convection, and water vapor diffuses through the boundary layer. Ranz and Marshall (Reference 19) provide correlations for both mass and heat transfer. After diffusing through the boundary layer, the steam will condense on the drop surface and the latent heat of condensation will act as a surface heat source.

The assumption of a rigid drop implies the longest time for the drop to heat up, thus providing the most conservative case.

The following equations were used in the analysis:

$$\frac{1}{\rho^2}\frac{\partial}{\partial\rho}\left(\rho^2\frac{\partial\tau}{\partial\rho}\right)=\frac{\partial\tau}{\partial\theta} \tag{1}$$

$$\frac{k_g}{D_V}\frac{P_{BM}}{P}\frac{M_m}{}\frac{RTr_o}{}=1+0.3(Re)^{1/2}(Sc)^{1/3} \tag{2}$$

$$\frac{hr_o}{k}=1+0.3(Re)^{1/2}(Pr)^{1/3} \tag{3}$$

where

$$\tau=\frac{T\infty-T}{T\infty-T_o}$$

$$\theta = \frac{\propto t}{r_o^2}$$

$$\rho = r/r_o$$

$$\mathrm{Re} = \frac{2 V_t r_o \rho_a}{\mu_a}$$

$$V_t = 0.153\, g^{0.714}\, D_p^{1.142} \left(\rho_p - \rho_a\right)^{0.714} / \rho_a^{0.286}\, \mu^{0.428}$$

and $D_p = drop\ diameter$

$\rho_a = density\ of\ drop$

$\rho_p = density\ of\ mixture$

$\mu = viscosity$

$g = acceleration\ of\ gravity$

$K_g = mass\ transfer\ coefficient$

$V_t = thermal\ velocity\ of\ the\ drop$

$R = gas\ constant\ for\ mixture$

$P = total\ pressure\ of\ mixture$

$\propto = thermal\ diffusivity\ for\ mixture$

$t = time$

$P_{BM} = average\ pressure\ of\ inert\ in\ boundary\ layer$

$M_m = average\ molecular\ weight\ of\ boundary\ layer$

$D_v = diffusivity\ of\ water\ vapor$

$h = heat\ transfer\ coefficient$

$k = thermal\ conductivity$

Re, Pr, Sc =Reynolds, Prandtl and Schmidt numbers

Equation (1) describes the temperature behavior of the drop; equation (2) describes the mass transfer coefficient; and equation (3) the convective heat transfer coefficient at the drop boundary.

Parsly (Reference 20) has solved this set of equations numerically using a finite difference method for a range of drop sizes from 500 to 4000 microns in diameter, containment temperatures of 212°F and 266°F, and initial drop temperatures of 86°F, 122°F and 176°F. Using these results for an average spray nozzle height of 134 feet and initial drop temperature of 100°F, equilibration time is much shorter than the time required for the drops of essentially all sizes to reach the containment sump. Table 6.2-81 presents the parametric results.

It can be concluded from Table 6.2-81 that even the largest spray drops attain the containment temperature at times far shorter than the time required to reach the containment sump. As a result, the spray effectiveness value of 1.0 is fully justified in the case of a LOCA. The effectiveness of the sprays following an MSLB, when the containment atmosphere is superheated, is discussed in Appendix 15B.

d. Net Positive Suction Head Available

Adequate net positive suction head (NPSH) for the containment spray pumps is assured under all postulated operating conditions by analysis of the suction head available and vendor testing of the completed pumps.

Maximum calculated flow under the most limiting NPSH conditions, i.e., during recirculation, is 3660 gpm. NPSH available at this flow is 23.76 feet versus a maximum required NPSH of 23.6 feet. The CBS pump analysis of available NPSH conservatively assumes that each residual heat removal pump (which shares a common suction on a train basis with each CBS pump) is also operating at design cold leg recirculation flow of 4388 gpm and considers the suction flow path with the highest hydraulic resistance. The formulas and flow resistance data in Reference 21 were used, along with GE strainer test data, to compute NPSH available.

Table 6.2-78 lists the values of containment pressure head, vapor pressure head of pumped fluid, suction head, and friction head used in the analysis.

e. Integrated Energy, Content of the Containment Atmosphere and Recirculation Water

Figure 6.2-87 and Figure 6.2-88 show the integrated energy content of the containment atmosphere and recirculation water, respectively, as functions of time following the postulated design basis loss-of-coolant accident. The integrated energy absorbed by the structural heat sinks and removed by the containment heat removal heat exchangers is shown in Figure 6.2-89 and Figure 6.2-90, respectively.

f. Debris

The major source of debris that could be generated during a loss-of-coolant accident is insulation. The thermal insulation being used inside the containment will be both stainless steel reflective insulation and fiberglass insulation of the type commercially known as Nukon, manufactured by Owen's-Corning Fiberglass, with a stainless steel jacket over the outside surface of the insulation. Nukon is consistent with the recommendations of Regulatory Guide 1.36.

6.2.2.4 Testing and Inspection

The preoperational testing of the containment heat removal system verified the functional capability of the individual systems under operational conditions.

Testing and inspection of the ECCS systems is discussed in Section 6.3 and Chapter 14.

The preoperational testing of the Containment Spray System verified the operational parameters of the spray pumps during recirculation to the RWST. This testing included a demonstration of system response to ESF signals and the ability of the sump to supply the containment spray and residual heat removal pumps. Flow testing of the nozzles was performed by the manufacturer and was not performed in the field. An air flow test was performed to verify that no nozzles are plugged.

Operability of the gravity feed system was demonstrated during preoperational testing of the ECCS Performance Test (Table 14.2-3, Item 8). The preoperational test will demonstrate the draw-down characteristics of the RWST and SAT during the different flow conditions of the ECCS Performance Test.

The Containment Spray System will be inspected and tested periodically in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI and the requirements of the Technical Specifications.

The containment recirculation sumps will be visually inspected periodically to insure that they are free of debris and all strainers are intact.

6.2.2.5 Instrumentation

The Containment Heat Removal System is provided with instruments and controls to allow the operator to monitor the status and operation of the spray system and to allow the automatic or manual initiation of the injection and recirculation modes of operation.

The manual spray actuation consists of four momentary controls (see Figure 7.2-1, sh.8). Actuation occurs only if two associated controls are operated simultaneously. This prevents inadvertent spray initiation as a result of operator error. The automatic initiation is by coincidence of 2 out of 4 protection set loops, monitoring the containment pressure. The spray actuation signal starts the containment spray pumps and positions all valves to their operating configuration. The design details of the Engineered Safety Features Actuation System are presented in Section 7.3.

The details of the interlocks involved in the suction valve realignment from the RWST to the containment sump during the switchover from injection to recirculation mode are presented in Table 6.3-7. RWST instrumentation is discussed in Subsection 6.3.5. Indications of pump operation are provided by pump status indication lamps and the pressure indications at the main control room. Alignment of automatic valves is indicated by the valve status indications. Additionally, a separate status monitoring indication system is provided at the control room for both modes of the spray system. This enables the operator to evaluate the extent to which the valves are open and if the system is operating effectively. Alarms are also provided to indicate that either train of the Containment Spray System is inoperative. The design features of the bypass and inoperable status alarm system which provide system level indication, in compliance with Regulatory Guide 1.47, are presented in Subsection 7.1.2.6.

Abnormal conditions of RWST level and temperature, RWST enclosure temperature, containment sump level, pump discharge pressure, pump motor temperatures, and heat exchanger outlet temperature are alarmed at the main control room to alert the operator. The design details of the Accident-Monitoring Instrumentation System are presented in Section 7.5.

The Control and Display Instrumentation System is designed to operate under all normal and abnormal conditions, including loss-of-coolant accident and loss of power. Diversity, redundancy of the sensors, circuitry and actuating devices meet the requirements of IEEE-Standard 279 and ensure that minimum system function is provided under postulated abnormal conditions. No single failure of the control and instrumentation will prevent the spray system minimum safety function. The design details of the instrumentation system are presented in Section 7.1.

A comparison of the containment water level instrumentation design with each of the five clarification points of NUREG-0737, Position II.F.1.2, (Page II.F.1.16), is addressed below:

a. The Seabrook design for containment water level complies with this requirement. Refer to clarification c. below for a discussion of the narrow range qualification.

b. The wide-range level measurement is designed to monitor water levels that correspond to all the water from the primary and safety systems and one-half the condensate storage tank. This capability exceeds a liquid volume of 600,000 gallons.

c. The narrow-range water level monitors are not required to operate after their respective sumps have been flooded as their function is to monitor operational leakage. They will only be exposed to a mild environment as any leakage that would cause a harsh environment would flood their sumps and would be detectable by the wide-range (recirculation) sump level indicators and instruments monitoring the containment atmosphere.

   The narrow-range containment sump level instrumentation will be covered by the maintenance/surveillance for equipment that is located in a mild environment.

d. This requirement is not applicable to Seabrook.

e. The functions of the wide-range level indication are:

1. Verify the existence of water in the containment as corroboration of detection of a LOCA
2. Verify that the water injected from the RWST is accumulating in the containment
3. Verify that there is adequate NPSH for the containment spray pumps
4. Verify that the containment water level is less than the design basis flood level.

The accuracy of this indication has been determined, has been reviewed against the functions listed above, and has been found acceptable to support each function.

6.2.3 Secondary Containment Functional Design

The function of the secondary containment (containment enclosure) is to collect any fission products which could leak from the primary containment structure into the containment enclosure and contiguous areas following a loss-of-coolant accident (LOCA). The containment enclosure provides a low leakage rate barrier between the containment and the environment to control all leakage from the containment boundary. The system is comprised of (a) a structural barrier surrounding the containment, adjacent vaults and penetration areas; and (b) a containment enclosure emergency cleanup system which maintains a pressure lower than ambient in the enclosure to prevent uncontrolled releases of radioactivity into the environment.

6.2.3.1 Design Basis

a. Containment Enclosure

1. The containment enclosure is designed for 3 psig differential pressure.
2. The containment enclosure is designed to withstand the transient pressure and temperature conditions produced in the annulus between the containment and the enclosure as a result of either a LOCA within the containment or a high-energy pipe rupture within the containment enclosure annulus.
3. The containment enclosure is capable of withstanding the external pressure conditions resulting from the maximum wind pressure postulated for the site, the external pressure drop resulting from a tornado, and tornado-generated missiles.
4. The containment enclosure is designed to withstand a safe shutdown earthquake.

b. Containment Enclosure Emergency Cleanup System

1. The system is capable of reducing the containment enclosure pressure to negative 0.25 inches w.g. (water gauge) following an accident and maintaining it at or below that level uniformly for up to one year.
2. The system is capable of processing the atmosphere of the containment enclosure space while maintaining the design negative pressure differential.
3. The system is designed to permit periodic inspection and monitoring of functional capability.
4. This system is designed in accordance with Regulatory Guide 1.52.
5. The system is designed to seismic Category I and Safety Class 2 requirements.
6. The system is designed to retain functional capability while experiencing a loss of offsite power concurrent with a LOCA and any single active component failure.

6.2.3.2 System Design

a. Containment Enclosure

The containment enclosure is comprised of a right cylindrical structure with a hemispherical dome and other penetration and equipment areas as described in Table 6.2-82. These structures completely enclose the containment, forming a second barrier to the uncontrolled escape of radioactive sources in the event of an accident. The inside diameter of the cylinder, constructed of reinforced concrete, is 158 feet. The vertical wall varies in thickness from 15 to 36 inches, and the dome is 15 inches. The inside of the dome is 5'-6" above the top of the containment structure. Design and performance data are listed in Table 6.2-82. The annular cylinder formed by the containment and the enclosure is shown on Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, Figure 1.2-5 and Figure 1.2-6. Codes, standards and guides applied in the design of this structure are discussed in Subsection 3.8.4.2.

b. Features in Support of the Containment Enclosure

All piping penetrating the containment structure is sealed and anchored at the containment structure and at the containment enclosure so as not to be overstressed by thermal or seismic-induced motion. Electrical penetrations are sealed and anchored at the containment structure.

The containment recirculation sump lines are enclosed in a sleeve which extends out to a vessel which encloses the first isolation valve outside the containment. This enclosure serves to contain any leakage from the sump line and first isolation valve.

The containment enclosure isolation features are discussed in Subsections 9.4.6.2 and 6.5.1.

All personnel doors and equipment hatches in the containment enclosure are under administrative control. The doors are provided with position indicators and alarms having readout and alarm capability at the primary and secondary alarm stations. These doorways and hatches must be closed to insure a negative pressure in the containment enclosure.

c. Containment Enclosure Emergency Cleanup System

This system has two functions: (1) to produce a negative pressure post accident in the annular, cylindrical volume between the containment and the containment enclosure, and (2) to collect any hazardous materials that might leak into these areas from the containment structure or equipment/systems located within the enclosure (ECCS) so that they may be disposed of in a controlled manner. Both these functions are performed by redundant filter trains, redundant fans, dampers and controls, and a common discharge ductwork system to the unit plant vent.

Each exhaust fan has a sufficient capacity, with a clean and dirty filter train, to remove the in-leakage into the entire containment enclosure area calculated to occur at the design negative differential pressure, maintaining the required differential pressure. Subsection 6.2.3.3a discusses the performance of the fans. The presence of the containment enclosure and the use of the exhaust fans to produce a slightly negative pressure between it and its external surroundings minimize the direct leakage from the containment structure to the environment.

The redundant filter trains contain moisture separators, upstream HEPA filters, carbon adsorber bank and a downstream HEPA filter bank. The use of HEPA and charcoal filters in the exhaust from the containment enclosure reduces the discharges of radioactive iodine so that offsite doses following a LOCA are within the guidelines of 10 CFR 100.

All components of the Containment Enclosure Emergency Cleanup System required to operate following an accident are Safety Class 2, seismic Category I. The system does not have provision for recirculation flow. Additional details are presented in Subsections 6.5.1 and 6.5.3.

d. Containment Enclosure Bypass Leakage

The maximum allowable leakage from the containment structure following an accident is 0.15 percent of the mass of its atmosphere per day. This would occur at maximum pressure. During the first 24 hours following a LOCA, the containment heat removal systems reduce the pressure, the driving force behind the leakage, to less than one-half the maximum value. As discussed in the preceding section, the direct leakage to the environs of radioactive contaminants from the containment is within the guidelines of 10 CFR 100.

Although, as discussed in the preceding section, a containment enclosure emergency cleanup system has been provided to minimize leakage to the environs, a significant number of lines penetrate the containment and terminate in areas not treated by this cleanup system. Therefore, all leakage attributed to penetrations and isolation valves, requiring Type B and Type C Test per 10 CFR 50, Appendix J, is conservatively assumed to bypass the cleanup system. The total allowable leakage for Type B and Type C Tests and for combined bypass leakage is discussed in Subsection 6.2.6.3. This is in accordance with Appendix J acceptance criteria.

6.2.3.3 Design Evaluation

The containment enclosure system design is evaluated from two viewpoints. Subsection 6.2.3.3a investigates the adequacy of the structure and associated equipment to achieve its functional goal, a negative pressure differential. Subsection 6.2.3.3b considers the vulnerability of the system to damage from a high-energy line rupture within the enclosure.

a. Containment Enclosure Analyses

One train of the Containment Enclosure Emergency Cleanup System is required to be able to draw down the entire Containment Enclosure Area to a negative differential pressure of 0.25 iwg. This differential pressure is required to be established between all areas that comprise the Containment Enclosure Area and their external surroundings. The areas that comprise the Containment Enclosure Area are listed in Table 6.2-82. This negative differential pressure has to be established within 8 minutes following a LOCA. The radiological dose analyses for a LOCA described in Section 15.6.5.4 begins to take credit for filtration of radioactive contaminants that leak into the Containment Enclosure Areas at 8 minutes following an accident. Per Appendix 15B, no credit is taken for filtering out any of the radioactive contaminants released into the Containment Enclosure Areas for the first 8 minutes following an accident. The filter removal efficiencies are set at 0 during this time.

Analysis has shown that one containment enclosure exhaust filter fan is capable of drawing down the entire containment enclosure area to the design negative differential pressure in less than 8 minutes after the initiation of a design basis LOCA. This analysis takes into account the engineered safety feature actuation system signal delay time, delay time for the diesel generator to supply power in the event of a simultaneous loss of offsite power, and the time for the filter fan to come up to speed.

In the event of a LOCA, the containment could experience an increase in volume on the order of 11,500 cubic feet because of thermal and pressure expansion. This is determined using the data listed in Table 6.2-82, and considers the swelling of the containment structure due to the design pressure of 52 psig. This would result in a decrease in the free volume of the Containment Enclosure Building of less than 1 percent with a similar corresponding rise in its pressure. The time for an exhaust filter fan to bring the Containment Enclosure Area back down to atmospheric pressure to compensate for the swelling of the containment structure is included in the analysis of the Containment Enclosure Area draw down time.

The analysis also includes the time required for an exhaust filter fan to reduce the Containment Enclosure Area to the required negative differential pressure. A separate analysis has determined that it is necessary to establish a negative differential pressure of 0.685 iwg at the 21' -0" elevation of the Containment Enclosure Ventilation Area. This will ensure that a negative differential pressure of 0.25 iwg exists at the top elevation of the Containment Enclosure for the full range of design basis outside ambient temperatures. The draw down analysis conservatively includes the time required for one filter exhaust fan to remove enough air to draw down the entire Containment Enclosure Area to an internal pressure of negative 0.685 iwg.

The analysis to verify this draw down time also takes into account a total calculated maximum in-leakage of 1025 cfm. This is the in-leakage through various air flow paths such as electrical, piping and duct penetrations, concrete structure, construction joints, doors, seal plates, metal partitions, ducts and floor drains. Air in-leakage was determined using data from the penetration sealant supplier, analytical calculation and experimental leakage data provided in "Conventional Buildings for Reactor Containment NAA-SR-10100 (1965)," issued by Atomics International, a Division of North American Aviation Incorporated. For conservatism, this in-leakage is calculated at the maximum differential pressure of 0.685 iwg. The calculated maximum in-leakage also includes leakage from the primary containment at a rate of 0.2% of the primary containment volume for the first day following a design bases LOCA. This assumption is conservative because primary containment leakage is limited to 0.15 percent by weight. This analysis also assumes a low airflow of 1890 cfm from a single fan. This is the minimum value acceptable during system surveillance testing. When all of these factors are taken into account, one filter exhaust fan is still capable of achieving the design negative differential pressure in less than the required design basis draw down time of 8 minutes.

Test or sampling connections in lines penetrating both the containment and containment enclosure are protected by either two isolation valves or by a locked-closed valve and one isolation valve so that no single failure can compromise the ability to achieve negative pressure by allowing a source of suction fluid to the exhaust fans other than the atmosphere of the containment enclosure.

The analyses of the pressure/temperature response of the containment to a LOCA, performed for Subsection 6.2.1, have demonstrated that there is never any significant change in the temperature on the outside of the containment wall. Accordingly, the temperature in the containment enclosure is determined by the heat generated by the equipment present inside it and energy removal by the containment enclosure cooling units which function both during normal plant operations and in the event of a LOCA. The cooling coils have been sized to continuously maintain the temperatures, in the areas to be cooled, for normal, abnormal and accident conditions as discussed in Section 3.11(B).

b. High Energy Line Rupture

The main steam, feedwater, and steam generator blowdown lines pass through the containment enclosure, but not directly. The enclosure boundary terminates on the fluid portion of the penetration for these pipes on main steam tunnel sides. The residual heat removal line also passes through the containment enclosure, but is classified as a moderate energy line because of its short operational period. Therefore, ruptures of these lines within the containment enclosure are not considered.

Failure of a high-energy line would result in pressurization of the containment enclosure due to the mass and energy release. The high-energy lines that penetrate the containment and traverse the Enclosure Building without guard pipes are:

1. Sample lines from the pressurizer
2. Sample lines from reactor coolant loops
3. Excess letdown line
4. Letdown line.

The sample lines (items 1 and 2 above) are normally isolated, and are only opened daily for short duration to purge the line and collect samples. Since such lines are under the direct control of the operator taking samples, their failure could be immediately detected by lack of pressure at the sample sink, and isolation effected by the operator. Also, the sample line isolation valves will automatically close on a containment isolation "T" signal.

The excess letdown line (item 3 above) is normally isolated. During operation, the liquid letdown is cooled to approximately 165°F by primary component cooling water in the excess letdown heat exchanger. Accordingly, the failure of this line within the containment enclosure is less severe than for the normal letdown line. The flow rate in the excess letdown line, when in use, is approximately 2100 lb/min of subcooled water.

Because of the mitigating factors associated with the rupture of other lines, a complete double-ended guillotine break of the 3-inch letdown line (item 4 above) becomes the design basis accident for the pressurization of the containment enclosure.

This break is also the most severe small line rupture outside containment from the standpoint of radiological consequences, and is discussed in Subsection 15.6.2. As described there, this break leads to a flow of 140 gpm of water at 380°F (having been cooled from 560°F by the regenerative heat exchanger) for a period of less than 30 minutes, by which time the operator will have isolated the rupture. With the secondary containment at 104°F, 10 percent relative humidity, and atmospheric pressure, the water flashes at the break into a steam-water mixture which would pressurize the containment enclosure to 1.5 psig within the first half hour. Whenever the flow is terminated, the pressure would begin to drop. No credit is taken for ventilation and purging, and only the containment enclosure annulus free volume was available for pressurization for conservatism. The design pressure for the containment enclosure structure is 3 psig.

6.2.3.4 Tests and Inspections

Preoperational testing of the containment enclosure and its associated exhaust system is discussed in Subsection 6.5.1.4 and Chapter 14.

Periodic testing of the containment enclosure exhaust fans is discussed in Subsection 6.5.1.4. This periodic testing will also include a visual surveillance of containment enclosure penetration and other seals.

6.2.3.5 Instrumentation Requirements

The system monitoring instrumentation and controls are provided in the main control room. Instrumentation associated with the Containment Enclosure Emergency Cleanup System is described in Subsection 6.5.1.5. The system is automatically initiated on a 'T' (containment isolation phase A) signal. The Engineered Safety Features Actuation System is described in detail in Section 7.3. The logic, controls and instrumentation of this engineered safety feature system function so that a single failure of any component will not result in the loss of functional capability for the system. Further information on safety-related instrumentation is included in Chapter 7. Area radiation and airborne radioactivity monitoring instrumentation details are presented in Subsection 12.3.4.

6.2.4 Containment Isolation System

The Containment Isolation System is comprised of the valves, piping and actuators required to isolate the containment following a LOCA or steam line rupture. The systems establish and/or maintain isolation of the containment from the outside environment to prevent the release of fission products, and to ensure that the public is protected in accordance with 10 CFR 100 guidelines.

Each piping penetration of the containment, except penetrations associated with engineered safety features equipment, is required to maintain or establish isolation of the containment under any loss-of-coolant accident or main steam pipe rupture that will initiate the containment isolation signals.

6.2.4.1 Design Bases

The design bases for the containment isolation system are as follows:

a. Minimize and limit the atmospheric release of radioactive materials in the event of a LOCA by isolating those lines penetrating the containment which are not required for the operation of the engineered safety feature systems.

b. Avoid the reactivity effects that could result from excessive cooldown of the Reactor Coolant System in the event of a steam line break accident, and prevent the overpressurization of the containment during such an occurrence by isolating the containment as well as the steam generators, as may be required to fulfill these objectives.

c. Provide double barrier protection for all lines that penetrate the containment, where a barrier may consist of a valve, a closed system, or a diaphragm depending upon its location and application.

   A closed system is one which satisfies all of the following requirements:

   1. The system does not communicate with either the Reactor Coolant System or the containment atmosphere.
   2. The system is protected against missiles and pipe whip.
   3. The system is designated seismic Category I.
   4. The system is classified Safety Class 2.
   5. The system is designed to withstand temperatures at least equal to the containment design temperature.
   6. The system is designed to withstand the external pressure from the containment structural acceptance test.
   7. The system is designed to withstand the environment and transient conditions resulting from either a loss-of-coolant accident or a main steam line break.

   Sealed-closed barriers which replace automatic isolation valves include blind flanges and locked-closed isolation valves. These barriers, which remain closed after a LOCA, will be managed through administrative controls.

d. The containment isolation system design shall comply with the requirements of 10 CFR 50, Appendix A, General Design Criterion 54, and other applicable criteria as follows:

1. Lines that are part of the reactor coolant pressure boundary and penetrate the containment, or are connected to the containment atmosphere, have their penetrations designated as Type I and are provided with valves as follows:

   (a) One locked-closed isolation valve inside and one locked-closed isolation valve outside containment; or

   (b) One automatic isolation valve inside and one locked-closed isolation valve outside containment; or

   (c) One locked-closed isolation valve inside and one automatic isolation valve outside containment; or

   (d) One automatic isolation valve inside and one automatic isolation valve outside containment.

   These provisions are in accordance with General Design Criteria 55 and 56.

   A simple check valve is considered an automatic isolation valve only on the inside of the containment on lines with flow coming into the containment.

2. Lines that penetrate the containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere have their penetrations designated as Type II. These lines are provided with valves as follows:

   (a) One automatic isolation valve outside containment; or

   (b) One locked-closed isolation valve outside containment; or

   (c) One isolation valve outside containment capable of remote manual operation.

   The second isolation barrier on these lines is provided by the closed system inside containment, as defined in Subsection 6.2.4.1c.

   These provisions are in accordance with General Design Criterion 57.

   A simple check valve is considered an automatic isolation valve only on the inside of the containment on the lines with flow coming into the containment. Therefore, a stop-check valve is provided on the emergency feedwater line outside the containment.

3. Connections/lines only for test purpose are provided with manual isolation valves that are opened only for testing. These valves are always closed when not testing.

4. Instrument lines which are connected to the RCS are designed so that a break within these boundaries results in a relatively small flow that can be made up with the normal charging systems, which is in accordance with Regulatory Guide 1.11. The penetrations of these lines and lines connected to the containment atmosphere are designated as Type III, and are provided with isolation barriers in accordance with 10 CFR 50, Appendix A, General Design Criteria 55 and 56. See Subsection 6.2.4.2m(4) for exceptions to Regulatory Guide 1.141. The containment pressure, reactor coolant wide-range pressure, reactor vessel level and core differential pressure instrument lines are designed in accordance with the provisions of Regulatory Guides 1.141 and 1.151. For the containment pressure transmitters, isolation from the containment atmosphere is provided by a sealed bellows arrangement located immediately inside the containment wall, and is connected to the pressure transmitter outside containment by sealed, fluid-filled, tubing. Isolation outside containment is provided by the diaphragm in the pressure transmitter.

   The RCS wide-range pressure, level and core differential pressure transmitters have a sealed bellows connected to the RCS and a second sealed bellows-type isolator outside the containment to provide two barriers in addition to the diaphragm in the transmitters. All components are connected by sealed, fluid-filled, tubing.

e. Relief valves are used as isolation valves and their relief setpoint is greater than 1.5 times the containment design pressure, in accordance with Standard Review Plan 6.2.4.

f. The containment isolation systems are designed to remain functional following a safe shutdown earthquake.

g. Containment isolation valve closure speeds and leak tightness will prevent radiological effects from exceeding the guidelines established by 10 CFR 100.

h. Classification of essential and nonessential systems that penetrate containment is given in Table 6.2-83. Essential systems are defined as those piping systems penetrating containment which are necessary for mitigating the consequences of an accident; nonessential systems are classified as those piping systems penetrating containment which provide auxiliary service functions for operation of the plant, and which are not required for mitigation of accidents.

i. Although it is not specifically required to isolate containment in response to Station Blackout, the capability to establish containment integrity is provided (see Section 8.4.4.5).

6.2.4.2 System Design

a. General Description

Schematic representations of the isolation valving systems which define the fluid systems penetrating the containment wall, including instrument lines, are shown on Figure 6.2-91, sh.1, Figure 6.2-91, sh.2, Figure 6.2-91, sh.3, Figure 6.2-91, sh.4, Figure 6.2-91, sh.5, Figure 6.2-91, sh.6, Figure 6.2-91, sh.7, Figure 6.2-91, sh.8, Figure 6.2-91, sh.9, Figure 6.2-91, sh.10, Figure 6.2-91, sh.11. All valves and piping are fabricated of suitable stainless and/or carbon steel to conform to the ASME Boiler and Pressure Vessel Code, Section III, Code Class NC. Special attention is given to materials to ensure there are no radiolytic or pyrolytic decomposition products to interfere with the safe operation of any engineered safety features. Section 6.1 includes further discussion of the materials.

b. Component Description

A summary of the fluid system lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 6.2-83. Each process valve is described as to type, open or closed status during normal operation, shutdown and accident conditions, and closure/opening time. Information is also presented on valve preferential failure mode, position indication, isolation signal and location relative to containment. The test, vent and drain (TVD) valves associated with penetrations are not specified in Table 6.2-83, but are shown on Figure 6.2-91, sh.1, Figure 6.2-91, sh.2, Figure 6.2-91, sh.3, Figure 6.2-91, sh.4, Figure 6.2-91, sh.5, Figure 6.2-91, sh.6, Figure 6.2-91, sh.7, Figure 6.2-91, sh.8, Figure 6.2-91, sh.9, Figure 6.2-91, sh.10, Figure 6.2-91, sh.11. All TVD valves located between the containment isolation valves are identified. For the remaining TVD valves (located outside the containment isolation valves), only the test valves are schematically shown as arrows to identify all containment isolation valves, demonstrate the ability to perform the Type C test (if required), and yet provide a clear, unobstructed schematic representation of containment penetrations. For further clarity, alphabetic suffixes were added to the individual lines of the multiple-line penetrations. These suffixes do not appear in other design documents.

c. Valve Actuation Signals

The design of the system providing the signals for containment isolation complies with the following general requirements:

1. The containment isolation signal overrides all signals for actuations of containment isolation valves for nonessential systems.
2. Phased isolation is used. With phased isolation, all systems except engineered safety features and non-engineered safety features-related systems are automatically isolated. Only those engineered safety feature-related systems that can be justified to remain operational shall not be automatically isolated during the initial phase.
3. Diverse parameters are used wherever possible for developing isolation signals.
4. Concurrent containment isolation occurs coincident with initiation of emergency core cooling.
5. All valves that receive a containment isolation signal cannot be reopened until the isolation signal is reset and manual action is taken to reopen the valve. The controls are separated so that only one valve, or group of valves associated with a penetration, open for each manual action.

Automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals ("T" Signal) is derived in conjunction with automatic safety injection actuation on Hi-1 containment pressure, and trips the majority of the automatic isolation valves. The Hi-1 setpoint is the lowest practical and includes margin for containment pressurization, instrument error and operating margin. These are valves in the nonessential process lines which do not increase the potential for damage to in-containment equipment when isolated.

This is defined as "phase A" isolation, and the valves are designated by the letter "T" in the isolation diagrams of Figure 6.2-91. The second, or "phase B," containment isolation signal ("P" Signal) is derived from Hi-Hi containment pressure and/or actuation of the Containment Spray System, and trips the automatic isolation valves in the other process lines (which do not include safety injection lines) penetrating the containment. These isolation valves are designated by the letter "P" in the isolation diagrams.

Containment air purge (CAP) and containment online purge (COP) system lines which provide an open path from the containment environs are equipped with radiation monitors that are capable of isolating these lines upon receipt of a high radiation signal, in addition to automatic safety injection actuation, manual containment spray actuation and manual phase "A" isolation signals. Further discussion of containment isolation signals is found in Section 7.3. (Refer to Figure 7.2-8.)

d. Valve Closure Time

The objective in establishing valve closure time is to limit the release of radioactivity from the containment to as-low-as-is-reasonably-achievable. Consideration is given to the fluid system requirements (e.g., water hammer) in determining the valve closure time, the effect of closure time on valve reliability, as well as the containment isolation requirements.

These considerations have been addressed in the design of the containment isolation system, within the context and requirements of the guidelines and applicable criteria presented in Subsection 6.2.4.1, Design Bases.

Isolation valve closure times for the Containment Isolation System are presented in Table 6.2-83. The valves listed there reflect the maximum time required to isolate a system so that radioactive release to the environs during a design basis accident is within limits in 10 CFR 100. Refer to Subsection 9.4.5 for discussion of containment online purge line isolation.

e. Operability of Valves Inside Containment

Isolation valves located inside containment are subject to the high pressure, high temperature, steam-laden atmosphere resulting from an accident. Operability of these valves in the accident environment is ensured by proper design, construction and installation, as reflected by the following considerations:

1. All components in the valve installation, including valve bodies, trim and moving parts, actuators, instrument air control and power wiring are constructed of materials sufficiently temperature and humidity resistant to be unaffected by the accident environment. Special attention is given to electrical insulation, air operator diaphragms and stem packing material. Section 3.11 discusses the qualification of this equipment for operation in the containment atmosphere during an accident condition.

2. In addition to normal pressures, the valves are designed to withstand maximum pressure differentials in the reverse direction imposed by the accident conditions. Additionally, a review was performed to ensure that the containment isolation valves inside the containment were not overpressurized due to trapped fluid in adjacent piping in the post-LOCA environment.

3. The containment structure online purge subsystem is designed to prevent debris from entering the exhaust and supply lines to ensure the operability of the isolation valves. This is accomplished by debris screens installed in the ends of the lines. Each debris screen consists of heavy-bar stainless steel grating, banded and welded to the exhaust and supply ends of the lines. Both the exhaust and inlet piping have two 90° bends and a minimum of 14 feet. This design greatly reduces the possibility of direct impingement of debris on the valves. The pipe, screens and supports are seismic Category I. The screens will be capable of withstanding the differential pressure resulting from a LOCA up to the point of containment isolation.

Operability of valves and their operators within containment atmosphere is addressed by qualifying this equipment to IEEE Standard 382-1972, Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations, and NRC Regulatory Guide 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants. Updated FSAR Section 3.9 provides additional information on valve operability.

Containment isolation valve operators have been provided to ensure adequate reliability for the operation of the valves. Lines penetrating the containment that serve the engineered safety features as well as their associated support systems are provided with motor operators. To verify that motor operators have sufficient torque to operate the valves, the valves are tested by opening and closing under full system pressure. Operability during and after a LOCA is thus assured.

Isolation valves outside containment, which are not required to function during a post-LOCA condition, are provided with air operators and spring return to the fail-safe position.

All motor-operated valves have manual handwheel operators and can be closed or opened manually on loss of primary power.

f. Location of Valves Outside Containment

All isolation valves outside the containment are located as close to the containment as practical. All exterior isolation valves are located within 15 feet of containment, except for the main steam, feedwater and combustible gas control isolation valves. The main steam isolation valves are located within 75 feet of penetrations X-1 (westside) and X-2 (eastside), and 85 feet away from penetrations X-4 (westside) and X-3 (eastside). The feedwater isolation valves are located within 30 feet of the containment. The combustible gas control isolation valves are located within 50 feet of containment.

The distances of the main steam isolation valves from the containment mentioned above are required to accommodate the main steam safety valves and power-operated relief valves. The distance of the feedwater and combustible gas control isolation valves from the containment accommodates the required physical piping arrangement and provides accessibility to the valves for maintenance.

Two main steam and two feedwater lines emerge from the containment structure and enclosure on each side of the building, 180° from each other. On each side, the four lines turn and run parallel toward the Turbine-Generator Building. These lines and their isolation valves are enclosed in a seismic Category I structure and are shielded from tornado-generated missiles. The structure is designed with sufficient vent openings to the external atmosphere so that any possible pipe failure inside will not cause failure of the building by overpressurization.

g. Actuation and Control Equipment

Containment isolation valves are provided with actuation and control equipment appropriate to the valve type. For example, globe and diaphragm valves are generally fitted with air diaphragm operators which will fail in the safe position on loss of operating air. Gate valves are generally fitted with motor operators and are powered from emergency buses. On loss of offsite power, the power source is automatically switched to the diesel generators which feed the emergency buses. Motor-operated valves fail in the as-is position. No manual operation is required for immediate isolation.

If actuating power is lost, the automatic air-operated isolation valves assume the position (open or closed) that provides greater safety. Motor-operated isolation valves fail "as is." Manual control switches provide a secondary mode of actuation for the automatic isolation valves. The positions of the automatic isolation valves and remote manual valves are displayed in the main control room at both their control switch location and as part of the Post-Accident Monitoring System except for the main steam isolation valve (MSIV) bypass valves (MS-V-204, MS-V-205, MS-V-206, and MS-V-207). The circuit breakers for these valves are administratively controlled locked open (off) with the valves locked closed during power operation except for surveillance testing and to equalize pressure across the MSIVs before they are opened. With the circuit breakers open, the indicating lights at the control switch are off but the Post-Accident Monitoring System lights are still operable.

h. Seismic Design

Protection for containment isolation systems and components against loss of function due to seismic event forces is provided. Containment isolation valves and their operators are designated as seismic Category I. Containment isolation provisions are capable of maintaining the isolation function during and after the Safe Shutdown Earthquake (SSE). The valves are capable of being realigned after the Design Basis Event and to withstand seismic aftershocks following the SSE.

The containment isolation valves, their operators, and supports are designed to assure that they are capable of withstanding the Safe Shutdown Earthquake as recommended by Regulatory Guide 1.29.

To assure their adequacy in this respect:

1. Valves are located in a manner to reduce their accelerations. Valves suspended on piping spans are designed for the loads to which the span would be subjected. Valves are mounted in the position recommended by the manufacturer.

2. Valve yokes are designed for adequacy and strengthened as required for the response of the valve operator to seismic loads.

3. Where valves are required to operate during seismic loading, the operator forces are factored in the design to assure that system function is preserved.

4. Control wires and piping to the valve operators are designed to assure that the flexure of the line does not endanger the control system. Appendages to the valve, such as position indicators and operators, are designed for structural adequacy.

Additional information regarding the seismic design of systems and components is presented in Sections 3.7 and 3.9.

i. Missile Protection

All containment isolation valves, actuators, and controls are located so as to be protected against accident-generated missiles including those caused by a loss-of-coolant accident. The isolation valve gallery outside the containment is partitioned into four areas by a vertical wall and platform. In addition to providing missile protection, this arrangement provides for separation of redundant safety-related lines and serves as radiation shielding and a work platform during maintenance periods.

Inside the containment, missile protection is provided by a missile shield wall and accumulator tanks. No extraneous equipment is placed in the valve galleries. Only valves so protected are considered to qualify as containment isolation valves. Details regarding the probability of missile generation and design features to prevent the formation of missiles are given in Section 3.5.

j. Potential Leakages

All nonautomatic isolation valves which are left open to perform post-accident functions, except MS-V393 and MS-V394, have potential leak paths into the containment enclosure only. The design of the enclosure recirculation and filtration system is such that all possible leaks have been considered in determining the capacity of the system to maintain the potential radioactive discharge within the limits of 10 CFR 100 subsequent to a LOCA.

Provisions are made to detect and minimize the leakage from the engineered safety features and auxiliary systems located outside containment. Leakage monitors are installed to provide the plant staff with the current knowledge of the system leakage rates. Multiple monitors for noble gas effluents are installed with an extended range designed to function during accident as well as normal operating conditions.

The potential sources of bypass leakage past the containment enclosure are listed in Table 6.2-83. Details of leakage acceptance limits and testing are given in Subsection 6.2.6.

k. Physical Separation

Physical separation of containment isolation systems and components is provided where required in accordance with design guidance contained in "Physical Protection for Systems and Components Important to Safety, "ANS-58.3/N182 - 1977, "Standard Criteria for Separation of Class 1E Equipment and Circuits," Trial-Use, IEEE-384-1974, and in NRC Regulatory Guide 1.75, "Physical Independence of Electric Systems."

l. Fire Protection

Fire protection for containment isolation provisions against loss of function from fire is provided in accordance with "Generic Requirements for Light Water Nuclear Power Plant Fire Protection," ANSI/ANS-59.4-1979, "Standard for Fire Protection for Nuclear Power Plants, NFPA report 803-1978, and "Seabrook Station Fire Protection System Evaluation and Comparison to APCSB 9.5-1 Appendix A."

Additional information on fire protection is presented in Subsection 9.5.1.

m. Clarifications

All characteristics of the containment isolation system design conform to the criteria, regulatory guidelines, and other standards mentioned above, with the following clarification:

1. Only one isolation valve outside the reactor containment is provided on each of the two lines between the containment building recirculation sump and the suction of the residual heat removal and containment spray pumps. This reduces the probability of the valve failing to open when called upon to function. The pipe between the containment sump and the isolation valve is jacketed, and the isolation valve is enclosed in protective chambers so that failure of the pipe or the valve body will not result in release to the environment of radioactive fluid or gases.

2. Closed system (as defined in Subsection 6.2.4.1c) is used as a second isolation barrier for the containment penetrations of the following systems: residual heat removal, safety injection, chemical volume and control (SI portion only), containment building spray, main steam, feedwater, steam generator blowdown, and primary component cooling water (thermal barrier portion only).

(a) The containment penetrations for the main steam, feedwater, steam generator blowdown and primary component cooling water (thermal barrier portion only) are designated as Type II penetrations. The definition of the Type II penetrations is provided in Subsection 6.2.4.1d.2.

(b) The remaining penetrations which use a closed system as a second isolation barrier are designated as Type E. The following provides the details for the individual penetrations:

(1) The residual heat removal system utilizes two normally closed, pressure-interlocked valves in series for each suction line inside the containment. Further details on interlocks are discussed in Chapter 7. This arrangement decreases the probability of release to the environment of radioactive fluid or gases by eliminating a potential leakage point, and retains redundant isolation capability should a residual heat removal system pipe rupture occur outside the containment.

The valve which is located closer to the RCS is not considered a containment isolation valve. The second valve defines the containment isolation barrier inside containment and is considered to be sealed closed. This containment isolation arrangement is as described in ANS 56.2/ANSI N271-1976 and endorsed by Regulatory Guide 1.141.

(2) The arrangement of CBS/ECCS suction penetrations is described in Item 1 above.

(3) The discharge of all engineering safety features systems utilizes the check valves located inside containment as automatic isolation barriers.

3. Each supply line to the hydrogen analyzer portion of the combustible gas control system is provided with two closed manual isolation valves outside containment. The isolation valve inside containment is locked open to allow for post-LOCA operation of the analyzers. The first isolation valve outside containment is locked closed except when the hydrogen analyzers are in operation. An additional isolation barrier is provided by a closed system outside containment.

The locked closed containment isolation valves are not required to be Type C tested as per Table 6.2-83. These lines form a closed, seismic Category I system outside containment. The integrity of the closed system is maintained by performing periodic surveillance testing as described in Subsections 6.2.5.1.g and 6.2.6.

4. The sealed, fluid-filled, instrument systems which penetrate the containment boundary are designed to ANSI B31.1, Seismic Category I, not ASME Section III. The penetration is designed to ASME Section III, Code Class 2. See Subsection 7.1.2.12 for a discussion of compliance to Regulatory Guide 1.151. These sealed sensing lines, shown on Figure 6.2-91, sh.9, have no isolation valves, but because of the isolation barriers provided as part of the sealed, fluid-filled, system, a postulated severing of the line during either normal operation or accident conditions will not result in any radioactive release from the containment. This containment isolation arrangement is as described in ANS 56.2/ANSI N271-1976 and endorsed by Regulatory Guide 1.141. See Subsection 6.2.4.1d for additional discussion.

5. Not used.

6. Some of the pressure indicators and transmitters on the main steam line outside the containment (such as PT 3001 to 3004, 3173, 3174, 3178, 3179, and PI 3051 to 3054) are not qualified since they are not required for engineered safety systems, are on secondary (noncontaminated fluid) loops, and are on small lines whose breaks will not result in any significant steam leaks. This is consistent with the requirements of Regulatory Guide 1.11.

7. The CAP System penetrations each have a blind flange using a resilient double o-ring design installed on the penetration outside containment during Modes 1, 2, 3, and 4. The design includes a test groove between the o-rings for Type "B" testing in accordance with 10 CFR 50, Appendix J. The penetrations terminate in the containment enclosure during plant Modes 1, 2, 3, and 4, thereby allowing any leakage to be collected and processed by the Containment Enclosure Ventilation System.

6.2.4.3 Design Evaluation

The Containment Isolation System has been designed in accordance with Regulatory Guide 1.11, Design Criteria 54, 55, 56, 57.

Accordingly, it has been specifically designed to:

a. Isolate lines penetrating the containment, which are not required for the operation of the engineered safety feature systems, in the event of a LOCA;

b. Isolate lines penetrating the containment, which are not required for the operation of the engineered safety feature systems, in the event of a main steam line break;

c. Shut the isolation valves in the CAP and COP systems upon detection of high radioactivity.

Isolation of the containment in the event of a LOCA or a steam line rupture is initiated upon receipt of isolation signals as discussed in Subsection 6.2.4.2c and Section 7.3.

6.2.4.4 Tests and Inspections

During preoperational testing, tests are performed on the containment isolation system to verify valve response to containment isolation signals, and to determine valve leakage rates. Chapter 14 contains a further discussion of the preoperational tests performed on this system.

Periodic testing of the containment isolation system is performed in accordance with Technical Specification requirements. Subsection 3.9.6.2 describes the in-service inspection program for these valves.

6.2.5 Combustible Gas Control in Containment

Following a loss-of-coolant accident (LOCA) hydrogen gas may be generated inside the containment by reactions such as zirconium fuel cladding with reactor coolant, corrosion of metals of construction by solutions used for emergency core cooling or containment spray, and by radiolysis of aqueous solution in the core and sump. To ensure that the containment integrity is not compromised by burning or explosion of this hydrogen, a combustible gas control system (CGCS) has been provided to mix the containment atmosphere, monitor combustible gas concentrations within the containment regions, and reduce the combustible gas concentrations within the containment by recombination of the free hydrogen with the oxygen in the containment air.

6.2.5.1 Design Bases

a. The CGCS is capable of continuously monitoring the hydrogen concentration in the containment during and after a design basis LOCA. The CGCS is also capable of monitoring hydrogen concentration in containment during a beyond-design-basis accident for accident management, including emergency planning. The operator in the main control room is alerted of the need to activate systems to reduce combustible gas concentrations, when required, by an alarm from the operating hydrogen monitor.

b. The containment mixing portion of the CGCS is designed to uniformly mix the containment atmosphere for as long as is necessary during and following an accident which generates hydrogen. Mixing of the containment atmosphere prevents high concentrations of hydrogen from accumulating locally.

c. The CGCS, containment mixing subsystem, meets the redundancy and power source requirements for engineered safety features. It is designed to withstand a single active mechanical component failure and a passive electrical failure; no single failure will incapacitate the containment mixing system.

d. Components of the CGCS located within the containment are protected against damage from internally generated missiles or jet impingement in the post-LOCA environment. Moreover, such components have been subjected to qualification tests to demonstrate their capability to remain operable in the LOCA environment for as long as may be required.

e. The CGCS, including its foundations and supports, is designed to withstand the effects of an SSE without loss of function.

f. The CGCS design will permit periodic in-service inspection, operability testing and leak rate testing of the system or its components.

g. The unit is provided with its own permanently installed combustible gas control equipment.

h. In the absence of containment isolation "T" signal, the CGCS is capable of purging the containment in the event that more than a single failure of active elements of the system occurs.

i. The CGCS is classified as a seismic Category I system, and is designed, fabricated, erected and tested to Safety Class 2 quality standards, except for the containment structure recirculating filter system which is Safety Class 3. The hydrogen analyzers, which are normally isolated from the containment, are comprised of Class 1E, seismic Category I components.

j. The design of the combustible gas control system is in accordance with NRC Regulatory Guides 1.7, 1.22, 1.26 and 1.29, General Design Criteria 5, 41, 42, 43, and 50 and SECY 03-0127, "Final Rulemaking Risk-Informed 10 CFR 50.44, "Combustible Gas Control in Containment."

6.2.5.2 System Design

The CGCS consists of subsystems which monitor the combustible gas concentrations in the containment, and which possess the capability for maintaining a mixed containment atmosphere to ensure that hydrogen concentrations remain below flammable levels following a loss-of-coolant accident. The overall system is depicted on Figure 6.2-92. A portion of the CGCS, the fans installed to mix the containment atmosphere, are not shown on this figure, but are discussed in Subsection 6.2.5.2b.

a. Hydrogen Monitoring Subsystem

The containment atmosphere is monitored by two completely independent hydrogen sampling and analysis systems which are started after an accident. The hydrogen analyzer design parameters are summarized in Table 6.2-84.

The suction intakes are located at elevation 183'-6", at opposite sides of the containment dome, terminating in 90° elbows pointing downward to minimize entry of spray into the sample lines. To prevent condensation of moisture in the suction lines to the analyzers, the lines are heat-traced and maintained at a temperature of 278° or greater, after penetrating the containment wall. This ensures that the gas sample is maintained above the steam saturation temperatures postulated to occur during design basis accidents (ref. Figure 3.11-1).

The analyzers are located outside containment at the 22'-0" level of the Main Steam and Feedwater Pipe Chase Building, and take suction through a heavy-walled tube of approximately ¼" bore, with lengths varying from 150 feet to 300 feet inside the containment.

To maintain the requirement for separation of the two redundant monitoring subsystems, each analyzer is powered from a different electrical train, and has its own distinct discharge piping to return the sampled gas to the containment.

The hydrogen analyzers are normally isolated from the containment-related piping by Safety Class 2 manual valves at the analyzers.

Since the analyzers are located outside the containment, periodic inspection and testing are facilitated. The less strenuous conditions expected in the Main Steam and Feedwater Pipe Chase Building, rather than the post-accident environment inside the containment, are all the analyzers need be qualified for.

The analyzers work on the principle of thermal conductivity of hydrogen at various concentrations. Grab sampling provision is available in the vicinity of the hydrogen analyzers on a per train basis.

b. Containment Atmosphere Mixing

Mixing of the containment atmosphere to prevent localized buildup of hydrogen concentrations is provided by the Containment Spray System, described in Subsection 6.2.2. This aspect of mixing is discussed further in Subsection 6.2.5.3.

Subcompartments not exposed to the sprays have been designed so that no pocketing of hydrogen in their upper levels is possible. The top of the entrance to these subcompartments is always flush with their ceiling, and the elevation of the ceiling immediately outside of them is never lower than the entrance way. Compartments located beneath the operating floor connect with the free containment volume by their doorways and via piping penetrations through which they also communicate with each other. Hydrogen vent ports have also been included in the ceiling of such subcompartments at the end opposite from their entrances to prevent any dead-ending which could inhibit natural dispersion by convection and diffusion.

The associated Containment Structure Recirculation Filter System (CSRFS) is described in Subsection 9.4.5. Following a LOCA, the two 4000 scfm fans are started by an engineered safety feature actuation signal, as discussed in Chapter 7. The fans take suction from the apex of the dome and discharge below the operating floor. The fans, the dampers, and the suction and discharge ductwork are capable of withstanding the physical, chemical and radiological environment to which they will be subjected in the event of a LOCA. These elements of the CSRFS are redundant, separate, and built to Safety Class 2 and seismic Category I standards. The location of the ductwork and the fans is shown on Figure 6.2-93.

c. Hydrogen Recombiners

One means of combustible gas control in the containment is through the use of electric hydrogen recombiners. The codes, standards and regulatory guides employed in their design are listed in Table 6.2-85. The unit has a pair of recombiners, located at the perimeter of the operating floor inside the containment. Thus, there is no need to protect personnel from radiation in the vicinity of an operating recombiner. The separateness extends beyond the physical distinction to the independence of instrumentation, control circuits and power supply so that no single failure can impede the operation of more than one recombiner. Table 6.2-86 summarizes the recombiner design parameters.

The recombiner, Figure 6.2-94, consists of an inlet preheater section, a heater-recombiner section, and a discharge mixing chamber. The inlet preheater section is a thermally insulated vertical metal duct positioned around a central heater section to take advantage of heat losses from the heater section. The heater section consists of four vertically stacked assemblies of electric heaters, each assembly containing individual heating elements. An outer enclosure provides protection against containment spray water. The overall assembly is mounted on structural steel framing which provides a substantial foundation free of normal operating vibration.

The recombiner is fabricated of a corrosion-resistant high temperature material, except for the base which is of carbon steel. The heaters are commercial-type electric resistant heaters sheathed with Incoloy-800, an excellent corrosion-resistant material for this purpose. These heaters are operated at significantly lower power densities than in commercial service.

Air is first drawn into the preheater section by natural convection, where it is warmed. It then passes through an orifice, plate, and enters the electric heater section where it is heated to approximately 1150 to 1400°F, thus causing recombination between the oxygen and hydrogen. The efficiency of recombination is 99.9 percent, minimum, at all hydrogen concentrations between 2 and 4 percent.

Tests have verified that the recombination is not a catalytic surface effect associated with the heaters, but occurs as a result of the increased temperature of the process gases. Since the phenomenon is not a catalytic effect, saturation of the unit does not occur. Operation of the recombiner is manually controlled from a panel located in the main control room (the recombiner, power supply panel and control panel are shown schematically in Figure 6.2-95). The recombiner power panel contains an isolation transformer plus an SCR controller to regulate power into the recombiner. This equipment is mounted outside the containment and thus, is not exposed to the post-LOCA environment. To control the recombination process, the correct power input for bringing the recombiner above the threshold temperature for recombination is set on the controller. The correct power required for recombination depends upon containment atmosphere conditions, and is determined when recombiner operation is required. For equipment tests and periodic checkouts, a thermocouple readout instrument is also provided in the control panel for monitoring temperatures in the recombiner. Reference 22 further describes these recombiners and their qualification testing.

d. Backup Purge System

The capability for purging of the containment at a controlled rate is also provided. Purging is accomplished by replacing the purged gas with clean compressed air.

The purge system, Figure 6.2-92, consists of a flow meter, a throttle valve and associated piping. After the flow meter, the piping terminates adjacent to the inlet screens of the emergency cleanup filters located within the containment enclosure. From the filters, there is a direct path to the unit plant vent. The purge line separates close to the emergency filters, and each of the two lines ends in a pipe reducer bell close to the inlet screen of one of the filter units. Each of these two lines includes an isolation valve, so that the purge gas flow can be directed to whichever filter unit is operating. The purge system is aligned and the purge flow is initiated manually. The Containment Enclosure Emergency Cleanup System is discussed in Subsection 6.2.3; the filter system's design requirements are detailed in Subsection 6.5.1.

Components and ductwork associated with the filters are classified as Safety Class 3 and seismic Category I. All piping inside the containment and the containment penetration connections associated with the purge system are duplicated to provide independent and redundant capability and to prevent a single failure from stopping the containment purge/flow vent. These are Safety Class 2. External to the containment, a single line runs from the penetration area to the containment enclosure emergency filters.

6.2.5.3 Design Evaluation

A beyond design bases accident could generate hydrogen. Turbulence created by the rupture of a coolant line serves to thoroughly mix the atmosphere of the containment in the early phases of the post-LOCA period. The release to the containment of the hydrogen produced by radiolysis of cooling water in the core would be through the break, and would be associated with the local mixing caused by the break flow. The hydrogen generated by corrosion of metals by spray water would, in turn, be mixed with the free containment volume by that spray. In addition, the aluminum and zinc in the containment are widely distributed. The inventories of these corrodible metals are shown in Table 6.2-87 and Table 6.2-88, respectively.

In the long term, after the temperature in the containment has been reduced by the containment heat removal system, and the rate of hydrogen production by corrosion is diminished, the principal source of combustible gas buildup is the radiolysis of the water injected into the reactor vessel and sprayed into the containment. Hydrogen originating from core radiolysis is deemed to be distributed in the region of the break by the dispersion caused by the mass flow from it. The radiolysis of sump water by fission products released from the core produces hydrogen at a lower rate than radiolysis in the core a day or two after the accident. Hydrogen generated by sump water radiolysis is distributed by two means, the second of which serves to insure that combustible gas concentrations, in general, are homogeneous within the containment, including recombiner or sampling locations:

a. The minimum post-LOCA spray rate is 3000 gpm. With a maximum possible sump water volume of 475,000 gallons the entire amount of sump water is sprayed through the containment atmosphere approximately every three hours. The fission products are uniformly distributed in the sump water, being mixed by turbulence caused by injection, break flow and spraying, but the means of release to the containment atmosphere must be across the water-gas interface. Hydrogen atoms, not bubbles, are formed by radiolysis. They diffuse through the sump water forming molecules, which themselves diffuse and escape to the atmosphere at the surface once the relatively low solubility of hydrogen in water is passed, saturating the sump water. Ultimately, the release rate of hydrogen to the containment free space equals the production rate from radiolysis of sump water. Using the data concerning spray droplets, fall height, etc., detailed in Subsection 6.5.2, it is calculated that the surface-to-mass ratio for the water in the sprays exceeds that of the water in the sump by a factor of more than 5000. Thus, virtually all of the hydrogen generated by radiolysis caused by fission products outside the core is released by the water while it is being sprayed. The spray pattern is designed so that, even if only one train of sprays is available, the volume of the containment above the operating floor is uniformly exposed to the sprays.

b. To insure that the atmosphere of the containment is mixed, so that hydrogen concentrations are virtually identical, the fans in the containment structure recirculation filter system discussed above recirculate the gases in the containment more rapidly than twice a day, even with only one fan available. The discharge of the fan flow below the operating floor promotes a flow from those regions upward to the suction of the fans at the apex of the dome, with the containment atmosphere passing through the sprayed region. Thus, no stratification of hydrogen is possible. The system meets the requirements of Regulatory Guide 1.7.

The pertinent data for evaluation of the control aspect of the combustible gas control system is summarized in Table 6.2-89. Although Subsection 15.6.5.3c.1 indicates that only < 1.0 percent of the zirconium would react, for conservatism, 5 percent is assumed to participate in a reaction with the coolant to produce hydrogen coincident with the accident. To prevent the concentration of hydrogen from reaching 4 volume percent in the containment, the recombiners would be turned on at any time before, or when, the concentration reached 3.5 volume percent. If only one recombiner is turned on 278 hours after a LOCA when the 3.5 volume percent setpoint is reached, the hydrogen concentration will begin to drop immediately. The combustible gas concentration in the containment as a function of time is shown on Figure 6.2-96, both with and without the operation of a single recombiner, at an efficiency of 99.9 percent. The backup purge system, as described in Subsection 6.2.5.2d, could be operated at rates up to 1000 scfm. This far exceeds that of a single recombiner which processes only 100 scfm. Although actuation of the recombiners could be delayed, the 3.5 volume percent is selected to have margin for detection of the failure of both recombiners to function. In this event, the backup purge system would be started at the 3.8 volume percent level. This would leave approximately 41 hours to start purging. Figure 6.2-97 shows the containment hydrogen concentration with neither recombiner nor purge, and with purging at a rate of only 3 percent of the containment volume per day starting at the 3.8 volume percent mark, 319 hours after the LOCA. The actual purge rate used will be based on an analysis of the containment atmosphere following a LOCA. Projected offsite doses resulting from containment purging, if required, are described in Subsection 15.6.5.4.

6.2.5.4 Testing and Inspection

a. Hydrogen Analyzer

The hydrogen analyzer is shop-tested using a gas mixture closely simulating the containment post-LOCA atmosphere expected at the time the units would be placed into service, with the temperature, pressure, humidity and hydrogen concentration conditions approximated.

During preoperational testing, the hydrogen analyzers are calibrated and checked for proper operation. System integrity will be maintained by performing periodic surveillance testing as described in Subsection 6.2.6. Periodic calibration tests will be performed in accordance with Technical Requirements.

b. Hydrogen Recombiner

The electric hydrogen recombiners have undergone extensive testing in the Westinghouse development program. These tests encompassed the initial analytical studies, laboratory proof-of-principal tests, and full-scale prototype testing. The full-scale prototype tests included the effects of:

1. Varying hydrogen concentrations
2. Alkaline spray atmosphere
3. Steam
4. Convection currents
5. Seismic activity

A detailed discussion of these tests is given in Reference 22.

During preoperational testing, the functional operability of the recombiner and its control will be demonstrated. Periodic testing of the recombiners will be performed in accordance with plant procedures.

c. Backup Purge System

The containment purge function of the CGCS will be demonstrated during preoperational testing. Initial and periodic tests of the containment enclosure emergency exhaust filters is described in Subsection 6.5.1.

d. Containment Structure Recirculation Filter System

The operability of containment atmospheric mixing fans and dampers will be verified during preoperational testing. Periodic tests of the fans and dampers will be conducted in accordance with Technical Specification requirements.

6.2.5.5 Instrumentation Requirements

With the occurrence of a LOCA, the ESF actuation signal will start the Containment Recirculation Fan System. Two independent hydrogen analyzers which monitor containment hydrogen concentrations after an accident, are located outside the containment. Off-normal conditions, such as low temperature, low sample flow and pressure, and cell failure are alarmed at the control room on a system level as a common alarm, and individually indicated at the local panel. The output of either channel of the hydrogen analyzers is available to the operator, both locally and at the MCB. When either channel indicates a hydrogen concentration at or prior to 0.5 volume percent below the limiting hydrogen concentration, this fact is alarmed at the main control board (MCB). The operator would then start the recombiners, if he had not done so already.

The recombiners are thermal electric types. Temperature sensors located at the recombiner input to the MCB temperature indication to maintain the recombiner temperature at an optimum value for efficient recombination of H2 and O2 into water. The power input to the recombiner is manually set from the MCB. Recombiner temperature and power input are indicated at the MCB. The temperature indication is used for equipment tests and periodic checkouts and is not required for the safety-related function of the hydrogen recombiners.

In the event that neither recombiner starts, the hydrogen concentration in the containment would continue to rise. If it has risen significantly above the first alarm point but is still below the limiting hydrogen concentration, this fact would be alarmed at the MCB as a signal to the operator to initiate the purging of the containment.

The Containment Purge System is normally closed, and is isolated from the containment by four valves: CGC-V14 and CGC-V28 inside the containment are remotely controlled, motor-operated valves that close on a "T" signal; valves CGC-V15 and CGC-V36 located outside the containment are manually operated and normally locked closed. Initiating purge flow is a manually controlled operation, with a combination of both local and remote control from the MCB. Valve status for the remotely operated valves CGC-V14 and CGC-V28 is provided near the associated control switches. Pressure and flow instruments are provided in the purge line. Purge flow indication is available at the MCB. Air for the purge system is supplied from the Compressed Air System.

6.2.6 Containment Leak Rate Testing

The reactor containment structure, the containment penetrations and the containment isolation barriers are designed to permit periodic Type A integrated leakage rate testing. The reactor containment and its leakage limiting barriers are also designed to permit periodic inspection of important areas such as penetrations. Penetrations with resilient seals or expansion bellows are designed to permit periodic leakage testing at pressures up to the containment design pressure. Piping systems penetrating the reactor containment are provided with the capability of redundant isolation, as dictated by their importance to safety functions. These systems are designed so that their isolation capabilities can be periodically tested for operability and leakage to ensure compliance with the established leakage rate limits. The foregoing are intended to be in full compliance with General Design Criteria 52, 53 and 54 of 10 CFR 50, Appendix A.

All portions of the above systems (RHR, CBS, SI and CS) are located within the containment enclosure boundary except piping associated with the injection phase of ECCS and a minor amount of charging pump piping used during the recirculation mode. Any leakage from these systems following a LOCA is therefore filtered by the containment enclosure emergency exhaust filters prior to release to the environment.

The piping that lies outside the containment enclosure boundaries includes the pump suction lines from the refueling water storage tank to the RHR, SI and CBS pumps (lines 1201-1-151-14" and 1202-1-151-14", see Figure 6.2-74) and the centrifugal charging pumps (lines 1205-1-151-8" and 1206-1-151-8", including valves LCV-112D and LCV-112E). These lines are used only during the injection phase of post-accident operation. They are isolated within the containment enclosure, will not be contaminated during recirculation, and will not present a release path. A portion of line CS-374-1-2501-4" is run outside the containment enclosure. This line is used during the recirculation phase of post-accident operation; however, there are no valves or equipment in this pipe segment. The leakage potential for this line is therefore small.

The RHR pumps are periodically tested via a recirculation path. This will function as a leak test because operating in the recirculation mode exposes the discharge pipe and pump to a pressure higher than that experienced during a LOCA. The RHR pumps are tested at least once every 18 months according to the Technical Specifications.

CBS equipment is periodically tested via a recirculation path. This will function as a leak test, as pressure during the recirculation mode exposes the discharge pipe and pump to a pressure higher than that experienced in a LOCA. The containment spray pumps are tested at least once every 18 months per Technical Specification 4.6.2.1.

The safety injection pumps are tested via a recirculation miniflow test line to the refueling water storage tank. The pressure in the pump and pump discharge line is greater than that seen in a LOCA. This test is performed at least every 18 months per Technical Specification 4.5.2.

Portions of the RHR, CBS, and SI systems are periodically inspected for leakage under the Leakage Reduction Program in accordance with Technical Specifications 6.7.6.a.

The leakage from ECCS and containment spray system equipment has been evaluated in Subsection 15.6.5.4d and the impact of this leakage is factored into the offsite dose calculations.

The Containment Leakage Rate Testing (CLRT) Program is described in the Leakage Test Reference (SLTR). Periodic Type A, B, and C leakage tests will be performed and reported in compliance with the intent of 10 CFR 50, Appendix J. Deviations from the wording of Appendix J are included in the Technical Specifications.

The testing program included a complete series of Type B and Type C tests as presented in Subsections 6.2.6.2 and 6.2.6.3, prior to fuel loading. During plant operation, each penetration requiring testing will be tested periodically to ensure continued compliance with leakage limits.

6.2.6.1 Containment Integrated Leakage Rate Test - Type A Test

The initial containment integrated leakage rate test (Type A test) was performed after completion of construction of the containment structure and prior to initial fuel loading. Periodic integrated leakage rate tests shall be conducted in accordance with Subsection 6.2.6.4 and Technical Specifications 4.6.1.2.

The maximum allowable integrated leakage rate, La, at the calculated peak accident pressure, Pa, is 0.15 weight percent per 24 hours. The calculated peak accident pressure, Pa, is 49.6 psig.

Prior to conducting the initial Type A test all penetrations were installed and all systems penetrating the containment were complete, up to and including all automatic isolation valves external to the containment. Deviations from this schedule were documented and properly considered when reporting final leakage rate test results.

The structural integrity test (SIT) preceded the initial Type A test. A minimum of 24 hours elapsed from the time the containment was in excess of 85 percent Pa for the SIT and the commencement of the Type A test, to assure sufficient time for outgassing from the internal structure.

The structural integrity of the containment vessel and of the containment enclosure building shall be determined in accordance with the Containment Leakage Rate Testing Program. Any abnormal degradation detected during these inspections will be reported as part of a special report as required by Technical Specifications.

Systems will be aligned for the Type A test based on the requirements of the Containment Leakage Rate Testing Program. Table 6.2-90 lists systems typically vented prior to and during the conduct of the Type A test. Table 6.2-91 lists those systems typically not vented and drained, and the justification thereof.

The required tests including applicable pre-test requirements, data analysis methods, test acceptance criteria, test schedule requirements, and reporting requirements are discussed in the Containment Leakage Rate Testing Program.

6.2.6.2 Containment Penetration Leakage Rate Test - Type B Test

Type B tests are required on all containment penetrations with resilient seals, gaskets, or expansion bellows. These include, but are not limited to, air locks, air lock door seals, piping penetrations with expansion bellows and blind flanges, and electrical seals. Those penetrations which are seal-welded are exempt from this testing requirement. Table 6.2-92 lists all containment penetrations falling into this category.

All penetrations requiring Type B testing will be tested in accordance with the Containment Leakage Rate Testing Program. This is in accordance with Appendix J of 10 CFR 50.

6.2.6.3 Containment Isolation Valve Leakage Rate Test - Type C Test

Type C tests are required on all lines that penetrate the primary containment and present a potential leakage path between the inside and outside atmospheres of the primary containment under postulated accident conditions. These include lines: (1) that provide a direct connection between containment atmosphere and the outside, like purge and vent lines; (2) whose isolation valves are required to close automatically upon receipt of a containment isolation signal to isolate containment atmosphere or the Reactor Coolant System; or, (3) whose isolation valves are required to operate intermittently under post-accident conditions to isolate containment atmosphere or the Reactor Coolant System. Table 6.2-83 lists all lines penetrating the containment and, where applicable, the containment isolation valves associated with those lines. Those lines not considered as requiring testing are noted. Containment isolation valves which are not Type C tested, and the reasons thereof, can be categorized as follows:

a. Valves that isolate lines which form a closed system inside containment satisfying the criteria of Updated FSAR Subsection 6.2.4.1c are not Type C tested. These systems, and therefore their respective containment isolation valves, will not communicate with containment atmosphere or the reactor coolant system under post-accident conditions. These systems include main steam, feedwater, and steam generator blowdown, and component cooling water supply and return for the thermal barrier heat exchangers.

b. Certain ECCS containment isolation valves are not Type C tested. The primary function of many of these valves is not to isolate containment following an accident, but rather to direct emergency core cooling water as desired. In fact, most of the valves will be open during one or more of the three ECCS post-accident modes. In addition, the valves are part of Safety Class 2/seismic Category I systems that are closed outside containment and liquid-filled, with an assured post-accident 30-day water supply. A water seal at a pressure greater than 1.10 Pa will be maintained at the containment penetrations associated with these isolation valves for the 30-day post-accident period. This seal precludes leakage of containment atmosphere.

c. The containment isolation valves on the CGC hydrogen analyzer lines are not required to be Type C tested. These lines form a closed, seismic Category I system outside containment. The integrity or the closed system will be maintained by performing periodic surveillance testing as described in Subsections 6.2.4.2.m.3 and 6.2.5.1.g.

Type C tests will be performed in accordance with the Containment Leakage Rate Testing Program. For valves tested in this manner, a radiological assessment will be made to establish the leakage limits. This form of testing meets the intent of 10 CFR 50, Appendix J (III.C.2), and no exemption is noted.

As discussed in 6.2.3.2.d, a containment enclosure emergency cleanup system has been provided to minimize leakage to the environs. A significant number of lines penetrate the containment and terminate in areas not treated by this cleanup system. All leakage attributed to penetrations and isolation valves, requiring Type B and Type C leakage rate tests per 10 CFR 50, Appendix J, is conservatively assumed to bypass the cleanup system.

6.2.6.4 Scheduling and Reporting of Periodic Tests

The Type A test schedule and reporting requirements will be in accordance with the Containment Leakage Rate Testing Program.

6.2.6.5 Special Testing Requirements

This section addresses the special requirements associated with the secondary containment surrounding the primary containment. The maximum allowable leakage rate and in-leakage limits are discussed in Subsection 6.2.3.3(a) and Technical Specification 3/4.6.5.2.

6.2.7 References

1. "CONTRAST-S MOD1 - A Digital Computer Program to Predict Containment Pressure-Temperature Response," UEC-TR-006-SUP, June 1979, A Supplement to UE&C Topical Report UEC-TR-006-0, March 1976.
2. Roshenow, W.M. and Hartnett, J.P., "Handbook of Heat Transfer," McGraw-Hill, New York (1973)
3. Brazlay, M.E., "Range of Interface Thermal Conductance for Aircraft Joints," NASA TND-426, May 1960.
4. Tagami, T., et al., "Studies for Safety Analysis of Loss-of-Coolant Accidents in Light Water Power Reactors," NSJ-TR-223, March 1968, Japan Atomic Energy Research Institute, Japan.
5. Uchida, H., et al., "Evaluation of Post-Accident Cooling Systems of Light Water Power Reactors," Third International Conference on the Peaceful Uses of Atomic Energy, New York, 1965.
6. "Preliminary Safety Analysis Report," Public Service of New Hampshire, Seabrook Station Units 1 & 2, Docket-Nos. 50-443 and 50-444.
7. Clausing, A.M., "Practical Techniques for Estimating the Accuracy of Finite-Difference Solutions to Parabolic Equations," J. Appl. Mech., March 1973.
8. Shepard, R. M., et al., "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8312-A, Rev. 2 and WCAP-8264-P-A (Proprietary) February 1974.

9. Niyogi, K.K., Bhatnager S.K. and Rathi, S.S., "COMPRESS - A Code for Calculating Subcompartment Pressure Response," UEC-TR-004-1, July 1976.

10. Idel' Chik, I.E., "Handbook of Hydraulic Resistance, Coefficients of Local Resistance and of Friction," Israel Program for Scientific Translations, Jerusalem, 1966.

11. NS-TMA-2075 letter from T. M. Anderson, Westinghouse to J. F. Stolz, NRC, April 25, 1979, Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version.

12. Westinghouse ECCS Evaluation Model, February 1978 Version, WCAP-9220, February 1978.

13. F. M. Bordelon, et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-6174, June 1974.

14. G. Collier, et al., "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Nonproprietary), June 1974.

15. Land, R.E., "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822, September 1976.

16. Deleted.

17. Kirk-Othmer, Encyclopedia of Chemical Technology, 2nd Edition Volume 3, Page 6.9, Figure 4.

18. Transactions of the Electrochemical Society, 1935, Volume 47, Page 88-100.

19. Ranz, W.E. and Marshall, W. R. Jr., Chemical Engineering Progress, 48, 4p. 173 (1952).

20. Parsly, L. F., Design Considerations of Reactor Containment Spray Systems - Part VI, ORNL-TM-2412.

21. Flow of Fluids through Valves, Fittings, and Pipe; Crane Technical Paper No. 410, 13th Printing - 1973.

22. J. F. Wilson, "Electric Hydrogen Recombiner for PWR Containments," Westinghouse WCAP-7709-L (Proprietary) and WCAP-7820 (Non-Proprietary), Supplements 1, 2, 3, 4 and 5.

23. Niyogi, K. K. and Lin, S. D., "An Analysis of Two Stream Gravity Feed Mixing Process," ASME 77-DE-44, February 14, 1977.

24. Deleted.

25. Deleted.

26. Deleted.

27. SECY 03-0127, "Final Rulemaking Risk-Informed 10 CFR 50.44, "Combustible Gas Control in Containment," dated July 24, 2003.

28. Calculation No. 4.3.16-13F, Revision 3, "Free Hydrogen in Containment."

29. NAI 8907-2, Revision 13, "GOTHIC Containment Analysis Package User Manual," Version 7.0, July 2001.

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 Design Bases

The Emergency Core Cooling System (ECCS) consists of the centrifugal charging pumps, safety injection pumps, a refueling water storage tank, the residual heat removal pumps, the residual heat removal heat exchanger, the safety injection accumulators, and the associated valves and piping.

Plants listed in Subsection 1.3.1 have similar Emergency Core Cooling Systems to that of Seabrook.

The primary function of the ECCS following an accident is to remove the stored and fission product decay heat from the reactor core so that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented.

The ECCS is designed to cool the reactor core as well as to provide additional shutdown capability following initiation of the following accident conditions:

a. Loss-of-Coolant Accident (LOCA) including a pipe break or a spurious relief or safety valve opening in the Reactor Coolant System (RCS) which would result in a discharge larger than that which could be made up by normal makeup system.

b. Rupture of a control rod drive mechanism causing a Rod Cluster Control Assembly (RCCA) ejection accident.

c. Steam or feedwater system break accident including a pipe break or a spurious relief or safety valve opening in the secondary steam system which would result in an uncontrolled steam release or a loss of feedwater.

d. A steam generator tube rupture.

The acceptance criteria for the consequences of each of these accidents is described in Chapter 15 in the respective accident analysis sections.

The bases used in design and for selection of ECCS functional requirements are derived from Appendix K Limits for fuel cladding temperature, etc., following any of the above accidents as delineated in 10 CFR 50.46. The subsystem functional parameters are selected so that, when integrated, the Appendix K requirements are met over the range of anticipated accidents and single failure assumptions.

NOTE: All drawings referenced that are not included were provided separately to the NRC as required in Updated FSAR Section 1.7.

The reliability of the ECCS has been considered in selection of the functional requirements, selection of the particular components and location of components and connected piping. Redundant components are provided where the loss of one component would impair reliability. Valves are provided in series where isolation is desired, and in parallel when flow paths are to be established for ECCS performance. Redundant sources of the safety injection actuation signal are available so that the proper and timely operation of the ECCS will be ensured. Sufficient instrumentation is available so that a failure of an instrument will not impair readiness of the system. The active components of the ECCS are powered from separate buses which are energized from offsite power supplies. In addition, redundant sources of auxiliary onsite power are available through the use of the emergency diesel generators to ensure adequate power for all ECCS requirements. Each generator is capable of driving all pumps, valves, and necessary instruments associated with one train of the ECCS.

All valves required to be actuated during ECCS operation are located to prevent vulnerability to flooding. Repositioning of valves due to spurious actuation coincident with a LOCA has been analyzed and is not considered credible for a design basis.

The environmental qualification of active ECCS equipment is discussed in Section 3.11.

Protection of the ECCS from missiles is discussed in Section 3.5. Protection of the ECCS against dynamic effects associated with ruptures of piping is described in Section 3.6. Protection from flooding is also discussed in Section 3.4.

The elevated temperature of the sump solution during recirculation is well within the design temperature of all ECCS components. In addition, consideration has been given to the potential for corrosion of various types of metals exposed to the fluid condition prevalent immediately after the accident or during long-term recirculation operations.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Flow diagrams of the ECCS are shown in Figure 6.3-1, Figure 6.3-2, Figure 6.3-3, Figure 6.3-4, Figure 6.3-5 and Figure 6.3-6.

a. System Operation

Upon the initiation of a safety injection "S" signal, the following automatic actions are initiated to commence the injection phase of emergency core cooling:

1. Centrifugal charging pumps start (see Dwg. NHY-503335).
2. Refueling water storage tank suction valves to charging pumps open (see Dwg. NHY-503335).
3. Normal charging path valves close (see Dwg. NHY-503337).
4. Charging pump miniflow valves close (see Dwg. NHY-503380 & NHY 503398).
5. Safety injection pumps start (see Dwg. NHY-503900).
6. Residual heat removal pumps start (see Dwg. NHY-503761).
7. Any closed accumulator isolation valves open. These valves will open only if power is available to the normally de-energized motor control centers E 522 and E 622 (see Dwg. NHY-503907).
8. Volume control tank outlet isolation valves close. Valves are interlocked with the RWST suction valves to the charging pumps (see Dwg. NHY-503341).
9. High head safety injection valves open (see Dwg. 1-NHY-503903).

During the injection phase, the two centrifugal charging pumps (CCPs) operate to inject into the cold legs of all four loops. The source water to the CCPs is the refueling water storage tank (RWST).

Once the Reactor Coolant System (RCS) pressure is below shutoff head of the two safety injection pumps (SIPs), they begin to take borated water from the RWST and deliver it to the cold legs of the four loops. This is done through the residual heat removal (RHR) injection/accumulator discharge lines. In the case of a steam line break or small RCS break, the system pressure remains high for a long period of time, and the CCPs and SIPs supply core cooling.

When the RCS pressure drops below the pressure of the four safety injection accumulator tanks, they discharge their contents in the four RCS cold legs. These accumulators contain borated water and are pressurized with nitrogen. This portion of the ECCS is most effective in the case of large RCS breaks where system pressure drops rapidly to the accumulator pressure.

The two residual heat removal pumps (RHRPs) take water from the RWST and inject it into the cold legs of all four RCS loops via the accumulator discharge lines once system pressure drops below the shutoff head of the pumps.

Therefore, upon the initiation of the safety injection "S" signal, borated water is injected into the RCS via the CCPs, SIPs, accumulator tanks and RHRPs. The point at which these various injection modes commence operating is controlled by the rate at which the reactor coolant is lost and system pressure drops.

The RWST supplies the borated water used for the injection phase of the ECCS. When the RWST water level drops to the low-low-1 level alarm point, the injection phase is discontinued and the cold leg recirculation phase is initiated.

The changeover from the injection mode to recirculation mode is initiated automatically and completed manually by operator action from the main control room. Protection logic is provided to automatically open the two containment recirculating sump isolation valves when two out of four refueling water storage tank level channels indicate a refueling water storage tank level less than a low-low-1 level setpoint in conjunction with the initiation of the engineered safeguards actuation signal ("S" signal). This automatic action would align the two residual heat removal pumps to take suction from the containment sump and to deliver directly to the RCS. It should be noted that the residual heat removal pumps would continue to operate during this changeover from injection mode to recirculation mode.

The two charging pumps and the two safety injection pumps would continue to take suction from the refueling water storage tank following the above automatic action, until manual operator action is taken to align these pumps in series with the residual heat removal pumps.

Valves SI-V138 and SI-V139, in the injection path, are normally closed (but not de-energized). A safety injection ("S") signal then opens the valves in the injection line.

The refueling water storage tank level protection logic consists of four level channels, with each level channel assigned to a separate process control protection set. Four refueling water storage tank level transmitters provide level signals to corresponding normally de-energized level channel bistables. Each level channel bistable would be energized on receipt of a refueling water storage tank level signal less than the low-low-1 level setpoint.

The two-out-of-four coincident logic is utilized in both protection cabinets A and B to ensure a trip signal in the event that two out of the four level channel bistables are energized. This trip signal, in conjunction with the "S" signal, provides the actuation signal to automatically open the corresponding containment sump isolation valves.

The low-low-1 refueling water storage tank level signal is also alarmed to inform the operator to initiate the manual action required to realign the charging and safety injection pumps for the recirculation mode. The manual switchover sequence that must be performed by the operator is delineated in Table 6.3-7. Following the automatic and manual switchover sequence, the two residual heat removal pumps would take suction from the containment sump and deliver borated water directly to two RCS cold legs. A portion of the Number 1 residual heat removal pump discharge flow would be used to provide suction to the two charging pumps which would also deliver directly to the RCS cold legs. A portion of the discharge flow from the Number 2 residual heat removal pump would be directed to the RCS cold legs. As part of the manual switchover procedure (see Table 6.3-7, Step 5), the suctions of the safety injection and charging pumps are cross connected so that one residual heat removal pump can deliver flow to the RCS and both safety injection and charging pumps, in the event of the failure of the second residual heat removal pump.

After approximately 5 to 6 hours, cold leg recirculation is terminated and hot leg recirculation is initiated. This is done to terminate any boiling in the core should the break be in one of the RCS cold legs. During this phase of recirculation, the SIPs discharge is aligned to supply water to all four RCS hot legs and the RHRPs discharge is aligned to supply water to RCS hot legs 1 and 4. The CCPs do not have the capability to feed the hot legs and continue to supply the cold legs.

6.3.2.2 Equipment and Component Descriptions

The component design and operating conditions listed in Table 6.3-1 are specified as the most severe conditions to which each respective component is exposed during either normal plant operation, or during operation of the ECCS. For each component, these conditions are considered in relation to the code to which it is designed. By designing the components in accordance with applicable codes, and with due consideration for the design and operating conditions, the fundamental assurance of structural integrity of the ECCS components is maintained. Components of the ECCS are designed to withstand the appropriate seismic loadings in accordance with their safety class as given in Table 3.2-2.

Descriptions of the major mechanical components of the ECCS follow:

a. Accumulators

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. During normal operation, each accumulator is isolated from the RCS by two check valves in series. Should the RCS pressure fall below the accumulator pressure, the check valves open and borated water is forced into the RCS. One accumulator is attached to each of the cold legs of the RCS. Mechanical operation of the swing disc check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

Connections are provided for remotely adjusting the level and boron concentration of the borated water in each accumulator during normal plant operation as required. Accumulator water level may be adjusted either by draining to the primary drain tank or by pumping borated water from the refueling water storage tank to the accumulator. Alternatively, the boron concentration may be adjusted by a feed and bleed process. This would involve pumping borated water into the accumulator from the refueling water storage tank, as described above, with simultaneous draining of the accumulator through the sample connection to the Containment Building drainage sump. Technical Specification actions related to out-of-service equipment during this evolution would be applied. Samples of the solution in the accumulators are taken periodically for checks of boron concentration.

Accumulator pressure is provided by a supply of nitrogen gas, and can be adjusted as required during normal plant operation; however, the accumulators are normally isolated from this nitrogen supply. Gas relief valves on the accumulators protect them from pressures in excess of design pressure.

The accumulators are located within the containment but outside of the secondary shield wall which protects them from missiles.

Accumulator gas pressure is monitored by indicators and alarms. The operator can take action as required to maintain plant operation within the requirements of the Technical Specification covering accumulator operability.

b. Residual Heat Removal Pumps

In the event of a LOCA, the residual heat removal pumps are started automatically on receipt of an "S" signal (see Dwg. NHY-503761). The residual heat removal pumps take suction from the refueling water storage tank during the injection phase and from the containment sump during the recirculation phase. Each residual heat removal pump is a single-stage vertical-position centrifugal pump.

A minimum flow bypass line is provided for the pumps to recirculate and return the pump discharge fluid to the pump suction should these pumps be started with the RCS pressure above their shutoff head. Once flow is established to the RCS, the bypass line is automatically closed (see Dwg. NHY-503764). This line prevents deadheading of the pumps and permits pump testing during normal operation.

The safety intent of Regulatory Guide 1.1 is met by the design of the ECCS so that adequate net positive suction head is provided to system pumps. In addition to considering the static head and suction line pressure drop, the calculation of available net positive suction head in the recirculation mode assumes that the vapor pressure of the liquid in the sump is equal to the containment ambient pressure. This assures that the actual available net positive suction head is always greater than the calculated net positive suction head.

The residual heat removal pumps are discussed further in Subsection 5.4.7. A pump performance curve is given in Figure 6.3-7. Available and required net positive suction head are shown in Table 6.3-1.

c. Centrifugal Charging Pumps

In the event of an accident, the charging pumps are started automatically on receipt of an "S" signal (see Dwg. NHY-503330), and are automatically aligned to take suction from the refueling water storage tank during injection (see Dwg. NHY-503335). During recirculation, suction is provided from the residual heat removal pump discharge.

These high head pumps deliver flow to the RCS at the prevailing RCS pressure. Each centrifugal charging pump is a multi-stage diffuser design, barrel-type casing with vertical suction and discharge nozzles.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the pump suction after cooling via the seal water heat exchanger during normal plant operation. Each minimum flow bypass line contains a motor-operated valve which closes on receipt of the safety injection signal. The miniflow isolation valve for each charging pump will re-open if the flow through its respective pump drops below the low flow setpoint. The valve will reclose in the presence of an "S" signal when the flow increases beyond the high flow setpoint (see Dwg. NHY-503398 & 503380). This signal also closes the valves to isolate the normal charging line and volume control tank and opens the charging pump/refueling water storage tank suction valves to align the high head portion of the ECCS for injection (see Dwg. NHY-503335). The charging pumps may be tested during power operation via the minimum flow bypass line.

A pump performance curve for the centrifugal charging pumps is presented in Figure 6.3-8. Available and required net positive suction head are shown in Table 6.3-1.

d. Safety Injection Pumps

In the event of an accident, the safety injection pumps are started automatically on receipt of an "S" signal (see Dwg. NHY-503900).

The pumps deliver water to the RCS from the refueling water storage tank during the injection phase, and from the containment sump via the residual heat removal pumps during the recirculation phase.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event that the pumps are started with the RCS pressure above pump shutoff head. This line also permits pump testing during normal plant operation. Two parallel valves in series with a third, downstream of a common header, are provided in this line. These valves are manually closed from the control room as part of the ECCS realignment from the injection to the recirculation mode.

The common recirculation header for the safety injection pumps is seismically analyzed and is seismically supported.

A pump performance curve is presented in Figure 6.3-9. Available and required net positive suction head are shown in Table 6.3-1.

e. Residual Heat Exchangers

The residual heat exchangers are conventional shell and U-tube type units. During normal cooldown operation, the residual heat removal pumps recirculate reactor coolant through the tube side while component cooling water flows through the shell side. During emergency core cooling recirculation operation, water from the containment sump flows through the tube side. The tubes are seal welded to the tube sheet.

A further discussion of the residual heat exchangers is found in Subsection 5.4.7.

f. Valves

Design features employed to minimize valve leakage include:

- Where possible, packless valves are used.
- Other valves which are normally open, except check valves and those which perform a control function, are provided with backseats to limit stem leakage.
- Normally closed globe valves are installed with recirculation fluid pressure under the seat to prevent stem leakage of recirculated (radioactive) water.
- Relief valves are enclosed, i.e., they are provided with a closed bonnet.

1. Motor-Operated Valves

The seating design of all motor-operated valves is of the Crane flexible wedge design. This design releases the mechanical holding force during the first increment of travel so that the motor operator works only against the frictional component of the hydraulic unbalance on the disc and the packing box friction. The disc is guided throughout the full disc travel to prevent chattering and to provide ease of gate movement. the seating surfaces are hard faced to prevent galling and to reduce wear.

Where a gasket is employed for the body to bonnet joint, it is either a fully trapped, controlled compression, spiral-wound asbestos gasket with provisions for seal welding, or it is of the pressure seal design with provisions for seal welding. Many of the valves stuffing boxes were originally designed with a lantern ring leakoff connection with packing configurations which minimize stem seal leakage to the full extent possible by the design. Exceptions to this criterion are gate valves that have been determined to be susceptible to pressure locking, which have been modified to utilize the valve stem leakoff connection as a vent path for the bonnet cavity. These valves use only a single packing set. Based on industry recommendations, many of the double packed stuffing boxes have been modified to a single packing configuration. The motor operator incorporates a "hammer blow" feature that allows the motor to impact the discs away from the backseat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed prior to impact. Valves which must function against system pressure are designed so that they function with a pressure drop equal to full system pressure across the valve disc.

2. Manual Globe, Gate and Check Valves

Gate valves employ a wedge design and are straight through. The wedge is either split or solid. All gate valves have backseat and outside screw and yoke.

Globe valves, "T" and "Y" style, are full-ported with outside screw and yoke construction.

Check valves are spring-loaded lift piston types for sizes 2 inches and smaller, swing type for sizes 2½ inches to 4 inches and tilting-disc type for sizes 4 inches and larger. Stainless steel check valves have no penetration welds other than the inlet, outlet and bonnet. The check hinge is serviced through the bonnet.

The stem packing and gasket of the stainless steel manual globe and gate valves are similar to those described above for motor-operated valves. Carbon steel manual valves are employed to pass nonradioactive fluids only and therefore do not contain the double packing and seal weld provisions.

3. Accumulator Check Valves (Swing-Disc)

The accumulator check valve is designed with a low pressure drop configuration with all operating parts contained within the body.

Design considerations and analyses which ensure that leakage across the check valves located in each accumulator injection line will not impair accumulator availability are as follows:

(a) During normal operation, the check valves are in the closed position with a nominal differential pressure across the disc of approximately 1650 psi. Since the valves remain in this position except for testing or when called upon to open following an accident, and are therefore not subject to the abuse of flowing operation or impact loads caused by sudden flow reversal and seating, they do not experience significant wear of the moving parts, and are expected to function with minimal back leakage. The back leakage can be checked via the test connection as described in Subsection 6.3.4.

(b) When the RCS is being pressurized during the normal plant heatup operation, the check valves are tested for leakage as soon as there is a stable differential pressure of about 100 psi or more across the valve. This test confirms the seating of the disc and whether or not there has been an increase in the leakage since the last test. When this test is completed, the accumulator discharge line motor-operated isolation valves are opened and the RCS pressure increase is continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

(c) The experience derived from the check valves employed in the emergency injection systems indicates that the system is reliable and workable; check valve leakage has not been a problem. This is substantiated by the satisfactory experience obtained from operation of the Robert Emmett Ginna and subsequent plants where the usage of check valves is identical to Seabrook.

(d) The accumulators can accept some in-leakage from the RCS without affecting availability. Continuous in-leakage would require, however, that the accumulator volume be adjusted periodically to Technical Specification requirements.

4. Relief Valves

Relief valves are installed in various sections of the ECCS to protect lines which have a lower design pressure than the RCS. The valve stem and spring adjustment assembly are isolated from the system fluids by a bellows seal between the valve disc and spindle. The closed bonnet provides an additional barrier for enclosure of the relief valves. Table 6.3-2 lists the system's relief valves with their capacities and setpoints.

5. Butterfly Valves

Each main residual heat removal line has an air-operated butterfly valve (RH-HCV-606 and 607) which is normally open and is designed to fail in the open position. The actuator is arranged so that air pressure on the diaphragm overcomes the spring force, causing the linkage to move the butterfly to the closed position. Upon loss of air pressure, the spring returns the butterfly to the open position. These valves are left in the full-open position during normal operation to maximize flow from this system to the RCS during the injection mode of the ECCS operation. These valves are used during normal residual heat removal system (RHRS) operation to control cooldown flow rate.

Each residual heat removal heat exchanger bypass line has an air-operated butterfly valve which is normally closed and is designed to fail closed. Those valves (RH-FCV-618 and 619) are used during normal cooldown to avoid thermal shock to the residual heat exchanger.

6. Accumulator Motor-Operated Valve Controls

As part of the plant shutdown administrative procedures, the operator is required to close these valves. This prevents a loss of accumulator water inventory to the RCS, and is performed shortly after the RCS has been depressurized below the safety injection unblock setpoint. The redundant pressure and level alarms on each accumulator would remind the operator to close these valves, if any were inadvertently left open. Power is disconnected after the valves are closed.

During plant startup, the operator is instructed via procedures to energize and open these valves when the RCS pressure reaches the safety injection unblock setpoint. Monitor lights in conjunction with an audible alarm will alert the operator should any of these valves be left inadvertently closed once the RCS pressure increases beyond the safety injection unblock setpoint (see Dwg. NHY-503907). Once open, power is disconnected and remains off except during valve testing.

The accumulator isolation valves are not required to move during power operation or in a post-accident situation, except for valve testing. For a discussion of limiting conditions for operation and surveillance requirements of these valves, refer to the Technical Specifications.

For further discussion of the instrumentation associated with these valves, refer to Subsections 6.3.5, 7.3.1b and 7.6.4.

7. Motor-Operated Valves and Controls

Remotely operated valves for the injection mode which are under manual control (i.e., valves which normally are in their ready position and do not require a safety injection signal) have their position indicated on the control board. If a component is out of its proper position, its monitor light will indicate this on the control panel. At any time during operation, when one of these valves is not in the ready position for injection, this condition is shown visually on the board, and an audible alarm is sounded in the control room on a system basis, as part of the Bypass and Inoperable Status alarms.

The ECCS delivery lag times are given in Chapter 15. The accumulator injection time varies as the size of the assumed break varies since the RCS pressure drop will vary proportionately to the break size.

Inadvertent mis-positioning of a motor-operated valve due to a malfunction in the control circuitry in conjunction with an accident has been analyzed and found not to be a credible event for use in design.

Table 6.3-3 is a listing of motor-operated isolation valves in the ECCS showing interlocks, automatic features, and position indications.

8. Motor-Operated SI Isolation Valves and Controls

SI-V93, the combined recirculation isolation valve from both safety injection pumps, is a normally open, motor-operated valve. This valve is closed by the operators, from the control room, during the switch over to the recirculation mode of safety injection.

Red/green valve position indication and valve full-closed monitor light is provided on the main control board. Additionally, any time SI-V93 leaves the full open position, an annunciator alarms for both the "SI Train A Inoperative" and the "SI Train B Inoperative" status monitoring alarms.

To prevent spurious operation or operator error, the control circuit for the motor operator is equipped with a dual contactor arrangement (see Updated FSAR Figure 8.3-45). This circuit requires two separate operator actions, involving the normal valve control switch plus a separate key-operated switch, to reposition the valve.

6.3.2.3 Applicable Codes and Classifications

Applicable industry codes and classifications for the ECCS are discussed in Section 3.2.

6.3.2.4 Material Specifications and Compatibility

Materials employed for components of the ECCS are given in Table 6.3-4. Materials are selected to meet the applicable material requirements of the codes in Table 3.2-2 and the following additional requirements for compatibility with the reactor coolant during the recirculation phase following a LOCA:

a. All parts of components in contact with borated water are fabricated of or clad with austenitic stainless steel or equivalent corrosion- resistant material.

b. All parts of components in contact (internal) with sump solution during recirculation are fabricated of austenitic stainless steel or equivalent corrosion-resistant material.

c. Valve seating surfaces are hard faced with Stellite Number 6 or equivalent to prevent galling and to reduce wear.

d. Valve stem materials are selected for their corrosion resistance, high tensile properties, and resistance to surface scoring by the packing.

6.3.2.5 System Reliability

Reliability of the ECCS is considered for all aspects of the system, from initial design through periodic testing of the components during plant operation. The ECCS is a two train, fully redundant standby safeguard feature. The system has been designed and proven by analysis as having the ability to withstand any single credible active failure during injection, or any active or passive failure during recirculation, while still meeting the performance objectives described in Subsection 6.3.1.

Two trains of pumps, heat exchangers, and flow paths are provided for redundancy, as only one train is required to satisfy the performance requirements. The initiating signals for the ECCS are derived from independent sources, as measured from process (e.g., low pressurizer pressure) or environmental variables (e.g., containment pressure). Redundant as well as functionally independent variables are measured to initiate the safeguards signals. Each train is physically separated and protected where necessary so that a single event cannot initiate a common failure. Power sources for the ECCS are divided into two independent trains supplied from the emergency buses which can receive power either from onsite or offsite power sources. Sufficient diesel generating capacity is maintained onsite to provide power to each train. The diesel generators and their auxiliary systems are completely independent, and each supplies power to one of the two ECCS trains.

The reliability program extends to the procurement of the ECCS components such that only designs which have been proven by past use in similar applications are acceptable for use. The quality assurance program, as described in Chapter 17, assures receipt of components only after manufacture and test to the applicable codes and standards.

The preoperational testing program assures that the ECCS as designed and constructed will meet the functional requirements calculated in its design.

The ECCS is designed with the ability for online testing of most components so the availability and operational status can be readily determined.

In addition to the above, the integrity and operability of the ECCS is assured through examination of critical components during the routine in-service inspection.

a. Active Failure Criteria

The ECCS is designed to accept a single failure following the incident without loss of its protective function. The system design will tolerate the failure of any single active component in the ECCS itself or in the necessary associated service systems at any time during the period of required system operations following the incident.

A Failure Mode and Effect Analysis is presented in Table 6.3-5, which demonstrates that the ECCS can sustain the failure of any single active component in either the short or long term and still meet the level of performance for core cooling.

Since the operation of the active components of the ECCS following steam line rupture is identical to that following a LOCA, the same analysis is applicable. The ECCS can sustain the failure of any single active component and still meet the level of performance for the addition of shutdown reactivity.

b. Passive Failure Criteria

The following design philosophy assures the necessary redundancy in component and system arrangement to meet the intent of the General Design Criteria on single failure as it specifically applies to failure of passive components in the ECCS. Thus, for the long term, the system design is based on accepting either a passive or an active failure.

1. Redundancy of Flow Paths and Components for Long-Term Emergency Core Cooling

In design of the ECCS, Westinghouse utilizes the following criteria:

(a) During the long-term cooling period following a loss of coolant, the emergency core cooling flow paths shall be separable into two subsystems, either of which can provide minimum core cooling functions and return spilled water from the floor of the containment back to the RCS.

(b) Either of the two subsystems can be isolated and removed from service in the event of a leak outside the containment. Redundant motor-operated valves arranged in series are provided for this isolation function.

(c) Adequate redundancy of check valves is provided to tolerate failure of a check valve during the long term as a passive component.

(d) Should one of these two subsystems be isolated in this long-term period, the other subsystem remains operable.

(e) Provisions are also made in the design to detect leakage from components outside the containment, to collect this leakage, and to provide for maintenance of the affected equipment.

A single passive failure analysis is presented in Table 6.3.6. It demonstrates that the ECCS can sustain a single passive failure during the long-term phase and still retain an intact flow path to the core to supply sufficient flow to maintain the core covered and affect the removal of decay heat. The procedure followed to establish the alternate flow path also isolates the component which failed.

Thus, for the long-term emergency core cooling function, adequate core cooling capacity exists with one flow path removed from service.

2. Subsequent Leakage from Components in Safeguards Systems

With respect to piping and mechanical equipment outside the containment, considering the provisions for visual inspection and leak detection, leaks will be detected before they propagate to major proportions. A review of the equipment in the system indicates that the largest sudden leak potential would be the sudden failure of a pump shaft seal. Evaluation of leak rate, assuming only the presence of a seal retention ring around the pump shaft, showed that flows less than 50 gpm would result. Piping leaks, valve packing leaks, or flange gasket leaks tend to build up slowly with time and are considered less severe than the pump seal failure.

Larger leaks in the ECCS are prevented by the following:

(a) The piping is classified in accordance with ANS Safety Class 2 and receives the ASME Class 2 quality assurance program associated with this safety class.

(b) The piping, equipment and supports are designed to ANS Safety Class 2 seismic classification, permitting no loss of function for the design basis earthquake.

(c) The system piping is located within a controlled area on the plant site.

(d) The piping system receives periodic pressure tests, and is accessible for periodic visual inspection.

(e) The piping is austenitic stainless steel which, due to its ductility, can withstand severe distortion without failure.

(f) Instrument tubing is designed to the requirements of Regulatory Guide 1.151 as discussed in Subsection 7.1.2.12.

Based on this review, the design of the Primary Auxiliary Building and related equipment was verified for its ability to handle leaks up to a maximum of 50 gpm. Leakage would drain to and collect in the primary auxiliary building sump. Automatic initiation of the sump pumps at a predetermined setpoint would be indicated at the main control board and would alert the operator to an abnormal condition. Corrective action would include determining the location of the leak by visual inspection, and remote or manual isolation of the leak point from the rest of the system within 30 minutes.

c. Potential Boron Precipitation

Boron precipitation in the reactor vessel can be prevented by a backflush of cooling water through the core to reduce boil-off and resulting concentration of boric acid in the water remaining in the reactor vessel. This is accomplished by a switch from cold to hot leg recirculation about 5 to 6 hours following an accident.

The minimum Hot Leg Recirculation flow meets decay heat removal requirements at this time.

Three flow paths are available for hot leg recirculation of sump water. Each safety injection pump can discharge to two hot legs with suction taken from residual heat removal pump discharge either directly or indirectly via the charging pump cross connect. The residual heat removal pump(s) will also be aligned to deliver flow to the hot leg injection header.

Loss of one pump or one flow path will not prevent hot leg recirculation since redundant methods are available for use.

d. Submerged Valve Motors

All electrically operated valves in the ECCS required to be functional during and following a LOCA are located outside containment. All other electrical equipment in the ECCS that is required during post-LOCA is either located outside containment or above the maximum calculated water level inside containment.

6.3.2.6 Protection Provisions

The provisions taken to protect the system from damage that might result from dynamic effects are discussed in Section 3.6. The provisions to protect the system from missiles are discussed in Section 3.5. The provisions to protect the system from seismic damage are discussed in Sections 3.7, 3.9 and 3.10. Thermal stresses on the RCS are discussed in Section 5.2.

6.3.2.7 Provisions for Performance Testing

Test lines are provided for performance testing of the ECCS as well as individual components. These test lines and instrumentation are shown in Figure 6.3-1. All pumps have miniflow lines for use in testing operability. Additional information on testing can be found in Subsection 6.3.4.2.

6.3.2.8 Manual Actions

No manual operator actions are required for proper operation of the ECCS during the injection mode of operation. Only limited manual actions are required by the operator to realign the system for the cold leg recirculation mode of operation, and, after approximately 5 to 6 hours, for the hot leg recirculation mode of operation. These actions are delineated in Table 6.3-7.

The transfer from the injection mode to recirculation mode is initiated automatically and completed manually by operator action from the main control room. Protection logic is provided to automatically open the two containment recirculation sump isolation valves when two out of four refueling water storage tank level channels indicate a refueling water storage tank level less than a low-low level setpoint in conjunction with the initiation of the engineered safeguard actuation signal ("S" signal - see Dwg. NHY-503258). The automatic action would align the two residual heat removal pumps to take suction from the containment sump and deliver directly to the Reactor Coolant System. The automatic action also aligns the two containment building spray pumps to take suction from the containment sump and deliver to the containment building spray headers. It should be noted that the residual heat removal and containment building spray pumps would continue to operate during this transfer from injection mode to recirculation mode.

The two charging pumps and the two safety injection pumps would continue to take suction from the refueling water storage tank following the above automatic action, until manual operator action is taken to align these pumps in series with the residual heat removal pumps.

The consequences of the operator failing to act altogether will be loss of high head safety injection pumps and charging pumps.

The refueling water storage tank level protection logic consists of four level channels, with each level channel assigned to a separate process control protection set. Four refueling water storage tank level transmitters provide level signals to corresponding normally de-energized level channel bistables. Each level channel bistable would be energized on receipt of a signal that the refueling water storage tank level is less than the low-low level setpoint.

The two-out-of-four coincident logic is utilized in both protection cabinets A and B to ensure a trip signal in the event that two out of the four level channel bistables are energized. This trip signal, in conjunction with the "S" signal, provides the actuation signal to automatically open the corresponding containment sump isolation valves.

The first manual actions are required of the operator after the "Lo-Lo" level signal actuates the opening of the sump valves. The "Lo-Lo" signal also alarms at the MCB to alert the operators to the need for action.

Figure 6.3-10 shows a schematic of the tank, the level instrumentation setpoints and the various water volume allowances.

All level setpoints were selected to ensure a minimum required injection volume, adequate transfer volumes and, at the same time, to avoid any spurious alarm actuations.

The "high level" alarm setpoint was selected to assure that, during filling of the RWST, the operator is alerted prior to a possible spillage of water from refueling water storage tank via overflow.

The "tech spec" level alarm setpoint was selected to assure that the minimum required injection volume remains above the "Lo-Lo" level (transfer) setpoint. If the water level drops below the "tech spec" setpoint, the alarm sounds and the plant will be placed in the mode mandated by the plant technical specifications. A "tech spec approach" alarm is provided to alert the operator to a need for makeup prior to reaching a water level requiring plant shutdown.

Spurious alarm actuation is prevented by an adequate separation between the instrument error bands associated with each setpoint. In addition to this, temperature compensation is employed for "high level," "tech spec" and "tech spec approach" alarms to account for level changes due to temperature fluctuations.

Instrument error bands were calculated accounting for uncertainties such as measurement accuracy, calibration accuracy, signal drift, environment changes, etc.

The time from accident initiation to the first required manual actions is dependent on the initial tank water level, the flow rate out of the tank and the "Lo-Lo" level setpoint. The minimum time is calculated based on injecting the minimum required injection volume of 350,000 gallons. As shown on Figure 6.3-10, the minimum injection allowance is contained between extreme low range of the "tech spec" alarm error band and the extreme upper range of the "Lo-Lo" signal error band. The maximum flow rate out of the RWST is based on the "Maximum Safeguards" ECCS case depicted on Figure 15.6-48 and described in Section 15.6.5.3. The mass flow rates depicted on Figure 15.6-48 do not include spillage out of the postulated broken line. This spillage flow is added, and Containment Building Spray flow from both trains is added to the mass flow rates of Figure 15.6-48 to determine the maximum flow rate of 13,200 gpm out of the tank. Based on injecting the minimum injection allowance at a maximum rate of 13,200 gpm, the first manual operator actions are not required until approximately 26 minutes after accident initiation.

It should be noted that the entire injection volume is assumed to come from the RWST, neglecting the additional volume available in the spray additive tank.

Figure 6.3-10 depicts two additional allowances of RWST inventory. The transfer allowance is the volume of RWST inventory set aside to allow the operators sufficient time to complete the transfer from the injection mode to the recirculation mode of ECCS operation. This transfer allowance is the volume of water contained between the low range of the RWST Lo-Lo level setpoint and the high range of the RWST EMPTY alarm setpoint. The RWST EMPTY alarm setpoint is based on the postulated single failure discussed below. The shutoff or single failure allowance is the volume of RWST inventory between the RWST EMPTY alarm setpoint and the calculated level at which vortexing is possible for the flow rate from the RWST associated with the postulated worst case single failure that could occur during this transfer.

In the event of the design bases LOCA, the containment sump isolation valves open automatically upon actuation of the RWST LO-LO level signal. The combination of containment pressure and elevation head from the sump seats, the check valves, in the lines between the RWST, CBS, and RHR pumps (CBS V3, -V7, -V55, -V56), reducing the flow rate out of the RWST to that of the safety injection and charging pumps; or approximately 1500 gpm. At his flow rate, the operators would have over 27 minutes to complete the transfer before the RWST EMPTY alarm could actuate. It should be noted that the RWST EMPTY alarm setpoint is based on the higher flow rate associated with the single failure discussed below, and the safety injection and charging pumps would be susceptible to vortexing at he RWST level. However, operators would still secure these pumps if the transfer was not completed before the RWST EMPTY alarm was actuated.

In an accident for which the RWST water is at the minimum allowed temperature, the containment heat sinks are at a low temperature and the heat transfer rate in the containment is high, the containment pressure may be high enough to actuate spray, but not high enough to seat the check valves in the suction lines from the RWST to the RHR and CBS pumps. For this case, the high flow rate out of the RWST would continue until the tank suction isolation valves (CBS-V2, -V5) are closed by the operators. The operators would have over 3 minutes to close these valves before vortexing is calculated to occur. Assuming the operators take the full time allotted to close these two valves, this would still leave an additional 5 minutes to complete the transfer.

The limiting single failure for this design is the failure of one of the RWST isolation valves to close (CBS-V2, -V5). Note that the failure of one of the containment sump isolation valves to open has the same affect with respect to flow rate out of the tank, however, it occurs earlier in the transfer and the operator has additional time to respond. If one of the RWST isolation valves does not close, the flow rate out of the RWST is reduced from 13,000 gpm to 7400 gpm, not the 1500 gpm when both valves close. The RWST EMPTY alarm is designed for this single failure. If the operators are not able to get the affected valve closed, the RWST EMPTY alarm will sound, alerting them to immediately shut off any pumps still taking suction from the RWST. The shutoff allowance is established to give the operators 1 minute of operation between the EMPTY alarm and the level at which vortexing could potentially occur for shutting off the pumps. Note that though all of the pumps still taking a suction from the RWST are shut off when this alarm sounds, the operators only have to stop the affected RHR and CBS pumps in order to prevent vortexing. The safety injection and charging pumps could operate for over 11 additional minutes before reaching their calculated vortex level. After the pumps are stopped, the transfer is completed and all available pumps are restarted.

Following the automatic and manual transfer sequence, the two residual heat removal pumps would take suction from the containment sump and deliver borated water directly to two RCS cold legs. A portion of the Number 1 residual heat removal pump discharge flow would be used to provide suction to the two charging pumps which would also deliver directly to the RCS cold legs. A portion of the discharge flow from the Number 2 residual heat removal pump would be used to provide suction to the two safety injection pumps which would also deliver directly to the RCS cold legs. As part of the manual transfer, the suctions of the safety injection and charging pumps are cross connected so that one residual heat removal pump can deliver flow to the RCS and both safety injection and charging pumps, in the event of the failure of the second residual heat removal pump.

See Section 7.5 for process information available to the operator in the control room following and accident.

6.3.3 Performance Evaluation

Chapter 15 accidents that result in ECCS operation are as follows:

a. Inadvertent opening of a steam generator relief or safety valve (see Subsection 15.1.4)

b. Small break LOCA (see Subsection 15.6.5)

c. Large break LOCA (see Subsection 15.6.5)

d. Major secondary system pipe failure (see Subsection 15.1.5)

e. Steam generator tube failure (see Subsection 15.6.3).

Safety injection is actuated from any of the following:

a. Low pressurizer pressure

b. Low steamline pressure

c. High containment pressure

d. Manual initiation.

A safety injection signal will rapidly trip the main turbine, close all feedwater control valves, trip the main feedwater pumps, and close the feedwater isolation valves.

Following the actuation signal, the suction of the centrifugal charging pumps is diverted from the volume control tank to the refueling water storage tank. Simultaneously, the valves isolating the charging pumps from the injection header automatically open. The safety injection pumps also start automatically but operate at shut off head when the RCS is at normal pressure. The passive injection system (accumulators) and the low head system (residual heat removal pumps) also provide no flow at normal RCS pressure.

Figure 6.3-6 is a simplified illustration of the ECCS. The notes provided with Figure 6.3-6 contain information relative to the operation of the ECCS in its various modes. The modes of operation illustrated are full operation of all ECCS components, cold leg recirculation with residual heat removal pump Number 2 operating, and hot leg recirculation with residual heat removal pump Number 1 operating.

Lag times for initiation and operation of the ECCS are limited by pump startup time and consequential loading sequence of these motors onto the safeguard buses. Most valves are normally in the position conducive to safety; therefore, valve opening time is not considered for these valves. If there is no power blackout, all pump motors and valve motors are started immediately upon receipt of the "S" signal. In the case of a blackout, a 10 second delay is assumed for diesel startup, then pumps and valves are loaded according to the sequencer. The charging pumps will be applied to the buses in 10 seconds, the safety injection pumps will start in 15 seconds, and the residual heat removal pumps in 20 seconds. These times refer to time after receipt of the "S" signal, and include time for attainment of speed of the diesel generators. See Subsection 8.3.1 for details of diesel generator operation. The time sequence of ECCS components is also discussed in Chapter 15 with the appropriate accident analysis.

6.3.3.1 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The assumed steam release is typical of the capacity of any single steam dump, relief or safety valve. The boron solution provides sufficient negative reactivity to maintain the reactor well below criticality. The cooldown for this case is more rapid than the actual case of steam release from all steam generators through one steam dump, relief, or safety valve. The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the steam generators. Since the transient occurs over a period of about 5 minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown. The analysis shows that there will be no return to criticality after reactor trip, assuming a stuck Rod Cluster Control Assembly, with offsite power available, and assuming a single failure in the engineered safety features. Since the reactor does not return to criticality, a Departure from Nucleate Boiling Ratio (DNBR) less than the safety analysis limit value does not exist.

6.3.3.2 Small Break LOCA

A LOCA is defined as a rupture of the RCS piping or of any line connected to the system. Ruptures of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the RCS through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is adequate to sustain pressurizer level and pressure for a break through a 0.375 inch diameter hole. This break results in a loss of approximately 17.5 lb/sec (127 gpm at 130°F and 2250 psia).

The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates emergency feedwater flow by starting emergency feedwater pumps.

The small break analyses deal with breaks of up to 1.0 ft^2 in area, where the safety injection pumps play an important role in the initial core recovery because of the slower depressurization of the RCS.

The RCS depressurization water level transients show that for a break of approximately 3.0 inch equivalent diameter, the transient is turned around and the core is recovering prior to accumulator injection. For a 3.5 inch equivalent diameter break, the core remains uncovered with a decreasing level until accumulator action. Thus, the maximum break size showing core recovery prior to accumulator injection will be approximately 3.0 inch equivalent diameter. Accumulator injection commences when pressure reaches 600 psia, i.e., approximately 1200 seconds for the 3.0 inch break size.

The analysis of this break has shown that the high head portion of the ECCS, together with accumulators, provide sufficient core flooding to keep the calculated peak clad temperature below required limits of 10 CFR 50.46. Hence, adequate protection is afforded by the ECCS in the event of a small break LOCA.

6.3.3.3 Large Break LOCA

A major LOCA is defined as a rupture 1.0 ft^2 or larger of the RCS piping including the double-ended rupture of the largest pipe in the RCS or of any line connected to that system. The boundary considered for LOCA as related to connection piping is defined in Section 3.6.

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. Reactor trip occurs and the Safety Injection System is actuated when the pressurizer low pressure trip setpoint is reached. Reactor trip and safety injection system actuation are also provided by a high containment pressure signal. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection provide additional negative reactivity insertion to supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

b. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

When the pressure falls below approximately 600 psia, the accumulators begin to inject borated water. The conservative assumption is made that injected accumulator water bypasses the core and goes out through the break until the termination of the blowdown phase. This conservatism is again consistent with the Final Acceptance Criteria.

The pressure transient in the reactor containment during a LOCA affects ECCS performance in the following ways. The time at which end of blowdown occurs is determined by zero break flow which is a result of achieving pressure equilibrium between the RCS and the containment. In this way, the amount of accumulator water bypass is also affected by the containment pressure, since the amount of accumulator water discharged during blowdown is dependent on the length of the blowdown phase and RCS pressure at end of blowdown. During the reflood phase of the transient, the density of the steam generated in the core is dependent on the existing containment pressure. The density of this steam affects the amount of steam which can be vented from the core to the break for a given downcomer head, the core reflooding process, and thus, the ECCS performance. It is through these effects that containment pressure affects ECCS performance.

For breaks up to and including the double-ended severance of a reactor coolant pipe, the ECCS will limit the clad temperature to well below the melting point and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved. See Table 15.6-1 for ECCS sequence of events.

For breaks up to and including the double-ended severance of a reactor coolant pipe, the ECCS will meet the Acceptance Criteria as presented in 10 CFR 50.46. That is:

a. The calculated peak fuel element clad temperature provides margin to the requirement of 2200°F.

b. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

c. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17 percent are not exceeded during or after quenching.

d. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

6.3.3.4 Major Secondary System Pipe Failure

The steam release arising from a rupture of a main steam pipe would result in energy removal from the RCS causing a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. There is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem. The core is ultimately shut down by the borated water injection delivered by the Safety Injection System.

Minimum capability for injection of the high concentration boric acid solution is assumed corresponding to the most restrictive single active failure in the Safety Injection System. The calculated transient delivery times for the borated water are listed in Table 15.1-1. In all cases, injection of the refueling water is preceded by the low concentration borated water, which is swept from the lines.

For the cases where offsite power is available, the sequence of events in the Safety Injection System is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 30 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept, of course, before the refueling water reaches the core. This delay, described above, is inherently included in the modeling.

In cases where the offsite power is not available, and additional 10 second delay is assumed to start the diesels and to load the necessary safety injection equipment onto them.

The analysis has shown that even assuming a stuck Rod Cluster Control Assembly with or without offsite power, and assuming a single active failure in the engineered safeguards, the core remains in place and intact. Radiation doses will not exceed 10 CFR 100 guidelines.

Although Departure from Nucleate Boiling (DNB) (with possible clad perforation) following a steam pipe rupture is not necessarily unacceptable and not precluded in the criterion, the above analysis, in fact, shows that no Departure from Nucleate Boiling occurs for any rupture assuming the most reactive Rod Cluster Control Assembly stuck in its fully withdrawn position.

6.3.3.5 Steam Generator Tube Failure

The accident examined is the complete severance of a single steam generator tube, assuming it to take place at power.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube failure:

a. Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before the trip as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that unit.

b. Continued loss of reactor coolant inventory leads to a reactor trip and safety injection signals generated by an OTDT signal. The resultant plant cooldown also leads to a rapid change in pressurizer level. The safety injection signal automatically terminates normal feedwater supply and initiates emergency feedwater supply and initiates emergency feedwater addition.

c. The four steam generator blowdown and flash tank concentrates liquid monitors and the condenser off-gas radiation monitor will alarm indicating a sharp increase in radioactivity in the secondary system. A high radioactivity signal from any one of the four steam generator blowdown radiation monitors or the flash tank concentrates radiation monitor will automatically isolate the concentrates discharge from the blowdown flash tank.

d. The reactor trip automatically trips the turbine and, if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of a coincident station blackout, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere through the steam generator safety and/or power-operated relief valves.

e. Following reactor trip, the continued action of emergency feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provide a heat sink which absorbs some of the decay heat. Thus, steam bypass to the condenser, or in the case of loss of offsite power, steam relief to atmosphere, is attenuated during the recovery procedure leading to isolation.

f. Safety injection flow results in increasing pressurizer water level. The time after trip at which the operator can clearly see returning level in the pressurizer is dependent upon the amount of operating auxiliary equipment.

A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed even assuming simultaneous loss of offsite power.

6.3.3.6 Existing Criteria Used to Judge the Adequacy of the ECCS

Criteria from 10 CFR 50.46:

a. Peak clad temperature calculated shall not exceed 2200°F.

b. The calculated total oxidation of the clad shall nowhere exceed 17% of the total clad thickness before oxidation.

c. The calculated total amount of hydrogen generated from the chemical reaction of the clad with water or steam shall not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the clad cylinders surrounding the fuel, excluding the clad around the plenum volume, were to react.

d. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

e. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by long-lived radioactivity remaining in the core.

In addition to and as an extension of the Final Acceptance Criteria, two accidents have more specific criteria as shown below.

In the case of the inadvertent opening of a steam generator relief or safety valve, an additional criteria for adequacy of the ECCS is: assuming a stuck Rod Cluster Control Assembly, offsite power unavailable, and a single failure in the engineered safety features, there will be no return to criticality after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the larger of a single steam dump, relief, or safety valve.

For a major secondary system pipe failure, the added criteria is: assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safeguards, the core remains in place and intact.

6.3.3.7 Use of Dual Function Components

The ECCS contains components which have no other operating function, as well as components which are shared with other systems. Components in each category are as follows:

a. Components of the ECCS which perform no other function are:

1. One accumulator for each loop which discharges borated water into its respective cold leg of the reactor coolant loop piping.

2. Two safety injection pumps, which supply borated water for core cooling to the RCS. (May be used during check valve testing also.)

3. Associated piping, valves and instrumentation.

b. Components which also have a normal operating function are:

1. Residual Heat Removal Pumps and the Residual Heat Exchangers

These components are normally used during the latter stages of normal reactor cooldown and when the reactor is held at cold shutdown for core decay heat removal or for flooding the refueling cavity. However, during all other plant operating periods, they are aligned to perform the low head injection function.

2. Centrifugal Charging Pumps

These pumps are normally aligned for charging service. As a part of the Chemical and Volume Control System, the normal operation of these pumps is discussed in Subsection 9.3.4.

3. Refueling Water Storage Tank

This tank is used to fill the refueling canal for refueling operations and to provide makeup to the spent fuel pool. However, during all other plant operating periods, it is aligned to the suction of the safety injection pumps, and the residual heat removal pumps. The charging pumps are automatically aligned to the suction of the refueling water storage tank upon receipt of the safety injection signals or a volume control tank low level signal. During normal operation they take suction from the volume control tank.

An evaluation of all components required for operation of the ECCS demonstrates that either:

a. The component is not shared with other systems, or

b. If the component is shared with other systems, it is either aligned during normal plant operation to perform its accident function or, if not aligned for its accident function, two valves in parallel are provided to align the system not utilized for injection and two valves in series are provided to isolate portions of the system not utilized for injection. These valves are automatically actuated by the safety injection signal.

Table 6.3-8 indicates the alignment of components during normal operation, and the realignment required to perform the accident function.

In all cases of component operation, safety injection has the priority usage so that an "S" signal will override all other signals and start or align systems for injection.

6.3.3.8 Limits on System Parameters

The analyses show that the design basis performance characteristic of the ECCS is adequate to meet the requirements for core cooling following a LOCA with the minimum engineered safety features equipment operating. In order to ensure this capability in the event of the simultaneous failure to operate any single active component, Technical Specifications are established for reactor operation.

Normal operating status of ECCS components is given in Table 6.3-9.

The ECCS components are available whenever the coolant energy is high and the reactor is critical. During low temperature operation there is a negligible amount of stored energy in the coolant and low decay heat; therefore, an accident comparable in severity to accidents occurring at operating conditions is less probable and fewer ECCS components are required.

The principal system parameters and the number of components which may be out of operation or in test, quantities and concentrations of coolant available, and allowable time in a degraded status are illustrated in the Technical Specifications. If efforts to repair the faulty component are not successful, the plant is placed into a lower operational status, i.e., hot standby to hot shutdown, hot shutdown to cold shutdown, etc.

6.3.4 Tests and Inspections

6.3.4.1 Preoperational Performance Tests

Preoperational performance testing of the ECCS is discussed in Chapter 14.

6.3.4.2 Reliability Tests and Inspections

Capability is provided for routine periodic testing of the ECCS components and all necessary support systems at power. Valves which operate after a LOCA are operated through a complete cycle, and pumps are operated individually in this test on their miniflow lines. The charging pumps also can be tested by their normal charging function. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under certain conditions. These conditions include considerations such as the period within which the component should be restored to service and the capability of the remaining equipment to provide the minimum required level of performance during such a period.

The operation of the remote stop valve and the check valve in each accumulator tank discharge line may be tested by opening the remote test line valves just downstream of the stop valve and check valve. Flow through the test line can be observed on instruments, and the opening and closing of the discharge line stop valve can be sensed on this instrumentation.

Where series pairs of check valves form the high pressure to low pressure isolation barrier between the RCS and safety injection system piping, periodic testing of these check valves must be performed to provide assurance that certain postulated failure modes will not result in a loss of coolant from the low pressure system outside containment with a simultaneous loss of safety injection pumping capacity.

The safety injection system test line subsystem provides the capability for determination of the integrity of the pressure boundary formed by series check valves. The tests performed verify that each of the series check valves can independently sustain differential pressure across its disc, and also verify that the valve is in its closed position. The required periodic tests are to be performed after each refueling just prior to plant startup, after the RCS has been pressurized.

Lines in which the series check valves are to be tested are the safety injection pump cold and hot leg injection lines and the residual heat removal pump cold and hot leg injection lines.

To implement the periodic component testing requirements, Technical Specifications have been established. During periodic system testing, a visual inspection of pump seals, valve packings, flanged connections, and relief valves is made to detect leakage. In-service inspection provides further confirmation that no significant deterioration is occurring in the ECCS fluid boundary.

Design measures have been taken to assure that the following testing can be performed:

a. Active components may be tested periodically for operability (e.g., pumps on miniflow, certain valves, etc).

b. An integrated system actuation test* can be performed when the plant is cooled down and the RHRS is in operation.

c. An initial flow test of the full operational sequence can be performed.

The design features which assure this test capability are specifically:

a. Power sources are provided to permit individual actuation of each active component of the ECCS.

b. The safety injection pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided.

c. The residual heat removal pumps are used every time the RHRS is put into operation. They can also be tested periodically when the plant is at power using the miniflow recirculation lines.

d. The centrifugal charging pumps are either normally in use for charging service or can be tested periodically on miniflow.

e. Remote operated valves can be exercised during routine plant maintenance.

f. Level and pressure instrumentation is provided for each accumulator tank for continuous monitoring of these parameters during plant operation.

g. Flow from each accumulator tank can be directed through a test line to determine valve operability.

h. A flow indicator is provided in the charging pump, safety injection pump, and residual heat removal pump headers. Pressure instrumentation is also provided in these lines.

---

* Details of the testing of the sensors and logic circuits associated with the generation of the safety injection signal, together with the application of this signal to the operation of each active component, are given in Section 7.2.

i. An integrated system test can be performed when the plant is cooled down and the RHRS is in operation. This test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry including diesel starting and the automatic loading of ECCS components of the diesels (by simulating a loss of offsite power in conjunction with an SI actuation test signal).

See the Technical Specifications for the selection of test frequency, acceptability of testing and measured parameters. ECCS components and systems are designed to meet the intent of the ASME Code, Section XI for in-service inspection.

6.3.5 Instrumentation Requirements

Instrumentation and associated analog and logic channels employed for initiation of ECCS operation are discussed in Section 7.3. This section describes the instrumentation employed for monitoring ECCS components during normal plant operation and ECCS post accident operation. All alarms are annunciated in the control room.

6.3.5.1 Temperature Indication: Residual Heat Exchanger Temperature

The fluid temperature at both the inlet and the outlet of each residual heat exchanger is recorded in the control room.

6.3.5.2 Pressure Indication

a. Charging Pump Inlet and Discharge Pressure

There is local pressure indication at the suction and discharge of each centrifugal charging pump.

b. Safety Injection Pump Suction Pressure

There is a locally mounted pressure indicator at the suction of each safety injection pump.

c. Safety Injection Pump Discharge Pressure

Safety injection pump discharge pressure is indicated in the control room for both pumps.

d. Accumulator Pressure

Duplicate pressure channels are installed on each accumulator. Pressure indication in the control room and high and low pressure alarms are provided by each channel.

e. Test Line Pressure

A local pressure indicator used to check for proper seating of the system check valves between the injection lines and the RCS is installed on the leakage test line. The pressure is also indicated on the control board.

f. Residual Heat Removal Pump Suction Pressure

Local pressure indication is provided at the inlet to each residual heat removal pump.

g. Residual Heat Removal Pump Discharge Pressure

Residual heat removal pump discharge pressure for each pump is indicated in the control room. A high pressure alarm is actuated by each channel.

h. Containment Building Spray Pumps Discharge Pressure

Containment building spray pump discharge pressure is indicated in the control room for both pumps.

6.3.5.3 Flow Indication

a. Charging Pump and Injection Header Flow

Flow for each centrifugal charging pump and the total centrifugal charging pump injection flow are indicated in the control room. A low-flow alarm for each pump is provided on the main control board.

b. Safety Injection Pump Flow

Injection flow for each of the safety injection pumps is indicated in the control room. A low-flow alarm for each pump is provided on the main control board.

c. Safety Injection Pump Minimum Flow

A flow indicator is installed in the safety injection pump minimum flow line.

d. Test Line Flow

Local and main control board indication of the leakage test line flow is provided to check for proper seating of the system check valves between the injection lines and the RCS.

e. Residual Heat Removal Pump Cold Leg Injection Flow

The flow from each residual heat removal subsystem to the RCS cold legs is indicated in the control room. These instruments also control the residual heat removal bypass valves, maintaining constant return flow to the RCS during normal cooldown.

f. Residual Heat Removal Pump Minimum Flow

A flowmeter installed in each residual heat removal pump discharge header provides control for the valve located in the pump minimum flow line. A low-flow alarm is provided on the main control board.

6.3.5.4 Level Indication

a. Refueling Water Storage Tank Level

Three separate level indicators are available at the main control board, from three separate level transmitters. Two channels are recorded. Apart from this, four channels of RWST level are provided to the ESF portion of the protection system. A low-low level, 2/4 channels, in combination with the "S" signal will generate an "ECCS/CONTN SPRAY RECIRC" signal. This signal is used to open the containment sump isolation valves. The following alarms are available through the computer:

1. RWST Level High
2. RWST Level Approaching Technical Specification
3. RWST Level Technical Specification Limit
4. RWST Level Lo-Lo at Recirc Setpoint
5. RWST Empty

Recirculation actuation is indicated on the control board by a monitoring light grouped with the appropriate valve position lights.

b. Accumulator Water Level

Duplicate water level channels are provided for each accumulator. Both channels provide indication in the control room and actuate high and low water level alarms.

c. Spray Additive Tank Level

Two level indication channels are provided in the control room.

d. Containment Flood Level

Level indication is provided in the control room on a per train basis.

e. Encapsulated Tank for Containment Recirculation Sump Valves

Water accumulation inside this tank is alarmed in the control room. This indicates leakage from the containment sump recirculation valves.

6.3.5.5 Valve Position Status Monitoring Indication

In addition to red (open) and green (closed) position indicating lights located above the associated control switch, the positions of the valves in Table 6.3-3 are also indicated by arrays of monitoring status lights at the main control board. These accident DBE status lights monitor the status of various engineered safeguard equipment following an accident and/or other Design Basis Events (DBE). The grouping of these lights provides the operator with a quick and easy identification of the required status of the important equipment for the various operational modes following an accident. The lights are grouped and are designed to go on upon these events occurring. Deviations are quickly identified by the light(s) failing to go on.

Isolation valves are provided with red (open) and green (closed) position-indicating lights located above the associated control switch. These lights are actuated by valve motor-operator limit switches or stem-mounted switches, as applicable. A monitor light for each isolation valve that is on when the valve is in isolation position, is provided in an array of monitor lights located on the MCB. For valves whose position indication lights are powered independently from the control circuit, control power availability is indicated by an indicating light at the MCB.

6.4 HABITABILITY SYSTEMS

The habitability systems include the control room complex which houses the controls to operate the plant safely under normal conditions and maintain it in a safe condition under all postulated accident conditions, as well as the supporting equipment, supplies and procedures for an emergency team of 10 operating and 25 technical support personnel. The Technical Support Center is in the control room complex and has the capacity to accommodate 25 technical support personnel.

6.4.1 Design Bases

The control room complex design provides specific areas for continuous occupancy by the station operating personnel and the technical support center personnel during postulated emergency conditions, including equipment and materials to which the operators may require access.

The structural design of the control room complex together with its supporting systems will ensure access and occupancy under accident conditions without occupants receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The control room complex has sanitary facilities and a first aid kit.

The Control Room Ventilation System, which includes redundant emergency makeup air/filtration subsystems, will prevent the buildup of airborne particulates and radioactive iodines within the control room complex during an accident.

Equipment, including necessary instrumentation, controls and procedures are provided at appropriate locations outside the control room with the design capability for (1) initiating prompt hot shutdown of the reactor and maintenance of the unit in a safe hot shutdown condition and (2) with a capability for subsequent cold shutdown of the reactor.

The ventilation design will allow unit operation from the isolated control room without makeup or exhaust air for a period of time of at least thirty hours before the carbon dioxide buildup would reach the concentration of 0.6 percent by volume. (Carbon dioxide concentration would not exceed 3 percent within 290 hours. At 1.5 percent, basic performance and physiological functions are not affected, but concentrations of 3 percent by volume should not be exceeded. Per ASHRAE Handbook, 1978 Applications, Chapter 12.)

Both control room makeup air intakes are located at a distance sufficiently away from each other and from major potential accident release points to minimize control room contamination in the event of a release of airborne radioactivity or other toxic substances.

The control room complex is maintained at a positive pressure of at least ⅛" w.g. with respect to outside and the adjacent cable spreading room. This positive pressure prevents the infiltration of hazardous contaminants. Self-contained breathing apparatus is supplied within the complex to provide breathing protection in the presence of hazardous contaminants.

The control room envelope boundary is designed and maintained so that unfiltered air inleakage is limited to ≤ 150 cfm during the emergency mode of operation.

Redundant air conditioning systems are provided to ensure that the control room atmosphere is maintained within acceptable temperature and humidity limits for equipment operability and personnel comfort.

Meteorological information instruments are provided. These instruments aid the operator in deciding to isolate a contaminated makeup air intake if necessary.

The habitability of the control room complex will not be compromised by the simultaneous occurrence of an SSE, a loss of offsite power and a loss-of-coolant accident.

The active and passive components of the Control Room Ventilation System provide an environment that is consistent with the requirements of General Design Criterion 19, which are to provide adequate ventilation air, maintain positive pressure to prevent infiltration of contaminants, temper the air for operator comfort, remove equipment heat, and maintain the control room below the guideline radiation dose of 5 rem TEDE.

6.4.2 System Design

The habitability systems include:

a. A concrete radiation shielded control room (Subsection 12.3.2)

b. Heating, Ventilating and Air Conditioning Systems (Subsection 9.4.1)

c. Redundant normal makeup air fans and associated discharge dampers

d. Redundant emergency makeup air fans and associated HEPA/carbon/HEPA filtering units for airborne particulate and iodine removal from all makeup air and a portion of the total control room recirculation air under accident conditions

e. Radiation and smoke monitored dual remote air intakes (Subsection 12.3.4)

f. Meteorological Information System (Section 2.3)

g. Fire Protection System (Subsection 9.5.1)

h. Full-face emergency breathing apparatus

i. Communications (Subsection 9.5.2)

j. Normal and emergency lighting (Subsection 9.5.3)

k. Toilet facilities

l. Medical supplies

m. Kitchen area.

6.4.2.1 Definition of Control Room Envelope

The control room occupies the entire 75'-0" level of the Control Building, and includes the main control room area, computer room, Technical Support Center, office, conference room and library, emergency storage room, HVAC equipment room, kitchen and sanitary facilities, as shown in Figure 1.2-32. All controls, equipment and materials to which the control room operator would require access during an emergency are contained within this envelope, except for the makeup air intakes' manual isolation valves which are located on elevation 51'-6" of the Diesel Generator Building.

6.4.2.2 Ventilation System Design

Details of the control room complex ventilation and filtration systems are described in Subsections 9.4.1 and 6.5.1. An air flow diagram of the Control Room Ventilation System which identifies equipment, ducting, dampers, instrumentation and air flow rates for both normal and emergency modes is shown in Figure 9.4-1, Figure 9.4-2, Figure 9.4-3, Figure 9.4-25; major components and their major design parameters are included in Table 9.4-1 and Table 6.5-6.

Two remote air intakes (east and west) are provided to furnish makeup air to the control room complex. The locations were selected to preclude both intakes from being susceptible to accident-generated airborne radioactivity or toxic gases at the same time. The east makeup air intake is located approximately 380 feet northeast from the center of the Unit 1 containment structure. The west intake is located approximately 500 feet southwest of the Unit 1 containment structure (see Figure 1.2-1).

The east air intake consists of a vertical 12-inch diameter carbon-steel pipe terminating in a tee-section. Both openings are protected by ½-inch square stainless steel wire mesh welded to the inside diameter of the respective opening. Protection of the intake against tornado missiles is provided by a reinforced concrete slab (see Figure 6.4-1). The west air intake consists of a vertical 12-inch diameter carbon-steel pipe terminating in a 180° bend. The opening is protected by ½-inch square screen and security grating welded to the inside diameter. A portion of the west intake pipe above grade adjacent to the cooling tower wall and approximately a 2-foot vertical section just below grade is not protected against tornado missiles. The unprotected 12-inch pipe, the vertical section above and below grade, and the 180° bend (see Figure 6.4-2) have an equivalent length of about 15 feet. This low effective target area results in a low mean value probability, calculated in the range of 2×10^{-9} to 3×10^{-7} per year, for tornado missile impact. Smoke monitoring equipment for the east intake, which alarms in the control room, is installed in a vault beneath the slab and within the fence enclosed area. Radiation monitoring equipment for the east air intake, which alarms in the control room, is located within the pipe in the Diesel Generator Building. Radiation and smoke monitoring equipment for the west intake, which also alarms in the control room, is located within the pipe in the Diesel Generator Building. The Diesel Generator Building is located within the protected area security fence, and access to the building is controlled by security doors which are part of the station access control system. Environmental conditions within the east intake vault is maintained by convection heaters and a sump pump.

The makeup air is transported via heavy wall carbon steel pipes from the remote air intakes to the control room HVAC equipment room (see Figure 9.4-1, Figure 9.4-2, Figure 9.4-3). The pipe enters the control room complex through the floor of one of the redundant filter units. The makeup air enters a tee located in a compartment of the filter unit upstream of the various filter components. The makeup air divides so that a portion of the air discharges through an isolation damper then enters this compartment, while the remainder of the air enters the branch leg of the tee to the interconnecting ductwork of the redundant filter unit. The filter units and connecting ductwork are located within the control room envelope.

During normal operations, makeup air is drawn from both remote intakes and delivered to the control room complex by one of the two redundant normal makeup air fans. The normal makeup air fans and associated discharge dampers are located outside the control room envelope on the 50'-0" elevation of the Diesel Generator Building. The air passes through medium efficiency prefilter(s) and electric heater(s) in both emergency filter units prior to discharging through an orifice into the control room HVAC equipment room. The prefilters remove dust and other airborne particulates and the heaters operate continuously to maintain the carbon filter relative humidity at or below 70 percent thereby optimizing carbon adsorber efficiency and life.

Under emergency conditions, makeup air is drawn from both remote air intakes and delivered to the control room complex by two fully redundant emergency filtration system fans. One hundred percent of the makeup air passes through the prefilter and heater and a HEPA-Carbon-HEPA filter configuration in either or both emergency filter units prior to discharging into the control room HVAC equipment room. In addition, approximately 2 percent of the total control room complex recirculation air flow, (i.e., including the air conditioning system flow rate) is drawn through the HEPA-Carbon-HEPA filter configuration in either or both emergency filter units. The HEPA filter(s) and carbon adsorber(s) are designed to remove radioactive airborne particulates and iodines (see Subsection 6.5.1 for filter design specifications). Under emergency conditions, the normal makeup air fans are automatically tripped off and their associated discharge dampers closed. The makeup air is transported to the control room via piping and backdraft dampers configured in parallel which bypass the normal makeup air fans and dampers. The backdraft dampers preclude short cycling of air during normal operations.

The exhaust air and supply air registers are adequately separated to preclude recycling stale air and other noxious gases. The outside makeup air intake and the point of discharge for the control room exhaust are also adequately separated to preclude recycling stale air and other noxious gases.

During normal operations, the modulating damper in the exhaust ductwork controls the amount of air being exhausted thereby maintaining a positive pressure within the control room complex. The damper is under the control of three static pressure sensing devices. The first pressure sensing point for the complex is in the HVAC equipment room, which is slightly lower in pressure than the remainder of the control room complex. The HVAC equipment room is maintained at least ⅛" w.g. above the outside atmospheric pressure, the second sensing point, and at least ⅛" w.g. above the cable spreading room, the third pressure sensing point.

Under emergency conditions, the exhaust system isolates by automatic trip of the exhaust fan and closure of the modulating damper and redundant isolation damper. The emergency makeup air is adequate to maintain the complex at a pressure at least ⅛" w.g. greater than the outside atmospheric and cable spreading room pressures. Air is exhausted from the complex by exfiltration.

The following system components are powered or controlled from the Emergency Electrical Distribution System, to ensure operating power during all modes of operation:

- The normal makeup air fans and associated discharge dampers
- Emergency makeup air fans and associated discharge dampers
- Filter system air heaters
- Radiation Monitoring Instrumentation System
- Exhaust system isolation dampers (Isolation Control System only).

The normal makeup air fans are electrically "cross-trained" with their associated discharge dampers, that is, the damper configured immediately downstream of the Train A fan is powered from the Train B vital bus and is controlled by the Train B control system. The damper configured immediately downstream of the Train B fan is powered from the Train A vital bus and is controlled by the Train A control system. This design ensures isolation of the normal makeup air system (i.e., trip of each makeup air and/or closure of its associated discharge damper) under emergency conditions regardless of any single active failure.

All automatic system dampers are pneumatically actuated and are designed to fail in the safe position (emergency mode configuration) on loss of instrument air.

Controls for habitability system components are located in the control room complex.

The normal makeup air fans and discharge dampers are controlled from the main control board (MCB). The components are manually actuated. Detection of high radiation in either remote intake will automatically isolate the normal makeup air subsystem. Actuation of the emergency makeup air and filtration subsystem fans will automatically isolate the normal makeup air subsystem. Failure of a vital bus or loss of instrument air will also isolate the system.

The emergency makeup air and filtration subsystem fans and discharge dampers are automatically actuated upon detection of high radiation in either remote intake or upon generation of a safety injection 'S' signal. The filtration subsystem can also be normally actuated from the MCB.

The exhaust subsystem fan and discharge dampers are controlled from the MCB. The subsystem will automatically isolate upon loss of control room pressurization, detection of high radiation in either remote intake, or actuation of the filtration subsystem fans.

The safety-related active components of the system are designed to seismic Category I requirements, and satisfy the design criteria of IEEE Standard 279 and other industry standards for electrical equipment, as defined in Subsection 8.1.4. No single failure of any of the active components will degrade the system's performance, as shown in Table 6.4-1.

The safety-related passive components (i.e., pressure boundary) of the makeup air system, ducts and filters, are also designed to seismic Category I requirements.

All safety-related active and passive components of the system are contained in missile-protected buildings, are underground, or in the case of some piping, consist of such a small exposed area that the possibility of being struck by a tornado missile is negligibly small. Specifically, the mean value probability of a missile impacting the unprotected portion of the west air intake is in the range of 2×10^{-9} to 3×10^{-7} per year. No internally generated missiles which could impair the system's ability to perform its safety-related functions are credible.

The system is designed to meet the intent of 10 CFR 50, Appendix A, General Design Criterion (GDC) 19, NUREG-0800 (Standard Review Plan) Section 6.4 and Subsection 6.5-1, and Regulatory Guides 1.52, 1.78, and 1.196.

The system is monitored and regulated by a Control Room Envelope Habitability Program which ensures that the control room and its Operators are in a condition to assure public safety during all modes of operation. It provides measures to allow the control room habitability systems (CRHS) to maintain a habitable environment for operators under normal and abnormal conditions, and identifies compensatory measures if habitability is in question.

6.4.2.3 Leak Tightness

The only openings in the control room envelope boundaries are the sealed cable penetrations, two personnel accesses, the exhaust air duct with isolation dampers, and building construction joints. The primary personnel access way is a double-door configuration. The total complex outleakage with the exhaust subsystem isolated is calculated to be 165 cfm at a pressure of (+) ⅛" w.g.

The control room envelope (CRE) boundary may be opened intermittently under administrative control. This applies only to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings that exceed the allowable opening size, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated. Unfiltered air inleakage testing (tracer gas testing) will be performed, as part of the control room habitability program, at a frequency of 6 years to ensure that the CRE boundary is meeting the design requirements.

6.4.2.4 Interaction with Other Zones and Pressure-Containing Equipment

a. Interaction with Other Zones

There are no potential adverse interactions between the control room complex ventilation zones and adjacent zones that could transfer toxic or radioactive contaminants into the control room. No ducts or ventilation piping from any other zone penetrates the control room envelope.

Only the remote air intakes, makeup air piping and associated isolation valves, and normal makeup air fans and discharge dampers are located outside the control room envelope. This portion of the system is heavy carbon steel piping with primarily welded construction to minimize infiltration. The emergency makeup air and filtration subsystem fans are located within the control room envelope and downstream of their associated filter components. This configuration maintains the filter units at a negative pressure precluding makeup air from bypassing the filter components under emergency conditions.

The exhaust and static pressure control subsystem functions to maintain a positive pressure in the control room with respect to the outside and cable spreading room during normal operations. The pneumatically operated modulating damper in the exhaust ductwork controls the amount of air being exhausted and, thereby, maintains a positive pressure within the control room complex. The damper is under the control of three static pressure sensing devices. The pressure sensing point for the complex is in the HVAC equipment room, which is slightly lower in pressure than the remainder of the control room envelope. The mechanical equipment room is kept at least ⅛" w.g. above the outside atmospheric pressure, the second pressure sensing point, and at least ⅛" w.g. above the cable spreading room at all times, the third pressure sensing point.

Under emergency conditions, the exhaust duct and associated control room envelope penetration are isolated by redundant dampers configured in series.

During normal operation, 1000 cfm of makeup air will be delivered to the control room complex. Approximately 145 cfm will be exfiltrated and the remaining 855 cfm will be exhausted. Under emergency conditions, approximately 600 cfm of makeup air will be delivered to the control room complex all of which will be exfiltrated.

An SSE will not provide an exfiltration path for control room complex air that will negate the ability to maintain a positive pressure since the Diesel Generator Building and Control Building are seismic Category I structures. The intake air equipment and welded piping to the control room complex are also seismic Category I.

b. Pressure-Containing Equipment

The pressure-containing equipment in the Control Room Complex consists of refrigerant lines, Computer Room Halon system, fire extinguishers, instrument air operated equipment, and self-contained breathing apparatus.

The Computer Room Air Conditioning Refrigerant System is nonsafety-related, nonseismic Category I and contains 10 pounds of Refrigerant 22.

The Uniform Mechanical Code, 1976 Edition, Section 1505 allows 22 pounds of Refrigerant 22 per 1,000 cubic feet of occupied space. The control room complex envelope is 246,000 cubic feet. The refrigerant charge of 10 pounds in the Computer Room Air Conditioning System is well below the allowable 22 pounds per 1,000 cubic feet.

The computer room Halon 1301 system is capable of a total concentration of 5 percent of the computer room volume of 13,340 cubic feet. The release of the entire volume of Halon into the control room complex envelope would result in a concentration of 0.30 percent. The National Fire Protection Association Code, Section 12A, lists 5 to 7 percent concentration as producing minimal, if any, central nervous system effects for exposures of approximately five minutes' duration.

Fire extinguishers of various types (Halon, dry chemical, and CO_2) are installed in the Control Room complex. All of the fire extinguishers are Underwriters Laboratories listed and/or Factory Mutual approved.

If all of the Halon fire extinguishers were discharged in the Control Room, the total amount of Halon released would be less than the total amount of Halon that would be released by the Computer Room Halon system. The release of the Computer Room Halon system will not adversely affect Control Room habitability. Therefore, the release of the lesser amount of Halon contained within the fire extinguishers would also not affect the Control Room complex habitability.

The operation of the dry chemical and/or CO_2 fire extinguishers would not affect Control Room habitability due to low volume of gas released from the fire extinguishers as compared to the total volume of the complex.

Control Room self-contained breathing apparatus (SCBA) - The release of the contents of any or all of the SCBA air packs or spare bottles into the Control Room atmosphere would not pose a detriment to Control Room habitability.

Instrument air operated components in the Control Room, primarily air operated dampers, have a normal air consumption rate. This air is released into the Control Room area. A review of these components has determined that the amount of air released is approximately 1 CFM. This is a contributor to the unfiltered inleakage assumed in Control Room dose analysis, [See Appendix 15C.]

The makeup air and exhaust air systems will serve to dilute any gaseous concentration of refrigerants or Halon below the already safe levels.

6.4.2.5 Shielding Design

The design basis loss-of-coolant accident (LOCA) establishes the shielding requirements for the Control Building. The control room shielding design is discussed in Subsection 12.3.2, and is evaluated from design bases LOCA source terms and doses which are presented in Subsection 15.6.5.4.

The external walls and roof of the control room are 2-foot thick reinforced concrete. These shield thicknesses in conjunction with the habitability systems will limit the integrated dose to the operators to less than 5 rem TEDE, in conformance with General Design Criterion 19 of Appendix A of 10 CFR, Part 50.

6.4.3 System Operational Procedures

6.4.3.1 Normal Mode

a. Normal Makeup Air Subsystem

During normal plant operation, the control room normal makeup air subsystem is aligned to deliver approximately 1000 cfm of outside air from both remote intakes (500 cfm per intake). With one normal makeup air fan operating and its associated discharge damper open, the intake isolation valves are positioned to allow equal amounts of air to be drawn from the east and west intakes. The normal makeup air flows through the prefilter and heater for each emergency filter unit and discharges through an orifice into the HVAC equipment room. The heater for each unit operates continuously to limit humidity to less than or equal to 70 percent. The prefilters are periodically replaced when particulate buildup causes the differential pressure across the filters to increase to a predetermined value.

In the event normal makeup air fails or is isolated for reasons other than those delineated in Subsections 6.4.3.2 and 6.4.3.3, appropriate operator action will be taken to re-establish makeup air. If makeup air is lost because of fan failure, the redundant normal makeup air fan and its discharge damper will be manually actuated. If makeup air is lost because of a vital bus outage or failure, or a loss of instrument air supply to the dampers, the emergency makeup air and filtration subsystem will be manually actuated.

b. Emergency Makeup Air and Filtration Subsystem

During normal plant operation, the emergency makeup air and filtration subsystem fans are idle and their associated discharge dampers are closed. Normal makeup air flows through each filter unit's prefilter and heater as discussed above. In the event this subsystem must be manually actuated during normal operation, it functions similarly to emergency mode operation.

c. Exhaust and Static Pressure Control Subsystem

During normal plant operation, the control room exhaust fan is operating and its discharge control damper modulates to maintain the control room complex at a pressure of at least (+) ⅛" w.g. with respect to the outside and adjacent areas. The redundant exhaust isolation damper remains fully open.

d. Cooling/Recirculation Subsystem

The non-safety related control room air conditioning subsystem will normally operate. However the, the safety related Trains may be placed in operation during normal plant operation. In the event of a malfunction in the non-safety related subsystem, or during a loss of offsite power, one of two 100% capacity safety-related trains of control room air conditioning will be placed in service manually. Following a loss of off-site power with the non-safety related subsystem de-energized, one of the two redundant safety related Trains will automatically start via the emergency diesel generator load sequence. Subsection 9.4.1 provides a more detailed description of this subsystem and its operation.

The unit heaters are not required to maintain the operation of the control room. Redundant unit heaters are not provided. If the unit heaters should fail during operation of the control room in the wintertime, the space temperature may drop below the normal comfort temperature. However, heat loads generated internally by electrical equipment would preclude excessively low temperatures.

6.4.3.2 Emergency Mode

a. Normal Makeup Air Subsystem

Following an accident, when high radiation is detected in either remote air intake or when the emergency makeup air and filtration subsystem fans are actuated, the normal makeup air fans automatically trip off and their associated discharge dampers automatically close. The control systems for these fans and dampers are "cross-trained," that is, the discharge damper associated with the Train A fan is controlled by the Train B control loop and vice versa. This configuration ensures isolation of the normal makeup air subsystem by fan trip and/or damper closure regardless of any single active failure.

Detection of smoke in either remote intake is alarmed only. Operator action is required to initiate the filter recirc. mode. Operations may, at their discretion, manually isolate the smoke-contaminated intake and re-establish makeup air from the unaffected intake to the control room complex via the emergency makeup air and filtration subsystem (see Subsection 6.4.3.3).

b. Emergency Makeup Air and Filtration Subsystem

Following an accident, when high radiation is detected at either remote intake or upon generation of an 'S' signal, both redundant emergency makeup air fans and their associated discharge damper are automatically actuated. Although the redundant filter system fans are designed to operate coincidentally and stabley in their parallel configuration, Operations may, at their discretion, shut down one of the systems during the course of the accident. Each filter system may also be initiated manually from the control room at anytime. The subsystem may be initiated manually upon detection of smoke in either remote intake (see Subsection 6.4.3.2).

Each emergency makeup air and filtration subsystem has a nominal capacity of 1100 cfm. This capacity is comprised of 600 cfm makeup air and 500 cfm recirculation air. These system flow rates have been calculated assuming both remote intake isolation valves are open to a throttle position allowing for 300 cfm makeup air from each intake. Following an accident, a contaminated remote intake does not have to be manually isolated. Design base analyses indicate that the makeup air dilution factor (i.e., 50 percent makeup air from "clean" intake, 50 percent makeup air from contaminated intake) and the radioactive particulate and iodine removal capacity of the filters together are adequate to maintain control room doses below allowable limits for the 30-day accident mitigation period.

c. Exhaust and Static Pressure Control Subsystem

Detection of high radiation in either remote makeup air intake or operation of either emergency makeup air and filtration subsystem fan will automatically isolate the exhaust and static pressure control subsystem. Under emergency conditions, the exhaust subsystem remains isolated at all times.

6.4.3.3 Smoke Removal Mode

The control room ventilation intake is provided with smoke detection capability to automatically alarm. Upon receipt of a smoke alarm from either remote intake, the operator will manually initiate the filter recirc. mode. The control room filter recirc. mode signal (CRFRM) will activate the emergency makeup air and filtration subsystem and isolate the normal control room makeup air and the exhaust and static pressure control subsystem. The HEPA filters associated with this system will remove smoke from the incoming air. Manual isolation of the smoke-laden intake can be accomplished by closing the appropriate 1-CBA-V9 or 2-CBA-V9 valve, locally. Additional venting of the control room could be accomplished by opening the doors and using portable exhausters.

6.4.3.4 Isolation Mode

The system operational procedure will be the same as in emergency mode, discussed in Subsection 6.4.3.2.

6.4.4 Design Evaluations

6.4.4.1 Radiological Protection

Radiological protection for the control room operators during accident conditions is provided by the control room shield walls and habitability system. An evaluation of the protection offered by the shielding and habitability systems is presented in Chapter 15 (see Subsection 15.6.5.4, Radiological Consequences of a LOCA).

6.4.4.2 Toxic Gas Protection

As stated in Subsection 2.2.3.1, no significant quantity of toxic gases is stored at any industrial facility in the vicinity of the site. No toxic gases are transported by the Boston and Maine Railroad. The distance to the nearest highway is one mile, so it is unlikely that any toxic chemical spill along a road will endanger control room habitability and, in addition, the control room design and operator training offer inherent protection against the consequences of a spill in the site vicinity.

An evaluation of chemical hazards identified six chemicals onsite that had the potential to provide a hazard to control room habitability. The six chemicals are propane, hydrogen (trailers), sodium hypochlorite (NaOCl), carbon dioxide (refrigerated liquid), nitrogen (gas) and sulfur hexafluoride (SF6). These chemicals were evaluated and it was determined that none of the chemicals would affect maintaining the control room habitable if released under the postulated accident scenarios (Reference 4). This evaluation also addressed potential off-site chemical hazards. No off-site chemical hazards were identified that would affect control room habitability. As part of the Control Room Envelope Habitability Program, an updated list of potential chemical hazards both on and off-site is available.

Toxic gas protection for control room operators in not required since no potential chemical hazards have been identified either on or off-site that could potentially create a hazardous chemical environment in the control room.

6.4.5 Testing and Inspection

During preoperational system testing, air systems are balanced to achieve design flow rates. In addition, operability of all equipment and control functions are verified. Periodic verification of equipment operability and certain system parameters will be performed in accordance with plant Technical Specifications.

Preoperational testing of the emergency filter units and their associated components will be performed in accordance with ANSI-N510 (1980). Subsequent periodic testing of the filter units will be performed in accordance with Technical Specifications which invoke certain ANSI-N510 (1980) in-place testing requirements and acceptance criteria.

Air system pressure boundaries are leak tested following installation.

A Control Room Envelope Habitability Program has been implemented. Four (4) types of tests are required to ensure Seabrook's Control Room is maintained habitable: an integrated tracer gas test, an administrative evaluation, a Control Room Envelope pressure test and equipment TS surveillances.

6.4.6 Instrumentation Requirements

6.4.6.1 Normal Makeup Air Subsystem

The controls for the normal makeup air fans and discharge dampers are located on the main control board (MCB). During normal operations, one of the two normal makeup air fans is running and its associated discharge damper is open.

Isolation of the normal makeup air subsystem is automatically initiated upon detection of high radiation in either remote air intake or upon actuation of the emergency makeup air fans.

Each remote air intake is provided with fully redundant radiation monitoring systems. The east intake radiation monitors are located at the intake. The west intake radiation monitors are located on the intake piping in the Diesel Generator Building. Details of the RDMS system are provided in Subsection 12.3.4.

The control scheme of the normal makeup air fans and dampers is "cross-trained" to ensure automatic isolation regardless of any single active failure, that is, detection of high radiation in either intake by Train A monitor or actuation of the Train A emergency makeup air fan will trip the Train A normal makeup air fan and close the discharge damper downstream of the Train B normal makeup air fan. Detection of high radiation in either intake by either Train B monitor or actuation of the Train B emergency makeup air fan will trip the Train B normal makeup air fan and close the discharge damper downstream of the Train A normal makeup air fan. This "cross-trained" design also ensures that the normal makeup air subsystem automatically isolates on an outage or failure of either vital electrical bus.

Status lights are provided on the MCB for the normal makeup air fans and discharge dampers. Position indication is provided for the remote intake manual isolation valves.

The following alarms are provided at the MCB:

- Control room normal makeup air isolation
- Loss of control room makeup air
- East air intake contaminated - smoke, radiation

- West air intake contaminated - smoke, radiation.

6.4.6.2 Emergency Makeup Air and Filtration Subsystem

The controls for the emergency makeup air fans and dampers are located on the MCB. The fans and dampers are automatically actuated upon detection of high radiation in either remote air intake or upon generation of an 'S' signal. The fans and dampers may also be manually actuated from the MCB.

The filter unit heaters operate continuously.

Status lights for the emergency makeup air fans and discharge dampers are provided on the MCB. Grouped status lights are also provided for the fans.

Differential pressure across each filter component is indicated locally in the control room HVAC equipment room. The temperature for each filter unit is indicated locally. Relative humidity, differential pressure and air flow for each filter unit are indicated and recorded on the station computer.

The following alarms are provided at the MCB:

- High filter unit differential pressure
- High filter temperature
- High filter relative humidity
- High filter carbon monoxide levels (early fire detection)
- High and low filter flow
- Filter isolation damper closed.

6.4.6.3 Exhaust and Static Pressure Control Subsystem

The exhaust and static pressure control subsystem fan and discharge control damper are controlled from the MCB. During normal operations, the fan is running and the control/isolation damper modulates to maintain a positive control room pressure. This modulating feature is controlled by a differential pressure control loop. This control system senses pressure in the Control Room HVAC Equipment Room, Cable Spreading Room and the outside atmosphere. The damper is modulated automatically to maintain the HVAC Room at greater than or equal to ⅛" w.g. positive pressure with respect to atmosphere and the Cable Spreading Room. The redundant exhaust isolation damper (CBA-DP-1058) is fully open during normal plant conditions. Loss of normal makeup air and/or loss of control room pressurization will close CBA-DP-28 and trip CBA-FN-15. Interlocks are provided so that isolation dampers CBA-DP-28, -1058 and exhaust fan CBA-FN-15 are isolated whenever a high radiation signal is present or fans 16A or 16B are running.

Status indication for CBA-FN-15, CBA-DP-28, and CBA-DP-1058 is provided on the MCB. Indication of differential pressure between the control room HVAC room and outside atmosphere is provided in the HVAC room. This differential pressure, as well as the differential pressure between the Cable Spreading Room and the HVAC room, is recorded on the station computer.

The following alarms are provided at the MCB:

- Low control room/outside atmosphere differential pressure
- Low control room/cable spreading room differential pressure
- CBA-FN-15 tripped.

6.4.6.4 Cooling/Recirculation Subsystem

All principal components of the safety related chilled water system (namely water chillers, chilled water pumps and air conditioning unit fans and dampers), except the chiller condenser exhaust fans are controlled from the MCB. The chiller condenser exhaust fans are controled from the control room air conditioning panel located in the control room air conditioning equipment room on elevation 75' within the control room pressure envelope. Additional instrumentation and control features are discussed in Subsection 9.4.1.

The following alarms are provided at the MCB:

- Control room high and low temperature
- Computer room high and low temperature.
- Safety related chiller trouble
- Condenser exhaust fan trip
- Condenser exhaust fan bypassed

6.4.7 References

1. N. Irving Sax, "Dangerous Properties of Industrial Chemicals," 5th Edition, Van Nostrand Reinhold, 1979.
2. "Handbook of Compressed Gases," Compressed Gas Association, Inc., Van Nostrand Reinhold, 1966.
3. "Hydrogen Chloride," Safety Data Sheet SD-39, Manufacturing Chemists Association, Washington, D.C., Quoted in Reference 2; also extracts supplied by personal communication with Miss Mott of AICE.
4. "Evaluation of On-Site and Off-Site Toxic Chemicals - Control Room Habitability 2008 Update (SBC-1069)," Document No. 32-9098365-000 December 18, 2008. Areva NP Inc.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 Engineered Safety Feature (ESF) Filter Systems

The following engineered safety feature filter systems are atmospheric cleanup systems provided to remove fission products and to retain the radioactive material following a design basis accident.

a. Containment Enclosure Emergency Air Cleaning System

b. Fuel Storage Building Emergency Air Cleaning System

c. Control room emergency makeup air and filtration subsystem.

These systems are secondary systems, as defined in NRC Regulatory Guide 1.52.

The following non-engineered safety-feature filter systems are described in other sections of the UFSAR:

a. Primary Auxiliary Building Air Cleaning System (Subsection 9.4.3)

b. Waste Processing Building Air Cleaning System (Subsection 9.4.4)

c. Containment Structure Recirculating Air Cleaning System (Subsection 9.4.5)

d. Containment Structure Purge Air Cleaning System (Subsection 9.4.5)

The fans, ducting and dampers of the containment structure recirculating air cleaning system perform a post-accident H_2 mixing function (see Subsection 6.2.5).

6.5.1.1 Design Bases

a. Containment Enclosure Emergency Air Cleaning System (CEEACS)

The Containment Enclosure Emergency Air Cleaning System is designed to maintain a negative pressure within the containment enclosure following an accident, to remove and retain airborne particulates and radioactive iodine, and to exhaust filtered air to the unit plant vent.

1. The CEEACS is designed to maintain a negative pressure of greater than or equal to 0.25 inches of water, following a design basis accident, in the annular region defined by the containment structure and the containment enclosure, as well as in the additional building volumes associated with the electrical penetration areas, mechanical piping penetration area and engineered safeguard equipment cubicles, so that any fission products leaking from these systems and from the primary containment will be retained in these areas and eventually processed through the filters. In order to ensure a negative pressure of greater than or equal to 0.25 inches of water is maintained at the top of the containment enclosure for the entire range of design outside ambient temperatures, a negative pressure greater than or equal to 0.685 inches of water has to be maintained at the 21' -0" elevation of the containment enclosure. The filter unit also accepts the discharge from the post-LOCA containment hydrogen purging duct, as discussed in Subsection 6.2.5.2.

2. The exhaust capacity is based on a conservative leak rate of 0.20 percent/day of the containment air mass at maximum internal pressure following a design basis LOCA as given in Table 6.5-7. Each containment enclosure exhaust fan is designed to exhaust at the rate of 2100 SCFM, which is equivalent to a volumetric inleakage rate of 325 percent/day from the containment structure to the containment enclosure annulus.

3. Sizing of the high efficiency particulate air filters (HEPA) and carbon adsorbers is based on the volumetric flow rate required to maintain the negative pressure in the containment enclosure annulus and connected penetration and engineered safeguard areas, and for fission product removal capability employing the conservative inventories given in Subsection 15.6.5.

4. The Containment Enclosure Emergency Air Cleaning System is a seismic Category I, Safety Class 2, system.

b. Fuel Storage Building Emergency Air Cleaning System

1. The Fuel Storage Building Emergency Air Cleaning System is designed to maintain a negative pressure of 0.25 inches of water within the Fuel Storage Building while in the irradiated fuel handling mode, to remove and retain airborne particulates and radioactive iodine, and to exhaust filtered air to the unit plant vent following a fuel handling accident.

2. The Exhaust Filter System is designed to remove and retain airborne particulate and radioactive iodine, and to exhaust the filtered air to the unit plant vent following a fuel-handling accident while either or both filters are operating.

3. Sizing of the HEPA filter and carbon adsorbers is based on the volumetric flow rates required to maintain the required negative pressure in the Fuel Storage Building for both the normal fuel handling mode and the fuel handling accident mode, and for fission product removal capability employing the conservative inventories presented in Subsection 15.6.5.

4. The Fuel Storage Building Emergency Air Cleaning System is a seismic Category I, Safety Class 3 system.

c. Control Room Emergency Makeup Air and Filtration Subsystem

The control room emergency makeup air and filtration subsystem is designed to supply makeup air from two remotely located intakes to the control room complex following an accident and/or a release of radiological contaminants or smoke, to maintain a positive pressure within the control room complex, and to remove and retain airborne particulates and radioactive iodines from all makeup air and a portion of recirculated air.

1. The control room emergency makeup air and filtration subsystem is designed to maintain a positive pressure of greater than or equal to 0.125 inches of water in the control room complex relative to the outdoors and to the cable spreading room. The system will operate following a design basis accident or other abnormal operating scenarios such as smoke contamination of a remote air intake (see Section 6.4 for additional details). The control room complex occupies the entire 75'-0" elevation of the Control Building and includes the main control room area, computer room, technical support center, offices, conference room and library, emergency storage room, HVAC equipment room, kitchen and sanitary facilities, as shown in Figure 1.2-32. The positive pressure will preclude infiltration of radiological and other hazardous contaminants to maintain habitability of the complex for a safe plant operation and shutdown, as necessary.

2. The makeup air portion of the nominal filter capacity (600 cfm) is conservatively greater than the minimum required ventilation rate for occupancy/breathing purposes (175 cfm) and the calculated complex outleakage at +0.125" W.G. (165 cfm). The recirculation air portion of the nominal filter capacity (500 cfm) is based on a desired filter system decontamination factor (see Subsection 15.6.5).

3. Sizing of the high efficiency particulate air (HEPA) filters and carbon adsorbers is based on the volumetric flow rate required to maintain the positive pressure within the control room complex, to satisfy ventilation requirements for occupancy/breathing purposes, and for fission-product removal capability employing the conservative inventories given in Subsection 15.6.5.

4. The control room emergency makeup air and filtration subsystem is a seismic Category I, Safety Class 3 system, except for some instrumentation which does not provide vital control or monitoring.

6.5.1.2 System Design

a. Containment Enclosure Emergency Air Cleaning System

The filter system consists of redundant filter trains, fans, dampers and controls and a common ductwork system. The air flow required to maintain a negative pressure in the Containment Enclosure Building is passed through demisters, which also function as prefilters, and through HEPA filters located both upstream and downstream of the carbon filter prior to exhausting through the plant vent (see Figure 9.4-7 for an air flow diagram).

A ductwork cross-connection is provided between the two filter trains at a point between the downstream HEPA filter and the fan inlet. Should the operating fan fail, this cross-connection will insure a continued air flow by manual startup of the redundant fan.

Each redundant filter train is complete, separate and independent from both electrical and control standpoints. Each filter train fan is supplied power from an independent ESF power train source, which will furnish power to its fan during abnormal and post-accident conditions. The operation of mechanical equipment is controlled and monitored in the plant unit control room, as discussed in Section 7.3.

The HEPA filters have a certified test efficiency of 99.97 percent based on DOP smoke test. For impregnated carbon filter efficiencies see Table 6.5-4. The evaluation of offsite effects due to potential accidents has been made in accordance with Appendix 15B, assuming minimum carbon filter efficiencies of 85 percent for organic iodines and 95 percent for elemental iodines for the conservative case. The carbon filters use a deep bed design which provides a gas residence time of approximately 0.5 second.

b. Fuel Storage Building Emergency Air Cleaning System

Separate redundant carbon filter exhaust systems are provided for filtering the building air prior to its exhaust to the unit plant vent. One carbon filter train will be operated whenever irradiated fuel not in a sealed cask is being handled. The filter unit, together with dampers and controls, will maintain the Fuel Storage Building at a negative pressure with respect to atmosphere so that any airborne particulate or radioactive iodine will be retained in the building and eventually processed through the filters (see Figure 9.4-4 for an air flow diagram).

The Filter Cleanup System consists of redundant filter trains, redundant fans, ductwork, dampers and controls. Each filter train consists of demisters, which also function as prefilters, heaters, medium efficiency filter, HEPA filters both upstream and downstream of the guard bed and carbon filters.

A ductwork cross-connection is provided to connect the two filter train systems at a point between the downstream HEPA filter and the fan inlet. Should the operating fan fail, this cross-connection will ensure continued air flow through a partially loaded or fully loaded filter by manual startup of the redundant fan. The cross-connection is sized so the temperature of a fully loaded filter bed will not rise above 200°F (see Subsection 6.5.1.3c).

Each redundant filter train is complete, separate and independent, from both electrical and control standpoints. Each filter train is supplied power to its fan during abnormal and post-accident conditions. The operation of mechanical equipment is controlled and monitored in the plant unit control room, as discussed in Section 7.3.

The HEPA filter manufacturer's minimum test efficiency of 99.97 percent is based on the DOP smoke tests. The impregnated carbon filters have a manufacturer's rated minimum test efficiency of 97 percent for methyl iodide (CH_3I-131) and 99.9 percent for elemental iodine as delineated in Table 6.5-5. The carbon filter has a deep bed design having a gas residence time of 0.5 seconds.

Prior to the start of handling of irradiated fuel not in a sealed cask, one exhaust filter system is placed in operation, the normal exhaust system is shutdown, and the normal exhaust damper is closed. Therefore, the system will be in operation should a fuel handling accident occur.

c. Control Room Emergency Makeup Air and Filtration Subsystem

The filter system consists of redundant filter trains, fans, dampers, and controls. The redundant filter trains share a common suction makeup air piping system. The makeup air pipe penetrates the floor of a plenum integral to the Train A filter unit and upstream of the internal filter components. This plenum is connected to a systematically similar suction plenum integral to the Train B filter unit via ductwork. Both filter units and the connecting ductwork are located entirely within the control room complex. Only the makeup air pipe penetrates and enters the complex envelope.

Each filter train consists of an isolation damper, a medium efficiency prefilter, an electric air heater, an upstream HEPA filter bank, a carbon adsorber section, a downstream HEPA filter bank, a backdraft damper, a fan, and a discharge control damper. Each filter train is also designed with an orifice plate downstream of the air heater and upstream of the first HEPA filter bank for bypass/recirculation air flow. During normal operations, when the filter system is idle, makeup air passes into the suction plenum of each filter, flows through the prefilter and heater, and discharges into the control room HVAC equipment room via the orifice. Under emergency conditions when the filter system is operating, makeup air is drawn into each filter suction plenum, through the prefilter and heater, and then mixes with a portion of the recirculation air which enters via the orifice plate. This air mixture then flows through the HEPA-carbon-HEPA configuration, and is discharged into the HVAC equipment room via the filter train's associated fan and dampers. The electric heater for each filter operates continuously to ensure that the relative humidity within the carbon adsorber banks is equal to or less than 70 percent at all times.

A pipe with a manual isolation valve is provided to cross-connect the two filter units at a point downstream of the carbon adsorber sections. The isolation valve is normally open/throttled to a preset position. During single train operation, this alignment will provide a small amount of air flow through the carbon adsorber section of the inoperable train. This air flow will remove decay heat satisfying fire protection concerns. The isolation valve will be closed for train isolation in the event a fire is detected in one of the units.

Each redundant filter train for both electrical and control requirements is complete, separate, and independent. Each filter train fan, air heater, and essential instrumentation and control components are powered from an independent ESF power train source, which will furnish power to the essential components during abnormal and accident conditions. Operation of the various system components is controlled and monitored in the plant unit control room, as discussed in Subsection 6.5.1.5c and Section 7.3.

The HEPA filters have a certified test efficiency of 99.97 percent based on DOP smoke test. The carbon adsorber efficiencies are provided in Table 6.5-6. Filter efficiencies utilized to evaluate post-accident control room doses are provided in Appendix 15B. The conservative analysis assumes minimum filter efficiencies of 95 percent for organic and elemental iodines and 99 percent for particulate iodines. The carbon adsorber design provides an average gas residence time of 0.25 seconds.

6.5.1.3 Design Evaluation

a. Containment Enclosure Emergency Air Cleaning System (CEEACS)

The containment enclosure exhaust filter trains are redundant, to insure the maintenance of a negative pressure in the containment enclosure and related areas and to insure cleanup of the exhaust air following an accident. All safety-related equipment and ductwork supports have been designed and seismically analyzed to withstand and function through a Safe Shutdown Earthquake (SSE). The system is designed to limit offsite post-accident doses to values below those specified in 10 CFR 100 (see Subsection 15.6.5 for evaluation of system performance). A single component failure will not result in loss of function of this ESF system.

In the unlikely event that an accident requiring filter operation occurs, both of the redundant filter train fans will be automatically started on the "T" signal (see Drawing NHY-503515) to provide an air flow velocity of approximately 45 fpm through their associated carbon filter beds. In the further unlikely event of failure of one operating fan, the ductwork cross-connection will provide redundant air flow from the redundant fan across the partially loaded or fully loaded filter bed.

See Section 6.2.3.3.a for a description of the analyses performed to demonstrate the capability of the system to draw down the Containment Enclosure Building to the design negative pressure within the design basis draw down time.

HEPA filters and carbon adsorbers were tested at the expected accident environmental conditions for this secondary system. Results indicated no degradation of filtering efficiency. Subsection 15.6.5 analysis conservatively assumes lower efficiencies.

The systems are designed to meet the intent of Regulatory Guides 1.4 and 1.52. See Table 6.5-1 for a discussion relative to conformance with Regulatory Guide 1.52, Rev. 2.

b. Fuel Storage Building Emergency Air Cleaning System (FSBEACS)

The fuel storage building exhaust filter trains are redundant to insure cleanup capability and the ability to maintain a negative pressure following a fuel handling accident. All safety-related air handling equipment, equipment support and ductwork supports are designed to operate during and following an SSE. The system is designed to limit offsite post-accident doses to values not exceeding the requirements of 10 CFR 100 (see Chapter 15). Loss of one emergency exhaust filter train will not prevent the safety function from being performed. During fuel handling, only one set of filters and fan will normally be operating. In the unlikely event of an accident, the second set of filters and fan can be manually started to provide redundancy. The operating filter will provide an air flow velocity of approximately 40 fpm through its associated carbon filter bed. In the further unlikely event of failure of the operating fan, the ductwork cross-connection will provide redundant air flow across the partially loaded or fully loaded filter bed. [Note]

HEPA filters and carbon adsorbers have been tested at the expected accident environmental conditions for this secondary system. Results indicated no degradation of filtering efficiency; however, conservative parameters based on Regulatory Guide 1.25 were used in the conservative analysis in Subsection 15.7.4.

The systems are designed to meet the intent of Regulatory Guides 1.25 and 1.52. See Table 6.5-2 for a discussion of conformance with Regulatory Guide 1.52, Rev. 2.

c. Control Room Emergency Makeup Air and Filtration Subsystem

The control room emergency filter trains are fully redundant to ensure the maintenance of a positive pressure within the control room complex, the provision of adequate supply makeup air for breathing/occupancy purposes, and the filtration of this makeup air and a percentage of recirculation air following an accident or smoke contamination of an intake given any single active failure. All safety-related passive and active components have been designed and supported to withstand and function during and following a Safe Shutdown Earthquake (SSE).

[Note] All drawings referenced in this section were provided under a separate submittal to the NRC (see Section 1.7).

The system is designed to limit post-accident control room doses to values below those specified in 10 CFR 50, Appendix A, GDC 19 and Section 6.4 of the Standard Review Plan (see Subsection 15.6.5 for accident dose analyses).

In the unlikely event that an accident requiring filter system operation occurs, both of the redundant filter train fans and discharge control dampers will be automatically actuated upon generation of an "S" signal or detection of high radiation in either remote air intake (see Subsection 6.5.1.5c).

The following analysis has been performed to demonstrate the capability of the system to maintain the control room complex at a positive pressure of 0.125 inches of water gauge:

1. Air Out-Leakage Analysis

   A calculation was performed to determine the control room complex envelope air out-leakage through various air flow paths such as electrical penetrations, concrete structure, construction joints, doors, and the worst-case exhaust isolation damper. The analysis was developed utilizing vendor information, ASHRAE data and methodology, and analytical and experimental leakage data provided in "Conventional Buildings for Reactor Containment-NAA-SR-10100(1965)," issued by Atomics International, a division of North American Aviation Incorporated.

   The calculated maximum out-leakage is approximately 165 cfm at a control room complex positive pressure of 0.125" W.G. The nominal makeup air capacity of the emergency makeup air and filtration subsystem is 600 cfm.

   The Control Room Emergency Air and Makeup Filtration System is controlled by the Control Room Envelope (CRE) Integrity Program. This program ensures the control room and its Operators are in a condition to assure public safety during all modes of operation. It provides measures to allow the control room habitability systems (CRHS) to maintain a habitable environment for operators under normal and abnormal conditions, and identifies compensatory measures if habitability is in question.

d. Analysis of Heat Generation in Charcoal Beds

A detailed analysis was performed to determine the highest attainable temperatures which could be reached by the charcoal beds in the Containment Enclosure System, Fuel Storage Building Filter System, and control room emergency makeup air subsystem due to decay heat from accumulated fission products resulting from a design basis accident. The filter loadings and other parameters used in the analysis for each filter system, as well as the results, are given below.

1. Pertinent Parameters for Analysis

Item	CEEACS	FSBEACS	CREMAFS
Initial Filter			
Temperature	100°F	100°F	105°F
Thermal	0.55 Btu/	0.55 Btu/	0.55 Btu/
Conductivity	hr ft °F	hr ft °F	hr ft °F
Specific Heat	0.242 Btu/lb °F	0.242 Btu/ lb °F	0.242 Btu/ lb °F
Energy Absorption			
Coefficient	3.05/ft	-	3.05/ft
Charcoal Density	0.38 gm/cc	0.38 gm/cc	0.38 gm/cc
Element Length	61 inches	61 inches	*30 inches/ 21 inches
Width	66 inches	120 inches	*24 inches/ 38 inches
Thickness	4 inches	2 inches	* 2 inches/ 4 inches
Number of Filter			
Elements	4	7	* 6/4

*1-CBA-F-38/1-CBA-F-8038

2. Filter Loading

Isotope	CEEACS	FSBEACS	CREMAFS
I-131	3.3×10^{3}	6.3×10^{2}	1.89×10^{-1}
I-132	4.1×10^{0}	nil	2.04×10^{-1}
I-133	3.4×10^{3}	6.8×10^{1}	3.66×10^{-1}
I-135	6.0×10^{2}	5.0×10^{-2}	3.16×10^{-1}

3. Results

This analysis indicates that desorption temperatures will not be reached for the Containment Enclosure Emergency Air Cleaning System, the control room emergency makeup air subsystem and the fuel storage building emergency air cleaning system charcoal filter beds, even for the case of no air flow due to fan failure and unavailability of air flow via the ductwork cross-connections between the two redundant filter trains of each system.

6.5.1.4 Tests and Inspections

a. Containment Enclosure Emergency Air Cleaning System

Fan discharge ductwork was leak-tested during installation. Air system balancing and adjustment to design air flow was completed prior to system preoperational testing.

The entire system underwent preoperational testing as described in Subsection 14.2.11.

Periodic testing will verify the ability of the system to maintain the containment annulus at the required negative pressure. Periodic filter testing will be performed to meet the intent of Regulatory Guide 1.52, as clarified in Table 6.5-1.

b. Fuel Storage Building Emergency Air Cleaning System

Fan discharge ductwork was leak-tested during installation. Air system balancing and adjustment to design air flow was completed during plant preoperational testing.

The installed systems underwent preoperational testing, as described in Subsection 14.2.11, prior to storage of fuel in the Fuel Storage Building.

Prior to refueling operations, the ability of each fan to maintain a fuel storage building negative pressure greater than or equal to 0.25" H_2O will be verified. Periodic filter testing will be performed to meet the intent of Regulatory Guide 1.52, as clarified in Table 6.5-2.

c. Control Room Emergency Makeup Air and Filtration Subsystems

Fan discharge ductwork was leak-tested during installation. Local leak testing is performed at potential leak paths on the makeup air piping (e.g., damper frame seals, cover plate gaskets, bolted flange seals, etc.). Air system balancing and adjustment to design air flow was completed prior to system preoperational testing.

Acceptance and periodic filter testing will be performed in accordance with Regulatory Guide 1.52 except that the 1980 issues of ANSI N509 and ANSI N510 will be utilized (see Table 6.5-3).

Testing will be done following the Control Room Envelope Habitability Program. Four (4) types of tests are required to ensure Seabrook's Control Room is maintained habitable: an Integrated Tracer Gas Test (performed every six years), an Administrative Evaluation (performed every 18 months), a Control Room Envelope Pressure Test (performed every 18 months), and equipment TS surveillances (performed as specified by Technical Specifications).

6.5.1.5 Instrumentation Requirements

a. Containment Enclosure Emergency Air Cleaning System

This post-accident cleanup system is designed to function automatically upon receipt of an engineered safety feature actuation "T" signal (containment isolation phase - A). Manual controls for the fans and dampers are also provided at the MCB (see Drawing NHY-503515). The containment enclosure (CE) pressure differential deviation is alarmed at the MCB. The discharge flow of the system is indicated locally, as well as recorded at the MCB. High and low flow alarms are also provided at the MCB. A high pressure drop condition across each filter train is recorded and alarmed at the MCB. In addition, the pressure differential of each filter component is indicated locally. Carbon adsorber discharge air high temperature and inlet air high humidity are alarmed at the MCB. Air temperature upstream of carbon adsorber and downstream of second HEPA filter is indicated locally.

b. Fuel Storage Building Emergency Air Cleaning System

The operation of the Fuel Storage Building (FSB) Emergency Air Cleaning System is controlled and monitored in the plant control room (see Drawing NHY-503543). FSB supply air is controlled through DP 13A/B (see Drawing NHY-503541). During fuel handling operations, the FSB is maintained at a slightly negative pressure with respect to the atmosphere through the FSB exhaust system (see Drawing NHY-503542). FSB high and low differential pressure is alarmed at the MCB. Also, the FSB temperature and relative humidity are monitored, and high deviations are alarmed at the MCB. Each filter train is monitored for differential pressure and high deviations are alarmed at the MCB. Each cleaning unit is provided with a temperature switch which alarms high temperature on the MCB. Temperature switches for automatic control of cleaning unit heaters are also provided. Independent low-flow instrumentation is provided to alarm at MCB the discharge air flow through each of the redundant air cleanup filter units (see Drawing NHY-506452). In addition, the differential pressure across each individual filter unit is indicated locally. Local temperature indicators are also provided both upstream and downstream of the carbon filters.

c. Control Room Emergency Makeup Air and Filtration Subsystem

The emergency makeup air and filtration subsystem fans and discharge dampers are also controlled from the MCB. In the auto mode, CBA-DP-27A and 27B will open upon receipt of a remote intake high radiation signal or a safety injection ("S") signal. Opening of these dampers will automatically start associated fans CBA-FN-16A and 16B. A high radiation signal or starting fans 16A and 16B will trip CBA-FN-27A, 27B, and 15 and isolate dampers CBA-DP-53A, 53B, 28 and 1058. Each filter air heater operates continuously. High and high-high temperature controls are provided. The high-high temperature control requires local/manual reset. Status lights are provided at the MCB for the fans and discharge dampers. Status lamps are also provided for the fans in the Accident Monitoring Instrumentation arrays.

Differential pressure indication across each filter component is provided locally. High differential pressure across each filter unit generates a VAS alarm at the MCB. The temperature for each filter unit is indicated locally. High temperature generates a VAS alarm. Relative humidity for each filter unit is recorded on the station computer. High and low air flow generate VAS alarms. Two carbon monoxide detectors per filter (one each at the inlet and outlet of the carbon adsorber banks) provide for early fire detection. These devices monitor the filtration systems and generate an alarm in the control room.

6.5.1.6 Materials

Listed in Table 6.5-4, Table 6.5-5 and Table 6.5-6 by commercial name, quantity and chemical composition are the materials used in or on the filter systems. Extremes in temperature or radiation that could promote radiolytic or pyrolytic decomposition of materials are not applicable to the Containment Enclosure Building or the fuel storage for normal or postulated accident conditions.

6.5.2 Containment Spray System

This section presents a detailed description of the fission product removal function of the Containment Building Spray (CBS) System.

6.5.2.1 Design Bases

The spray system provides for iodine removal to limit the consequences of a LOCA to the limits of 10 CFR 100 by providing a rapid reduction in containment elemental iodine concentration following a LOCA. This greatly reduces the amount of iodine that can leak from the primary containment into the containment enclosure structure. The combination of iodine removal by the sprays and the containment enclosure exhaust filters limits the offsite radiological consequences resulting from the design basis accident. These consequences and the system performance are discussed in Subsection 15.6.5.

Table 15.6-17, prepared in accordance with Regulatory Guide 1.4, lists the fission products that the CBS system is designed to remove during a LOCA. A discussion of the extent to which credit is taken for fission product removal by the CBS system is included in Subsection 15.6.5.

The system uses the principles of hydrodynamic equilibrium to provide a reliable means for buffering the spray solution with an iodine removal agent. The spray rings are designed to provide a high degree of spray coverage to maximize the spray effectiveness for iodine removal.

6.5.2.2 System Design (for Fission Product Removal)

Iodine removal inside the containment following a LOCA is accomplished by 3010 gpm of spray (assuming only one train available) with a boric acid-sodium hydroxide spray solution. The design details of the spray system are presented in Subsection 6.2.2.

The spray system initially takes suction from the refueling water storage tank (RWST) and continues to add a fresh spray additive solution into the containment until the inventory of the RWST is depleted. Upon a low level in the RWST, the suctions of the spray pumps are automatically transferred to the containment sump. The time of transfer is a function of the number of emergency core cooling pumps and containment spray pumps in operation. At design flow rates and minimum pumps in operation (3575 gpm injection and 3010 gpm spray) the transfer to the recirculation mode takes place in 53.7 minutes, assuming a 10-second delay for injection and a 62 second delay in spray flow. With maximum safeguards in operation (6600 gpm injection and 6600 gpm spray), the transfer to the containment sump occurs at approximately 26 minutes, assuming a 10-second delay for injection and a 62 second delay in spray flow. The spray pumps remain in operation for as long as is necessary to control the containment pressure within the required limits. The maximum delay in delivery of the NaOH solution to the spray nozzles is 2.47 minutes.

The chemical additive for the spray is stored in the spray additive tank (SAT) located adjacent to the RWST. When a containment spray actuation signal occurs, two valves in parallel lines provide redundant flow paths for supplying the chemical additive to the RWST and spray pump suction. The chemical additive is stored as a 20 wt percent NaOH solution. The chemical spray additive flows by gravity into a mixing chamber in the RWST. The design is such that the pH of the solution leaving the mixing chamber is averaging between 9.0 and 9.6. The total amount of chemical supplied will result in a containment sump liquid having a pH range between 8.8 and 9.4 during the recirculation phase.

The spray is delivered to the containment through 198 SPRACO 1713A nozzles per flow train, each having a flow rate of 15.2 gpm at containment design pressure and 40 psi differential operating pressure. The nozzles produce a drop size spectrum with a conservatively estimated volumetric drop diameter of 1000 microns.

The nozzles are spaced to provide a uniform spray pattern across the containment cross section. The nozzles of the redundant spray trains are uniformly spaced between the nozzles of the other spray train, so that either subsystem will provide uniform coverage. The SPRACO 1713A nozzles have been used extensively in other nuclear plants and various iodine removal experiments.

The location of the spray nozzles in the dome of the containment is shown on Figure 6.2-76 and Figure 6.2-77. The flow weighted average fall height of the spray drops is a minimum of 134 ft. for headers 1 and 2. For headers 3 and 4, the average fall height is 144 ft. The operating floor is at an elevation of 25 ft.

The following is a tabulation of important spray iodine removal parameters:

Spray fall height	134 ft
Total containment free volume	2.715×10^6 ft^3
Sprayed containment free volume	2.310×10^6 ft^3
Containment pressure, design	52.0 psig
Containment temperature, design	296°F
Spray flow rate per train design/minimum	3010/2808 gpm
Spray solution pH, minimum	9.0
Average	9.0 to 9.6
Spray additive flow rate	
Maximum safety injection	240 gpm
Minimum safety injection	150 gpm

These parameters are chosen to minimize credit taken for iodine removal by the spray system and provide a conservative basis for calculating offsite doses.

Two headers are provided for each 3010 gpm train. Headers No. 1 and 3 are supplied by one of the CBS pumps while Headers No. 2 and 4 are supplied by the other pump.

Header No. 1 has 118 nozzles spaced approximately 3°apart. Fifty-nine of these nozzles are oriented vertically downward. The remaining 59 are oriented at 45° on the inside of the header pointing inward and downward.

Header No. 3, which is the companion header to header No. 1, has 80 nozzles spaced approximately 4° 30' apart. Forty of these nozzles are at a 45° angle pointing downward toward the containment wall, 16 are at a 45° angle pointing downward and inward, 20 nozzles point directly downward and 4 point horizontally toward the containment vertical centerline.

Header No. 2 contains 162 nozzles spaced approximately 2° 15' apart. Eighty are mounted at a 45° angle pointing downward toward the containment wall, 41 point directly downward and 41 are at a 45° angle pointing inward.

Header No. 4, which is the companion header to header No. 2, contains 36 nozzles spaced approximately 10° apart. Eighteen are mounted at a 45° angle pointing downward toward the containment wall. The remaining 18 point directly downward.

Figure 6.2-84, Figure 6.2-85 and Figure 6.2-86 show the spray drop distribution, accumulated number percentage, and accumulated volume percentage for the SPRACO No. 1713A nozzle. These distributions are the basis for evaluation of iodine removal effectiveness.

Approximately 85 percent of the containment net free volume is covered by spray. Figure 6.2-80, Figure 6.2-81, Figure 6.2-82, Figure 6.2-83 show plan and elevation views of the expected spray patterns.

Table 6.2-80 lists the regions of sprayed volumes in the containment and the volume of each sprayed region. It can be seen from Figure 6.2-80 and Figure 6.2-81 that more than 95 percent of the cross section of the containment at the operating floor (elevation 25') is covered by spray.

6.5.2.3 Design Evaluation

The Containment Spray System has been evaluated by both conservative and realistic models. Subsection 15.6.5 presents the parameters used in both the conservative and realistic analyses and the results of these analyses.

The initial spray solution will be boric acid buffered with sodium hydroxide to a pH of approximately 9.0 to 9.6. Sump pH after recirculation commences will be above 8.5 to assure high iodine partition factors and prevent re-evolution of iodine from the spray solution.

Using the method described by Parsly (Reference 1) a fresh spray drop is calculated to be less than 10 percent saturated during its fall from the spray nozzles. The removal of iodine should therefore be controlled by gas phase resistance. Steam condensation and drop coalescence have been discussed by Pasedag and Gallagher (Reference 2) and by Parsly (Reference 1). Pasedag shows that the total reduction in drop mass transfer surface area over a 100-foot fall due to these phenomena is only about 10 percent. Parsly recommends for calculational purposes that a mean drop diameter of 25 percent larger than the actual mean be used to conservatively compensate for coalescence and size distribution. The calculations in Subsection 15.6.5 assume a mean drop diameter of 1250 microns, which conservatively estimates the iodine removal half-life.

Table 6.2-80 lists the sprayed regions in the containment. The spray headers are designed to directly spray approximately 85 percent of the containment free volume. In addition the unsprayed areas are designed to allow, as much as possible, the free exchange of air with the sprayed area to prevent the pocketing of fission products.

Due to the relatively low surface to volume ratio in the reactor containment it is not expected that wall effects will play a large part in the removal of iodine by spray solution. No wall effects are included in the calculation in Subsection 15.6.5.

The mathematical models used in calculating iodine removal by spray are presented in Subsection 15.6.5. These models assume that the spray removal function is effective throughout the containment sprayed volume and that the effectiveness is constant. Credit for mixing between the containment sprayed and unsprayed volumes is described in Subsection 15.6.5, in conjunction with the discussion of the multi-compartment spray model used to calculate iodine removal rate constants.

6.5.2.4 Tests and Inspections

Subsection 6.2.2.4 discusses the tests and inspections performed to verify the functional capability of the spray system and components, including active valves, pumps, and the spray nozzles, to deliver the required flow for containment heat removal.

Demonstration of those drawdown characteristics of the SAT and RWST which provide flow at the proper pH for effective iodine removal is also discussed in Subsection 6.2.2.4.

The spray additives are sampled on a periodic basis to verify their continued state of readiness.

6.5.2.5 Instrumentation

The system is provided with instrumentation and control to allow the operator to monitor the status and operation of the Containment Spray System from the control room to allow the automatic or manual initiation of the injection mode of operation. The instrumentation details are presented in Subsection 6.2.2.5.

The SAT is steam heated, with the steam flow controlled by a temperature controller, which modulates a pneumatic-operated flow control valve. Low temperature of the tank is alarmed at the main control board (MCB). The tank level is indicated at the MCB. Refueling water storage tank instrumentation is described in Subsection 6.3.5.

Each of the SAT discharge valves may be opened either manually by the operator or automatically upon receipt of the containment spray signal.

The valve positions are indicated at the MCB, above the control switches and also as a part of a separate status monitoring indication system.

6.5.2.6 Materials

The spray solution is stored in the spray additive tank (SAT), and the RWST holds a boron solution. Both tanks are constructed of stainless steel which has been shown to be resistant to chemical attack by the respective stored solutions of 2400 to 2600 ppm boron (as boric acid) and 19 to 21 percent by weight of sodium hydroxide. The spray solution is not susceptible to radiolytic or pyrolytic decomposition under the conditions anticipated in the post-accident environment. The corrosion properties of the spray solution are discussed in detail in Subsection 6.2.2.2.

6.5.3 Fission Product Control Systems

This section provides a discussion of the operation of all fission product control systems following the design basis loss-of-coolant accident (LOCA).

6.5.3.1 Primary Containment

The primary containment is a reinforced concrete structure with a hemispherical dome. The concrete thicknesses of the vertical wall and the dome are 4½ feet and 3½ feet respectively. A welded steel liner plate is anchored to the inside face of the concrete to function as a leaktight membrane. Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, Figure 1.2-5, Figure 1.2-6 show typical sections through the containment and the location of the hydrogen purge system equipment and ductwork. Table 6.5-7 summarizes the operation of the containment following a LOCA. Calculation of offsite doses following a LOCA is discussed in Subsection 15.6.5. For the conservative case, 0.20 percent of the containment air mass is assumed to leak to the containment enclosure during the first 24 hours following an accident, and 0.10 percent/day thereafter. For the realistic case, 0.10 percent of the containment air mass is assumed to leak during the first 24 hours and 0.05 percent/day thereafter. The offsite doses presented in Subsection 15.6.5 are based on a containment free volume of 2.715×10^6 ft^3.

The iodine removal function and effectiveness of the Containment Building Spray System is discussed in Subsection 6.5.2. This system will begin operation within 62 seconds after receipt of a LOCA-generated "P" signal, as described in Subsection 6.2.1.1.

The function of the containment isolation systems is discussed in Subsection 6.2.4.

No credit is taken for iodine removal by the Containment Online Purge System since it is only operated intermittently during normal plant operation and will isolate on a containment isolation signal if operating at the onset of an accident. Radiological consequences of this occurrence are addressed in Subsection 15.6.5.

The combustible gas control system hydrogen recombiners, permanently located inside the containment, are designed to be operational within seven days following a DBA as described in Subsection 6.2.5. Should both recombiners be inoperable for 50 days after the DBA, hydrogen concentration in the containment will be controlled by use of the hydrogen purge line to the plant vent via the Containment Enclosure Emergency Cleanup System described in Subsection 6.5.1.

6.5.3.2 Secondary Containment

The secondary containment is comprised of a reinforced concrete cylindrical structure with a concrete hemispherical dome, the Engineered Safety Features (ESF) equipment cubicles, and the pipe and electrical penetration areas.

The release of airborne contamination following an accident due to leakage into the containment enclosure (the secondary containment) is controlled by a filtered exhaust system which maintains a subatmospheric pressure in each subcompartment. The Containment Enclosure Emergency Cleanup System (CEECS) is the only fission product control system in the secondary containment. This system directs a nominal 2000 cfm of charcoal filtered exhaust air to the plant vent. Actual exhaust flow rate will be the sum of the primary containment leakage (see Subsection 6.5.3.1) and the inleakage from the surrounding environment. See Section 6.2.3.3.a for a description of the analyses performed to demonstrate the capability of the system to draw down the Containment Enclosure Building to the design negative pressure within the design bases draw down time. Iodine removal efficiency of the charcoal beds is assumed to be 85 percent for organic iodide and 95 percent for inorganic iodine for the conservative case; for the realistic case, the iodine removal efficiency is assumed to be 95 percent for organic iodines and 99 percent for inorganic iodines. CEECS fans are shown in Figure 6.5-1.

The air recirculation system for the ESF cubicles serves only for cooling and provides no fission product control function; therefore, no mixing fraction was assumed. The gross volume of each compartment in the containment enclosure is listed in Table 6.5-7.

Anticipated and conservative assumptions concerning the fission product control functions of the secondary containment and the time sequence of events assumed in offsite dose estimates are discussed in Chapter 15.

Figure 9.4-5, Figure 9.4-6, Figure 9.4-7 and Figure 9.4-8, show each secondary containment area, the ventilation system associated with that area and the locations of exhaust intakes.

There are no non-ESF filter systems used to control pressure and fission product release in these areas.

6.5.4 References

1. Parsly, L.F., "Design Considerations of Reactor Containment Spray Systems - Part VII, A Method for Calculating Iodine Removal by Sprays," ORNL-TM-2412 - Part VII, Feb. 1970.

2. Pasedag, W.F. and Gallagher, J.L., "Drop Distribution and Spray Effectiveness" Nuclear Technology, Vol. 10, April 1971.

6.6 IN-SERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

The in-service inspection program for Class 2 and 3 components is defined in the Technical Specifications. In general, this program meets all the requirements of the ASME Code, Section XI, Edition and Addenda as required in 10 CFR 50.55a. The references to ASME Section XI contained in paragraphs 6.6.6 are from the 1983 Edition Summer 1983 Addenda.

6.6.1 Components Subject to Examination

The welds selected for examination in Class 2 piping systems have been determined in accordance with the requirements of the 1974 Edition of ASME Section XI with Addenda through Summer of 1975 as allowed or required by 10 CFR 50.55a, Paragraph (b)(2).

All Class 2 components that do not meet the exemption requirements of IWC-1220, except the containment spray piping beyond the last downstream valve, will be examined in accordance with the requirements of IWC-2000, using the methods listed in Table IWC-2500-1 or the alternatives allowed by the Code Case entitled "Alternative Rules for Examination of Class 2 Piping."

Containment spray piping beyond the last downstream valve is exempted from in-service inspection for the following reasons:

a. These lines are normally empty and are required to function only in the unlikely event of a major LOCA or a main steam line break.

b. These lines are not subject to fluctuating stresses, so that propagation of cracks due to defects in the pipe is unlikely.

c. Spray piping welds are examined by radiography during fabrication, so that large defects that could cause catastrophic failure during system operation will be detected and repaired during fabrication and installation of the piping.

d. This piping contains hundreds of open nozzles so that splits or cracks which do not directly cause catastrophic failure would not have a marked effect on the function of the system.

e. System pressure is low, so that pressure stresses in the piping would be unlikely to cause failure in the presence of small defects.

All Class 3 components shall be examined per the requirements of Subsection IWD.

Safety classifications of all components are presented in Subsection 3.2.

6.6.2 Accessibility

The design and arrangement of Class 2 and Class 3 components provides adequate clearances to conduct the required examinations at the Code-required inspection interval. In general, access has been provided for examination of welds by proper layout of piping and equipment. Where surface or volumetric examination is required, insulation has been designed to be quickly and easily removed and replaced. Inaccessible Class 3 lines have been provided with taps to permit pressure decay tests in lieu of visual examination, as permitted in IWD-2000. Provisions for pump tests, as required by IWP-2000, have been made by incorporating suction and discharge pressure-measuring devices or flow measurement orifices adjacent to each safety class pump. Clearances around the pumps have been maintained to permit access for rotation speed and vibration measurements. Provisions for valve tests, as required by IWV-2000, have been incorporated by including valve position indicators in the design, as required, and by maintaining clearances for access to the valves.

6.6.3 Examination Techniques and Procedures

In general, where volumetric examination is required to be performed, manual ultrasonic techniques are employed rather than radiography. Where equipment configuration, layout, or other considerations, make it undesirable to perform a manual ultrasonic examination, an automatic technique is employed. Radiography is employed if ultrasonic examination fails to give meaningful results. For components of the reactor coolant pressure boundary, see Subsection 5.2.4.

Where surface examination is required, carbon steel components are typically examined by the magnetic particle method, and stainless steel components by the liquid penetrant method. The liquid penetrant method may be used when component size, geometry or space restriction prevents magnetic particle examination of carbon steel components.

The steam generator tubing is examined by the eddy current method (see Subsection 5.4.2.5).

All examination techniques are qualified to Section V or Section XI of the Code, using qualified personnel.

6.6.4 Inspection Intervals

The inspection schedule for Class 2 and Class 3 components is in accordance with subarticles IWC-2400 and IWD-2400, and is included in the Technical Specifications.

6.6.5 Examination Categories and Requirements

The examination categories and requirements for Class 2 components are in agreement with Section XI, Article IWC-2000.

For Class 3 components, the requirements of Section XI, Article IWD-2000, are complied with, as applicable.

6.6.6 Evaluation of Examination Results

a. Evaluation of Class 2 examination results will comply with the requirements of IWC-3000 of Section XI which applies the rules of IWB-3000. In general, indications detected during in-service examination that exceed the acceptance standards of IWB-3000 of the 1983 Edition, with addenda up to and including summer 1983, will require repair in accordance with IWC-4000 of Section XI.

b. Repair procedures for Class 2 components comply with the requirements of IWC-4000 of Section XI.

c. Evaluation of Class 3 components is consistent with IWD-3000 of Section XI which also applies the rules of IWB-3000. Visual examinations of Class 3 components consist of evaluations for leakage and integrity of structural attachments. Defects will be evaluated per IWB-3000 and repairs performed in accordance with the requirements of IWD-4000 of Section XI.

   In general, evaluation of leakage in Class 3 piping will be consistent with the intent of IWA-3000, since defects resulting in perceptible leakage are not acceptable.

d. Defects in Class 3 pressure boundary components will be removed or reduced to acceptable size by grinding, cutting, or drilling. If the defect and the repair do not encroach on the minimum wall, then no repair welding will be required. Repaired surfaces will be smoothly blended into the surrounding material with no discontinuities. If the defect or its removal encroaches upon the minimum wall thickness, the area will be built up to the minimum wall thickness by welding, and the surfaces smoothly blended into the surrounding surface. If the built-up material thickness of carbon steel components that have been repaired by welding is greater than ¾", post-weld heat treatment is required.

6.6.7 System Pressure Tests

The system pressure testing program for Class 2 components complies with the requirements of IWC-5000 of Section XI.

The system pressure testing program for Class 3 components complies with the requirements of IWD-5000 of Section XI.

6.6.8 Augmented In-Service Inspection to Protect Against Postulated Piping Failures

As stated in Section 3.6(B).2.1.a.4, for main steam and feedwater piping penetrating containment, no breaks were postulated between the first pipe whip restraint inside the containment and the five-degree restraint outside containment. To protect against postulated piping failures, this piping is subject to augmented inservice inspection as defined in the Risk-Informed ISI Break Exclusion Region (BER) evaluation (EE-07-035). The augmented lines are:

MS-4000-02-30"	FW-4606-03-18"
MS-4000-41-30"	FW-4606-04-16"
MS-4001-02-30"	FW-4607-03-18"
MS-4001-41-30"	FW-4607-04-16"
MS-4002-02-30"	FW-4608-03-18"
MS-4002-37-30"	FW-4608-04-16"
MS-4003-02-30"	FW-4609-03-18"
MS-4003-37-30"	FW-4609-04-16"

6.7 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (BWR)

Not applicable to Seabrook.

6.8 EMERGENCY FEEDWATER SYSTEM

6.8.1 Design Bases

The Emergency Feedwater (EFW) System provides the capability to remove heat from the Reactor Coolant System during emergency conditions when the Main Feedwater System is not available, including small LOCA cases. The EFW System operates over a time period sufficient to cool down the Reactor Coolant System to temperature and pressure levels at which the Residual Heat Removal System can operate. Under certain design basis scenarios, condensate storage tank inventory may be depleted prior to affecting the transition to RHR cooling. However, sufficient steam generator inventory will exist to maintain cooling until the RHR system can be placed in operation. For Station Blackout, the EFW turbine-driven pump will operate during the four-hour coping duration to cool down and maintain the secondary side pressure at about 250 psig (see Section 8.4.1). During all other modes of plant operation, including startup, hot standby, and normal operation up to full power load, the EFW system has zero flow.

The EFW system is designed to meet the following safety-related functional requirements:

a. A malfunction or single active failure of a system component or nonessential equipment does not reduce the performance capabilities of the system.

b. The functional performance of system components is not affected by adverse environmental occurrences, abnormal operational requirements, and off-normal conditions such as small breaks in the Reactor Coolant System or the loss of offsite power.

c. System components and piping have sufficient physical separation and shielding to protect against the effects of internally and externally generated missiles.

d. The functional performance of the system is not affected by pipe whip and jet impingement that may result from high or moderate energy piping breaks or cracks.

e. The system possesses diversity in motive power sources so that the system performance requirements are met with either power source.

f. The system design precludes the occurrence of fluid flow instability during normal plant operation and during upset or accident conditions.

g. Provisions are included to verify correct system operation, to detect and control system leakage, and to isolate portions of the system in case of excessive leakage or component malfunctions.

h. The system is capable of automatically initiating flow upon receipt of a system actuation signal. The system is also capable of manual actuation to provide protective action and for operational testing independent of the automatic signal. Single failure of the manual circuit will not result in loss of the system function.

i. The system design possesses the capability to automatically terminate flow to a depressurized steam generator, while providing flow to intact steam generators.

j. For Station Blackout analysis, no other active or passive failures, other than those causing the Station Blackout, nor other design basis events such as seismic or line breaks are assumed to occur (see Section 8.4.1).

The Emergency Feedwater System is designed in accordance with ASME Code Section III, Class 3; IEEE Standards 323-1974 and 344-1975, Class 1E; and Seismic Category I requirements. System components are located within Seismic Category I structures and are thereby protected against effects of natural phenomena.

The design of the EFW system was reviewed subsequent to the issuance of the NRC's March 10, 1980 letter to near-term operating license applicants (TMI-2 Task Action Plan, NUREG-0737, Item II.E.1.1). The review addressed the following areas:

a. Subsection 10.4.9 of the Standard Review Plan

b. Branch Technical Position ASB 10-1

c. Generic short-term and long-term requirements applicable to the EFW system design, and operating procedures

d. A reliability evaluation of the EFW system as outlined in NUREG-0611.

An item-by-item discussion of the EFW system's compliance with each requirement was provided by applicants letter to NRC, dated July 27, 1982.

6.8.2 System Description

Upon loss of normal feedwater flow, the reactor is tripped, and the decay and sensible heat is transferred to the steam generators by the Reactor Coolant System via the reactor coolant pumps or by natural circulation when the pumps are not operational.

Heat is removed from the steam generators via the main condensers or the main steam safety and/or steam generator atmospheric relief valves. Steam generator water inventory is maintained by water makeup from the Emergency Feedwater System. The system will supply feedwater to the steam generators to remove sufficient heat to prevent the over-pressurization of the Reactor Coolant System, and to allow for eventual system cooldown.

The Emergency Feedwater System is comprised of two full-sized pumps (one motor- and one turbine-driven) whose water source is the Condensate Storage Tank (CST). Suction lines are individually run from the CST to each pump. A common EFW pump recirculation line discharges back to the Condensate Storage Tank. This return line functions for recirculation pump testing and ensures minimum flow to prevent pump damage for any system low-flow operating condition. Both pumps feed a common discharge header, which in turn supplies the four emergency feed lines. The common discharge header includes normally open gate valves between each branch connection to provide isolation in the event of a pipe break or for maintenance. Each emergency feed line is connected to one of the main feedwater lines downstream of the feedwater isolation valve. Each main feedwater line enters the containment through a single penetration and feeds a single steam generator. For a diagram of the Emergency Feedwater System see Figure 6.8-1 and Figure 6.8-2.

Additional redundant pumping capability is provided by the startup feed pump in the feedwater system. The head and capacity curves for these pumps, which are plotted on the same sheet, show that the startup pump has sufficient capacity to serve as backup for the emergency feedwater pumps (see Figure 6.8-3). The startup feed pump (and steam generator wet-lay-up pumps) discharge line inside the Emergency Feedwater Pump Building and Main Steam and Feedwater Pipe Chase are seismically supported. All connections from these lines to other plant piping include normally closed valves. Valve and pump operation is administratively controlled. During normal plant operation, these lines are not pressurized. Valves V156 and V163 are normally closed, and are furnished with a motor operator.

Figure 10.4-6 and Figure 10.4-7 also shows a valve in the branch line from the EFW turbine-driven pump suction line for use as a connection for emergency makeup from the condensate storage tank to the spent fuel pool or from the fire protection system to the condensate storage tank. This valve is normally closed, and is administratively opened. Additional makeup to the CST is available from the demineralized water storage tank (DM-TK-259) using the portable cooling tower makeup pump (SW-P-329) via demineralized water system valves DM-V-677, V-800, and V-490 (see Figure 9.2-15).

A dedicated 196,000 gallons of demineralized water is maintained in the lower half of the condensate storage tank for the exclusive use of the Emergency Feedwater System. For a description of the condensate storage facility see Subsection 9.2.6. Makeup to the tank is provided by the Demineralized Water Makeup System (see Subsection 9.2.3).

The motor-driven pump and pump controls are powered from an emergency bus. The startup feed pump is also capable of being powered from an emergency bus (see Subsection 8.3.1.1.b.9.(a)), and diesel generator capacity is available to start this pump while carrying the maximum load listed in Table 8.3-1 (See Subsection 8.3.1.1.e.6). Steam for the turbine-driven pump is supplied from either of two main steam headers via branch lines connected upstream of the main steam isolation valves. Each branch line includes an air-operated, fail-open valve. The common EFW steam header contains one air-operated fail open valve (see Subsection 10.3.2.5). A summary of pump data is provided in Table 6.8-1.

The branch lines to each steam generator include a manual gate isolation valve, two motor-operated flow control valves, a flow venturi, and a flow orifice. The flow control valves are normally in the open position when the system is not operating and are automatically closed during system operation in the event of a pipe break. These valves can be operated remotely as described in Subsection 6.8.5 to control steam generator water level. Two valves in series are provided for redundancy and are powered from different trains. Each valve is also provided with a handwheel to permit manual operation. The open position of the flow control valves for system limiting conditions will be set to insure the minimum required flow of 470 gpm to three steam generators and a minimum total flow of 650 gpm to four steam generators with one EFW pump operational.

6.8.3 Safety Evaluation

The Emergency Feedwater (EFW) System components, instrumentation, and power supplies are sized and designed with sufficient redundancy to maintain the system's safety-related functions under all credible accident conditions. The combination of one turbine-driven pump and one motor-driven pump provides a diversity of power sources to assure delivery of feedwater under emergency conditions.

The system has been designed to provide the required flow following a single active failure and a loss of offsite power and has been evaluated for a Station Blackout event (see Section 8.4.1). The common discharge header is not normally pressurized during normal plant operation.

Figure 10.4-8 and Figure 10.4-9 shows the Main Feedwater System, the tie-in of the EFW and four individual stop-check valves. These check valves (FW-V76, -V82, -V88 and V94) prevent backflow of feedwater from the main feedwater system or the steam generator (SG).

Steam supply for the turbine-driven emergency feedwater pump is from either of two main steam headers via branch lines connected upstream of the main steam isolation valves. The branch lines contain normally closed, fail-open air-operated EFW isolation valves MS-V393 and MS-V394. These valves also serve a containment isolation function. These valves are controlled by Class 1E 125V DC solenoids. Branch isolation valve MS-V393 is controlled by a Train A solenoid valve and the alternate branch isolation valve, MS-V394, is controlled by redundant Train A and Train B solenoid valve. The common EFW steam supply line contains a normally closed, fail-open air-operated valve (MS-V395). This valve is controlled by both Train A and Train B Class 1E solenoids. Isolation valves MS-V393 and MS-V394 open in response to an EFW initiation signal to admit steam to MS-V395. Valve MS-V395 is timed to sequentially open following opening of either MS-V393 or MS-V394. The sequential opening of MS-V395 allows for pressurizing the EFW steam header to discharge accumulated condensate via system drains prior to introducing steam to the turbine governor valve. Approximately 45 seconds following an EFW initiation signal, MS-V395 is full open and supplying steam to the mechanical hydraulic turbine governor valve.

The east and west EFW steam supply branches and the common header each contains a condensate drain pot and steam trap arrangement. The common header drain pot contains a constant steam vent line. The composite drain assemblies provide for minimizing accumulated steam header condensate during an EFW System standby, startup, and operational condition.

The outside containment Steam Generator Blowdown isolation valves close automatically whenever an EFW pump is in operation. Blowdown isolation during EFW pump operation is a required support function that preserves Steam Generator and Condensate Storage Tank inventory. The outboard SB isolation valves are fail closed, air operated valves which are controlled by independent A-train and B-train pilot solenoid valves. The close signals for EFW pump operation derive from either position limit switches on MS-V393 (Train A) and MS-V394 (Train A and Train B), or motor-driven EFW pump breaker position.

The EFW supply has two safety-grade flow control valves for each steam generator. One valve in each supply is powered by the A train emergency power source and the other valve is powered from the B train. The primary (or normal) flow control valves for the A and C steam generator will be powered by the A train with B and D steam generators' valves powered by the B train. Backup flow control valves will be powered from the opposite emergency power train. These valves can be controlled from either the main control board or the remote shutdown panel using safety-grade controls.

The five valves in the EFW pump discharge header are furnished with gear operators so that a concern for power diversity is not applicable.

The design and operation of the EFW system has been reviewed regarding the occurrence of hydraulic instabilities, characterized as water hammer. The EFW system is connected to the main feedwater system through stop-check valves outside the containment. The flow regulating valves in each EFW line are normally open, and are sized to pass the required flow under accident conditions. The only action required to establish EFW flow is to start the pumps. One pump has sufficient capacity to furnish the required flow to the Nuclear Steam Supply System.

An analysis of the EFW system has established that its function and performance are not affected by the common causes for loss of flow resulting in water hammer, such as pump trip, or rapid valve closure. A pressure transient in the main feedwater system resulting from a pipe break, pump trip, and/or valve closure would be dissipated before flow is established in lines to each SG. A trip of one pump will not affect the capability of the other pump to provide flow to the intact SGs. The only automatic valve closure in the EFW system would occur in the line to a faulted SG. During operation of the EFW system, the plant operators can initiate any changes in flow to each SG, as required.

The EFW pumps' supply and recirculation line piping runs from nozzles on the Condensate Storage Tank (CST) to the EFW pumphouse. The CST nozzles and adjacent piping are protected by a seismic Category I structure which is part of the CST enclosure and tornado-missile shield. The piping is routed underground and runs below grade into the EFW pumphouse, also a seismic Category I structure.

The EFW pump recirculation line to the CST, is designed to ASME Code Section III, Class 3 seismic Category I requirements. Valves FW-V346 and FW-V347 are administratively opened for EFW pump-surveillance testing and as required ensuring minimum EFW pump flow during system operation.

The water lines to the oil cooler are designed to seismic Category I requirements. Water lines from the oil cooler are designed to ASME Code Section III, Class 3, seismic Category I requirements. The breakdown orifice in the line to the oil cooler limits the flow to 2-3 gpm. This flow was considered in sizing the pump capacity. In the unlikely event of pipe failure, this flow will easily be handled by the pump room floor drains.

An accident analysis for this system in conjunction with the loss of the Main Feedwater System is provided in Chapter 15. A failure analysis of the Emergency Feedwater System following a feedwater pipe break is provided in Table 6.8-2.

6.8.4 Tests and Inspections

Prior to initial plant startup, the Emergency Feedwater System is hydrostatically tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 3, and pre-operationally tested as described in Chapter 14.

Periodic testing in accordance with Technical Specifications will be performed during normal plant operation. During periodic surveillance testing of the EFW pumps, manual valve alignments will be required. Only one EFW pump will be tested at a time. Because each EFW pump is capable of providing 100 percent of required flow, full system flow requirements will be available at all times. Automatic indication of EFW pump inoperable status is provided as discussed in subsection 7.1.2.6.

6.8.5 Instrumentation Requirements and Controls

The Emergency Feedwater System will be actuated automatically on loss of offsite power, low-low level in any of the steam generators, safety injection signals or an ATWS mitigation system actuation signal. The engineered safety feature actuation system details are presented in Section 7.3. The ATWS mitigation system is discussed in Section 7.6. Manual controls for the turbine-driven pump steam supply valves are located at the main control board (MCB), as well as at the remote safe shutdown (RSS) panel. For the motor-driven pump, the controls are located at the MCB and in the switchgear room. The suction and discharge pressures of both pumps are indicated locally. Pump discharge pressure and CST level indication are provided on the MCB. Low suction pressures are alarmed at the MCB.

Flow indications for all four individual emergency feedwater lines are provided. Safety grade flow orifice instrumentation readouts are displayed at the MCB. The instruments are powered from the safety grade inverters - A and C steam generators on the Train A inverter, and B and D steam generators on the Train B inverter. These instrumentation channels meet or exceed the requirements for Design Category 2 instrumentation as provided in Subsection 7.5.5. The design details of the accident monitoring instrumentation are presented in Table 7.5-1. Flow venturi are also provided in each emergency feedwater line. Two of the four flow venturi instrumentation readouts are displayed at a RSS panel, and the remaining two flow venturi instrumentation readouts are displayed at a second RSS panel.

A sustained high-flow condition in any of the lines is indicative of a line break. A break isolation is incorporated so that the affected line will be isolated by automatically closing the motor-operated valves on high flow signals from redundant flow instrumentation. Break isolation is required to conserve CST inventory and to provide the required flow to the intact steam generators. High flow alarms are also provided to alert the operator to this condition. The break isolation system is designed so that a single failure will not prevent break isolation or prevent emergency feedwater flow to at least two steam generators. Manual override provisions are also incorporated at the MCB as well as at the RSS panels, along with the open/close valve position indication. Each of the motor-operated control valves in each branch line is provided with fully independent power supplies, instrumentation, and controls to ensure that at least one of the valves in each branch line can be closed when needed. All eight valves can be operated from the MCB. Four of the valves, one in each branch, can also be operated from an RSS panel and the remaining four valves, one in each branch, can be operated from a second RSS panel. The associated control switches are located on the MCB and the RSS panel so that the valves normally used to control flow have the same train assignment as the flow indication for the steam generator (i.e., A and C SG are normally controlled with the A train control valves and B and D SG with B train valves). Thus, complete redundancy is provided to control flow or to isolate any steam generator in the event of pipe breaks.

The review of the emergency feedwater flow indication was performed as part of the detailed control room design review (DCRDR). As part of this effort, a determination of the needed characteristics of this display to support the emergency operating procedures (EOPs) was made. A comparison of the needed characteristics against the available instrumentation was made, and no deficiencies were found.

Each EFW pump recirculation line contains a normally closed motor-operated valve powered from its respective train.

Associated remote-manual control switches are located at the MCB and at RSS panels CP-108A and CP-108B for EFW pumps P-37A and P-37B respectively.

A flow orifice and associated instrumentation are provided in the common pump discharge recirculation path to the CST. This instrumentation is provided to permit periodic testing of the pumps to verify proper head-flow characteristics during periodic pump testing and to monitor recirculation flow during normal operation. Alarms are also provided to indicate that either train of the Emergency Feedwater System is inoperative. The design features of the bypass and operable status alarm system which provide system level indications in compliance with Regulatory Guide 1.47 are presented in Subsection 7.1.2.6.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 6
ENGINEERED SAFETY FEATURES

TABLES

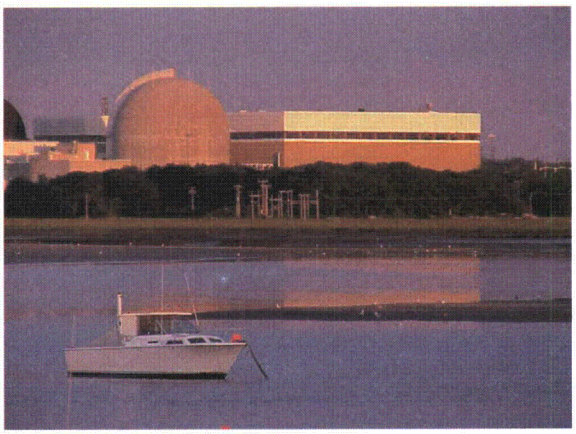


TABLE 6.1(B)-1 PRINCIPAL ENGINEERED SAFETY FEATURES METALLIC MATERIALS *

Valves	
Bodies	SA-182, Grade F304 or F316 SA-351, Grade CF3M or CF8M
Bonnets	SA-182, Grade F304 or F316 SA-240, TP 304 SA-351, Grade CF3M or CF8M
Discs	SA-182, Grade F316 SA-479, TP 316
Closure bolting/studs	SA-453, Grade 660 SA-564, Grade 630
Nuts	SA-193, Grade B6 SA-194, Grade 6, 8F, or 8M SA-453, Grade 660
Tanks	
Shells and heads	SA-240, TP 304; SA-479, TP 304
Flanges and nozzles	SA-276, TP 304; SA-182, TP 304 SA-312, TP 304
Closure bolting and nuts	SA-193, Grade B8; SA-194, Grade 6
Pumps	
Pump casing	SA-351, TP 316
Piping	SA-213, TP 316
Pipe fittings	
Pipe plug, cross, bushing, & male connector	SA-182, TP 316
Nipple	SA-312, TP 316
Orifice	SA-479, TP 316
Closure studs and nuts	SA-193, Grade B8M; SA-194, Grade 8M or B8M

* Excluding NSSS-supplied

Heat Exchangers	
Shells and shell heads	SA-285, Grade C
Shell nozzles	SA-106, Grade B
Shell flanges	SA-105 (normalized)
Shell couplings	SA-105
Channel	SA-240, TP 304
Channel facing	SA-182, Grade F 304
Channel nozzles	SA-312, TP 304
Channel cover	SA-516, Grade 70 lined with SA-240, TP 304
Channel couplings	SA-479, TP 304
Tubes	SA-249, TP 304 (.05% max. carbon)
Tubesheets	SA-182, Grade F 304
Closure bolting and nuts	SA-193, Grade B7; SA-194, Grade 2H
Piping	SA-312, TP 304 or SA-358, TP 304
Fittings	SA-182, TP 304 or SA-403, Grade WP 304 or WP 304W
Flanges	SA-182, TP F304
Closure Bolting and Nuts	SA-193, Grade B8M or B8; SA-194, Grade 6
Weld Materials	SFA-5.1, 5.4, 5.9, 5.17, 5.18, 5.20

Table 6.1(B)-2 COATED AREAS ON MAJOR EQUIPMENT AND STRUCTURES INSIDE CONTAINMENT *

Component/Structures	Painted Surface Area (ft^2)
Containment Cylinder	63,774
Containment Dome	30,778
Miscellaneous Concrete	96,514
Structural Steel	64,688
Polar Gantry Crane	19,920
Floor and Sump	11,640
Equipment Hatch	3,250
Equipment Steel	2,345
Personnel Hatch	360
Containment Recirc. Unit	705

* Excluding NSSS-supplied

TABLE 6.1(B)-3 DELETED

Table 6.1(B)-4 OTHER ORGANIC MATERIALS IN CONTAINMENT

Item	Material	Amount
Reactor coolant pump lubricant	Petroleum base oil	1,060 gal.
Cable insulation		50,000 lbs.
Power cable	Ethylene propylene rubber with Hypalon jacket	
	Silicone Rubber	
Control and instrument cable	Cross-linked polyethylene insulation with Hypalon jacket	

Table 6.1(N)-1 ENGINEERED SAFETY FEATURE MATERIALS

Valves	
Bodies	SA-182, Grade F316; SA-351, Grade CF8 or CF8M
Bonnets	SA-182, Grade F316; SA-351, Grade CF8 or CF8M
Discs	SA-182, Grade F316; SA-564, Grade 630; SA-351, Grade CF8 or CF8M
Pressure retaining bolting	SA-453, Grade 660
Pressure retaining nuts	SA-453, Grade 660; SA-194, Grade 6
Auxiliary Heat Exchangers	
Heads	SA-240, Type 304
Nozzle necks	SA-182, Grade F304; SA-312, Grade TP304; SA-240, Type 304
Tubes	SA-213, Grade TP304; SA-249, Grade TP304
Tube Sheets	SA-182, Grade F304; SA-240, Type 304; SA-516, Grade 70 with Stainless Steel Cladding A-8 Analysis
Shells	SA-240 and SA-312, Grade TP304
Auxiliary Pressure Vessels, Tanks, Filters, etc.	
Shells and heads	SA-351, Grade CF8A; SA-240, Type 304; SA-264 Clad Plate of SA-537, Class 1 with SA-240, Type 304 Clad and Stainless Steel Weld Overlay A-8 Analysis
Flanges and nozzles	SA-182, Grade F304; SA-350, Grade LF2 with SA-240, Type 304 and Stainless Steel Weld Overlay A-8 Analysis
Piping	SA-312 and SA-240, TP304 or TP316 Seamless
Pipe fittings	SA-403, Grade WP304 Seamless
Closure bolting and nuts	SA-193, Grade B7; SA-194, Grade 2H

Auxiliary Pumps

Pump casing and heads	SA-351, Grade CF8 or CF8M; SA-182, Grade F304 or F316
Flanges and nozzles	SA-182, Grade F304 or F316; SA-403, Grade WP316L Seamless
Piping	SA-312, Grade TP304 or TP316 Seamless
Stuffing or packing box cover	SA-351, Grade CF8 or CF8M; SA-240, Type 304 or Type 316; SA-182, Grade F304 or F316
Pipe fittings	SA-403, Grade WP316L Seamless; SA-213, Grade TP304, TP304L, TP316 or TP316L
Closure bolting and nuts	SA-193 Grade B6, B7 or B8M; SA-194, Grade 2H or 8M; SA-453, Grade 660; and Nuts SA-194, Grade 2H; 6, 7 and 8M

Table 6.1(N)-2 PROTECTIVE COATINGS ON WESTINGHOUSE SUPPLIED EQUIPMENT INSIDE CONTAINMENT

Component	Painted Surface Area (ft^2)
RCS Component Supports	11,230
Reactor coolant pump assemblies	4,048
Accumulator tanks	5,400
Manipulator crane	2,600
Other refueling equipment	2,125
Remaining equipment (such as valves, auxiliary tanks and heat exchanger supports, transmitters, alarm horns, small instruments)	<1,300

Table 6.2-1 GENERAL INFORMATION AND INITIAL CONDITIONS

I.	General Information, Containment	
	Design Pressure, psig	52.0/-3.5
	Design Temperature, °F	296
	C. Free Volume, ft^3	2.704×10^6
	D. Design Leak Rate, % mass/day	0.15
II.	Initial Conditions	
	A. Reactor Coolant System	
	1. Reactor Power Level, Mw(t)	3652.8
	2. Average Coolant Temperature, °F	592.7
	3. Mass of Reactor Coolant System Liquid, lbm	492,269
	4. Mass of Reactor Coolant System Steam, lbm	12,370
	5. Liquid Plus Steam Energy, Btu	302.3×10^6
	B. Containment	
	1. Pressure, psig	1.5
	2. Inside Temperature, °F	120
	3. Outside Temperature, °F	110
	4. Relative Humidity, %	1
	5. Service Water Temperature, °F **	65
	6. Refueling Water Temperature, °F	98
	7. Spray Water Temperature, °F	100
	C. Stored Liquid	
	1. Refueling Water Storage Tank *, gal	350,000 min.
	2. Total Free Volume, Four Accumulators	3,800 nominal
	3. Spray Additive Tank, gal	10,700

** System analysis has been performed to permit plant operation up to a maximum ocean temperature of 68.5°F to accommodate occasional summer ocean temperature excursions.

* 475,000 gal maximum capacity

Table 6.2-2 ENGINEERED SAFETY FEATURE SYSTEMS INFORMATION

	Design Capacity	Values Used for Containment Analysis
A. Passive Safety Injection System		
1. Number of Accumulators (Safety Injection Tanks)	4	4
2. Pressure Set Point, psig	600	600
B. Active Safety Injection Systems		
1. Charging Pumps		
a. Number of Lines	2	1 (Min. SI), 2 (Max. SI)
b. Number of Pumps	2	1 (Min. SI), 2 (Max. SI)
c. Flow Rate per Pump, gpm	150 @ 2,518 psi	450 (Injection) 0 (Recirculation)
2. High Pressure Safety Injection		
a. Number of Lines	2	1 (Min. SI), 2 (Max. SI)
b. Number of Pumps	2	1 (Min. SI), 2 (Max. SI)
c. Flow Rate per Pump, gpm	440 @ 2,680 ft TDH	450 (Injection) 0 (Recirculation)
3. Low Pressure Safety Injection		
a. Number of Lines	2	1 (Min. SI), 2 (Max. SI)
b. Number of Pumps	2	1 (Min. SI), 2 (Max. SI)
c. Flow Rate per Pump, gpm	3,000 @ 375 ft TDH	4,000 (Injection) 3,000 (Recirculation)
4. Containment Spray System		
a. Number of Lines	2	1
b. Number of Pumps	2	1
c. Number of Headers	2	1
d. Flow Rate per pump, gpm	3,010 @ 550 ft TDH	2,808
5. Heat Exchangers		
a. Containment Spray Heat Exchanger		
(1) Type	Shell and Tube	

	Design Capacity	Values Used for Containment Analysis
(2) Number of Units	2	1
(3) Heat Transfer Area/Unit, ft^2	3,468	3,468
(4) Overall Heat Transfer Coefficient, Btu/hr - ft^2 - °F		(431 service)
(5) Flow Rates/Unit, gpm		
(a) Recirculation (Tube) Side		2,808
(b) Exterior (Shell) Side		4,800
(6) Source of Cooling Water	Component Cooling Water	
(7) Time of Recirculation, sec.	2755 (Min. SI)	1688 (Max. SI)
b. Residual Heat Removal Heat Exchanger		
(1) Type	Shell and Tube	
(2) Number of Units	2	1 (Min. SI)
(3) Heat Transfer Area/Unit, ft^2	5444	5444
(4) Overall Heat Transfer Coefficient, Btu/hr - ft^2-°F		385 (Service)
(5) Flow Rates/Units, gpm		
(a) Recirculation (Tube) Side		3,000
(b) Exterior (Shell) Side		5,000
(6) Source of Cooling Water	Component Cooling Water	
(7) Time of Recirculation, sec.	2755 (Min. SI)	1688 (Max. SI)
c. Component Cooling Water Heat Exchanger		
(1) Type	Shell and Tube	

	Design Capacity	Values Used for Containment Analysis
(2) Number of Units	2	1 (Min. SI)
(3) Heat Transfer Area/Unit, ft^2	24,444	24,444
(4) Overall Heat Transfer Coefficient Btu/hr - ft^2 - °F		539
(5) Flow Rates/Unit, gpm		
(a) Component Cooling Water(Shell) Side		11,727
(b) Station Service Water (Tube) Side	8,000 (Ocean) 11,000 (Tower)	7,500 (ocean) 9,851 (Tower)
(6) Source of Cooling Water	Station Service Water	
(7) Time of Recirculation, sec	2755 (Min. SI)	1688 (Max. SI)

Table 6.2-3 CONTAINMENT PASSIVE HEAT SINK DATA

Heat Sink	Material	One Side Exposed Area(ft^2)	Thickness (in)	Volume (ft3)	Sides Exposed
1. Concrete Cylinder	Steel-lined Concrete	63,774			one
	Concrete		0.018	1,993	
	Paint		0.375	286,983	
	Carbon Steel		0.625		
	Air Gap		54.0		
	Concrete				
2. Containment Dome	Steel-lined Concrete	30,788			one
	Concrete		0.018		
	Paint		0.50	1,282	
	Carbon Steel		0.0625		
	Air Gap		42.0	107,723	
	Concrete				
3. Miscellaneous	Painted Concrete	48,257			both
	Paint		0.059		
	Concrete		53.52	215,226	
	Paint		0.059		
4. Refueling Canal Floor	SS-lined Concrete	1,152			both
	Stainless Steel		0.25	24	
	Air Gap		0.0625		
	Concrete		18.0	1,728	
5. Refueling Canal	SS-lined Concrete	6,005			both
	Stainless Steel		0.1875	94	
	Air Gap		0.0625		
	Concrete		36.0	18,014	
6. Conduit	Galvanized Steel	11,728	0.154	150.5	one
7. Ducts & Trays	Galvanized Steel	55,927	0.0729	340	one
8. Structural Steel	Painted Concrete	32,344			both
	Paint		0.018		
	Steel		0.558	1,503	
	Paint		0.018		
9. Polar Crane, Equipment Hatch, Personnel Hatch	Painted Steel	23,529			one
	Paint		0.018		
	Steel		0.736	1,443	

Heat Sink	Material	One Side Exposed Area(ft^2)	Thickness (in)	Volume (ft3)	Sides Exposed
10. Equipment Steel	Painted Steel	2,345			one
	Paint		0.018		
	Steel		1.55	303	
11. Containment Floor and Sump	Painted Steel	11,639			one
	Paint		0.009		
	Concrete		47.43	46,003	
	Steel		0.25	243	
	Air Gap		0.0625		
	Concrete		120.0	116,390	

TABLE 6.2-4 PASSIVE HEAT SINK MATERIAL PROPERTIES

Material	Thermal Conductivity (Btu/hr-ft-°F)	Volumetric Heat Capacity (Btu/ft^3-°F)
Paint	0.88	24.12
Carbon Steel	27.0	58.8
Stainless Steel	10.0	56.0
Air Gap	0.0184	0.0173
Concrete	0.92	22.62

TABLE 6.2-5 BREAK LOCATIONS POSTULATED FOR LOSS-OF-COOLANT ACCIDENT ANALYSIS

Code No.	Break Location
1	Pump Suction
2	Cold Leg
3	Hot Leg

TABLE 6.2-6 SPECTRUM OF BREAKS POSTULATED FOR LOSS-OF-COOLANT ACCIDENT ANALYSIS

Code No.	Break Type and Size
1	Full Double-Ended Guillotine Rupture
2	Small Double-Ended Guillotine Rupture (0.6 of Full Size for Break at Pump Suction)
3	Split Rupture (3 ft^2 for Break at Pump Suction)

TABLE 6.2-7 SINGLE ACTIVE FAILURES POSTULATED FOR LOSS-OF-COOLANT ACCIDENT ANALYSIS

Code No.	SAF
1	Failure of One Diesel Generator (Minimum Safety Injection)
2	Failure of One Diesel Generator at Time of Containment Spray Actuation (Maximum Safety Injection)

TABLE 6.2-8 CALCULATED PEAK CONTAINMENT PRESSURE AND PEAK CONTAINMENT TEMPERATURE FOLLOWING LOSS-OF-COOLANT ACCIDENTS

Case	Accident Conditions	Peak Pressure (psig)	Time of Peak Pressure (sec)	Peak Temperature (°F)	Time of Peak Temperature (sec)	Energy Released to Containment Up To End of Blowdown (10^6 Btu)
1.1.1	Full DE Pump Suction Guillotine (Min. SI)	49.2	3,601	272	3,613	311.0
1.1.2	Full DE Pump Suction Guillotine (Max. SI)	49.6	3,603	273	3,614	311.0
1.2.1	0.6 DE Pump Suction Guillotine (Min. SI)	49.2	3,604	272	3,616	307.6
1.3.1	3 ft^2 Pump Suction Split (Min. SI)	35.6	33	251	32	306.9
2.1.1	Full DE Cold-Leg Guillotine (Min. SI)	36.6	18	254	18	305.9
3.1.1	Full DE Hot-Leg Guillotine * (Min. SI)	37.5	20	256	22	311.8

* This case has been analyzed for the blowdown period only.

Note: This table presents results related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the results of this analysis of record, as compared with the long-term LOCA containment response results at an analyzed core power level of 3659 MWt.

TABLE 6.2-9 LOSS-OF-COOLANT ACCIDENT CHRONOLOGY FOR VARIOUS BREAK LOCATIONS

	Time (sec)		
Event	Case 1.1.2	Case 2.1.1	Case 3.1.1 *
Pipe Rupture	0.0	0.0	0.0
Assumed Initiation of ECCS	10.0	10.0	10.0
Peak Containment Pressure During Blowdown	21.0	17.0	19.5
End of Blowdown	24.2	22.0	21.0
Beginning of Containment Spray	65.0	65.0	65.0
End of Reflood	155.1	212.1	-
End of Broken-Loop Steam Generator Energy Release	678.0	-	-
Peak Containment Pressure After Blowdown and Before Circulation	695.0	215.0	-
End of Injection and Beginning of Recirculation (350,000 gal. RWST Water Injected)	1688.0	2755.0	2755.0
Peak Containment Pressure During Recirculation	3603.0	6000.0	-
Containment Pressure at 50% of Design Value	28000.0	19000.0	-

* This case has been analyzed for the blowdown period only.

Note: This table presents results related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the results of this analysis of record, as compared with the long-term LOCA containment response results at an analyzed core power level of 3659 MWt.

6.2-10 INITIAL PLANT OPERATING POWER LEVELS POSTULATED FOR MAIN STEAM LINE BREAK ANALYSIS

Code No.	Power Level
1	102%
2	75%
3	50%
4	25%
5	0% (Hot Shutdown)

6.2-11 SPECTRUM OF RUPTURES POSTULATED FOR MAIN STEAM LINE BREAK ANALYSIS

Code No.	Break Type and Size	
1	Full Double-Ended Guillotine Rupture	
2	Small Double-Ended Guillotine Rupture*	
	Power Level	Break Size (sq..ft.)
	102%	0.60
	75%	0.55
	50%	0.45
	25%	0.33
	0%	0.20
3	Split Rupture	
	Power Level	Break Size (sq. ft.)
	102%	0.80
	75%	0.84
	50%	0.80
	25%	0.660
	0%	0.40
4	Additional Small DE Guillotine Rupture *	
	(0.25 ft^2 for 75% and 0.10 ft^2 for 0% only.)	

* Area given is for each of forward flow and reverse flow.

TABLE 6.2-12 SINGLE FAILURES POSTULATED FOR MAIN STEAM LINE BREAK ANALYSIS

Code No.	Single Failure
1	Failure of one Feedwater Isolation Valve (FWIV) to Close
2	Failure of Broken-Loop Main Steam Isolation Valve (MSIV) to Close
3	Failure of One Feedwater Control Valve (FCV) to Close
4	Failure of One Main Feedwater Pump Trip (MFPT)
5	Failure of Broken-Loop Emergency Feedwater Pump Runout Control (EFPRC)
6	Failure of One Containment Spray Train

TABLE 6.2-13 CALCULATED PEAK CONTAINMENT PRESSURE AND PEAK CONTAINMENT TEMPERATURE FOLLOWING FULL DOUBLE-ENDED GUILLOTINE RUPTURE IN MAIN STEAM LINE

Case [+]	Peak Pressure (psig)	Time of Peak Pressure [*] (sec)	Peak Temperature (°F)	Time of Peak Temperature (sec)	Total Energy Release to Containment (10^6 Btu)
1.1.2	33.6	173	364	82	324.72
1.1.6	32.9	173	360	98	301.77
5.1.6	36.1	281	349	112	345.13

[+] The first digit refers to the power levels listed in Table 6.2-10, the second digit is the break spectrum in Table 6.2-11, and the third digit is the single failure in Table 6.2-12.

[*] Same as steam generator dryout time in these cases.

Note: This table presents results related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the results of this analysis of record, as compared with the MSLB containment response results at an analyzed core power level of 3659 MWt.

TABLE 6.2-14 CALCULATED PEAK CONTAINMENT PRESSURE AND PEAK CONTAINMENT TEMPERATURE FOLLOWING SMALL DOUBLE-ENDED GUILLOTINE RUPTURES IN MAIN STEAM LINE

Case [+]	Peak Pressure (psig)	Time of Peak Pressure (sec)	Peak Temperature (°F)	Time of Peak Temperature (sec)	Total Energy Release to Containment (10^6 Btu)
5.2.6	28.8	1460*	317	612	332.92
5.4.5	20.6	1165	306	1165	445.50
5.4.6	23.0	2435*	305	1260	291.62

[+] The first digit refers to the power levels listed in Table 6.2-10, the second digit is the break spectrum in Table 6.2-11, and the third digit is the single active failure in Table 6.2-12.

* Steam generator dryout time.

Note: This table presents results related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the results of this analysis of record, as compared with the MSLB containment response results at an analyzed core power level of 3659 MWt.

TABLE 6.2-15 CALCULATED PEAK CONTAINMENT PRESSURE AND PEAK CONTAINMENT TEMPERATURE FOLLOWING RUPTURES IN MAIN STEAM LINE

Case ±	Peak Pressure (psig)	Time of Peak Pressure * (sec)	Peak Temperature (°F)	Time of Peak Temperature (sec)	Total Energy Release to Containment (10^6 Btu)
4.3.6	34.7	353*	349	144	340.57

± The first digit refers to the power levels listed in Table 6.2-10, the second digit is the break spectrum in Table 6.2-11, and the third digit is the single active failure in Table 6.2-12.

* Steam generator dryout time.

Note: This table presents results related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the results of this analysis of record, as compared with the MSLB containment response results at an analyzed core power level of 3659 MWt.

Table 6.2-16 POSTULATED BREAKS FOR SUBCOMPARTMENT ANALYSIS

Containment Subcompartment	Postulated Break	Break Location
Pressurizer Compartment	Full double-ended guillotine spray line rupture	EL 12'-0"; EL 24-6" EL 34'-0"; EL 45'-0" EL 54'-0 5/8"
Pressurizer Skirt Cavity	Full double-ended guillotine surge line rupture	Pressurizer surge line nozzle

TABLE 6.2-17a

(DELETED)

TABLE 6.2-17b

(DELETED)

TABLE 6.2-18

(DELETED)

TABLE 6.2-19

(DELETED)

TABLE 6.2-20

(DELETED)

TABLE 6.2-21a

(DELETED)

TABLE 6.2-21b

(DELETED)

TABLE 6.2-22

(DELETED)

TABLE 6.2-23a

(DELETED)

TABLE 6.2-23b

(DELETED)

TABLE 6.2-24

(DELETED)

TABLE 6.2-25

(DELETED)

TABLE 6.2-26

(DELETED)

TABLE 6.2-27

(DELETED)

TABLE 6.2-28

(DELETED)

TABLE 6.2-29

(DELETED)

Table 6.2-30 MASS AND ENERGY RELEASE RATES FOLLOWING SPRAY LINE RUPTURE

Time (sec)	Mass Rate (lb/sec)	Enthalpy (Btu/lb)
0.00000	0.0	611.71
.00502	5756.7	611.71
.01002	5615.6	611.65
.02003	6046.9	607.30
.03004	6377.2	604.57
.04002	6514.4	603.09
.05005	6255.2	603.74
.06003	6325.0	602.99
.07006	6345.5	602.60
.08003	6287.2	602.63
.09001	6040.9	603.77
.10002	6273.6	602.47
.11510	6520.5	601.18
.20010	6057.1	603.29
.30004	5925.2	604.01
.34000	5884.1	604.24
.35027	5901.4	604.12
.37001	5858.6	604.35
.38001	5828.3	604.53

Time (sec)	Mass Rate (lb/sec)	Enthalpy (Btu/lb)
.40002	5902.6	604.07
.41010	5928.7	603.90
.43002	5845.8	604.38
.45755	5785.9	604.73
.47751	5795.2	604.65
.51009	5749.2	604.88
.55000	5797.0	604.52
.65003	5717.6	604.86
.70010	5736.6	604.64
.81021	5684.6	604.78
.85000	5699.4	604.60
.90008	5683.0	604.62
.94003	5662.9	604.68
1.00005	5671.7	604.51
.00000	5535.8	604.06

Break Type : Full Double-Ended Spray Line
Break Location: See Notes

VENT PATH NO.	Vol. Node No.		AREA (ft^2)	$\frac{L}{A}$ Ratio (1/ft)	HEAD LOSS, K				
	From	To			FRICTION K, fL/d	TURNING AND OBSTRUCTION LOSS, K	EXPANSION K	CONTRACTION K	TOTAL
1	1	2	139.40	.05	.150		.010	.020	.180
2	1	23	92.00	.05	.057		1.0	.300	1.357
3	2	3	21.50	.20	.110		.130	.310	.550
					(.110)		(.390)	(.180)	(.680)
4	2	4	8.20	.35	.030		.110	.310	.450
					(.030)		(.390)	(.180)	(.600)
5	2	5	10.67	.46	.060		.110	.310	.480
					(.060)		(.390)	(.180)	(.630)
6	2	6	4.92	1.20	.130		.110	.310	.550
					(.130)		(.390)	(.180)	(.700)
7	2	7	4.81	1.20	.130		.110	.310	.550
					(.130)		(.390)	(.180)	(.700)
8	2	8	4.92	1.21	.130		.110	.310	.550
					(.130)		(.390)	(.180)	(.700)
9	2	23	18.90	.50	.050		1.0	.500	1.550

Break Type : Full Double-Ended Spray Line
Break Location: See Notes

VENT PATH NO.	Vol. Node No.		AREA (ft^2)	$\frac{L}{A}$ Ratio (1/ft)	HEAD LOSS, K				
10	3	4	70.70	.12	.600		7.730	.190	8.520
11	3	8	13.58	.38	.080	.110		.300	.490
12	3	9	37.00	.34	.160				.160
13	4	5	69.75	.10	.210			1.710	1.920
14	4	10	19.60	.64	.050				.050
15	5	6	12.36	.42	.080	.110		.300	.490
16	5	11	14.70	.86	.080				.080
17	6	7	12.36	.39	.080	.110			.190
18	6	12	5.32	2.35	.170				.170
19	7	8	13.69	.35	.080	.110			.190
20	7	13	5.32	2.35	.170				.170
21	8	14	5.32	2.37	.170				.170
22	9	10	71.22	.12	.600		7.73	.190	8.520
23	9	14	13.56	.38	.080	.110		.300	.490
24	9	15	9.43	.33	.110			.370	.480
					(.110)		(.550)	(0.0)	(.660)

Break Type : Full Double-Ended Spray Line
Break Location: See Notes

VENT PATH NO.	Vol. Node No.		AREA (ft^2)	$\frac{L}{A}$ Ratio (1/ft)	HEAD LOSS, K				
25	10	11	71.22	.10	.210			1.710	1.920
26	10	15	2.90	.84	.050			.430	.480
					(.050)		(.740)	(0.0)	(.790)
27	11	12	13.56	.38	.080	.110		.300	.490
28	11	16	8.13	.51	.060			.220	.280
					(.060)		(.200)	(0.0)	(.260)
29	12	13	13.56	.35	.080	.110			.19
					(.110)		(0.0)	(.050)	(.160)
30	12	16	5.32	1.25	.110	.010			.120
31	13	14	13.56	.35	.080	.110			.190
32	13	16	5.32	1.25	.110		.010		.120
					(.110)		(0.0)	(.050)	(.160)
33	14	15	5.32	1.27	.110		.010		.120
					(.110)		(0.0)	(.050)	(.160)
34	15	16	8.66	1.66	.636	.424	.880	.770	2.710
35	15	17	18.02	.20	.070		.540	.310	.920

Break Type : Full Double-Ended Spray Line
Break Location: See Notes

VENT PATH NO.	Vol. Node No.		AREA (ft^2)	$\frac{L}{A}$ Ratio (1/ft)	HEAD LOSS, K				
36	16	18	19.43	.15	.050		1.320	.270	1.640
37	17	18	29.80	.37	2.088	1.392	4.120	3.760	11.360
38**	17	19	1.20	.50	.050		.620	.170	.840
39*	18	20	12.30	.50	.050		.800	.410	1.260
40*	18	23	10.80	1.40	1.910		1.0	.400	3.310
41	19	20	37.90	.33	.120		.420	.470	1.010
42	19	21	91.60	.09	.040				.040
43	20	22	47.70	.17	.100		.060	.120	.280
44	20	23	114.50	.17	.040		1.0		1.040
45	21	22	45.80	.33	.120		.420	.470	1.010
46	22	23	119.10	.17	.040		1.0		1.040

* PRESSURE DEPENDENT AREAS.

TABLE 6.2-32 PRESSURIZER COMPARTMENT NODAL DESCRIPTION AND PEAK NODAL DIFFERENTIAL PRESSURE

Break Type : Full Double-Ended Spray Line
Break Location: See Notes

				Initial Conditions				Design	
Node Volume No. ( )	Description	Height (ft)	Volume (ft^3)	Temperature (°F)	Pressure (lb/in^2)(a)	Relative Humidity %	Calculated Peak Pressure Differential (lb/in^2(g))	Peak Pressure Differential (lb/in^2(g))	Margin %
1	Bounded by El.56'-6"-6 $\frac{11}{16}$" and El.63'-0"	6.44	902.1	120	15.2	90	5.37[1]	6.5	21.0
2	Bounded by El.50'-1 $\frac{7}{16}$" and El.56'-6 $\frac{11}{16}$"	6.44	719.0	120	15.2	90	7.48[1]	7.5	0.3
REGION BETWEEN EL.37'-6¾" AND EL.50'-1									
3	Bounded by Azimuth 60° and 150°	12.56	457.7	120	15.2	90	16.18[2]	18.5	14.3
4	Azimuth 150° and 180°	12.56	243.1	120	15.2	90	13.96[3]	18.5	32.5
5	Azimuth 180° and 240°	12.56	184.4	120	15.2	90	13.82[3]	18.5	33.9

Break Type : Full Double-Ended Spray Line
Break Location: See Notes

				Initial Conditions				Design	
6	Azimuth 240° and 300°	12.56	67.3	120	15.2	90	13.48[3]	18.5	37.2
7	Azimuth 300° and 360°	12.56	67.3	120	15.2	90	13.48[3]	18.5	37.2
8	Azimuth 0° and 60°	12.56	66.6	120	15.2	90	13.65[3]	18.5	35.5
REGION BETWEEN EL.25'-0" AND 37'-6¾"									
	Bounded by:								
9	Azimuth 60° and 150°	12.56	465.3	120	15.2	90	18.47[3]	21.5	16.4
10	Azimuth 150° and 180°	12.56	246.5	120	15.2	90	14.98[3]	21.5	43.5
11	Azimuth 180° and 240°	12.56	184.6	120	15.2	90	14.57[3]	21.5	47.6
12	Azimuth 240° and 300°	12.56	66.8	120	15.2	90	14.23[3]	21.5	51.1
13	Azimuth 300° and 360°	12.56	66.8	120	15.2	90	14.42[3]	21.5	49.1
14	Azimuth 0° and 60°	12.46	66.8	120	15.2	90	15.14[3]	21.5	42

Break Type : Full Double-Ended Spray Line
Break Location: See Notes

				Initial Conditions				Design	
REGION BETWEEN EL.22'-0" AND EL.25'-0"									
	Bounded by:								
15	Azimuth 0° and 180°	3.00	54.1	120	15.2	90	47.74[4]	50	5
16	Azimuth 180° and 360°	3.00	58.3	120	15.2	90	13.44[3]	25	86
REGION BETWEEN EL.16'-6" AND EL.22'-0"									
	Bounded by:								
17	Azimuth 0° and 180°	5.50	141.8	120	15.2	90	25.19[4]	25	0
18	Azimuth 180° and 360°	5.50	245.2	120	15.2	90	11.53[4]	25	117
REGION BETWEEN EL.8'-3" AND EL.16'-6"									
	Bounded by:								
19	Azimuth 0° and 180°	8.2	670.0	120	15.2	90	16.08[5]	25	55
20	Azimuth 180° and 360°	8.2	400.0	120	15.2	90	5.32[5]	25	370

Break Type : Full Double-Ended Spray Line
Break Location: See Notes

				Initial Conditions				Design	
REGION BETWEEN EL.0'-0" AND EL.8'-3"									
21	Bounded by: Azimuth 0° and 180°	8.25	746.0	120	15.2	90	15.25[5]	25	64
22	Azimuth 180° and 360°	8.25	518.0	120	15.2	90	7.71[5]	25	224
23	Containment		2.71×10^6	120	15.2	90	-		-

NOTES:

1 .Peak differential pressure occurs following a postulated rupture of the spray line at elevation 54'-0".
2. Peak differential pressure occurs following a postulated rupture of the spray line at elevation 45'-0".
3. Peak differential pressure occurs following a postulated rupture of the spray line at elevation 34'-0".
4. Peak differential pressure occurs following a postulated rupture of the spray line at elevation 24'-6".
5. Peak differential pressure occurs following a postulated rupture of the spray line at elevation 12'-0".

TABLE 6.2-33 MASS AND ENERGY RELEASE RATES SURGE LINE DOUBLE-ENDED GUILLOTINE

Time (sec)	Mass Rate (lb/sec)	Enthalpy (Btu/lb)
0.0	0.0	705.37
.001	8.66lE3	705.37
.01011	1.365E4	705.19
.01501	1.468E4	705.07
.02011	1.522E4	704.89
.02511	1.544E4	704.68
.03007	1.546E4	704.43
.03513	1.536E4	704.15
.04012	1.519E4	703.88
.045	1.498E4	703.64
.05012	1.475E4	703.44
.05513	1.453E4	703.3
.06011	1.433E4	703.24
.0651	1.417E4	703.24
.07009	1.403E4	703.29
.07506	1.394E4	703.37
.08008	1.389E4	703.46
.08525	1.386E4	703.54
.09013	1.385E4	703.6
.0951	1.386E4	703.63
.10009	1.388E4	703.63
.15005	1.382E4	703.04
.20014	1.348E4	701.91

TABLE 6.2-34 PRESSURIZER SKIRT CAVITY VENT PATH DESCRIPTION

Break Type: Full Double-Ended Surge Line
Break Location: Node #6

VENT PATH NO.	VOL. NODE NO.		AREA (ft^2)	$\frac{L}{A}$ Ratio (1/ft)	HEAD LOSS, K				
	From	To			FRICTION K, $\frac{fL}{d}$	TURNING AND OBSTRUCTION LOSS, K	EXPANSION K	CONTRACTION K	TOTAL
	1	2	21.75	.323			.421		.421
	1	6	22.86	.213				.127	.129
	2	3	13.93	.504	.064			.402	.466
	2	4	12.17	.209	.017		.56	.41	.987
	2	5	193.7	.033			.528		.528
	3	5	13.93	.47	.065		1.0		1.065
	4	5	30.4	.263			1.0		1.000
	5	6	.26	.2					1.770*

* Corresponds to an orifice coefficient = 0.6

TABLE 6.2-35 PRESSURIZER SKIRT CAVITY NODAL DESCRIPTION

Break Type: Full Double-Ended Surge Line
Break Location: Node #6

Node Volume No.( )	Description	Height (ft)	Volume (ft^3)	Initial Conditions			Calculated Peak Pressure Differential ($lb/in^2(g)$)	Design	
				Temperature (°F)	Pressure	Relative Humidity %		Peak Pressure Differential	Margin %
1	Volume from El.(-) 5'-11" to 0'-0"		128.7	120	15.2	90	186.4	N/A	N/A
2	Volume from El.26'-0" to (-)5'-11"		1262.8	120	15.2	90	83	N/A	N/A
3	Volume northeast of the skirt cavity at El.(-) 12'-0"		181.1	120	15.2	90	6.5	N/A	N/A
4	Volume southeast of the skirt cavity at El.(-) 12'-0"		121.6	120	15.2	90	5.8	N/A	N/A
5	Containment		$2.7x10^6$	120	15.2	90	-	N/A	N/A
6	Volume under pressurizer vessel up to El.0'-0"		166.0	120	15.2	90	190.8	N/A	N/A

CASES ANALYZED AND TABLES WITH RESULTS

	Case	Blowdown	Reflood	Post-Reflood	Balance Mass	Balance Energy
A	Double-Ended Pump Suction Min. S.I.	6.2-38	6.2-43	6.2-48	6.2-51	6.2-57
B	Double-Ended Pump Suction Max. S.I.	6.2-37	6.2-44	6.2-49	6.2-52	6.2-58
C	0.6 Doubled-Ended Pump Suction Min. S.I.	6.2-39	6.2-45	6.2-50	6.2-53	6.2-59
D	Three Square Foot Pump Suction Split Min. S.I.	6.2-40	6.2-46	6.2-46	6.2-54	6.2-60
E	Double-Ended Hot Leg Min. S.I.	6.2-41	-	-	6.2-55	6.2-61
F	Double-Ended Cold Leg Min. S.I.	6.2-42	6.2-47	6.2-47	6.2-56	6.2-62

NOTES:

1 Table 6.2-38 is a duplicate of 6.2-37. It is provided for continuity only. Safety injection has no effect on the blowdown portion of the transient.

2 Double Ended refers to the size of break. A "0.6" Double Ended Break is a guillotine break with a break equal to 60 percent of the double pipe area.

TABLE 6.2-37 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MAXIMUM SAFETY INJECTION) BLOWDOWN MASS AND ENERGY RELEASES

	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.00	0.0	0.0	0.0	0.0
.100	41585.6	23222.0	22323.9	12416.9
.201	42362.9	23834.2	24356.8	13556.8
.400	44284.6	25474.7	23645.2	13186.6
.601	44898.0	26473.8	21180.8	11822.1
.901	41387.1	25100.8	19367.0	10821.4
1.20	38232.0	23606.1	18928.0	10580.4
1.70	34320.6	21965.3	18905.3	10569.0
2.30	28812.4	19267.6	18555.8	10385.2
2.70	25338.9	17319.1	18824.2	10551.4
3.00	21136.7	14640.7	18262.6	10247.8
3.80	16809.7	11759.4	16575.7	9325.8
4.40	14536.5	10187.2	15689.3	8838.8
4.60	14136.0	9904.1	16858.6	9505.1
5.60	12420.5	8712.2	16234.2	9146.6
6.20	11982.8	8392.7	15921.2	8969.6
9.00	9684.1	6920.5	13988.9	7873.7
9.40	9877.1	7295.9	13944.1	7847.8
9.80	7860.5	6752.3	13413.0	7545.2
11.2	7195.3	5955.9	12055.2	6770.1
12.6	6392.5	5330.5	10878.9	6096.6
14.2	5270.8	4457.8	9458.3	5291.0
15.4	4624.5	4016.1	8372.2	4662.4
16.0	4276.8	3896.1	8226.6	4346.2
16.6	3798.5	3785.0	7662.0	3768.1
17.2	2917.2	3362.5	6194.4	2812.4
18.0	2096.5	2573.5	4799.2	1976.5
18.6	1633.2	2019.1	5400.4	2075.0
19.0	1355.9	1682.7	5037.7	1845.3
20.0	752.7	940.2	3059.0	969.9
21.0	338.0	423.7	67.7	19.5
22.4	257.7	323.5	0.0	0.0
22.6	224.5	282.0	23247.9	6138.9

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
22.8	218.1	274.0	0.0	0.0
24.2	0.0	0.0	0.0	0.0

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

[1]TABLE 6.2-38 **DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MINIMUM SAFETY INJECTION) BLOWDOWN MASS AND ENERGY RELEASES**

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
.100	41585.6	23222.0	22323.9	12416.9
.201	42362.9	23834.2	24356.8	13556.8
.400	44284.6	25474.7	23645.2	13186.6
.601	44898.0	26473.8	21180.8	11822.1
.901	41387.1	25100.8	19367.0	10821.4
1.20	38232.0	23606.1	18928.0	10580.4
1.70	34320.6	21965.3	18905.3	10569.0
2.30	28812.4	19267.6	18555.8	10385.2
2.70	25338.9	17319.1	18824.2	10551.4
3.00	21136.7	14640.7	18262.6	10247.8
3.80	16809.7	11759.4	16575.7	9325.8
4.40	14536.5	10187.2	15689.3	8838.8
4.60	14136.0	9904.1	16858.6	9505.1
5.60	12420.5	8712.2	16234.2	9146.6
6.20	11982.8	8392.7	15921.2	8969.6
9.00	9684.1	6920.5	13988.9	7873.7
9.40	9877.1	7295.9	13944.1	7847.8
9.80	7860.5	6752.3	13413.0	7545.2
11.2	7195.3	5955.9	12055.2	6770.1
12.6	6392.5	5330.5	10678.9	6096.6
14.2	5270.8	4457.8	9458.3	5291.0
15.4	4624.5	4016.1	8372.2	4662.4
16.0	4276.8	3896.1	8226.6	4346.2
16.6	3798.5	3785.0	7662.0	3768.1
17.2	2917.2	3362.5	6194.4	2812.4
18.0	2096.5	2573.5	4799.2	1976.0
18.6	1633.2	2019.1	5400.4	2075.0
19.0	1355.9	1682.7	5037.7	1845.0
20.0	752.7	940.2	3059.0	969.9
21.0	338.0	423.7	67.7	19.5

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
22.4	257.7	323.5	0.0	0.0
22.6	224.5	282.0	23247.9	6138.9
22.8	218.1	274.0	0.0	0.0
24.2	0.0	0.0	0.0	0.0

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-39 0.6 DOUBLE-ENDED PUMP SUCTION GUILLOTINE BLOWDOWN MASS AND ENERGY RELEASE

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
.100	37543.2	20967.3	20214.7	11237.1
.201	37944.1	21308.4	21975.7	12225.4
.401	36638.0	20904.7	21828.7	12163.4
.700	33054.7	19396.9	19709.0	11000.9
1.00	29205.3	17472.1	18881.6	10546.6
1.30	28590.9	17281.0	18523.2	10348.9
1.80	27872.1	17065.2	18556.9	10369.9
2.40	25425.5	15927.1	18594.2	10403.9
2.60	24340.3	15406.0	19417.0	10873.5
3.00	22073.9	14311.3	19071.6	10693.9
4.20	15784.4	10768.9	16796.7	9450.3
4.80	13599.1	9382.0	15784.7	8891.4
5.20	12779.3	8831.1	15259.8	8598.8
5.40	12360.3	8537.3	16045.8	9047.2
5.80	11786.2	8140.7	16003.2	9020.3
6.60	11030.8	7652.3	15476.1	8722.3
7.20	10707.8	7427.5	15174.0	8555.3
8.80	9628.9	6747.4	14262.7	8045.8
10.0	9104.3	6657.1	13591.9	7667.0
10.4	7141.4	5960.4	13369.8	7542.4
11.0	6719.3	5667.2	12860.2	7254.0
12.4	6245.8	5141.1	11696.9	6590.9
13.4	5760.0	4782.4	10939.6	6158.4
15.4	4607.1	3914.9	9283.6	5223.1
17.2	3763.9	3443.8	7877.2	4395.5
17.8	3350.2	3288.6	7820.0	4113.4
18.6	2336.4	2719.4	6629.6	3083.0
19.6	1659.1	2042.2	5097.7	2117.1
20.2	1337.2	1656.2	5887.9	2305.3
20.8	1054.2	1312.7	4080.9	1518.5

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
21.4	793.8	992.3	3612.4	1187.6
22.8	250.8	315.8	501.5	143.9
23.4	227.4	350.0	8.6	2.4
25.4	0.0	0.0	0.0	0.0

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-40 THREE-FOOT SQUARED PUMP SUCTION SPLIT BLOWDOWN MASS AND ENERGY RELEASES

BREAK PATH NO. 1 FLOW

TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC
0.000	0.0	0.0
.101	43200.0	24145.5
.301	42783.0	24131.5
.801	37700.1	21931.5
1.00	35905.1	21070.4
1.60	32918.7	19583.1
2.30	27472.4	16545.4
2.70	26963.0	16334.6
5.60	21735.5	13305.5
6.60	20831.6	12649.3
8.00	19175.3	11651.3
10.4	18279.7	11095.6
11.0	18588.4	11214.5
11.8	16835.9	10477.7
15.6	14537.2	9132.5
19.2	12060.2	7791.2
19.6	11802.9	7649.8
22.0	10229.8	6800.5
24.4	8417.9	5836.8
26.2	6696.8	5035.3
28.2	4227.0	3614.1
29.2	3443.9	2879.8
30.2	3043.1	2425.1
30.8	2476.7	1986.1
32.0	3360.7	1801.3
32.8	2831.4	1496.9
33.0	3522.0	1619.7
33.6	3227.6	1471.0
34.4	4423.5	1675.5
34.6	4552.6	1634.7
36.0	2084.9	700.7

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

BREAK PATH NO. 1 FLOW

TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC
36.4	1762.9	602.4
36.6	1002.9	336.6
36.8	1110.8	396.7
37.2	0.0	0.0

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-41 DOUBLE-ENDED HOT LEG GUILLOTINE BLOWDOWN MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
.100	42412.0	27585.7	26857.8	17356.3
.201	36857.8	24038.5	23284.6	14946.6
.301	36716.0	23961.5	20962.8	13271.1
.400	35842.5	23421.4	19773.5	12306.1
.600	34749.8	22805.0	18504.8	11159.1
.800	33776.8	22305.6	17743.6	10457.2
1.00	32541.3	21657.0	17433.2	10108.4
1.50	30093.3	20391.3	17866.4	10093.6
1.90	28232.4	19272.8	18090.5	10122.4
2.50	25753.2	17648.4	17757.9	9898.8
2.90	24401.8	16701.8	17349.0	9673.7
3.70	22280.4	15138.3	16388.6	9161.6
4.40	20976.4	14165.7	15338.1	8612.7
5.00	20574.8	13790.2	14392.2	8125.0
5.20	20586.1	13722.5	14353.9	8116.9
6.20	20286.9	13372.7	13168.3	7469.0
7.80	19363.8	12704.9	11142.1	6359.3
8.00	14861.3	10628.5	10887.7	6219.4
8.80	14379.7	10294.9	9891.5	5672.8
10.2	13371.4	9390.4	8200.0	4756.8
10.8	12710.4	8915.7	7477.7	4375.0
11.6	12053.0	8399.5	6581.5	3910.8
12.6	10953.0	7664.2	5594.4	3416.6
13.4	9880.5	7046.4	4931.1	3094.3
14.2	8707.2	6450.8	4369.5	2826.8
14.8	7604.6	5945.2	3991.6	2648.9
15.2	6516.0	5488.7	3717.0	2521.5
15.8	4549.6	4601.4	3218.8	2305.3
16.2	3592.6	3958.6	2828.8	2134.9
17.0	2419.4	2853.6	1901.9	1791.0

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
17.8	1602.7	1940.9	1067.1	1304.1
19.0	1036.5	1268.7	542.1	673.1
20.4	303.3	353.9	242.4	303.3
21.0	0.0	0.0	0.0	0.0

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-42 DOUBLE-ENDED COLD LEG GUILLOTINE BLOWDOWN MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW		SI SPILL PATH FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0	0.0	0.0
.100	25295.8	14098.9	54132.4	30247.2	4429.2	389.8
.201	23118.0	12892.3	60165.0	33618.5	4384.0	385.8
.401	23004.3	12839.9	59305.1	33129.2	4293.9	377.9
.901	22213.9	12526.9	54065.6	30190.6	4068.5	358.0
1.20	21352.5	12166.1	52444.1	29323.8	3933.5	346.1
1.50	20915.4	12052.1	49796.5	27885.8	3798.3	334.3
2.00	19832.2	11605.5	42660.9	23983.9	3572.7	314.4
2.20	17682.0	10389.3	40816.5	22980.5	3482.9	306.5
2.30	16563.3	9748.7	43663.7	24595.2	3463.0	304.7
2.40	15811.8	9320.4	37164.6	20951.3	3445.8	303.2
2.60	14854.2	8778.0	39095.1	22064.1	3411.2	300.2
2.80	14140.2	8372.4	35745.6	20192.2	3376.5	297.1
3.30	13004.1	7737.5	33471.6	18931.5	3290.1	289.5
3.70	12331.6	7389.3	28681.3	16221.2	3221.0	283.4

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW		SI SPILL PATH FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
3.90	12028.5	7246.4	27163.6	15356.4	3186.4	280.4
4.60	11000.3	6818.5	25391.6	14334.3	3055.6	268.9
6.00	8967.0	6022.2	23959.4	13546.2	2743.9	241.5
7.60	6796.1	5006.5	21012.0	12080.6	2547.5	224.2
8.80	6037.4	4350.0	18836.2	11072.6	2473.5	217.7
9.80	6077.5	4254.8	16674.0	10338.3	2316.1	203.8
10.2	6005.8	4468.8	14404.1	9361.3	2252.9	198.3
10.6	4750.8	4257.8	12026.7	8293.7	2189.8	192.7
11.2	4073.1	3795.5	9437.7	7068.3	2121.9	186.7
11.6	3887.8	3597.3	8317.5	6517.9	2114.0	186.0
13.8	2879.2	2879.0	5400.9	4716.3	2005.4	176.5
14.2	2658.9	2781.0	4022.0	4332.5	1953.6	186.0
15.0	1302.8	1577.4	3449.5	3296.8	1850.5	
16.2	919.6	1139.2	5502.0	2353.2	1799.6	
18.0	368.7	463.5	3956.5	1286.5	1756.5	

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW		SI SPILL PATH FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
18.8	221.2	280.3	3791.2	1120.4	1666.0	176.5
19.4	112.6	143.7	1522.3	438.2	1598.2	154.6
20.2	90.7	116.3	1031.1	293.5	1558.3	146.6
20.4	101.2	130.0	0.0	0.0	1561.0	137.4
22.0	0.0	0.0	0.0	0.0	1583.0	139.3
20.4	101.2	130.0	0.0	0.0	1561.0	137.4
22.0	0.0	0.0	0.0	0.0	1583.0	139.3

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-43 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MINIMUM SAFETY INJECTION) REFLOOD MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
24.2	0.0	0.0	0.0	0.0
24.7	.0	.0	146.7	12.9
25.0	.0	.0	146.7	12.9
25.3	44.3	52.3	146.7	12.9
27.3	135.9	160.7	146.7	12.9
28.3	164.2	194.1	146.7	12.9
29.3	610.0	725.5	5265.9	680.5
30.3	772.7	921.3	6546.7	920.9
31.3	755.9	901.0	6421.0	908.4
33.3	717.3	854.6	6123.1	874.5
35.3	682.1	812.2	5853.2	842.4
36.3	665.8	792.6	5724.2	827.4
38.3	635.8	756.5	5482.3	799.3
40.3	608.6	724.0	5259.5	773.5
42.3	583.9	694.3	5053.3	749.7
44.3	561.3	667.3	4861.9	727.5
46.3	540.6	642.3	4683.5	706.8
48.3	521.3	619.3	4516.5	687.4
50.3	503.4	597.8	4359.5	669.1
52.3	486.6	577.8	4211.5	651.7
56.3	456.0	541.2	3938.5	619.4

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
57.3	493.6	585.8	339.1	254.5
58.3	548.0	651.3	360.8	286.0
62.3	501.8	595.9	341.1	259.4
63.3	490.3	582.2	336.2	252.9
67.3	448.1	531.7	318.5	229.2
77.3	361.5	428.5	282.6	182.3
85.3	309.4	366.4	261.5	155.0
93.3	268.5	317.8	245.3	134.3
101.3	237.1	280.6	233.1	118.8
109.3	213.7	252.8	224.2	107.6
121.3	190.6	225.4	215.6	96.8
133.3	177.8	210.3	210.9	90.8
147.3	170.9	202.2	208.2	87.5
153.2	169.7	200.6	207.7	86.8

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-44 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MAXIMUM SAFETY INJECTION) REFLOOD MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
24.2	0.0	0.0	0.0	0.0
25.0	.0	.0	.0	.0
25.1	57.6	68.1	.0	.0
25.2	49.6	58.6	.0	.0
26.2	94.5	111.7	.0	.0
27.3	134.8	159.4	.0	.0
28.3	174.4	206.2	.0	.0
29.3	822.1	983.7	6923.0	941.1
30.3	830.2	990.6	6969.3	968.2
32.3	790.6	942.8	6675.9	937.2
33.3	772.0	920.4	6536.0	920.9
34.3	754.2	899.0	6401.2	905.2
36.3	721.5	859.6	6148.0	875.6
38.3	692.1	824.3	5916.0	848.7
40.3	665.6	792.4	5702.6	823.9
42.3	641.5	763.4	5505.6	801.1
43.3	630.2	749.9	5412.6	790.4
45.3	699.0	724.4	5236.1	769.9
47.3	589.3	700.8	5071.0	750.6
47.4	588.4	699.7	5063.0	749.7
49.3	571.0	678.9	4916.2	732.4

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
50.3	562.3	668.4	4842.3	723.7
52.3	545.7	648.6	4701.0	706.9
54.3	530.1	629.8	4567.4	690.8
56.3	515.3	612.1	4440.8	675.4
58.3	501.3	595.3	4320.5	664.7
59.4	201.1	237.9	1111.6	292.9
71.4	196.0	231.8	1119.7	289.7
80.4	192.6	227.8	1125.1	287.0
95.4	187.2	221.4	1134.3	282.4
99.4	185.8	219.8	1136.7	281.2
115.4	180.2	213.2	1146.6	276.0
119.4	178.9	211.5	1149.1	274.7
151.4	168.3	199.0	1168.7	263.5
155.1	167.1	197.6	1170.9	262.2

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-45 0.6 DOUBLE-ENDED PUMP SUCTION GUILLOTINE REFLOOD MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
25.4	0.0	0.0	0.0	0.0
25.9	.0	.0	146.7	12.9
26.3	42.0	49.6	146.7	12.9
28.5	120.2	142.1	146.7	12.9
29.5	144.9	171.3	146.7	12.9
30.5	455.8	541.1	4148.9	519.8
31.5	559.1	667.6	6207.2	897.6
32.5	317.2	378.3	6526.8	1138.9
32.7	538.4	642.9	6066.6	885.7
33.5	639.4	762.2	6244.6	825.6
34.5	502.8	600.1	5819.6	856.8
35.5	247.2	294.6	6136.8	1108.9
36.5	462.5	551.1	6212.9	716.4
36.8	414.1	493.5	6151.2	656.6
37.5	408.5	486.8	6102.7	631.8
38.6	586.9	698.8	5470.2	805.0
40.6	560.4	666.9	5245.5	779.0
42.6	537.1	638.9	5042.0	754.6
44.6	515.7	613.3	4852.6	731.9
46.6	496.2	589.9	4676.0	710.7
48.6	478.1	568.2	4510.6	690.9

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
49.6	469.6	558.0	4431.7	681.4
53.6	438.3	520.6	4138.2	646.1
57.6	410.8	487.6	3874.7	614.2
58.7	373.2	442.4	300.8	203.1
59.7	515.1	612.5	368.0	297.2
61.7	495.9	589.6	359.0	285.1
69.7	415.1	492.8	321.5	234.4
76.7	358.1	424.7	295.5	199.8
80.7	331.2	392.6	283.2	183.8
88.7	286.6	339.6	263.3	157.9
96.7	251.7	298.1	248.0	138.2
104.7	224.7	266.0	236.3	123.3
132.7	175.5	207.6	215.8	97.2
157.5	164.4	194.4	211.1	91.3

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-46 THREE-FOOT SQUARED PUMP SUCTION SPLIT REFLOOD MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC
37.2	0.0	0.0
37.7	146.7	12.9
39.3	248.4	133.8
41.2	308.7	206.1
42.2	3136.3	744.6
43.3	7095.2	1801.1
44.3	6988.0	1780.4
46.3	6679.0	1704.6
48.3	6386.4	1633.1
49.3	6248.8	1599.7
51.3	5990.4	1537.4
53.3	5752.2	1480.5
55.3	5531.5	1428.2
57.3	5326.5	1380.0
61.3	4956.3	1293.5
62.3	4870.9	1273.6
64.3	4707.5	1235.7
68.3	4406.5	1165.9
69.4	1018.4	1110.5
70.4	1351.3	1582.8
71.4	1320.5	1546.2
73.4	1106.1	1275.7
74.4	940.1	1022.8
75.4	813.5	845.6
80.4	673.9	662.2
84.4	613.4	585.1
88.4	563.8	522.3
98.4	472.0	407.4
106.4	425.0	349.2
118.4	384.2	299.0

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

BREAK PATH NO.1 FLOW

TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC
132.4	363.2	273.3
152.4	355.2	263.5
161.0	355.0	263.3
161.1	528.2	179.3
100000.0	528.2	70.2

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-47 DOUBLE-ENDED COLD LEG GUILLOTINE BLOWDOWN MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW		SI SPILL PATH FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
22.0	0.0	0.0	0.0	0.0	0.0	0.0
22.5	.0	.0	.0	.0	1649.6	145167.4
22.9	.0	.0	.0	.0	1632.2	143630.8
23.0	35.6	42.0	.0	.0	1618.7	142449.4
23.3	17.0	20.1	.0	.0	1608.7	141562.7
24.0	33.5	39.6	.0	.0	1577.5	138820.2
26.0	58.7	69.4	.0	.0	1498.9	131900.0
28.0	75.6	89.4	.0	.0	1425.9	125481.1
29.0	83.5	98.8	264.7	36.6	1390.9	122398.8
30.0	113.4	134.2	5873.9	836.3	1362.4	119889.6
31.0	115.1	136.3	6051.5	872.8	1329.8	117020.7
32.9	112.1	132.7	5757.4	844.1	1270.5	111804.2
34.0	110.5	130.8	5594.0	827.7	1237.6	108912.0
35.0	109.1	129.1	5452.4	813.4	1208.6	106357.2
36.0	107.8	127.5	5316.8	799.7	1180.3	103868.7
37.0	106.5	126.1	5187.0	786.5	1152.8	101442.3
39.0	104.2	123.3	4943.2	761.6	1099.6	96761.5

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW		SI SPILL PATH FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
40.0	103.1	122.0	4828.4	749.8	146.7	12908.2
42.0	101.1	119.7	4611.4	727.4	146.7	12908.2
44.0	99.3	117.5	4409.4	706.4	146.7	12908.2
46.0	97.6	115.5	4220.4	686.7	146.7	12908.2
47.1	99.4	117.6	236.4	279.5	146.7	12908.2
51.1	98.0	116.0	232.5	274.8	146.7	12908.2
68.1	93.5	110.6	219.3	259.3	146.7	12908.2
99.1	89.2	105.5	229.0	245.8	146.7	12908.2
103.1	88.7	104.9	230.7	244.3	146.7	12908.2
135.1	84.5	100.0	247.9	232.2	146.7	12908.2
137.1	84.3	99.7	249.1	231.4	146.7	12908.2
173.1	78.8	93.2	268.9	216.2	146.7	12908.2
203.1	73.9	87.4	283.2	202.7	146.7	12908.2
212.1	72.4	85.6	287.3	198.7	146.7	12908.2
212.2	114.1	134.8	322.6	28.4	146.7	12908.2
1000.0	79.0	93.3	357.8	31.5	146.7	12908.2
10000.0	41.7	49.3	395.0	34.8	146.7	12908.2
100000.0	21.7	25.7	415.0	36.5	146.7	12908.2

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-48 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MINIMUM SAFETY INJECTION) POST-REFLOOD MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
165.4	221.0	273.5	365.3	127.7
180.4	219.1	271.1	367.3	127.7
185.4	206.3	255.3	380.0	125.1
190.4	206.5	255.6	379.8	124.8
205.4	205.0	253.7	381.3	124.7
250.4	200.3	247.9	386.1	124.5
255.4	200.4	248.0	386.0	124.3
280.4	197.6	244.5	388.8	124.3
290.4	197.1	243.9	389.2	124.0
320.4	193.7	239.7	392.7	124.0
330.4	192.6	238.4	393.7	123.9
335.4	177.2	219.3	409.1	122.3
355.4	175.1	216.7	411.2	122.2
370.4	173.1	214.2	413.2	122.3
375.4	172.8	213.8	413.6	122.2
395.4	170.1	210.5	416.3	122.3
405.4	168.9	209.0	417.5	122.3
670.4	168.9	209.0	417.5	122.3
670.5	98.8	121.4	487.5	129.0
825.4	94.6	116.2	491.7	125.3
895.4	93.1	114.3	493.3	123.4
910.4	92.7	113.9	493.6	123.0
995.4	90.8	111.5	495.6	120.7

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
1065.4	89.6	110.1	496.7	118.6
1210.4	87.3	107.1	499.1	117.9
1305.4	85.7	105.2	500.6	114.2
1350.4	85.0	104.3	501.4	112.4
1565.4	81.8	100.4	504.6	108.9
1761.0	81.7	100.3	504.6	108.5
1761.1	87.4	112.2	493.9	55.4
3600.0	73.3	95.9	513.1	59.4
3600.1	57.8	68.2	528.6	46.8
10000.0	42.9	50.7	543.4	47.8
100000.0	22.3	26.4	564.0	49.6
1000000.0	10.7	12.7	575.7	50.7

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-49 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MAXIMUM SAFETY INJECTION) POST-REFLOOD MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
165.0	184.7	230.2	1276.8	216.8
173.0	183.8	229.1	1277.7	216.9
203.0	183.7	229.0	1277.8	215.7
208.0	183.0	228.1	1278.5	215.7
218.0	183.4	228.6	1278.1	215.2
233.0	183.0	228.1	1278.5	214.7
238.0	182.3	227.2	1279.2	214.7
258.0	182.7	227.7	1278.8	208.2
263.0	181.9	226.7	1279.6	208.2
278.0	182.0	226.8	1279.5	207.7
328.0	180.9	225.5	1280.6	206.2
333.0	181.5	226.2	1280.0	205.8
348.0	180.5	225.0	1281.0	205.5
378.0	180.6	225.1	1280.9	204.4
408.0	179.6	223.8	1281.9	198.1
413.0	180.0	224.3	1281.5	197.8
428.0	179.4	223.6	1282.1	197.5
438.0	179.7	224.0	1261.8	197.1
463.0	178.9	223.0	1282.6	196.5
678.0	178.8	222.8	1282.7	196.0
678.1	98.6	122.0	1362.9	212.1
688.0	98.4	121.6	1363.1	211.8
773.0	96.0	118.7	1365.5	209.3
808.0	95.0	117.5	1366.5	208.3
938.0	92.1	113.9	1369.4	204.0
1208.0	87.3	107.9	1374.2	197.3
1213.0	87.3	107.8	1374.2	197.1
1458.0	83.3	102.8	1378.2	190.3
1636.9	83.3	102.8	1378.2	190.3
1637.0	88.9	113.9	1372.6	130.3

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
3600.0	73.3	95.9	1388.2	134.6
3600.1	57.8	68.2	1403.7	123.5
10000.0	42.9	50.7	1418.6	124.8
100000.0	22.3	26.4	1439.2	126.6
1000000.0	10.7	12.7	1450.8	127.7

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-50 0.6 DOUBLE-ENDED PUMP SUCTION GUILLOTINE POST-REFLOOD MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
164.8	224.7	280.2	361.6	127.5
169.8	224.3	279.7	362.0	127.4
174.8	211.7	264.0	374.6	124.7
204.8	208.2	259.6	378.2	124.6
214.8	207.1	258.2	379.3	124.6
224.8	206.8	257.6	379.6	124.3
244.8	205.0	255.7	381.3	124.1
274.8	202.2	252.1	384.2	123.9
279.8	201.7	251.5	384.6	123.9
314.8	197.6	246.4	388.7	123.8
319.8	181.4	226.2	404.9	122.3
334.8	180.0	224.5	406.3	122.2
354.8	177.6	221.5	408.8	122.3
359.8	177.6	221.4	408.8	122.1
404.8	171.7	214.1	414.6	122.3
414.8	170.8	212.9	415.6	122.2
439.8	167.6	209.0	418.7	122.3
704.8	167.4	208.7	418.9	122.2
704.9	98.0	121.2	488.4	128.9
709.8	97.8	121.0	488.5	128.8
789.8	95.6	118.2	490.8	126.8
829.8	94.6	117.0	491.7	125.8
1074.8	89.6	110.8	496.8	118.9
1214.8	87.3	107.9	499.0	118.4
1314.8	85.7	105.9	500.7	114.5
1449.8	83.5	103.1	502.9	112.5
1454.8	83.4	103.0	502.9	112.2
1574.8	81.8	101.0	504.5	109.2
1769.8	81.8	101.0	504.6	108.9
1769.9	87.5	112.3	498.9	55.6

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
3600.0	73.4	96.2	512.9	59.5
3600.1	57.8	68.2	528.6	46.5
10000.0	42.9	50.7	543.4	47.8
100000.0	22.3	26.4	564.0	49.6
1000000.0	10.7	12.7	575.7	50.7

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-51 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MINIMUM SAFETY INJECTION) MASS BALANCE

	TIME (SECONDS)	.00	24.20	24.20	153.24	675.36	1761.02	3600.00
		MASS (THOUSAND LBM)						
INITIAL	IN RCS AND ACC	802.38	802.38	802.38	802.38	802.38	802.38	802.38
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	71.34	370.38	1006.97	2085.27
	TOTAL ADDED	0.00	0.00	0.00	71.34	370.38	1006.97	2085.27
*** TOTAL AVAILABLE ***		802.38	802.38	802.38	873.72	1172.76	1809.35	2887.65
DISTRIBUTION	REACTOR COOLANT	505.43	96.42	96.47	167.92	167.92	167.92	167.92
	ACCUMULATOR	296.95	208.05	208.01	-.00	0.00	0.00	0.00
	TOTAL CONTENTS	802.38	304.47	304.47	167.92	167.92	167.92	167.92
EFFLUENT	BREAK FLOW	0.00	497.90	497.90	705.79	1004.84	1641.43	2719.73
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	497.90	497.90	705.79	1004.84	1641.43	2719.73
*** TOTAL ACCOUNTABLE ***		802.38	802.37	802.37	873.71	1172.75	1809.34	2887.65

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-52 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MAXIMUM SAFETY INJECTION) MASS BALANCE

	TIME (SECONDS)	.00	24.20	24.20	155.09	682.99	1636.62	3600.00
		MASS (THOUSAND LBM)						
INITIAL	IN RCS AND ACC	802.38	802.38	802.38	802.38	802.38	802.38	802.38
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	184.61	937.28	2331.36	5200.49
	TOTAL ADDED	0.00	0.00	0.00	184.61	937.28	2331.36	5200.49
*** TOTAL AVAILABLE ***		802.38	802.38	802.38	986.98	1739.66	3133.74	6002.87
DISTRIBUTION	REACTOR COOLANT	505.43	96.42	96.47	177.04	177.04	177.04	177.04
	ACCUMULATOR	296.95	208.05	208.01	-.00	0.00	0.00	0.00
	TOTAL CONTENTS	802.38	304.47	304.47	177.04	177.04	177.04	177.04
EFFLUENT	BREAK FLOW	0.00	497.90	497.90	809.94	1562.61	2956.69	5825.82
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	497.90	497.90	809.94	1562.61	2956.69	5825.82
*** TOTAL ACCOUNTABLE ***		802.38	802.37	802.37	986.98	1739.65	3133.73	6002.86

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-53 0.6 DOUBLE-ENDED PUMP SUCTION GUILLOTINE MASS BALANCE

	TIME (SECONDS)	.00	25.40	25.40	157.46	709.83	1769.76	3600.00
		MASS (THOUSAND LBM)						
INITIAL	IN RCS AND ACC	802.38	802.38	802.38	802.38	802.38	802.38	802.38
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	72.58	392.14	1013.65	2086.82
	TOTAL ADDED	0.00	0.00	0.00	72.58	392.14	1013.65	2086.82
*** TOTAL AVAILABLE ***		802.38	802.38	802.38	874.95	1194.52	1816.02	2889.20
DISTRIBUTION	REACTOR COOLANT	505.43	89.41	100.05	171.60	171.60	171.60	171.60
	ACCUMULATOR	296.95	214.19	203.55	-.00	0.00	0.00	0.00
	TOTAL CONTENTS	802.38	303.61	303.61	171.60	171.60	171.60	171.60
EFFLUENT	BREAK FLOW	0.00	498.77	498.77	703.35	1022.92	1644.42	2717.60
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	498.77	498.77	703.35	1022.92	1644.42	2717.60
*** TOTAL ACCOUNTABLE ***		802.38	802.37	802.37	874.95	1194.51	1816.02	2889.19

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-54 THREE-FOOT SQUARED PUMP SUCTION SPLIT MASS BALANCE

	TIME (SECONDS)	.00	37.20	37.20	160.96
	MASS (THOUSAND LBM)				
INITIAL	IN RCS AND ACC	802.38	802.38	802.38	802.38
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	65.37
	TOTAL ADDED	0.00	0.00	0.00	65.37
*** TOTAL AVAILABLE ***		802.38	802.38	802.38	867.75
DISTRIBUTION	REACTOR COOLANT	505.43	109.11	109.15	180.01
	ACCUMULATOR	296.95	199.15	199.10	-.00
	TOTAL CONTENTS	802.38	308.25	308.25	180.01
EFFLUENT	BREAK FLOW	0.00	494.12	494.12	687.73
	ECCS SPILL	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	494.12	494.12	687.73
*** TOTAL ACCOUNTABLE ***		802.38	802.37	802.37	867.74

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-55 DOUBLE-ENDED HOT LEG GUILLOTINE MASS BALANCE

	TIME (SECONDS)	.00	21.00	21.00
	MASS (THOUSAND LBM)			
INITIAL	IN RCS AND ACC	802.38	802.38	802.38
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00
	TOTAL ADDED	0.00	0.00	0.00
*** TOTAL AVAILABLE ***		802.38	802.38	802.38
DISTRIBUTION	REACTOR COOLANT	505.43	96.42	96.50
	ACCUMULATOR	296.95	232.00	231.92
	TOTAL CONTENTS	802.38	328.42	328.42
EFFLUENT	BREAK FLOW	0.00	473.96	473.96
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	473.96	473.96
*** TOTAL ACCOUNTABLE ***		802.38	802.38	802.38

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-56 DOUBLE-ENDED COLD LEG GUILLOTINE MASS BALANCE

	TIME (SECONDS)	.00	22.00	22.00	212.09
	MASS (THOUSAND LBM)				
INITIAL	IN RCS AND ACC	802.38	802.38	802.38	802.38
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	110.90
	TOTAL ADDED	0.00	0.00	0.00	110.90
*** TOTAL AVAILABLE ***		802.38	802.38	802.38	913.28
DISTRIBUTION	REACTOR COOLANT	505.43	68.43	84.88	162.79
	ACCUMULATOR	296.95	176.40	159.95	-.00
	TOTAL CONTENTS	802.38	244.33	244.83	162.79
EFFLUENT	BREAK FLOW	0.00	504.84	504.84	548.39
	ECCS SPILL	0.00	52.70	52.70	101.89
	TOTAL EFFLUENT	0.00	557.54	557.54	750.27
*** TOTAL ACCOUNTABLE ***		802.38	802.37	802.37	913.06

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-57 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MINIMUM SAFETY INJECTION) ENERGY BALANCE

	TIME (SECONDS)	.00	24.20	24.20	153.24	675.36	1761.02	3600.00
	ENERGY (MILLION BTU)							
INITIAL ENERGY	IN RCS, ACC,S GEN	894.34	894.34	894.34	894.34	894.34	894.34	894.34
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	6.28	32.59	88.61	183.50
	DECAY HEAT	0.00	8.24	8.24	24.94	73.37	155.29	278.68
	HEAT FROM SECONDARY	0.00	-17.91	-17.91	-17.91	-9.20	-.33	-.33
	TOTAL ADDED	0.00	-9.66	-9.66	13.31	96.77	243.58	461.85
*** TOTAL AVAILABLE ***		894.34	884.68	884.68	907.66	991.11	1137.92	1356.19
DISTRIBUTION	REACTOR COOLANT	303.11	19.36	19.37	42.95	42.95	42.95	42.95
	ACCUMULATOR	26.13	18.31	18.30	-.00	0.00	0.00	0.00
	CORE STORED	26.85	13.84	13.84	5.09	4.91	4.32	3.33
	PRIMARY METAL	132.98	125.38	125.38	108.68	80.13	61.82	46.50
	SECONDARY METAL	121.87	118.64	118.64	108.10	89.06	64.06	47.45
	STEAM GENERATOR	283.40	278.28	278.18	246.96	204.34	153.83	117.22
	TOTAL CONTENTS	894.34	573.71	573.71	511.77	421.39	326.98	257.45

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

	TIME (SECONDS)	.00	24.20	24.20	153.24	675.36	1761.02	3600.00
	ENERGY (MILLION BTU)							
EFFLUENT	BREAK FLOW	0.00	310.98	310.98	395.93	569.76	810.98	1098.79
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	310.98	310.98	395.93	569.76	810.98	1098.79
*** TOTAL ACCOUNTABILITY ***		894.34	884.69	884.69	907.70	991.15	1137.96	1356.24
*** TOTAL ACCOUNTABILITY ***		894.34	884.69	884.69	907.70	991.15	1137.96	1356.24

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-58 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MAXIMUM SAFETY INJECTION) ENERGY BALANCE

	TIME (SECONDS)	.00	24.20	24.20	155.19	682.99	1636.86	3600.00
	ENERGY (MILLION BTU)							
INITIAL ENERGY	IN RCS, ACC,S GEN	894.34	894.34	894.34	894.34	894.34	894.34	894.34
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	16.25	82.48	205.16	457.64
	DECAY HEAT	0.00	8.24	8.24	25.15	73.93	146.81	279.47
	HEAT FROM SECONDARY	0.00	-17.91	-17.91	-17.91	-9.12	-.33	-.33
	TOTAL ADDED	0.00	-9.66	-9.66	23.49	147.30	351.64	736.79
*** TOTAL AVAILABLE ***		894.34	884.68	884.68	917.83	1041.64	1245.99	1631.13
DISTRIBUTION	REACTOR COOLANT	303.11	19.36	19.37	44.71	44.71	44.71	44.71
	ACCUMULATOR	26.13	18.31	18.30	-.00	0.00	0.00	0.00
	CORE STORED	26.85	13.84	13.84	5.09	4.91	4.39	3.33
	PRIMARY METAL	132.98	125.38	125.38	106.67	79.26	62.36	46.55
	SECONDARY METAL	121.87	118.64	118.64	108.18	87.65	64.36	47.54
	STEAM GENERATOR	283.40	278.18	278.18	247.16	200.80	154.57	117.46
	TOTAL CONTENTS	894.34	573.71	573.71	511.80	417.33	330.38	259.59
EFFLUENT	BREAK FLOW	0.00	310.98	310.98	406.09	624.37	915.67	1371.61
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	310.98	310.98	406.09	624.37	915.67	1371.61
*** TOTAL ACCOUNTABILITY ***		894.34	884.69	884.69	917.89	1041.70	1246.05	1631.19

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-59 0.6 DOUBLE-ENDED PUMP SUCTION GUILLOTINE ENERGY BALANCE

	TIME (SECONDS)	.00	25.40	25.40	157.46	709.83	1769.76	3600.00
	ENERGY (MILLION BTU)							
INITIAL ENERGY	IN RCS, ACC,S GEN	894.34	894.34	894.34	894.34	894.34	894.34	894.34
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	6.39	34.51	89.20	183.64
	DECAY HEAT	0.00	8.62	8.62	25.60	76.99	155.58	279.32
	HEAT FROM SECONDARY	0.00	-17.09	-17.09	-17.09	-7.79	.48	.48
	TOTAL ADDED	0.00	-8.48	-8.48	14.89	103.70	246.26	463.44
*** TOTAL AVAILABLE ***		894.34	885.87	885.87	909.23	998.05	1140.61	1357.78
DISTRIBUTION	REACTOR COOLANT	303.11	18.73	19.66	44.54	44.54	44.54	44.54
	ACCUMULATOR	26.13	18.85	17.91	-.00	0.00	0.00	0.00
	CORE STORED	26.85	13.79	13.79	5.09	4.91	4.33	3.33
	PRIMARY METAL	132.98	125.80	125.80	109.62	79.92	62.03	46.66
	SECONDARY METAL	121.87	119.72	119.72	109.46	88.73	64.37	47.73
	STEAM GENERATOR	283.40	281.41	281.41	250.76	204.08	154.60	117.94
	TOTAL CONTENTS	894.34	578.30	578.30	519.47	422.17	329.86	260.20
EFFLUENT	BREAK FLOW	0.00	307.58	307.58	389.94	576.05	810.92	1097.77
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	307.58	307.58	389.94	576.05	810.92	1097.77
*** TOTAL ACCOUNTABLE ***		894.34	885.88	885.88	909.41	998.22	1140.79	1357.96

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-60 THREE-FOOT SQUARED PUMP SUCTION SPLIT ENERGY BALANCE

	TIME (SECONDS)	.00	37.20	37.20	160.96
	ENERGY (MILLION BTU)				
INITIAL ENERGY	IN RCS, ACC,S GEN	894.34	894.34	894.34	894.34
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	5.75
	DECAY HEAT	0.00	14.31	14.31	29.85
	HEAT FROM SECONDARY	0.00	-15.58	-15.58	-15.58
	TOTAL ADDED	0.00	-1.27	-1.27	20.02
*** TOTAL AVAILABLE ***		894.34	893.08	893.08	914.37
DISTRIBUTION	REACTOR COOLANT	303.11	22.71	22.72	44.95
	ACCUMULATOR	26.13	17.52	17.52	-.00
	CORE STORED	26.85	12.29	12.29	5.09
	PRIMARY METAL	132.98	125.51	125.51	110.80
	SECONDARY METAL	121.87	121.47	121.47	111.52
	STEAM GENERATOR	283.40	286.72	286.72	256.62
	TOTAL CONTENTS	894.34	586.22	586.22	528.97
EFFLUENT	BREAK FLOW	0.00	306.86	306.86	385.43
	ECCS SPILL	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	306.86	306.86	385.43
*** TOTAL ACCOUNTABLE ***		894.34	893.08	893.08	914.40

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-61 DOUBLE-ENDED HOT LEG GUILLOTINE ENERGY BALANCE

	TIME (SECONDS)	.00	21.00	21.00
	ENERGY (MILLION BTU)			
INITIAL ENERGY	IN RCS, ACC,S GEN	894.34	894.34	894.34
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00
	DECAY HEAT	0.00	8.41	8.41
	HEAT FROM SECONDARY	0.00	-17.95	-17.95
	TOTAL ADDED	0.00	-9.54	-9.54
	*** TOTAL AVAILABLE ***	894.34	884.80	884.80
DISTRIBUTION	REACTOR COOLANT	303.11	23.95	23.95
	ACCUMULATOR	26.13	20.42	20.41
	CORE STORED	26.85	11.74	11.74
	PRIMARY METAL	132.98	124.74	124.74
	SECONDARY METAL	121.87	117.45	117.45
	STEAM GENERATOR	283.40	274.67	274.67
	TOTAL CONTENTS	894.34	572.97	572.97
EFFLUENT	BREAK FLOW	0.00	311.84	311.84
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	311.84	311.84
	*** TOTAL ACCOUNTABLE ***	894.34	884.81	884.81

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-62 DOUBLE-ENDED COLD LEG GUILLOTINE ENERGY BALANCE

	TIME (SECONDS)	.00	22.00	22.00	212.09
	ENERGY (MILLION BTU)				
INITIAL ENERGY	IN RCS, ACC,S GEN	894.34	894.34	894.34	894.34
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	9.76
	DECAY HEAT	0.00	6.35	6.35	29.85
	HEAT FROM SECONDARY	0.00	-17.03	-17.03	-17.03
	TOTAL ADDED	0.00	-10.68	-10.68	22.58
*** TOTAL AVAILABLE ***		894.34	883.67	883.67	916.92
DISTRIBUTION	REACTOR COOLANT	303.11	14.54	15.99	43.95
	ACCUMULATOR	26.13	15.52	14.08	-.00
	CORE STORED	26.85	16.34	16.34	5.09
	PRIMARY METAL	132.98	126.38	126.38	110.65
	SECONDARY METAL	121.87	119.53	119.53	112.12
	STEAM GENERATOR	283.40	280.82	280.82	259.05
	TOTAL CONTENTS	894.34	573.14	573.14	530.85
EFFLUENT	BREAK FLOW	0.00	305.90	305.90	377.14
	ECCS SPILL	0.00	4.64	4.64	8.97
	TOTAL EFFLUENT	0.00	310.54	310.54	386.11
*** TOTAL ACCOUNTABLE ***		894.34	883.68	883.68	916.96

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-63 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MINIMUM SAFETY INJECTION) PRINCIPAL PARAMETER TRANSIENTS

TIME SECONDS	FLOODING TEMP ° F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWNCOMER HEIGHT FT	FLOW FRACTION	INJECTION TOTAL (CUBIC FEET PER SECOND)	ACCUMULATOR (CUBIC FEET PER SECOND)	SPILL (CUBIC FEET PER SECOND)	ENTHALPY BTU/LBM
24.2	204.9	0.000	0.000	0.00	0.00	.250	0.0	0.0	0.0	0.00
24.8	202.5	27.110	0.000	.52	1.76	.000	12030.7	12353.9	0.0	88.00
25.0	200.4	35.797	.000	1.05	1.97	.000	12305.6	12223.9	0.0	88.00
26.0	199.1	3.133	.314	1.51	5.32	.339	12103.3	11516.6	0.0	88.00
27.0	198.9	3.013	.455	1.66	8.73	.360	11570.7	10234.0	0.0	88.00
30.3	197.7	6.744	.674	2.15	15.00	.659	7041.8	7503.7	0.0	88.00
31.2	197.1	6.429	.693	2.31	16.00	.658	7712.7	7257.8	0.0	88.00
33.3	196.1	5.899	.725	2.62	16.00	.653	7279.1	5310.9	0.0	88.00
37.3	195.2	5.313	.743	3.11	16.00	.642	6615.5	5130.0	0.0	88.00
41.3	195.1	4.923	.749	3.54	16.00	.630	6073.4	5575.8	0.0	88.00
46.3	195.7	4.547	.753	4.03	16.00	.616	5521.8	5003.7	0.0	88.00
52.3	197.1	4.193	.754	4.57	16.00	.601	4968.7	4437.4	0.0	88.00
56.3	198.3	3.922	.754	4.91	16.00	.591	4651.0	4112.2	0.0	88.00
57.3	196.5	4.251	.756	4.98	15.95	.622	536.7	0.0	0.0	88.00
58.3	199.0	4.594	.756	5.07	15.74	.623	515.7	0.0	0.0	88.00
68.3	201.3	4.069	.754	5.51	14.81	.623	522.6	0.0	0.0	88.00
70.3	205.3	3.543	.752	6.06	13.02	.614	545.2	0.0	0.0	88.00
77.3	210.3	3.120	.748	6.54	13.11	.604	555.2	0.0	0.0	88.00
85.3	215.2	2.740	.745	7.04	12.59	.501	363.2	0.0	0.0	88.00
95.3	223.7	2.331	.741	7.58	12.25	.575	560.0	0.0	0.0	88.00

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TIME SECONDS	FLOODING TEMP ° F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWNCOMER HEIGHT FT	FLOW FRACTION	INJECTION TOTAL ACCUMULATOR SPILL (CUBIC FEET PER SECOND)			ENTHALPY BTU/LBM
105.3	231.1	2.125	.733	8.07	12.17	.558	574.2	0.0	0.0	88.00
115.3	233.2	1.951	.737	8.52	12.27	.545	576.8	0.0	0.0	88.00
127.3	245.5	1.323	.737	9.01	12.54	.533	578.5	0.0	0.0	88.00
141.3	252.7	1.744	.733	9.55	12.99	.525	570.5	0.0	0.0	88.00
153.2	256.2	1.713	.741	10.00	13.42	.522	570.9	0.0	0.0	88.00

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-64 DOUBLE-ENDED PUMP SUCTION GUILLOTINE (MAXIMUM SAFETY INJECTION) PRINCIPAL PARAMETER TRANSIENTS

TIME SECONDS	FLOODING TEMP ° F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWNCOMER HEIGHT FT	FLOW FRACTION	INJECTION TOTAL (CUBIC FEET PER SECOND)	ACCUMULATOR (CUBIC FEET PER SECOND)	SPILL (CUBIC FEET PER SECOND)	ENTHALPY BTU/LBM
24.2	204.9	0.00	0.000	0.00	0.00	.250	0.0	0.0	0.0	0.00
24.8	202.5	27.621	0.000	.51	1.89	.000	13816.3	12353.7	0.0	88.00
25.0	200.4	37.282	.000	1.86	2.21	.000	13522.5	12219.6	0.0	88.00
25.9	199.2	3.314	.311	1.50	5.58	.339	13522.5	11559.9	0.0	88.00
25.9	198.9	3.113	.457	1.66	9.37	.360	12485.1	11022.5	0.0	88.00
29.3	198.0	7.453	.642	2.02	15.98	.672	8865.8	7721.2	0.0	88.00
30.7	196.8	6.853	.696	2.30	16.00	.670	8290.7	7149.1	0.0	88.00
32.3	195.9	6.406	.721	2.56	16.00	.667	7946.7	6780.6	0.0	88.00
36.3	194.3	5.764	.743	3.59	16.00	.657	7267.1	6049.9	0.0	88.00
40.3	194.0	5.349	.750	3.56	16.00	.647	6726.0	5470.8	0.0	88.00
45.3	194.3	4.961	.754	4.09	16.00	.636	6171.2	4879.5	0.0	88.00
50.3	195.2	4.650	.755	4.58	16.00	.625	5707.6	4390.2	0.0	88.00
55.3	196.5	4.385	.755	5.04	16.00	.616	5310.1	3972.2	0.0	88.00
59.4	197.8	2.527	.738	5.38	16.00	.420	1454.8	0.0	0.0	88.00
62.4	198.9	2.511	.739	5.55	16.00	.420	1454.7	0.0	0.0	88.00
71.4	202.8	2.456	.739	6.03	16.00	.422	1454.6	0.0	0.0	88.00
80.4	207.9	2.401	.739	6.51	16.00	.424	1454.6	0.0	0.0	88.00
91.4	213.9	2.334	.740	7.07	16.00	.426	1454.5	0.0	0.0	88.00
101.4	224.9	2.274	.741	7.57	16.00	.429	1454.4	0.0	0.0	88.00
111.4	226.2	2.214	.742	8.06	16.00	.432	1454.3	0.0	0.0	88.00
121.4	232.2	2.154	.742	8.53	16.00	.435	1454.3	0.0	0.0	88.00
133.4	239.2	2.082	.744	9.07	16.00	.439	1454.2	0.0	0.0	88.00
143.4	244.8	2.022	.745	9.51	16.00	.443	1454.1	0.0	0.0	88.00
155.1	250.8	1.954	.746	10.00	16.00	.448	1454.0	0.0	0.0	88.00

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-65 SYSTEM PARAMETERS

Plant model	4 loop, 12 ft core
Core power (Mwt)	3652.8
Core inlet temperature (°F)	566.5
Steam pressure (psi)	1000
Assumed containment backpressure (psia)	74.7

Note: This table presents mass and/or energy release data related to the long-term LOCA containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the long-term LOCA containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-66 FULL DOUBLE-ENDED MSLB AT HOT SHUTDOWN (WITH SAF IN CONTAINMENT SPRAY SYSTEM) MASS AND ENERGY RELEASES (FORWARD FLOW)

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
0.0	3229	1188.8
0.253	3145	1190.2
0.254	3145	1190.2
1.0	2896	1194.1
2.0	2549	1197.7
3.0	2272	1200.2
4.0	2044	1201.9
5.0	1855	1203.1
6.0	1698	1203.9
6.5	1631	1204.2
7.0	1565	1204.4
7.95	1457	1204.7
9.0	1354	1204.8
10.0	1268	1204.8
11.0	1208	1204.7
12.0	1199	1204.7
13.0	1187	1204.7
14.0	1171	1204.7
15.0	1154	1204.6
16.0	1134	1204.6
17.0	1113	1204.5
18.0	1092	1204.4
19.0	1071	1204.3
20.0	1050	1204.2
21.0	1045	1204.1
22.0	1025	1204.0
23.0	1006	1203.9
24.0	988	1203.8

Note: This table presents mass and energy release data related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the MSLB containment response at an analyzed core power level of 3659 MWt.

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
25.0	971	1203.7
26.0	954	1203.6
27.0	939	1203.5
28.0	924	1203.4
29.0	910	1203.2
30.0	897	1203.1
35.0	842	1202.6
40.0	800	1202.2
45.0	769	1201.8
50.0	746	1201.5
60.0	715	1201.0
70.0	698	1200.7
80.0	687	1200.6
90.0	681	1200.5
100.0	677	1200.4
120.0	671	1200.3
140.0	666	1200.2
160.0	661	1200.1
180.0	656	1200.0
200.0	650	1199.9
220.0	644	1199.8
240.0	635	1199.4
260.0	627	1199.3
280.0	615	1199.1
281.0	613	1199.1
281.2	32.46	1199.1
1800.0	32.46	1199.1
1800.1	0	0
∞	0	0

Note: This table presents mass and energy release data related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the MSLB containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-67 FULL DOUBLE-ENDED MSLB AT HOT SHUTDOWN (WITH SAF IN CONTAINMENT SPRAY SYSTEM) MASS AND ENERGY RELEASES (REVERSE FLOW)

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
0.0	9639	1188.8
0.253	9639	1188.8
0.254	4543	1189.4
1.0	4298	1191.3
2.0	4047	1193.5
3.0	3827	1195.3
4.0	3630	1196.8
5.0	3453	1198.1
6.0	3295	1199.2
6.5	3223	1199.4
7.0	2112	1199.4
7.95	0	1199.4
∞	0	0.0

Note: This table presents mass and energy release data related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the MSLB containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-68 FULL DOUBLE-ENDED MSLB AT 102% POWER (WITH SAF OF BROKEN LOOP MSIV) MASS AND ENERGY RELEASES (FORWARD FLOW)

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
0.0	2884	1193.2
0.250	2823	1194.0
0.251	2823	1194.0
1.0	2641	1196.2
2.0	2435	1198.3
3.0	2270	1200.0
4.0	2138	1200.9
5.0	2053	1201.5
6.0	2012	1201.8
6.5	1993	1202.0
7.0	1974	1202.1
8.0	1938	1202.4
9.0	1900	1202.6
10.0	1858	1202.9
11.0	1813	1203.2
12.0	1773	1203.4
13.0	1727	1203.6
14.0	1680	1203.9
15.0	1636	1204.0
16.0	1610	1204.2
17.0	1569	1204.4
18.0	1528	1204.5
18.6	1504	1204.5
19.0	1488	1204.6
20.0	1450	1204.7

Note: This table presents mass and energy release data related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the MSLB containment response at an analyzed core power level of 3659 MWt.

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
21.0	1414	1204.7
22.0	1379	1204.8
23.0	1346	1204.8
24.0	1315	1204.8
25.0	1285	1204.8
26.0	1257	1204.8
27.0	1231	1204.7
28.0	1206	1204.7
29.0	1183	1204.7
30.0	1161	1204.6
35.0	1071	1204.3
40.0	1007	1204.0
45.0	962	1203.7
50.0	931	1203.5
60.0	889	1203.2
70.0	862	1202.9
80.0	840	1202.7
90.0	823	1202.5
100.0	808	1202.3
120.0	787	1202.1
140.0	770	1201.8
160	755	1201.6
173.1	746	1201.5
173.1	32.46	1201.9
1800.0	32.46	1201.9
1800.1	0	0
∞	0	0

Note: This table presents mass and energy release data related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the MSLB containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-69 FULL DOUBLE-ENDED MSLB AT 102% POWER (WITH SAF OF BROKEN LOOP MSIV) MASS AND ENERGY RELEASES (REVERSE FLOW)

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
0.0	8611	1193.2
0.250	8611	1193.4
0.251	4058	1193.5
1.0	3944	1195.1
2.0	3813	1196.5
3.0	3711	1197.6
4.0	3630	1198.3
5.0	3562	1198.8
6.0	3505	1199.1
6.5	3479	1199.3
7.0	3335	1199.4
8.0	3098	1199.5
9.0	2760	1199.8
10.0	2473	1200.0
11.0	2185	1200.2
12.0	1898	1200.5
13.0	1610	1200.8
14.0	1323	1201.2
15.0	1035	1201.6
16.0	748	1202.2
17.0	460	1202.9
18.0	173	1203.8
18.6	0	1204.5
∞	0	1204.5

Note: This table presents mass and energy release data related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the MSLB containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-69A MASS AND ENERGY BALANCE PEAK CALCULATED CONTAINMENT PRESSURE FOR MSLB

ENERGY BALANCE (X 10^6 Btu)

	Initial 0 sec	Peak Pressure 281 sec	End of Blowdown 1800 sec
Containment atmosphere	3.2	200.5	122.2
Containment sump	1.1	20.9	161.1
Total Energy	4.3	221.4	283.3
Initial energy	4.3	4.3	4.3
Energy added by blowdown	0.0	260.3	345.1
Energy added by sprays	0.0	3.6	44.7
Energy removed by heat sinks	0.0	46.8	86.3
Total Energy	4.3	221.4	307.8

MASS BALANCE (X 10^3 lbm)

	Initial 0 sec	Peak Pressure 281 sec	End of Blowdown 1800 sec
Containment atmosphere	203.9	374.5	311.2
Containment sump	12.5	111.5	830.1
Total Mass	216.4	486.0	1141.3
Initial Mass	216.4	216.4	216.4
Mass added by blowdown	0.0	216.9	287.6
Mass added by sprays	0.0	52.3	657.5
Total Mass	216.4	485.6	1161.5

Note: This table presents mass and energy release data related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the MSLB containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-69B MASS AND ENERGY BALANCE PEAK CALCULATED CONTAINMENT TEMPERATURE FOR MSLB

ENERGY BALANCE (X 10^6 Btu)

	Initial 0 sec	Peak Temperature 82 sec	End of Blowdown 1800 sec
Containment atmosphere	3.2	155.0	57.2
Containment sump	1.1	4.7	264.6
Total Energy	4.3	159.7	321.8
Initial energy	4.3	4.3	4.3
Energy added by blowdown	0.0	175.1	324.7
Energy added by sprays	0.0	0.0	91.1
Energy removed by heat sinks	0.0	19.7	63.8
Total Energy	4.3	159.7	356.3

MASS BALANCE (X 10^3 lbm)

	Initial 0 sec	Peak Temperature 82 sec	End of Blowdown 1800 sec
Containment atmosphere	203.9	330.4	252.8
Containment sump	12.5	31.8	1545.2
Total Mass	216.4	362.2	1797.0
Initial Mass	216.4	216.4	216.4
Mass added by blowdown	0.0	145.8	270.2
Mass added by sprays	0.0	0.0	1340.0
Total Mass	216.4	362.2	1826.6

Note: This table presents mass and energy release data related to the MSLB containment response analysis of record. Section 6.2.1.8 contains a discussion of the mass and energy release data associated with this analysis of record, as compared with the mass and energy release data associated with the MSLB containment response at an analyzed core power level of 3659 MWt.

TABLE 6.2-70 DELETED

TABLE 6.2-71 DELETED

TABLE 6.2-72 DELETED

TABLE 6.2-73 DELETED

TABLE 6.2-74 DELETED

TABLE 6.2-75 CONTAINMENT SPRAY SYSTEM PARAMETERS

Spray Additive Tank	
Quantity	1
Type	Vertical cylinder
Volume	10,700 gal
Material	Austenitic stainless steel
Design code	ASME Section III, Class 3
ANSI N18.2 safety class	Class 3
Operating pressure	Atmospheric
Design temperature	100°F
Maximum fluid temperature	98°F
NaOH concentration	20% by weight
Containment Spray Pumps	
Quantity	2
Type	Centrifugal
Horsepower	600 Hp
Design flow	3010 gpm
Operating flow (minimum)	2808 gpm
NPSH required	21 ft
Operating flow (recirculation)	3660 gpm
NPSH required	23.6 ft.
NPSH available	23.76 ft.
Design temperature	300°F
Design pressure	350 psig
Design code	ASME III, Class 2
ANSI N18.2 safety class	Safety Class 2

Refueling Water Storage Tank	
Quantity	1
Capacity	475,000 gal
Material	Austenitic stainless steel
Type	Vertical cylinder
Design code	ASME III, Class 2
ANSI N18.2 safety class	Class 2
Concentration of boron	2400-2600 ppm boron
Design temperature	100°F
Maximum fluid temperature	98°F*
Operating pressure	Atmospheric
Containment Spray Heat Exchanger	
Quantity	2
Type	Shell and tube
Design codes:	
Shell side	ASME III, Class 3
Tube side	ASME III, Class 2
ANSI N18.2 safety class:	
Shell side	3
Tube side	2
Material:	
Shell side	Carbon steel
Tube side	Austenitic stainless steel
Design pressure:	
Shell side	150 psig
Tube side	346 psig
Design temperature:	
Shell side	200°F
Tube side	300°F
Containment Spray Headers and Nozzles	
Nozzle quantity	396
Material	Austenitic Stainless Steel
Design flow rate, per nozzle, gpm	15.2
Pressure drop, per nozzle, psi	40

* Maximum fluid temperature within the mixing chamber during the injection phase is 100°F due to the exothermal reaction between sodium hydroxide and boric acid.

TABLE 6.2-76 HEAT EXCHANGER PERFORMANCE DATA

Containment Spray Heat Exchanger

Type	Vertical U-tube
Number	2
Capacity (each)	96.7×10^6 Btu/hr
Heat Transfer Area (each)	3468 ft^2 (Effective)
Units Overall Heat Transfer Coefficient	431 (Service) 758 (Clean)
Tube (Hot) side flow rate (each)	1.5×10^6 lb/hr
Shell (Cold) side flow rate (each)	2.35×10^6 lb/hr
Tube side inlet temperature	244.8°F
Tube side outlet temperature	180.8°F
Shell side inlet temperature	119.9°F [(a)]
Shell side outlet temperature	161.5°F

[(a)] For the design basis accident, this equipment will experience a 6°F cooling water supply temperature transient (120°F to 126°F to 120°F) over a 1½ hour period, or 3°F for a period of 5 hours (cooling tower operation), which will have an insignificant impact on analysis of piping stresses.

TABLE 6.2-77 CONTAINMENT COOLING - ACTIVE COMPONENT FAILURE ANALYSIS

Component	Failure	Comments and Consequences
Suction or discharge valve for RHR pump	Fails to open	Two parallel paths; only one required. *
Suction or discharge valve for spray pump	Fails to open	Two parallel paths; only one required. *
RHR pump	Fails to start; mechanical or loss of power	Two parallel paths; only one required. *
Spray pump	Fails to start; mechanical or loss of power	Two parallel paths; only one required. *
Containment penetration line	Cracked or clogged	Two parallel paths; only one required. *
Spray nozzles	Clogged	The strainer in sump will prevent large particles from entering spray system. 198 nozzles/train ensure no significant decreases of flow.
Automatic valve between spray additive tank and RWST	Fails to open	Two valves in parallel; each capable of permitting full flow.

* 1 containment spray heat exchanger and 1 residual heat removal heat exchanger in operation provide 100% of minimum required cooling

TABLE 6.2-78 DELETED

TABLE 6.2-79 CODES, STANDARDS AND GUIDES APPLICABLE TO THE CONTAINMENT SPRAY SYSTEM

A. USNRC Regulatory Guides

Title	Reg. Guide No.
Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	1.1
Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for PWRs	1.4
Periodic Testing of Protection System Actuation Functions	1.22
Quality Group Classification and Standards	1.26
Seismic Design Classification	1.29
Control of Stainless Steel Welding	1.31
Protection Against Pipe Whip Inside Containment	1.46
Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	1.47
Design Limits and Loading Combinations for Seismic Category I Fluid System Components	1.48
In-service Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components	1.51
Application of the Single Failure Criterion to Nuclear Power Protection Systems	1.53
Manual Initiation of Protection Actions	1.62
Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants LWR Edition	1.70
Sumps for Emergency Core Cooling and Containment Spray Systems	1.82
Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Station will be as Low as is Reasonably Achievable	8.8

B. 10 CFR Part 100, Reactor Site Criteria

C. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code:

Material Specifications	Section II
Nuclear Power Plant Components	Section III
Nondestructive Examination	Section V
Pressure Vessels Division I	Section VIII
Welding and Brazing Qualifications	Section IX
Rules for In-service Inspection of Nuclear Power Plant	Section XI

D. Institute of Electrical and Electronics Engineers (IEEE) Standards:

Title	Std. No.
Criteria for Protection Systems for Nuclear Power Generating Stations	IEEE-279
Criteria for Class 1E Power Systems for Nuclear Power Generating Stations	IEEE-308
Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations	IEEE-323
Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	IEEE-336
Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems	IEEE-338
Trial-Use Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations	IEEE-344
Trial-Use Guide for General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems	IEEE-352
Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems	IEEE-379

E. American National Standard Institute (ANSI) Standards:

Pipe Threads	B2.1
Steel Pipe Flanges and Flanged Fittings	B16.5
Steel Butt Welding Fittings	B16.9
Steel Weld Fittings	B16.11
Butt Welding Ends	B16.25
Power Piping	B31.1
Nuclear Safety Criteria for Design of Stationary Pressurized Water Reactor Plants	N18.2
Protective Coatings for Light Water Nuclear Reactor Containment Facilities	N101.2

F. Tubular Exchanger Manufacturers Association (TEMA)

Mechanical Standards TEMA Class "R" Heat Exchangers

G. Hydraulic Institute

Standard for Rotary, Reciprocating, and Centrifugal Pumps

H. American Institute of Steel Construction (AISC)

Steel Construction Manual

I. American Society for Testing and Materials (ASTM)

Materials Specification

J. Nuclear Energy Institute (NEI)

	NEI No.
Pressurized Water Reactor Sump Performance Evaluation Methodology	04-07

TABLE 6.2-80 CONTAINMENT SPRAYED VOLUME

		Volume (10^6 ft^3)
a)	Bldg. volume above elevation 25'-0"	
	Cylinder	(+) 1.447
	Dome	(+) 0.718
b)	Annulus below elevation 25'-0"	(+) 0.132
c)	Refueling Canal	(+) 0.045
d)	Pressurizer cubicle above elevation 25'-0"	(-) 0.005
e)	Steam generators above elevation 25'-0"	(-) 0.015
f)	Missile shield shadow	(-) 0.010
g)	Equipment hatch	(-) 0.002
	Total Sprayed Volume	2.310

TABLE 6.2-81 EFFECT OF DROP SIZE ON SPRAY EFFICIENCY

Drop Size (microns)	% Drops of Smaller Size	Terminal Velocity (fps)	Residence Time (sec)	Equil. Time (sec)	Distance Traveled (ft)
500	68.9	6.7	20.0	0.5	< 3.3
1000	96	12.8	10.5	1.7	< 21.7
1500	99.99	17.9	7.5	3.8	< 68.0

TABLE 6.2-82 CONTAINMENT ENCLOSURE DESIGN AND PERFORMANCE DATA

Free Volume, ft^3	
Containment Enclosure Annulus	524,344
Electrical Penetration Areas	84,035
Mechanical Penetration Areas	70,320
RHR & SI Equipment Vaults	102,816
Containment Enclosure Equipment Area	92,568
Charging Pump Areas	12,000
Pressure, Inches H_2O gauge	
Normal Operation	0
Post Accident	-0.25*
Temperature, °F	
Normal Operation	50-104
Post Accident, Maximum	148
Exhaust Fans	
Number	2
Type	Centrifugal
Nominal Flow, scfm	2,000
Filters	
Number of Trains	2
Moisture Separator, per train	1
HEPA, per train	2
Carbon Adsorber, per train	1
Thickness of Containment, in.	42-54
Thickness of Containment Enclosure, in.	15-36
Containment Wall Characteristics	
Coefficient of Linear Expansion, in./in.-°F	6.5×10^{-6}
Modulus of Elasticity, psi	3.12×10^6 (3000 psi concrete)
	3.61×10^6 (4000 psi concrete)
Thermal Conductivity, Btu/hr-ft-°F	0.83
Thermal Capacitance, Btu/ft3-°F	29.0

* A negative pressure of 0.685 iwg has to be established at the 21' -0" elevation of the Containment Enclosure Equipment Area to ensure that the required design negative pressure differential of 0.25 iwg is established at the top of the Containment Enclosure Annulus for the full range of design outside ambient temperatures.

TABLE 6.2-83 CONTAINMENT ISOLATION SYSTEM DESIGN INFORMATION

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-1	II	Main Steam (Loop I)	Secondary Steam	30	Yes	No	9	30"x24"x30"-MS-V86	O	No	68-6	Gate	Elec-Hyd	O	CL	CL	FAI	MSIS, CSW	5(4)	A&B	Yes
				6				6"-MS-PV3001	O	No	68-9	Globe	Pneumatic	CL	-	CL	FC	CSW	70(4)	A	Yes
				4				4"-MS-V393	O	No	65-6	Globe	Pneumatic	CL	CL	O	FO	CSW	20(4)	A	Yes
				6				6"x10"-MS-V6,7,8,9,10	O	No	60-0	Safety	Self	CL	CL	CL	-	-	-	-	No
				4				4"x2-1/2"-MS-V204	O	No	72-0	Globe	Motor	LC	LC	LC	FAI	MSIS, CSW(13)	30(4)	A	Yes
				1				1"-MSD-V44	O	No	24-7	Globe	Motor	O	CL	CL	FAI	MSIS, CSW	25(4)	A	-
X-2	II	Main Steam (Loop II)	Secondary Steam	30	Yes	No	9	30"x24"x30"-MS-V88	O	No	71-9	Gate	Elec-Hyd	O	CL	CL	FAI	MSIS, CSW	5(4)	A&B	Yes
				6				6"-MS-PV3002	O	No	66-9	Globe	Pneumatic	CL	-	CL	FC	CSW	70(4)	B	Yes
				4				4"-MS-V394	O	No	68-8	Globe	Pneumatic	CL	CL	O	*FO	CSW	20(4)	A&B	Yes
				6				6"x10"-MS-V22.23, 24,25,26	O	No	86-0	Safety	Self	CL	CL	CL	-	-	-	-	No

* On Train A control power failure, valve position is FC; On Train B control power failure, valve position is FO; On instrument air failure, valve position is FO.

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
				4				4"x2-1/2"-MS-V205	O	No	112-0	Globe	Motor	LC	LC	LC	FAI	MSIS, CSW(13)	30(4)	A	Yes
				1				1"-MSD-V45	O	No	24-7	Globe	Motor	O	CL	CL	FAI	MSIS, CSW	25(4)	A	-
X-3	II	Main Steam (Loop III)	Secondary Steam	30	Yes	No	9	30"x24"x30"-MS-V90	O	No	77-7	Gate	Elec-Hyd	O	CL	CL	FAI	MSIS, CSW	5(4)	A&B	Yes
				6				6"-MS-PV3003	O	No	74-7	Globe	Pneumatic	CL	-	CL	FC	CSW	70(4)	A	Yes
				6				6"x10"-MS-V36,37,38,39,40	O	No	72-0	Safety	Self	CL	CL	CL	-	-	-	-	No
				4				4"x2-1/2"-MS-V206	O	No	82-0	Globe	Motor	LC	LC	LC	FAI	MSIS, CSW(13)	30(4)	A	Yes
				1				1"-MSD-V46	O	No	18-5	Globe	Motor	CL	CL	CL	FAI	MSIS, CSW	25(4)	A	-
X-4	II	Main Steam (Loop IV)	Secondary Steam	30	Yes	No	9	30"x24"x30"-MS-V92	O	No	83-5	Gate	Elec-Hyd	O	CL	CL	FAI	MSIS, CSW	5(4)	A&B	Yes
				6				6"-MS-PV3004	O	No	78-7	Globe	Pneumatic	CL	-	CL	FC	CSW	70(4)	B	Yes
				6				6"X10"-MS-V50,51,52,53,54	O	No	72-0	Safety	Self	CL	CL	CL	-	-	-	-	No
				4				4"x2-1/2"-MS-V207	O	No	82-5	Globe	Motor	LC	LC	LC	FAI	MSIS, CSW(13)	30(4)	A	Yes
				1				1"-MSD-V47	O	No	18-5	Globe	Motor	CL	CL	CL	FAI	MSIS, CSW	25(4)	A	-

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-5	II	Main Feed Water (Loop I)	Secondary Water	18	Yes	No	9	18"-FW-V30	O	No	13-7	Gate	Pneu/ Elec-Hyd	O	CL	CL	FAI	S,RXT, SG,CSW	10(4)	A&B	Yes
				4				4"-FW-V76	O	No	26-7	Stop/ Check	Manual/ Self	CL	CL	O	-	-	10	-	No
X-6	II	Main Feed Water (Loop II)	Secondary Water	18	Yes	No	9	18"-FW-V39	O	No	14-9	Gate	Pneu/ Elec-Hyd	O	CL	CL	FAI	S,RXT, SG,CSW	10(4)	A&B	Yes
				4				4"-FW-V82	O	No	18-0	Stop/ Check	Manual/ Self	CL	CL	O	-	-	10	-	No
X-7	II	Main Feed Water (Loop III)	Secondary Water	18	Yes	No	9	18"-FW-V48	O	No	28-6	Gate	Pneu/ Elec-Hyd	O	CL	CL	FAI	S,RXT, SG,CSW	10(4)	A&B	Yes
				4				4"-FW-V88	O	No	20-6	Stop/ Check	Manual/ Self	CL	CL	O	-	-	10	-	No
X-8	II	Main Feed Water (Loop IV)	Secondary Water	18	Yes	No	9	18"-FW-V57	O	No	27-7	Gate	Pneu/ Elec-Hyd	O	CL	CL	FAI	S,RXT, SG,CSW	10(4)	A&B	Yes
				4				4"-FW-V94	O	No	20-6	Stop/ Check	Manual/ Self	CL	CL	O	-	-	10	-	No
X-9	E	Residual Heat Removal Pump Suction (Loop I/ Hot Leg)	Reactor Coolant	12	No	No	2	12"-RC-V23	I	Yes	21-7	Gate	Motor	CL	O	CL	FAI	CSW	120(4)	A	Yes
				3				3"x4"-RC-V24	I	Yes	8-4	Relief	Self	-	-	-	-	-	-	-	No

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-10	E	Residual Heat Removal Pump Suction (Loop IV/ Hot Leg)	Reactor Coolant	12	No	No	2	12"-RC-V88	I	Yes	21-0	Gate	Motor	CL	O	CL	FAI	CSW	120(4)	A	Yes
				3				3"x4"-RC-V89	I	Yes	9-0	Relief	Self	-	-	-	-	-	-	-	No
X-11	I,E	Residual Heat Removal	Borated Water	8	Yes	No	4	8"-RH-V14	O	No	0-8	Gate	Motor	O	CL	O	FAI	CSW	15(4)	A	Yes
				6				6"-RH-V31	I	No	106-2	Check	Self	-	-	-	-	-	-	-	No
				6				6"-RH-V15	I	No	109-5	Check	Self	-	-	-	-	-	-	-	No
				3/4				3/4"-RH-V28	I	No	35-0	Globe	Pneumatic	CL	CL	CL	FC	CSW,T	10	B	Yes
X-12	I,E	Residual Heat Removal	Borated Water	8	Yes	No	4	8"-RH-V26	O	No	2-8	Gate	Motor	O	CL	O	FAI	CSW	15(4)	B	Yes
				6				6"-RH-V30	I	No	130-6	Check	Self	-	-	-	-	-	-	-	No
				6				6"-RH-V29	I	No	128-1	Check	Self	-	-	-	-	-	-	-	No
				3/4				3/4"-RH-V27	I	No	34-6	Globe	Pneumatic	CL	CL	CL	FC	CSW,T	10	A	Yes

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-13	I,E	Residual Heat Removal	Borated Water	8	Yes	No	3	8"-RH-V32	O	No	6-6	Gate	Motor	O	CL	CL	FAI	CSW	40(4)	B	Yes
				3/4				3/4"-RH-V49	I	No	34-1	Globe	Pneumatic	CL	CL	CL	FC	CSW,T	10	A	Yes
				8				8"-RH-V70	O	No	6-5	Gate	Motor	CL	CL	CL	FAI	CSW	40(4)	A	Yes
				8				8"-RH-V50	I	No	53-5	Check	Self	-	-	-	-	-	-	-	No
				8				8"-RH-V51	I	No	52-11	Check	Self	-	-	-	-	-	-	-	No
X-14	I,E	Cont. Spray	Borated Water	8	Yes	No	7	8"-CBS-V11	O	Yes	2-8	Gate	Motor	CL	CL	O	FAI	CSW,CSAS	13(4)	A	Yes
				8				8"-CBS-V12	I	Yes	4-4	Check	Self	-	-	-	-	-	-	-	No
X-15	I,E	Cont. Spray	Borated Water	8	Yes	No	7	8"-CBS-V17	O	Yes	2-8	Gate	Motor	CL	CL	O	FAI	CSW,CSAS	13(4)	B	Yes
				8				8"-CBS-V18	I	Yes	4-4	Check	Self	-	-	-	-	-	-	-	No
X-16	I	Cont. On-Line Purge (Exhaust)	Cont. Atmosphere	8	No	Yes	10	8"-COP-V4	O	Yes	2-9	Butterfly	Pneumatic	CL	O	CL	FC	CVIS,CSW	2	A	Yes
				8				8"-COP-V3	I	Yes	1-4	Butterfly	Pneumatic	CL	O	CL	FC	CVIS,CSW	2	B	Yes

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-17	I	Equip. Vent (RCDT)	Vent Gas	2 2	No	Yes	7	2"-VG-FV-1661 2"-VG-FV-1712	O I	Yes Yes	12-5 13-7	Globe Globe	Solenoid Solenoid	O O	O O	CL CL	FC FC	T,CSW T,CSW	2 2	A B	Yes Yes
X-18	I	Cont. On-Line Purge (Supply)	Cont. Atmosphere	8 8	No	Yes	10	8"-COP-V1 8"-COP-V2	O I	Yes Yes	3-0 1-4	Butterfly Butterfly	Pneumatic Pneumatic	CL CL	O O	CL CL	FC FC	CVIS,CSW CVIS,CSW	2 2	A B	Yes Yes
X-19	I	Post Accident Monitoring Sample	Borated Water	1/2 1/2	No	Yes	1	1/2"-SS-FV-2857 1/2"-SS-V273	O I	Yes Yes	6-8 3-5	Globe Check	Solenoid Self	CL -	CL -	CL (11) -	FC -	T,CSW -	2 -	B -	Yes No
X-20	I	Primary Component Cooling Water Loop A (Supply)	Demineralized Water (Corrosion Inhibitor)	12 12 1-1/2	No	Yes	6	12"-CC-V168 12"-CC-V57 1-1/2"x2"-CC-V845	O I I	Yes Yes Yes	5-6 36-5 31-4	Butterfly Butterfly Relief	Pneumatic Pneumatic Self	O O -	O O -	CL CL -	FC** FC** -	P,CSW P,CSW -	10 10 -	B A -	Yes Yes No
X-21	I	Primary Component Cooling Water Loop A (Return)	Demineralized Water (Corrosion Inhibitor)	12 12 1-1/2	No	Yes	6	12"-CC-V122 12"-CC-V121 1-1/2"x2"-CC-V410	O I I	Yes Yes Yes	4-11 28-10 23-4	Butterfly Butterfly Relief	Pneumatic Pneumatic Self	O O -	O O -	CL CL -	FC** FC** -	P,CSW P,CSW -	10 10 -	B A -	Yes Yes No
X-22	I	Primary Component Cooling Water Loop B (Return)	Demineralized Water (Corrosion Inhibitor)	12 12 1-1/2	No	Yes	6	12"-CC-V257 12"-CC-V256 1-1/2"x2"-CC-V474	O I I	Yes Yes Yes	7-9 32-11 29-4	Butterfly Butterfly Relief	Pneumatic Pneumatic Self	O O -	O O -	CL CL -	FC** FC** -	P,CSW P,CSW -	10 10 -	A B -	Yes Yes No
X-23	I	Primary Component Cooling Water Loop B (Supply)	Demineralized Water (Corrosion Inhibitor)	12 12 1-1/2	No	Yes	6	12"-CC-V175 12"-CC-V176 1-1/2"x2"-CC-V840	O I I	Yes Yes Yes	9-4 52-5 44-2	Butterfly Butterfly Relief	Pneumatic Pneumatic Self	O O -	O O -	CL CL -	FC** FC** -	P,CSW P,CSW -	14 14 -	A B -	Yes Yes No
X-24	I,E	Safety Injection (Hi Head)	Borated Water	3 4 4 3/4	Yes	- No No -	3	3"-SI-V140 4"-SI-V138 4"-SI-V139 3/4"-SI-V158	I O O I	No No No No	5-0 7-8 9-2 12-6	Check Gate Gate Globe	Self Motor Motor Pneumatic	CL CL CL CL	CL CL CL CL	O O O CL	- FAI FAI FC	- CSW,S CSW,S CSW,T	- 10(4) 10(4) 10	- A B B	No Yes Yes Yes
X-25	I,E	Safety Injection (Hi Head)	Borated Water	4 2 2 3/4	Yes	No	3	4"-SI-V102 2"-SI-V106 2"-SI-V110 3/4"-SI-V160	O I I I	No No No No	2-0 74-2 51-7 42-3	Gate Check Check Globe	Motor Self Self Pneumatic	CL - - CL	CL - - CL	O - - CL	FAI - - FC	CSW - - T,CSW	12(4) - - 10	A - - B	Yes No No Yes

** On electric power failure, valve position is FAI; On instrument air failure, valve position is FC.

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-26	I,E	Safety Injection (Hi Head)	Borated Water	4	Yes	No	3	4"-SI-V77	O	No	10-10	Gate	Motor	CL	CL	O	FAI	CSW	12(4)	B	Yes
				2				2"-SI-V81	I	No	224-6	Check	Self	-	-	-	-	-	-	-	No
				2				2"-SI-V86	I	No	182-8	Check	Self	-	-	-	-	-	-	-	No
				3/4				3/4"-SI-V134	I	No	62-1	Globe	Pneumatic	CL	CL	CL	FC	CSW,T	10	A	Yes
X-27	I,E	Safety Injection (Hi Head)	Borated Water	4	Yes	No	4	4"-SI-V114	O	No	3-3	Gate	Motor	O	O	O	FAI	CSW	10(4)	A	Yes
				2				2"-SI-V118	I	No	75-0	Check	Self	-	-	-	-	-	-	-	No
				2				2"-SI-V122	I	No	144-4	Check	Self	-	-	-	-	-	-	-	No
				2				2"-SI-V126	I	No	72-7	Check	Self	-	-	-	-	-	-	-	No
				2				2"-SI-V130	I	No	108-2	Check	Self	-	-	-	-	-	-	-	No
				3/4				3/4"-SI-V131	I	No	33-11	Globe	Pneumatic	CL	CL	CL	FC	CSW,T	10	B	Yes
X-28	I,E	Chemical & Volume Control (RCP 1A Seal Water)	Borated Water	2	(10)	No	5	2"-CS-V166	O	No	3-9	Globe	Motor	O	O	O	FAI	CSW	10(4)	A	Yes
				2				2"-CS-V4	I	No	2-0	Check	Self	-	-	-	-	-	-	-	No
X-29	I,E	Chemical & Volume Control (RCP 1B Seal Water)	Borated Water	2	(10)	No	5	2"-CS-V162	O	No	2-10	Globe	Motor	O	O	O	FAI	CSW	10(4)	A	Yes
				2				2"-CS-V20	I	No	2-0	Check	Self	-	-	-	-	-	-	-	No
X-30	I,E	Chemical & Volume Control (RCP 1C Seal Water)	Borated Water	2	(10)	No	5	2"-CS-V158	O	No	3-9	Globe	Motor	O	O	O	FAI	CSW	10(4)	A	Yes
				2				2"-CS-V36	I	No	6-3	Check	Self	-	-	-	-	-	-	-	No
X-31	I,E	Chemical & Volume Control (RCP 1D Seal Water)	Borated Water	2	(10)	No	5	2"-CS-V154	O	No	3-3	Globe	Motor	O	O	O	FAI	CSW	10(4)	A	Yes
				2				2"-CS-V52	I	No	3-6	Check	Self	-	-	-	-	-	-	-	No
X-32	I	Equipment and Floor Drainage (RCDT)	Mildly Contaminated Waste Water	3	No	Yes	7	3"-WLD-V82	O	Yes	1-0	Globe	Pneumatic	O	O	CL	FC	T,CSW	10	A	Yes
				3				3"-WLD-V81	I	Yes	14-6	Globe	Pneumatic	O	O	CL	FC	T,CSW	10	B	Yes
				1-1/2				1-1/2"x2"-WLD-V213	I	Yes	13-4	Relief	Self	-	-	-	-	-	-	-	No
X-33	I,E	Chemical & Volume Control (Charging)	Borated Water	3	No	No	5	3"-CS-V143	O	No	3-5	Gate	Motor	O	O	CL	FAI	S,CSW	10	B	Yes
				3				3"-CS-V144	I	No	4-7	Check	Self	-	-	-	-	-	-	-	No

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-34	I	Equipment and Floor Drainage (RC Sump)	Mildly Contaminated Waste Water	2	No	Yes	6	2"-WLD-FV-8331	I	Yes	17-3	Globe	Solenoid	O	O	CL	FC	T,CSW	2	B	Yes
				2				2"-WLD-FV-8330	O	Yes	3-0	Globe	Solenoid	O	O	CL	FC	T,CSW	2	A	Yes
				3/4				3/4"x1"-WLD-V209	I	Yes	17-1	Relief	Self	-	-	-	-	-	-	-	No
X-35A	I	Safety Injection (Test Line)	Borated Water	3/4	No	Yes	3	3/4"-SI-V70	I	Yes	8-10	Globe	Pneumatic	CL	CL	CL	FC	T,CSW	10	B	Yes
				3/4				3/4"-SI-V62	O	Yes	3-11	Globe	Pneumatic	CL	CL	CL	FC	T,CSW	10	A	Yes
				3/4				3/4"x1"-SI-V247	I	Yes	10-0	Relief	Self	-	-	-	-	-	-	-	No
				3/4				3/4"-SI-V157	O	Yes	59-8	Globe	Pneumatic	CL	CL	CL	FC	T,CSW	10	A	Yes
X-35B	I	Reactor Coolant (Pressurizer Steam/ Liquid Sample)	Steam & Borated Water	1/2	No	Yes	8	1/2"-RC-FV-2830	I	Yes	61-5	Globe	Solenoid	CL	CL	CL	FC	T,CSW	2	B	Yes
				1/2				1/2"-RC-FV-2831	I	Yes	63-4	Globe	Solenoid	CL	CL	CL	FC	T,CSW	2	B	Yes
				1/2				1/2"-RC-FV-2840	O	Yes	4-10	Globe	Solenoid	CL	CL	CL	FC	T,CSW	2	A	Yes
				3/4				3/4"x1"-RC-V312	I	Yes	69-8	Relief	Self	-	-	-	-	-	-	-	No
X-35C	I	Reactor Coolant (RC Sample Loop I)	Borated Water	1/2	No	Yes	8	1/2"-RC-FV-2874	O	Yes	5-1	Globe	Solenoid	CL	CL	CL (11)	FC	T,CSW	2	B	Yes
				1/2				1/2"-RC-FV-2894	O	Yes	5-0	Globe	Solenoid	LC	LC	LC (11)	FC	CSW	2(4)	A	Yes
				1/2				1/2"-RC-FV-2832	I	Yes	84-9	Globe	Solenoid	CL	CL	CL (11)	FC	T,CSW	2	A	Yes
				3/4				3/4"x1"-RC-V314	I	Yes	75-11	Relief	Self	-	-	-	-	-	-	-	No
X-35D	I	Reactor Coolant (RC Sample Loop III)	Borated Water	½	No	Yes	8	1/2"-RC-FV-2876	O	Yes	7-5	Globe	Solenoid	CL	CL	CL (11)	FC	T,CSW	2	A	Yes
				1/2				1/2"-RC-FV-2896	O	Yes	11-10	Globe	Solenoid	LC	LC	LC (11)	FC	CSW	2(4)	B	Yes
				1/2				1/2"-RC-FV-2833	I	Yes	67-6	Globe	Solenoid	CL	CL	CL (11)	FC	T,CSW	2	B	Yes
				3/4				3/4"x1"-RC-V337	I	Yes	75-4	Relief	Self	-	-	-	-	-	-	-	No
X-36A	I	Demineralized Water	DM Water	1	No	Yes	2	1"-DM-V4	O	Yes	2-3	Gate	Manual	LC	-	-	-	-	-	-	No
				1				1"-DM-V5	I	Yes	16-5	Gate	Manual	LC	-	-	-	-	-	-	No
				1-1/2				1-1/2"x2"-DM-V18	I	Yes	15-10	Relief	Self	-	-	-	-	-	-	-	No
X-36B	I	Nitrogen Gas (H1 Pressure)	N_2 Gas	1	No	Yes	4	1"-NG-V13	O	Yes	3-9	Globe	Pneumatic	CL	CL	CL	FC	T,CSW	10	B	Yes
				1				1"-NG-V14	I	Yes	8-11	Globe	Pneumatic	CL	CL	CL	FC	T,CSW	10	A	Yes
X-36C	I	Reactor Makeup Water	DM Water	3	No	Yes	1	3"-RMW-V30	O	Yes	2-4	Globe	Pneumatic	O	CL	CL	FC	T,CSW	10	B	Yes
				3				3"-RMW-V29	I	Yes	8-11	Check	Self	-	-	-	-	-	-	-	No

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-37A	I	Chemical & Volume Control (Letdown)	Borated Water	3 3	No	Yes	5	3"-CS-V150 3"-CS-V149	O I	Yes Yes	3-5 113-0	Globe Gate	Pneumatic Motor	O O	O O	CL CL	FC FAI	T,CSW T,CSW	10 10	B A	Yes Yes
X-37B	I	Chemical & Volume Control (Excess Letdown)	Borated Water	2 2 3/4	No	Yes	5	2"-CS-V168 2"-CS-V167 3/4"x1"-CS-V794	I O I	Yes Yes Yes	7-10 4-10 8-1	Globe Globe Relief	Motor Motor Self	O O -	O O -	CL CL -	FAI FAI -	T,CSW T,CSW -	10 10 -	B A -	Yes Yes No
X-38/ 76A	I	Fire Protection	Dry	4 4	No	Yes	2	4"-FP-V592 4"-FP-V588	O I	Yes Yes	7-11 3-0	Gate Check	Manual Self	LC -	- -	- -	- -	- -	- -	- -	No No
X-38/ 76B	I	Combustible Gas Control	Containment Atmosphere	10 10 2 2	No	Yes	8	10"-CGC-V46 10"-CGC-V45 2"-CGC-V43 2"-CGC-V44	I O O O	Yes Yes Yes Yes	2-7 47-7 45-3 46-9	Check Gate Gate Gate	Self Manual Manual Manual	- LC LC LC	- - - -	- - - -	- - - -	- - - -	- - - -	- - - -	No No No No
X-39	I	Spent Fuel Pool Cooling and Cleanup	Dry	2 2 3/4	No	Yes	2	2"-SF-V86 2"-SF-V87 3/4"x1"-SF-V101	I O I	Yes Yes Yes	3-11 3-8 2-8	Gate Gate Relief	Manual Manual Self	LC LC -	- - -	- - -	- - -	- - -	- - -	- - -	No No No
X-40A	I	Nitrogen Gas (Low Pressure)	N_2 Gas	1 1	No	Yes	1	1"-NG-FV-4609 1"-NG-FV-4610	O I	Yes Yes	6-1 8-3	Globe Globe	Solenoid Solenoid	CL CL	CL CL	CL CL	FC FC	T,CSW T,CSW	2 2	A B	Yes Yes
X-40B	I	PRT Sample	Vent Gas	1/2 1/2	No	Yes	8	1/2"-RC-FV-2836 1/2"-RC-FV-2837	I O	Yes Yes	8-3 5-4	Globe Globe	Solenoid Solenoid	CL CL	CL CL	CL CL	FC FC	T,CSW T,CSW	2 2	B A	Yes Yes
X-41		Spare																			
X-42		Spare																			
X-43A		Spare																			

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-43B		Spare																			
X-43C		Spare																			
X-43D		Spare																			
X-43E		Spare																			
X-44		Spare																			
X-45		Spare																			
X-46		Spare																			
X-47A		Spare																			
X-47B		Spare																			
X-47C		Spare																			
X-47D		Spare																			

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-48A	II	Primary Component Cooling Water Thermal Barrier Loop B (Supply)	Demineralized Water (Corrosion Inhibitor)	6	No	No	6	6"-CC-V1092	O	No	5-5	Butterfly	Motor	O	O	O	FAI	CSW	-	B	Yes
X-48B	II	Primary Component Cooling Water Thermal Barrier Loop B (Return)	Demineralized Water (Corrosion Inhibitor)	6	No	No	6	6"-CC-V1095	O	No	5-1	Butterfly	Motor	O	O	O	FAI	CSW	-	B	Yes
X-49A	II	Primary Component Cooling Water Thermal Barrier Loop A (Supply)	Demineralized Water (Corrosion Inhibitor)	6	No	No	6	6"-CC-V1101	O	No	4-6	Butterfly	Motor	O	O	O	FAI	CSW	-	A	Yes
X-49B	II	Primary Component Cooling Water Thermal Barrier Loop A (Return)	Demineralized Water (Corrosion Inhibitor)	6	No	No	6	6"-CC-V1109	O	No	4-4	Butterfly	Motor	O	O	O	FAI	CSW	-	A	Yes
X-50	III E	Safety Injection PT-936	Demineralized Water		Yes	No	9	N/A		No											
X-51		Spare																			
X-52A	I	Air Sample Supply	Containment Atmosphere	1/2 1/2	No	No	11	1/2"-CAH-FV-6572 1/2"-CAH-FV-6573	O I	Yes Yes	1-0 1-0	Gate Gate	Solenoid Solenoid	O O	CL CL	CL CL	FC FC	T,CSW T,CSW	2 2	A B	Yes Yes
X-52B	I	Air Sample Supply	Containment Atmosphere	1/2 1/2	No	No	11	1/2"-CAH-FV-6574 1/2"-CAH-V12	O I	Yes Yes	1-0 1-0	Gate Check	Solenoid Self	O -	CL -	CL -	FC -	T,CSW -	2 -	A -	Yes No

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-53-56		(Numbers not used).																			
X-57A		Spare																			
X-57B	III E	Safety Injection PT-937	Demineralized Water		Yes	No	9	N/A		No											
X-57C		Spare																			
X-58		Spare																			
X-59		Spare																			
X-60	E	Containment Spray	Borated Water	16	Yes	No	2	16"-CBS-V14	O	No	2-0	Encap-sulated Gate	Motor	CL	CL	O	FAI	Recirc. CSW	34(4)	B	Yes
X-61	E	Containment Spray	Borated Water	16	Yes	No	2	16"-CBS-V8	O	No	2-0	Encap-sulated Gate	Motor	CL	CL	O	FAI	Recirc., CSW	34(4)	A	Yes
X-62***																					
X-63	II	Steam Generator Blowdown	Secondary Water	3	No	No	9	3"-SB-V9	O	No	7-2	Gate	Pneumatic	O	O	CL	FC	T,CSW	10	A&B	Yes
X-64	II	Steam Generator Blowdown	Secondary Water	3	No	No	9	3"-SB-V10	O	No	10-0	Gate	Pneumatic	O	O	CL	FC	T,CSW	10	A&B	Yes

*** Penetration for fuel transfer tube, Type B test applicable.

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-67	I	Service Air	Air	2 2	No	Yes	7	2"-SA-V229 2"-SAV-V1042	O I	Yes Yes	- 10-2	Gate Globe	Manual Manual	LC LC	LC LC	LC LC	- -	- -	- -	- -	No No
X-65	II	Steam Generator Blowdown	Secondary Water	3	No	No	9	3"-SB-V11	O	No	8-0	Gate	Pneumatic	O	O	CL	FC	T,CSW	10	A&B	Yes
X-66	II	Steam Generator Blowdown	Secondary Water	3	No	No	9	3"-SB-V12	O	No	6-4	Gate	Pneumatic	O	O	CL	FC	T,CSW	10	A&B	Yes
X-68		Instr. Air	Air	2	No	Yes	12	2"-IA-V530 2"-IA-V531	O I	Yes Yes	14-11	Globe Check	Pneumatic Self	CL CL	CL CL	CL CL	CL -	T -	10 -	A -	Yes No
X-69		Spare																			
X-70		Spare																			
X-71A/ 74A	E	Combustible Gas Train B H_2 Analyzer Inlet	Containment Atmosphere	1	No	No	8	1"-CGC-V32	O	No	10-7	Globe	Manual	LC	LC	O	-	-	-	-	No
X-71B/ 74B	I	Combustible Gas Train B H_2 Analyzer Return	Containment Atmosphere	1 1	No	No	8	1"-CGC-V24 1"-CGC-V25	O I	No No	10-7 10-2	Globe Check	Manual Self	LC -	LC -	O -	- -	- -	- -	- -	No No
X-71C/ 74C	I	Combustible Gas Containment Exhaust	Containment Atmosphere	2	No	No	8	2"-CGC-V36	O	Yes	27-6	Globe	Manual	LC	LC	O (12)	-	-	-	-	No
X-76/ 73				2				2"-CGC-V28	I	Yes	13-1	Globe	Motor	CL	CL	O (12)	FAI	T,CSW	12	B	Yes
X-71D/ 74D	I	Leak Detection	Containment Detection	1/2 1/2	No	Yes	11	1/2"-LD-V1 1/2"-LD-V2	I O	Yes Yes	2-1 1-6	Globe Globe	Manual Manual	LC LC	LC LC	LC LC	- -	- -	- -	- -	No No

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-71E/ 74E		Spare																			
X-72A/ 75A	E	Combustible Gas Train A H_2 Analyzer Inlet	Containment Atmosphere	1	No	No	8	1"-CGC-V10	O	No	10-7	Globe	Manual	LC	LC	O	-	-	-	-	No
X-72B/ 75B	I	Combustible Gas Train A H_2 Analyzer Return	Containment Atmosphere	1 1	No	No	8	1"-CGC-V3 1"-CGC-V4	O I	No No	10-7 13-3	Globe Check	Manual Self	LC -	LC -	O -	- -	- -	- -	- -	No No
X-72C/ 75C X-76/ 73	I	Combustible Gas Containment Exhaust	Containment Atmosphere	2 2	No	No	8	2"-CGC-V15 2"-CGC-V14	O I	Yes Yes	9-0 18-2	Globe Globe	Manual Motor	LC CL	LC CL	O (12) O (12)	- FAI	- T,CSW	- 12	- A	No Yes
X-72D/ 75D		Spare																			
X-72E/ 75E		Spare																			
X-72F/ 75F		Spare																			
HVAC-1 (9)	E	Containment Air-Purge	Containment Atmosphere	36 36	No		10				0-6 0-6										
HVAC-2 (9)	E	Containment Air-Purge	Containment Atmosphere	36 36	No		10 10				0-6 0-6										

Cnt. Penetration (8)	Appl. Gen. Design Criteria or Reg. Guide (1)	System Name	Fluid Contained	Line Size (in.)	Essential Sys.	Pot. Bypass or Through-Line Leakage	Figure 6.2-91 Sheet	Isolation Valve Number	Valve Loc Inside/ Outside (2)	Type C Tests	Length of Pipe (ft-in) (3)	Valve		Valve Position (5)				Containment Isolation Signal (6)	Valve Close Time (Sec) (7)	Power Source Bus A or B	Position Indication
												Type	Operator	Normal	Shutdown	Post Accident	Power Failure				
X-77A	III E	Reactor Coolant System LT-1321 LT-1322 LIS-1321	Demineralized Water		Yes	No	9	N/A		No											
X-77B	III E	Reactor Coolant System LT-1321 LT-1322 LIS-1322 PT-403	Demineralized Water		Yes	No	9	N/A		No											
X-77C	III E	Safety Injection PT-934	Demineralized Water		Yes	No	9	N/A		No											
X-77D	III E	Safety Injection PT-2576	Demineralized Water		Yes	No	9	N/A		No											
X-78A	III E	Reactor Coolant System LT-1311 LT-1312 LIS-1311	Demineralized Water		Yes	No	9	N/A		No											
X-78B	III E	Reactor Coolant System LT-1311 LT-1312 LIS-1312 PT-405	Demineralized Water		Yes	No	9	N/A		No											
X-78C	III E	Safety Injection PT-935	Demineralized Water		Yes	No	9	N/A		No											
X-78D	III E	Safety Injection PT-2577	Demineralized Water		Yes	No	9	N/A		No											

NOTES FOR TABLE 6.2-83

1. I. - General Design Criteria 55 and 56

   II - General Design Criterion 57

   III - Sealed fluid instrument lines (Section 6.2.4.1.d.3)

   E - Exception, as discussed in Subsection 6.2.4.2.m

2. I - Location inside the containment

   O - Location outside the containment

3. Length of pipe from containment to outermost isolation valve

4. The listed valve close time is not essential for containment isolation.

   This valve does not receive a containment isolation signal.

5. O - Open FAI - Fail as is LO - Locked open

   CL - Closed FC - Fail closed LC - Locked closed

6. Containment Isolation Signals

   T - Phase A containment isolation signal

   P - Phase B containment isolation signal

   CVIS - Containment ventilation isolation signal

   Other

   MSIS - Main steam line isolation signal

   S - Safety injection signal

   RXT - Reactor trip signal coincident with a low reactor coolant Tavg signal

   SG - Steam generator hi-hi signal

   CSW - Control Switch

   CSAS - Containment Spray actuation signal

   RECIRC - Refueling water storage tank lo-lo level signal coincident with an S signal.

7. The closing times given are those that are specified as maximum

8. Alpha designations on penetration numbers are for clarification only and are not reflected in other documents

9. Testable blind flanges are installed on the outboard side of penetrations HVAC-1 and HVAC-2 during modes 1, 2, 3, and 4. These blind flanges form part of the containment boundary and are subject to Type B testing.

10. The seal injection function is non-essential, but this seal injection path does provide an alternate passive path for boration and RCS inventory control, which is credited for safe shutdown.

11. These sampling valves may be open intermittently to obtain a post accident sample (not a safety-related function).

12. These CGC valves would only be opened post accident for a containment building purge, a backup function. This would only be required if both safety-related hydrogen recombiners failed or if the post LOCA hydrogen generation rate was significantly greater than the design basis generation rate.

13. The MSIV bypass valves are normally locked closed with the breakers locked open. During startup and surveillance testing, the valves may be open (breakers closed), which allows MSIS and CSW operation.

TABLE 6.2-84 HYDROGEN GAS ANALYZER DESIGN PARAMETERS

A.	General	
	Safety Classification	1E
	Sensor Type	Thermal catalytic
	Scale, %	0 to 10
	Indication	Local (4% H_2 concentration to be clearly indicated in red)
	Accuracy, %	2½ full scale
	Controls	Off-Standby-Analyze
	Response Time, seconds	60, 90% step change
	Min. Sample Flow, scfm	0.08 @ 60 psig containment pressure
	Max. Sample Transit Time, minutes *	10.0
B.	Normal Sampling	
	Temperature, °F	50 to 120
	Pressure, psig	-3.5 to 1.5
	Gamma Radiation, Rad/hr	50
	Relative Humidity, %	5 to 95
C.	LOCA and Post-LOCA Sampling	
	Temperature, °F	380 maximum
	Pressure, psig	-5 to 60
	Gamma Radiation, Rads – water	1.5×10^8
	Rads - air	6.3×10^7
	Water	1900 ppm boron, pH of 10.5
	Relative Humidity, %	100
D.	Normal Operating Environment	
	Temperature, °F	50 to 150
	Pressure, psig	0 to 3
	Humidity, %	10 to 95
	Radiation Dose, Rads in40 year period	5×10^6

* Time from sample entering the sample line in the containment until it reaches the hydrogen analyzer.

TABLE 6.2-85 APPLICABLE CODES, STANDARDS, AND REGULATORY GUIDES USED IN THE DESIGN OF THE ELECTRIC HYDROGEN RECOMBINER

1. NRC Regulatory Guides

   1.7 (March 10, 1971)

   1.28 (Safety Guide 28, 6/7/72)

   1.29 (Rev. 2, 2/76)

   1.38 (Rev. 1, 10/76)

2. 10 CFR 50, Appendix A, GDC 2, 41, 42, 43

3. Industry Codes

   ASME IX (Welding and Brazing Requirements)

   National Electric Code

   National Electric Manufacturing Association

   National Fire Protection Association

4. Underwriters' Laboratories, Inc.

5. Institute of Electrical and Electronics Engineers

   IEEE 308-1971

   IEEE 323-1974

   IEEE 334-1974

   IEEE 383-1974

   IEEE 344-1975

TABLE 6.2-86 ELECTRIC HYDROGEN RECOMBINER TYPICAL PARAMETERS

Power (maximum), kW *	75
Capacity (minimum), scfm	100
Heaters	
Number	4
Heater surface area/heater, ft^2	35
Maximum heat flux, Btu/hr-ft^2	2850
Maximum sheath temperature, °F	1550
Gas Temperature	
Inlet, °F	80 to 155
In heater section, °F	1150 to 1400
Materials	
Outer structure	300-Series SS
Inner structure	Incoloy-800
Heater element sheath	Incoloy-800
Dimensions, ft	
Height	8
Width	3.9
Depth	4.6
Weight, lb	4500

* Power can be controlled by SCR.

TABLE 6.2-87 CONTAINMENT BUILDING ALUMINUM INVENTORY

Item	Exposed Surface (ft^2) (Note 1 & 2)
Jib Crane	3
Nuclear Instrumentation	83
Flux Map Drive System	88
Rod Position Indicator Connectors	81
CRDM Connectors	69
Miscellaneous Valves	86
Containment Elevator Fans	127.9
Excore Detector Supports	28.6
CRD Fans, Blades, and Hubs	19.1
Polar Gantry Crane	40.8
I&C Transmitters & Misc.	34.9
Electrical Fixtures	27.6
Test Pump	3
Refueling Machine and Transfer System	26.4
Signs (HP, etc.)	8
Gaitronics System	140
Work Control Allocation (Note 2)	500
Contingency (Note 2)	433.7
Total	1800 (Note 1 & 2)

Note 1: The 1800 ft^2 total value is the design basis value for hydrogen generation. Allocation among the specified categories may vary due to design changes in progress. Refer to Calculation 4.3.16.13F for actual allocation.

Note 2: The aluminum contingency values in Table 6.2-87 are not the most restrictive values following OR12. Calculation C-S-1-83814, Seabrook Post Accident Chemical Product Formation, has more limiting values. Calculation C-S-1-83814 removed the Work Control Contingency. The limiting values are based on the effect of aluminum corrosion product on the containment sump strainers. Allocation of the design contingency may vary due to design changes in progress.

TABLE 6.2-88 CONTAINMENT BUILDING ZINC INVENTORY

Item	Exposed Surface (ft^2)
Ductwork, Angles and Supports	48,357
Decking	7,850
Grating	24,395
Cable Trays	19,295
Tray Supports	86,190
Conduit	23,387
Conduit Supports	98,256
Instrument Tube Trays and Supports	21,040
Electrical Box Supports	8,416
Wire Mesh Doors	1,182
Scaffolding Components	3,400
Wire Mesh Tool Crib	648
Misc. Sheet Metal and Structural Members	700
Refueling Machine and Fuel Transfer System	52
Debris Interceptor Hardware	25
Work Control Allocation	6,000
Contingency	5,598
Total	355,000

TABLE 6.2-89 PARAMETERS USED TO EVALUATE CGCS PERFORMANCE

Reactor Power Level, MW_t	3650.6
Initial Volume of Containment Atmosphere, scf	2.55×10^6
Mass of Zircaloy Cladding, lb	43,336
Hydrogen in Primary Coolant, scf	1127
Hydrogen Production Rates (scfh) from Corrosion:	
Per the methodology of Reference 27, Attachment II.	

TABLE 6.2-90 SYSTEMS OPEN TO CONTAINMENT ATMOSPHERE AND/OR VENTED PRIOR TO AND DURING TYPE "A" TESTING

1. Reactor Coolant
2. Containment Online Purge
3. Equipment Vent - Hydrogenated
4. Equipment and Floor Drainage
5. Combustible Gas Control (including H_2 Analyzers)
6. Fire Protection
7. Containment Air Purge
8. Sample (Including Post-Accident)
9. SI Accumulators (Including Test/Fill Line)
10. Demineralized Water
11. Nitrogen Gas
12. Reactor Makeup Water
13. CS Purification/Letdown
14. Letdown Return
15. Reactor Cavity Cleanup
16. Containment Air Handling
17. Service Air
18. Instrument Air

TABLE 6.2-91 SYSTEMS NOT VENTED DURING TYPE "A" TESTING

System	Justification
High Pressure Safety Injection	System is normally filled with water and operating under post-accident conditions
Low Pressure Safety Injection	System is normally filled with water and operating under post-accident conditions
Containment Spray	System is normally filled with water and operating under post-accident conditions. Portion of the system inboard of the isolation valves is automatically vented inside containment through the spray nozzles.
Chemical and Volume	System maintains plant in safe Control (including RCP Seal Water) condition during the test - normally filled with water
Residual Heat Removal	System maintains plant in safe condition during the test - normally filled with water
Cooling Water to Containment Fan Coolers	System is normally filled with water under post-accident conditions. Fan coolers are used to maintain temperature during the test.
Steam Generator Blowdown	Closed system inside containment
Main Steam	Closed system inside containment
Feedwater	Closed system inside containment
PCCW to Thermal Barrier Cooling	Closed system inside containment

TABLE 6.2-92 CONTAINMENT LINER PENETRATIONS*

Piping Penetration No.	Essential (E) or Nonessential (NE)	System	Selection Basis
X-1	E	Main Steam from SG E 11A	Decay Heat Removal
X-2	E	Main Steam from SG E 11B	Decay Heat Removal
X-3	E	Main Steam from SG E 11C	Decay Heat Removal
X-4	E	Main Steam from SG E 11D	Decay Heat Removal
X-5	E	Feedwater to SG E 11A	Decay Heat Removal
X-6	E	Feedwater to SG E 11B	Decay Heat Removal
X-7	E	Feedwater to SG E 11C	Decay Heat Removal
X-8	E	Feedwater to SG E 11D	Decay Heat Removal
X-9	NE	RHR Pump Suction from HL #1	
X-10	NE	RHR Pump Suction from HL #4	
X-11	E	RHR to Safety Injection	Low Pressure Injection
X-12	E	RHR to Safety Injection	Low Pressure Injection
X-13	E	RHR to Safety Injection	Hot Leg Injection
X-14	E	Containment Bldg. Spray	Containment Spray
X-15	E	Containment Bldg. Spray	Containment Spray
X-16	NE	Containment Online Purge	
X-17	NE	Hydrogenated Vent Hdr.	
X-18	NE	Containment Online Purge	
X-19	NE	Post-Accident Sample	Req'd for Post-Accident Sampling
X-20	NE*	Primary Component Cooling Water, Loop A	Desirable for Some Accidents; Isolate on Hi-2 Containment Pressure
X-21	NE*	Primary Component Cooling Water, Loop A	Desirable for Some Accidents; Isolate on Hi-2 Containment Pressure
X-22	NE*	Primary Component Cooling Water, Loop B	Desirable for Some Accidents; Isolate on Hi-2 Containment Pressure
X-23	NE*	Primary Component Cooling Water, Loop B	Desirable for Some Accidents; Isolate on Hi-2 Containment Pressure

* Although these systems are nonessential, they are valuable in accident monitoring and control.

Piping Penetration No.	Essential (E) or Nonessential (NE)	System	Selection Basis
X-24	E	SI from Charging Pumps	High Pressure SI
X-25	E	SI, High Head	Medium Pressure SI
X-26	E	SI, High Head	Medium Pressure SI
X-27	E	SI, High Head	Medium Pressure SI
X-28	NE	Seal Water to RC Pump 1A	
X-29	NE	Seal Water to RC Pump 1B	
X-30	NE	Seal Water to RC Pump 1C	
X-31	NE	Seal Water to RC Pump 1D	
X-32	NE	RC DRN TK to Primary DRN TK	
X-33	NE	Chemical and Volume Control	
X-34	NE	Floor Equipment Drain	
X-35	NE*	RCS Sampling	Required for Post-Accident Sampling Manual Bypass Operation
X-35	NE	SI Test	
X-36	NE	Demineralized Water	
X-36	NE	Nitrogen Gas	
X-36	NE	Reactor Makeup Water	
X-37	NE	Letdown HX	
X-37	NE	RCP Seal Water Return	
X-38	NE*	Combustion Gas Control	May be Required for Purging Following Some Accident-Manual Operation
X-38	NE	Fire Protection	
X-39	NE	Refueling Cavity Purification	
X-40	NE	Nitrogen to PRT	
X-40	NE	PRT Gas Sample	
X-41	NE	Spare	
X-42	NE	Spare	
X-43	E	Press. Protection Containment	Containment Monitoring
X-44	NE	Spare	
X-45	NE	Spare	

Piping Penetration No.	Essential (E) or Nonessential (NE)	System	Selection Basis
X-46	NE	Spare	
X-47	E	Press. Protection Monitoring	Containment
X-48	NE	PCCW-Thermal Barrier Cooling	
X-49	NE	PCCW-Thermal Barrier Cooling	
X-50	E	Press. Protection	Containment Monitoring
X-51	NE	Spare	
X-52	E	Containment Air Sample	Containment Monitoring
X-53-56		(Numbers Not Used)	
X-57	E	Press. Protection	Containment Monitoring
X-58	NE	Spare	
X-59	NE	Spare	
X-60	E	Containment Recirc. Sump CBSTK-10B	Containment Spray and SI Recirculation
X-61	E	Containment Recirc. Sump CBSTK-10A	Containment Spray and SI Recirculation
X-62	NE	Fuel Transfer Tube	
X-63	NE	Steam Generator Blowdown E11A	
X-64	NE	Steam Generator Blowdown E11B	
X-65	NE	Steam Generator Blowdown E11C	
X-66	NE	Steam Generator Blowdown E11D	
X-67	NE	Service Air	
X-68	NE	Instrument Air	
X-69	NE	Spare	
X-70	NE	Spare	
X-71	NE*	Combustible Gas Control	Required for H_2 Post-Accident Sampling
X-71	NE	Spare	
X-72	NE*	Combustible Gas Control	Required for H_2 Post-Accident Sampling
X-72	NE	Spare	
X-77	E	Reactor Vessel Level Indication System	Required for Post-Accident Monitoring
X-78	E	Reactor Vessel Level Indication System	Required for Post-Accident

Piping Penetration No.	Essential (E) or Nonessential (NE)	System	Selection Basis
			Monitoring
HVAC-1	NE	Containment Air Purge Supply	
HVAC-2	NE	Containment Air Purge	
E-1	NA	Electrical	NA
E-2	NA	Electrical	NA
E-3	NA	Electrical	NA
E-4	NA	Electrical	NA
E-5	NA	Electrical	NA
E-6	NA	Electrical	NA
E-7	NA	Electrical	NA
E-8	NA	Electrical	NA
E-9	NA	Spare Header	NA
E-10	NA	Spare Header	NA
E-11	NA	Electrical	NA
E-12	NA	Spare Header	NA
E-13	NA	Spare Header	NA
E-14	NA	Electrical	NA
E-15	NA	Spare Header	NA
E-16	NA	Electrical	NA
E-17	NA	Electrical (Conax)	NA
E-18	NA	Electrical	NA
E-19	NA	Electrical	NA
E-20	NA	Electrical	NA
E-21	NA	Electrical	NA
E-22	NA	Electrical	NA
E-23	NA	Electrical	NA
E-24	NA	Electrical	NA
E-25	NA	Electrical	NA
E-26	NA	Electrical	NA
E-27	NA	Electrical	NA
E-28	NA	Electrical	NA

Piping Penetration No.	Essential (E) or Nonessential (NE)	System	Selection Basis
E-29	NA	Electrical	NA
E-30	NA	Electrical	NA
E-31	NA	Electrical	NA
E-32	NA	Electrical	NA
E-33	NA	Spare Header	NA
E-34	NA	Electrical	NA
E-35	NA	Electrical	NA
E-36	NA	Spare Header	NA
E-37	NA	Spare Header	NA
E-38	NA	Electrical	NA
E-39	NA	Electrical	NA
E-40	NA	Electrical	NA
E-41	NA	Electrical	NA
E-42	NA	Electrical	NA
E-43	NA	Electrical	NA
E-44	NA	Spare Header	NA
E-45	NA	Electrical	NA
E-46	NA	Spare Header	NA
E-47	NA	Electrical	NA
E-48	NA	Electrical	NA
E-49	NA	Electrical	NA
E-50	NA	Electrical	NA
E-51	NA	Electrical (Conax)	NA
E-52	NA	Electrical	NA
E-53	NA	Electrical	NA
E-54	NA	Electrical	NA
E-55	NA	Electrical	NA
E-56	NA	Electrical	NA
E-57	NA	Instrument X-77	NA
E-58	NA	Mechanical Spare ***	NA

*** Flange welded to nozzle inside and outside containment. Blank test flange bolted to flange inside containment.

Piping Penetration No.	Essential (E) or Nonessential (NE)	System	Selection Basis
E-59	NA	Mechanical Spare ***	NA
E-60	NA	Unused **	NA
E-61	NA	Unused **	NA
E-62	NA	Unused **	NA
E-63	NA	Unused **	NA
E-64	NA	Instrument X-78	NA

** Plate welded to nozzle inside containment.

NOTE 1 Viewed from outside containment looking south.

NOTE 2 Electrical penetrations are Westinghouse except for E-17 and E-51 which are Conax.

TABLE 6.3-1 EMERGENCY CORE COOLING SYSTEM COMPONENT DESIGN PARAMETERS

Accumulators	
Number	4
Design pressure (psig)	700
Design temperature (°F)	300
Operating temperature (°F)	100 to 150
Normal operating pressure (psig)	650
Minimum operating pressure (psig)	585
Total volume (ft^3)	1350 each
Nominal operating water volume (ft^3)	850 each
Volume N_2 gas (ft^3)	500
Boric acid concentration (ppm)	2600-2900
Relief valve setpoint (psig)	700
Centrifugal Charging Pumps	
Number	2
Design pressure (psig)	2800
Design temperature (°F)	300
Design flow [a] (gpm)	150
Design head (ft)	5800
Maximum flow (gpm)	550
Head at maximum flow (ft)	1400
Discharge head at shutoff (ft)	6200
Motor rating (hp)	600
Required NPSH at maximum flow (ft)	28
Available NPSH (ft)	40
Safety Injection Pumps	
Number	2
Design pressure (psig)	1750
Design temperature (°F)	300
Design flow (gpm)	425

[a] Includes miniflow

Safety Injection Pumps	
Design head (ft)	2700
Maximum flow (gpm)	669
Minimum head at maximum flow (ft)	1700
Minimum head at shutoff (ft)	3545
Motor rating [b] (hp)	450
Required NPSH at maximum flow (ft)	18
Available NPSH (ft)	43.4
Residual Heat Removal Pumps	
Number	2
Design pressure (psig)	600
Design temperature (°F)	400
Design flow (gpm)	3000
Design head (ft)	375
Maximum flow (gpm)	5150 [c]
Minimum head at maximum flow (ft)	275
Minimum head at shutoff (ft)	460
NPSH required at 4388 gpm (ft) [d]	18
NPSH available at 4388 gpm (ft) [d]	25.61
Power (hp)	400
Residual Heat Exchangers	
(See Subsection 5.4.7 for design parameters)	
Motor-Operated Valves	Maximum Opening or Closing Time
Up to and including 8 inches, time (sec) Over 8 inches [e]	15 [f][g]

[b] 1.15 service factor not included.
[c] 5150 GPM corresponds to maximum pump flow on manufacturer's certified pump curve. Calculated maximum system flow is 4500 GPM during injection mode and 4388 GPM during recirculation.
[d] These conditions reflect the most limiting suction conditions for the residual heat removal pumps during post-LOCA recirculation from the containment sump.
[e] Closing time varies dependent upon size, type valve and type of actuator.
[f] Does not include valves RH-V32 and RH-V70.
[g] Active valves that do not receive an automatic signal for operation and that do not have a required stroke time in any analysis may have a longer maximum opening or closing time.

TABLE 6.3-2 EMERGENCY CORE COOLING SYSTEM RELIEF VALVE DATA

Description	Fluid Discharge	Fluid Inlet Temperature Normal (°F)	Set Pressure (psig)	Backpressure Constant (psig)	Maximum Total Backpressure (psig)	Capacity
Outside Containment N_2 supply to accumulators	N_2 gas	120	700	0	0	1500 scfm
Safety injection pump discharge	Water	120	1750	0 to 15	50	20 gpm
Residual heat removal pump safety injection line	Water	120	600	0 to 15	50	20 gpm
Safety injection pumps suction header	Water	100	220	0 to 15	50	25 gpm
Accumulator to containment	N_2 gas	120	700	0	0	1500 scfm
Inside Containment N_2 supply to accumulators (NNS-B31.1 Portion)	N_2 gas	120	800	0	50	1184 scfm

TABLE 6.3-3 MOTOR-OPERATED ISOLATION VALVES IN THE EMERGENCY CORE COOLING SYSTEM

Location	Valve Identification	Interlocks	Automatic Features	Position Indication	Alarms
Accumulator isolation valves (See Dwg. NHY-503907)	SI-V3, -V17, -V32, -V47	"S" signal, RCS pressure > SI unblock pressure	Opens on "S" signal closed, MCC power available, and RCS	MCB	Yes-out of position
Safety injection pump suction from RWST	CBS-V47, -V51 CBS-V49, -V53	None	None	MCB	Yes-out of position
RHR suction from RWST (See Dwg. NHY-503255)	CBS-V2, -V5	Cannot be opened manually by control switch unless sump valve closed and RHR recir. valve closed	Opens on "S" signal	MCB	Yes-out of position
RHR discharge to safety injection/charging	RH-V35, -V36	Cannot be opened unless None safety injection pump miniflow isolated and RHR suction valve from RCS closed	None	MCB	Yes-out of position*
Safety injection hot leg injection (See Dwg. NHY-503909)	SI-V77, -V102	None	None	MCB	Yes-out of position*
RHR hot leg injection (See Dwg. NHY-503769)	RH-V32, -V70	None	None	MCB	Yes-out of position*
Containment sump isolation valve (See Dwg. NHY-503252)	CBS-V8, -V14	Cannot be opened in normal operation unless RHR suction valves from RCS closed	Opens on RWST low-low-1 with "S" signal	MCB	Yes-out of position*
CVCS (See Dwg. NHY-503335) suction from RWST	LCV-112 D&E	"S" signal and CVC tank low level	Opens on "S" signal and CVS tank low level	MCB	Yes-out of position

- Out-of-position alarm is provided as part of the Bypass and Inoperable Status Alarm System

MCB - Main Control Board

Location	Valve Identification	Interlocks	Automatic Features	Position Indication	Alarms
Safety injection pump to cold leg (See Dwg. NHY-503909)	SI-V114	None	None	MCB	Yes-out of position
CVCS normal discharge (See Dwg. NHY-50337)	CS-V142, -V143	"S" signal	Closes on "S" signal	MCB	Yes-out of position
Cold leg isolation	SI-V138, -V139	"S" signal	Opens on "S" signal	MCB	Yes-out of position
Charging pump/safety injection pump crossover (See Dwg. NHY-503338)	CS-V460, -V461 CS-V475	None	None	MCB	Yes-out of position
RHR to RCS cold legs (See Dwg. NHY-503769)	RH-V14, -V26	None	None	MCB	Yes-out of position*
Safety injection pump miniflow (See Dwg. NHY-503911 and NHY-503901)	SI-V89, -V90, -V93	Cannot be opened unless None RHR discharge to safety injection and to charging pumps closed		MCB	Yes-out of position*
RHR cross connect (See Dwg. NHY-503765)	RH-V21, -V22	None	None	MCB	Yes-out of position*
Safety injection pump cross connect (See Dwg. NHY-503912)	SI-V111, -V112	None	None	MCB	Yes-out of position*
Charging pump miniflow (See Dwg. NHY-503398 & 503380)	CS-V196, -V197	"S" signal in conjunction with charging pump Hi flow	Closes on "S" signal in conjunction with charging pump Hi flow	MCB	Yes-out of position

* Out-of-position alarm is provided as part of the Bypass and Inoperable Status Alarm System

MCB - Main Control Board

TABLE 6.3-4 MATERIALS EMPLOYED FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Material
Accumulators	Carbon steel clad with austenitic stainless steel
Pumps	
Centrifugal charging	Austenitic stainless steel
Safety injection	Austenitic stainless steel
Residual heat removal	Austenitic stainless steel
Residual heat exchangers	
Shell	Carbon steel
Shell end cap	Carbon steel
Tubes	Austenitic stainless steel
Channel	Austenitic stainless steel
Channel cover	Austenitic stainless steel
Tube sheet	Austenitic stainless steel
Valves	
Motor-operated valves containing radioactive fluids	
Pressure containing parts	Austenitic stainless steel or equivalent
Body-to-bonnet bolting and nuts	Low alloy steel
Seating surfaces	
Stems	Stellite No. 6 or equivalent Austenitic stainless steel or 17-4 pH stainless
Motor-operated valves containing nonradioactive, boron-free fluids	
Body, bonnet and flange	Carbon steel
Stems	Corrosion resistant steel
Diaphragm valves	Austenitic stainless steel
Accumulator check valves	
Parts contacting borated water	Austenitic stainless steel
Clapper arm shaft	17-4 pH stainless

Relief valves

Stainless steel bodies	Austenitic stainless steel
Carbon steel bodies	Carbon steel
All nozzles, discs, spindles and guides	Austenitic stainless steel
Bonnets for stainless steel valves without balancing bellows	Stainless steel or plated carbon steel
All other bonnets	Carbon steel

Piping

All piping in contact with borated water	Austenitic stainless steel

TABLE 6.3-5 FAILURE MODE AND EFFECTS ANALYSIS – EMERGENCY CORE COOLING SYSTEM – ACTIVE COMPONENTS

Component	Failure Mode	ECCS Operation Phase	*Effect on System Operation	**Failure Detection Method	Remarks
1. Motor-operated gate valve LCV-112 B (LCV-112C analogous)	Fails to close on demand.	Injection - cold legs of RC loops.	Failure reduces redundancy of providing VCT discharge isolation. No effect on safety for system operation; isolation valves LCV-112C and CS-V192 provide backup tank discharge isolation.	Valve position indication(open to closed position change) at MCB. Valve closed position monitor light for group monitoring of components at MCB.	Valve is electrically interlocked with isolation valve LCV-112D. Valve closes on actuation by a SI "S" signal provided isolation valve LCV-112D is at a full open position.
2. Motor-operated gate valve LCV-112 D (LCV-112E analogous)	Fails to open on demand.	Injection - cold legs of RC loops.	Failure reducers redundancy of providing fluid flow from RWST to suction of HHSI/CH pumps. No safety effect on system operation. Alternate isolation valve (LCV-112E) opens to provide backup flow path to suction of HHSI/CH pumps.	Same methods of detection as that stated for item #1 except open position monitor light for group monitoring of components at MCB and closed to open position change indication at MCB.	Valve is electrically interlocked with the instrumentation that monitors fluid level of the VCT. Valve opens upon actuation by a SI "S" signal or upon actuation by a low-low-level VCT signal.
3. Centrifugal charging pump CS-P-2A(CS-P-2B analogous)	Fails to deliver working fluid.	Injection and recirculation cold legs of RC loops.	Failure reduces redundancy of providing emergency coolant to the RCS at prevailing incident RCS pressure. Fluid flow from HHSI/CH pump CS-P-2A will be lost. Minimum flow requirements at prevailing high RCS pressures will be met by HHSI/CH pump CS-P-2B delivery.	HHSI/CH pump discharge header flow (FI-917) at MCB. Trip/closed pump switchgear circuit breaker indication on MCB. Circuit breaker close position monitor light for group monitoring of components at MCB. Breaker trip alarm at MCB.	One HHSI/CH pump is used for normal charging of RCS during plant operation. Circuit breaker aligned to close on actuation by a SI "S" signal.
4. Motor-operated globe valve CS-V196(CS-V197 analogous)	Fails to close on demand.	Injection-cold legs of RC loops.	Failure reduces redundancy of providing isolation of HHSI/CH pump miniflow line. No effect on safety for system operation.	Valve position indication(open to closed position change) at MCB. Valve closed position monitor light and alarm for group monitoring of components at MCB.	Valve aligned close upon actuation to by a SI "S" signal.
5. Motor-operated gate valve CS-V143(CS-V14 2 analogous)	Fails to close on demand.	Injection-cold legs of RC loops.	Failure reduces redundancy of providing isolation of HHSI/CH pump discharge to normal charging line of CVCS. No effect on safety for system operation. Alternate isolation valve(CS-V142) provides backup normal CVCS charging line isolation.	Same method of detection as that stated for item #1.	Valve aligned to close upon actuation by a SI "S" signal.

* See list at end of table for definition of acronyms and abbreviations used.

** As part of plant operation, periodic tests, surveillance inspections and instrument calibrations are made to monitor equipment and performance. Failures may be detected during such monitoring of equipment in addition to detection.

Component	Failure Mode	ECCS Operation Phase	*Effect on System Operation	**Failure Detection Method	Remarks
6. Motor-operated gate valve SI-V138 (SI-V139 analogous)	Fails to open on demand.	Injection-cold legs of RC loops.	Failure reduces redundancy of fluid flow paths from HHSI/CH pumps to the RCS. No effect on safety for system operation. Alternate (SI-V139) open to provide backup flow path from HHSI/CH pumps to RCS.	Same method of detection as that stated for item #4.	Valve aligned to open upon actuation by a SI "S" signal.
7. Motor-operated globe valve RH-FCV-610 (RH-FCV-611 analogous)	a. Fails to close on demand. b. Fails closed.	Injection-cold legs of RC loops. Injection-cold legs of RC loops.	a. Failure reduces working fluid delivered to RCS from RHR pump RH-P-8A. Minimum flow requirements for LHSI will be met by LHSI/RHR pump RH-P-8B delivering working fluid to RCS. b. Failure results in an insufficient fluid flow through LHSI/RHR pump RH-P-8A for a small LOCA or steam line break resulting in possible pump damage. If pump becomes inoperative minimum flow requirements for LHSI will be met by LHSI/RHR pump HR-P-8B delivering working fluid to RCS.	a. Valve position indication(open to closed position change) at MCB. RHR pump return line to cold legs flow indication (FI-618) at MCB. b. Same as that stated above for failure mode"Fails to close on demand" except closed to open position change at MCB.	Valve is regulated by signal from flow transmitter located in pump discharge header. The control valve opens when the RHR pump discharge flow is less than~700 gpm and closes when the flow exceeds ~1400 gpm.
8. Residual heat removal pump RH-P-8A(Pump RH-P-8A) analogous)	Fails to deliver working fluid.	Injection-cold legs of RC loops.	Failure reduces redundancy of providing emergency coolant to the RCS from the RWST at low RCS pressure(195 psig). Fluid flow from LHSI/RHR pump RH-P-8A will be lost. Minimum flow requirements for LHSI will be met by LHSI/RHR pump RH-P-8B delivering working fluid.	RHR pump return line to cold legs flow indication (FI-618) at MCB. RHR pump discharge pressure (PI-614) at MCB. Open pump switchgear circuit breaker indication at MCB. Circuit breaker close position monitoring light and alarm for group monitoring of components at MCB. Breaker trip alarm at MCB.	The RHR pump is sized to deliver reactor coolant through the RHR heat exchanger to meet plant cooldown requirements and is used during plant cooldown and startup operations. The pump circuit breaker is aligned to close on actuation by a SI "S" signal.
9. Safety injection pump SI-P-6A(Pump SI-P-6B analogous)	Fails to deliver working fluid.	Injection-cold legs of RC loops.	Failure reduces redundancy of providing emergency coolant to the RCS from the RWST at high RCS pressure(1520 psi). Fluid flow from HHSI/SI pump SI-P-6A will be lost. Minimum flow requirements for HHSI will be met by HHSI/SI pump SI-P-6B delivering working fluid.	SI pumps discharge pressure(PI-919) at MCB. SI pump discharge flow (FI-918) at MCB. Open pump switchgear circuit breaker indication at MCB. Circuit breaker close position monitor light and alarm for group monitor ing of components at MCB. Breaker trip alarm at MCB.	Pump aligned too close on circuit breaker actuation by a SI "S" signal.
10. Motor-operated globe valve CBS-V8 (CBS-V14 analogous)	Fails to open on demand.	Recirculation-cold legs of RC loops.	Failure reduces redundancy of providing fluid from the Containment Sump to the RCS during recirculation. LHSI/ RHR pump RH-P-8A will not provide recirculation flow. Minimum LHSI flow requirements will be met through opening of isolation valve CBS-V14 and recirculation of fluid by LHSI/RHR pump RH-P-8B.	Same method of detection as that stated for item #6. In addition failure may be detected through monitoring of RHR pump return line to cold legs flow indication (FI-618) and RHR pump discharge pressure (PI-614) at MCB.	Valve is actuated to open by SI "S" SIGNAL in coincidence with two-out-of-four "Low low Level" RWST signals. Valve is electrically interlocked from remotely being opened from MCB by isolation valves CBS-V2, RC-V23 and RC-V22.

Component	Failure Mode	ECCS Operation Phase	*Effect on System Operation	**Failure Detection Method	Remarks
11. Motor- operated gate valve CBS-V2 (CBS-V5 analogous)	Fails to close on demand.	Recirculation - col d legs of RC loops.	Failure reduces redundancy of providing flow isolation of Containment Sump from RWST. No effect on safety for system operation. Alternate check isolation valve CBS-V55 provides backup isolation.	Same method of detection as that stated for item #4.	Valve is electrically interlocked with isolation valve CBS-V8 and RH-V35 and may not be opened unless these valves are closed, for manual operation from main control board. Valve opens automatically on "S" signal.
12. Motor- operated gate valve RH-V14.	Fails to close on demand.	Recirculation - col d legs of RC loops.	Failure reduces redundancy of providing LHSI/RHR pump discharge flow path isolation of RCS. No effect on safety for system operation. Alternate isolation valve RH-V26 will be closed to isolate alternate flow path to cold legs.	Same method of detection as that stated for item #4.	
13. Motor-operated globe valve SI-V93	Fails to close on demand.	Recirculation - col d legs of RC loops.	Failure reduces redundancy of providing isolation of HHSI/SI pump's miniflow line isolation from RWST. No effect on safety for system operation. Alternate isolation valves SI-V89 and SI-V90 in each pumps' miniflow line provide backup isolation.	Same method of detection as that stated for item #4.	Valve is electrically interlocked with isolation valves RH-V35 and RH-V36 and may not be opened unless these valves are closed.
14. Motor-operated globe valve SI-V90 (SI-V89 analogous)	Fails to close on demand.	Recirculation - cold legs of RC loops.	Failure reduces redundancy of providing isolation of HHSI/SI pump SI-P-6A mini- flow isolation from RWST. No effect on safety for system operation. Alternate isolation valve SI-V93 in main miniflow line provides backup isolation.	Same method of detection as that stated for item #4.	Same remark as stated for item #16.
15. Motor-operated gate valve RH-V35	Fails to open on demand.	Recirculation - cold legs of RC loops.	Failure reduces redundancy of providing NPSH to suction of HHSI/CH pumps from LHSI/RHR pumps. No safety effect on system operation. Minimum NPSH to HHSI/CH pump suction will be met by flow from LHSI/RHR pump RH-P-8B via cross-tie line and opening of isolation valve CS-V460 or CS-V461 and isolation valve RH-V36.	Same method of detection as that stated for item #4.	Valve is electrically interlocked with isolation valves SI-V90, SI-V89, SI-V93, RC-V23, RC-V22 and CBS-V8. Valve cannot be opened unless valve SI-V93 or SI-V90 and SI-V89 valves are closed; valve RCS-V23 or RCS-V22 is closed, and CBS-V8 is open.
16. Motor-operated gate valve RH-V36.	Fails to open on demand.	Recirculation - cold legs of RC loops.	Failure reduces redundancy of providing NPSH to suction of HHSI/SI pumps from LHSI/RHR pumps. No effect safety for system operation. Minimum NPSH to HHSI/SI pump suction will be met by flow from LHSI/RHR pump RH-P-8A via cross-tie line and opening of isolation valve CS-V460 or CS-V461 and isolation valve RH-V35.	Same method of detection as that stated for item #4.	Valve is electrically interlocked with isolation valves, SI-V90, SI-V89 SI-V93, and CBS-V14, RC-V88 and RC-V87. Valve cannot be opened unless valve SI-V93 or SI-V90 and SI-V89 valves are closed; valve RC-V88 or RC-V87 is closed and valve CBS-V14 is open.

Component	Failure Mode	ECCS Operation Phase	*Effect on System Operation	**Failure Detection Method	Remarks
17. Motor-operated gate valve CS-V460 (CS-V461 analogous)	Fails to open on demand.	Recirculation - cold legs of RC loops.	Failure reduces redundancy of providing fluid flow through cross-tie between suction of HHSI/CH pumps and HHSI/SI pumps. No effect on safety of system operation. Alternate isolation valve (CS-V461 opens to provide backup flow path through cross- tie line.	Same method of detection as that stated for item #4.	
18. Motor-operated gate valve CBS-V47 (CBS-V51 analogous)	Fails to close on demand.	Recirculation - cold legs of RC loops.	Failure reduces redundancy of providing flow isolation of HHSI/SI pump suction from RWST. No effect on safety for system operation. Alternate check isolation valve (CBS-V48) provides backup isolation.	Same method of detection as that stated for item #4.	
19. Motor-operated gate valve LCV-112D (LCV-112E analogous)	Fails to close on demand.	Recirculation - cold legs of RC loops.	Failure reduces redundancy of providing flow isolation of suction of HHSI/CH pumps from RWST. No effect on safety for system operation. Alternate check isolation valve (CBS-V58) provides backup isolation.	Same method of detection as that stated previously for failure of item during injection phase of ECCS operation.	
20. Residual heat pump RH-P-8A (pump RH-P-8B analogous)	Fails to deliver working fluid.	Recirculation - cold legs of RC loops.	Failure reduces redundancy of providing recirculation of coolant to the RCS from the Containment Sump. Fluid flow from LHSI/RHR pump RH-P-8A will be lost. Minimum recirculation flow requirements for LHSI flow will be met by LHSI/RHR pump RH-P-8B delivering fluid.	Same method of detection as that stated previously for failure of item during injection phase of ECCS operation.	
21. Safety injection pump SI-P-6A (pump SI-P-6B analogous)	Fails to deliver deliver working fluid.	Recirculation - cold or hot legs of RC loops.	Failure reduces redundancy of providing recirculation of coolant to the RCS from the Containment Sump to cold legs of RC loops via RHR and SI pumps. Fluid flow from HHSI/SI pump SI-P-6A will be lost. Minimum recirculation flow requirements for HHSI flow will be met by HHSI/SI pump SI-P-6B delivering working fluid.	Same method of detection as that stated previously for failure of item during injection phase to ECCS operation.	
22. Motor-operated gate valve RH-V14	Fails to close on demand.	Recirculation - hot legs of RC loops.	Failure reduces redundancy of providing recirculation of coolant to the RCS from the Containment Sump to hot legs of RC loops. Fluid flow from LHSI/RHR pump RH-P-8A will continue to flow to cold legs for RC loops. Closure of backup isolation valve RH-V22 permits minimum recirculation flow requirement to hot legs of RC loops to be met by LHSI/RHR pump RH-P-8B recirculating fluid to RC hot legs directly and via HHSI/SI pumps.	Same method of detection as that stated for item #4.	

Component	Failure Mode	ECCS Operation Phase	*Effect on System Operation	**Failure Detection Method	Remarks
23. Motor-operated gate valve RH-V32 (RH-V70 analogous)	Fails to open on demand.	Recirculation - hot legs of RC loops.	Failure reduces redundancy of providing recirculation of coolant to the RCS from the containment sump to the hot legs of RC loops. No effect on safety for system operation. Alternate isolation valve (RH-V70) opens to provide flow path to RCS hot legs via LHSI/RHR pumps.	Same method of detection as that stated for item #4.	
24. Motor-operated gate valve RH-V26.	Fails to close on demand.	Recirculation - hot legs of RC loops.	Failure reduces redundancy of providing recirculation of coolant to the RCS from the Containment Sump to hot legs of RC loops. Fluid flow from LHSI/RHR pump RH-P-8B will continue to flow to cold legs of RC loops. Closure of backup isolation valve RH-V21 permits minimum recirculation flow requirements to hot legs of RC loops to be met by LHSI/RHR pump RH-P-8A recirculating fluid to RC hot legs directly and via HHSI/SI pumps.	Same method of detection as that stated for item #4.	
25. Motor-operated gate valve SI-V112 (SI-V111 analogous)	Fails to close on demand.	Recirculation - hot legs of RC loops.	Failure reduces redundancy of providing flow isolation of HHSI/SI pump flow to cold legs of RC loops. No effect on safety for system operation valve SI-V114 provides backup isolation against flow to cold legs of RC loops.	Same method of detection as that stated for item #4.	
26. Motor-operated gate valve SI-V102 (SI-V77 analogous)	Fails to open on demand.	Recirculation - hot legs of RC loops.	Failure reduces redundancy of providing recirculation of coolant to the hot legs of RCS from the Containment Sump via HHSI/SI pumps. Minimum recirculation flow requirements to hot legs of RC loops will be met by LHSI/RHR pump RH-P-8A and RH-P-8B recirculating fluid from Containment Sump to hot legs of RC loops and HHSI/SI pump SI-P-6B recirculating fluid to hot legs 2 and 3 of RC loops through the opening of isolation valve SI-V77.	Same method of detection as that stated for item #6. In addition, SI pump discharge pressure (PI-919) and flow (FI-918) at MCB.	
27. Motor-operated gate valve SI-V114	Fails to close on demand.	Recirculation - hot legs of RC loops.	Failure reduces redundancy of providing flow isolation of HHSI/SI pump flow to cold legs of RC loops. No effect on safety for system operation. Alternate isolation valves SI-V112 and SI-V111 in cross-tie line between HHSI/SI pumps provides backup isolation against flow to cold legs of RC loops.	Same method of detection as that stated for item #4.	

Component	Failure Mode	ECCS Operation Phase	*Effect on System Operation	**Failure Detection Method	Remarks
28. Residual heat removal pump RH-P-8A (Pump RH-P-8B analogous)	Fails to deliver working fluid.	Recirculation - hot legs of RC loops.	Failure reduces redundancy of providing recirculation of coolant to the RCS from the Containment sump to the hot legs of RC loops. Fluid flow from LHSI/RHR pump RH-P-8A will be lost. Minimum flow requirements to hot legs directly and via HHSI/SI pumps.	Same method of detection as that stated previously for failure of item during injection phase of ECCS operation except flow indication is not available.	

List of abbreviations and acronyms

CH, CS	Charging	RC	Reactor Coolant
HHSI	High Head Safety Injection	RCS	Reactor Coolant System
LHSI	Low Head Safety Injection	RHR, RH	Residual Heat Removal
LOCA	Loss of Coolant Accident	RWST	Refueling Water Storage Tank
MCB	Main Control Board	SI	Safety Injection
NPSH	Net Positive Suction Head	VCT	Volume Control Tank
CBS	Containment Spray		

TABLE 6.3-6 EMERGENCY CORE COOLING SYSTEM RECIRCULATION PIPING PASSIVE FAILURE ANALYSIS LONG-TERM PHASE

Flow Path	Indication of Loss of Flow Path	Alternate Flow Path
Low Head Recirculation		
From containment sump to low head injection header via the residual heat removal pumps and the residual heat exchangers	Accumulation of water in a residual heat removal pump compartment or auxiliary building sump	Via the independent, identical low head flow path utilizing the second residual heat exchanger and residual heat removal pump
High Head Recirculation		
From containment sump to the high head injection header via residual heat removal pump, residual heat exchanger and the high head injection pumps	Accumulation of water in a residual heat removal pump and safety injection pump compartment or the auxiliary building sump or charging pump compartments	From containment sump to the high head injection headers via alternate residual heat removal pump, residual heat exchanger, safety injection or charging pump

TABLE 6.3-7 MANUAL ACTIONS REQUIRED FOR ECCS OPERATION

(BASED ON NO SINGLE FAILURE)

During the injection mode of ECCS operation, the operators are not required to take any manual actions. All equipment operates automatically as designed. The minimum time calculated fort he injection mode is approximately 26 minutes. During this time, the operators verify that all ECCS pumps are operating, and monitor the RWST and Containment Building recirculation sump levels in anticipation of the transfer to the recirculation mode of ECCS operation. Component cooling water flow to the residual heat removal heat exchangers is automatically initiated on a 'T' signal. The operator verifies that this has occurred before the transfer to recirculation begins.

This table summarizes the manual operator actions required to complete the transfer from the injection mode of ECCS operation to the recirculation mode. This table assumes that no single failures have occurred before the transfer begins and also that none occur during the transfer. The actions listed are also based on the design basis large break LOCA all ECCS pumps operating, and any loss of off-site power occurring at the initiation of the safety injection signal.

This table is not intended to be a summary of the plant procedures that are used during this event. It is simply a summary of the manual actions required to support ECCS operation without any single failures being assumed to occur. This table assumes that all equipment operates as designed.

TRANSFER TO COLD LEG RECIRCULATION

The RWST "Lo-Lo" level signal in conjunction with an 'S' single initiates the automatic opening of the containment sump isolation valves. Once operators are alerted to this, they perform the following manual actions:

1. Reset the 'S' signal.

2. Verify the containment sump isolation valves (CBS-V8/V14) are open and close both RWST suction valves (CBS-V2/V5) to the RHR and CBS pumps.

3. Close the three safety injection pump miniflow valves (SI-V89 /V90 /V93)

4. Restore power to one of the two RHR to cold leg injection isolation valves (CS-9787-1 for RH-V14 or CS-9787 for RH-V26).

5. Close the selected RHR to cold leg injection isolation valve (RH-V14 or RH-V26).

6. Open the two parallel safety injection pump to charging pump suction cross-connect valves (CS-V460 / V461).

7. Open both RHR pump discharge valves to the safety injection pump and charging pump suctions (RH-V35 / V36).

NOTE: After the completion of the manual actions outlined above, all ECCS pumps are operating with their suctions being provided from the containment sump. The ECCS is now aligned for cold leg recirculation with both RHR pumps taking suction from the containment sump and delivering flow directly to two RCS cold legs and to the suctions for the safety injection and charging pumps. The safety injection and charging pumps are delivering this flow the RCS cold legs.

The following actions provide additional isolation of the RWST from the recirculation fluid.

8. Close the two RWST suction valves to the safety injection pumps (CBS-V47 / V51).

9. Close the two RWST suction valves to the charging pumps (CS-LCV-112D and E) and de-energize.

10. Remove power from the RHR to cold leg isolation valves.

At approximately 5 to 6 hours after the initiation of the accident, ECCS operation is shifted from the cold leg recirculation mode to the hot leg recirculation mode via the following manual actions:

1. Restore power to the two motor control centers required (CS-9787 or CS-9787-1).

2. Close the second RHR to cold leg injection isolation valve (RH-V14 or V26).

3. Open both RHR to hot leg injection isolation valves (RH-V32 / V70).

4. Stop No. 1 Safety injection pump.

5. Close No. 1 safety injection pump discharge cross connect valve (SI-V112).

6. Open No. 1 safety injection pump discharge hot leg recirculation isolation valve (SI-V102).

7. Start No. 1 safety injection pump.

8. Stop No. 2 safety injection pump.

9. Close No. 2 safety injection pump discharge cross connect valve (SI-V111).

10. Close the safety injection pumps common discharge isolation valve (SI-V114).

11. Open No. 2 safety injection pump discharge hot leg recirculation isolation valve (SI-V77).

12. Start No. 2 safety injection pump.

13. De-engergize the two motor control centers

The ECCS is now aligned for hot leg recirculation with both RHR pumps taking suction from the containment sump and delivering flow directly to two RCS hot legs and to the suctions of the safety injection and charging pumps. The safety injection pumps are now delivering flow to two RCS hot legs and the charging pumps are delivering flow to two RCS cold legs.

TABLE 6.3-8 EMERGENCY CORE COOLING SYSTEM SHARED FUNCTIONS EVALUATION

Component	Normal Operating Arrangement	Accident Arrangement
Refueling water storage tank	Lined up to suction of safety injection and residual heat removal pumps	Lined up to suction of centrifugal charging, safety injection and residual heat removal pumps
Centrifugal charging pumps	Lined up for charging service; suction from volume control tank, discharge via normal charging.	Suction from refueling water storage tank, discharge lined up to cold leg injection. Valves for realignment meet single failure criteria.
Residual heat removal pumps	Lined up to cold legs of reactor coolant piping	Lined up to cold legs or hot legs or reactor coolant piping
Residual heat exchangers	Lined up to cold legs of reactor coolant piping	Lined up to cold legs or hot legs of reactor coolant piping

TABLE 6.3-9 NORMAL OPERATING STATUS OF EMERGENCY CORE COOLING SYSTEM COMPONENTS FOR CORE COOLING

Number of safety injection pumps operable	2
Number of charging pumps operable	2
Number of residual heat removal pumps operable	2
Number of residual heat exchangers operable	2
Refueling water storage tank volume (gal)	477,000 (min.)
Boron concentration in refueling water storage tank (ppm)	2,400-2,600
Boron concentration in accumulator (ppm)	2,300-2,600
Number of accumulators	4
Minimum accumulator pressure (psig)	585
Nominal accumulator water volume (ft^3)	850
System valves, interlocks, and piping required for the above components which are operable	All

TABLE 6.3-10 DELETED

TABLE 6.4-1 CONTROL ROOM COMPLEX SAFETY-RELATED VENTILATION SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Comments
Normal Makeup Air Fan	Fan trips	Two 100 percent capacity fans are provided for normal makeup air and control room pressurization requirements. The redundant fan may be manually actuated to supply the makeup air. Each fan is powered from a separate emergency bus. Loss of air flow is alarmed. Fan status lights are provided on the main control board.
Normal Makeup Air Discharge Damper	Damper fails to open, or fails to close	If the damper fails to open, the redundant damper may be opened and its associated fan actuated to provide the makeup air. If both normal, makeup air trains are unavailable because of a vital bus outage, the emergency makeup air and filtration subsystem may be utilized. If the damper fails to close, the crosstrain control scheme design ensures that the associated fan trips to ensure isolation. Both normal makeup air dampers are provided with manual handwheel override actuators. Indicating lights on the main control board monitor all damper positions via limit switches on the damper linkage.
Emergency Makeup Air Fan	Fan fails to actuate on high intake radiation, 'S' signal, or manual Actuation	Two 100 percent capacity fans are provided with a fully redundant filter unit associated with each fan. Both fans are automatically actuated on high radiation or an 'S' signal. If one fan fails on manual actuation, the redundant fan may be manually actuated. Flow indication and alarms are provided for each filter/fan train. Each fan is powered from separate emergency buses. Fan status lights are provided on the main control board.

Component	Malfunction	Comments
Emergency Makeup Air Discharge Damper	Damper fails to open on high intake radiation, 'S' signal or manual Actuation	Both dampers are automatically opened on high radiation or an 'S' signal. If one damper fails to open on manual actuation, the redundant damper and associated fan may be actuated. Indicating lights on the main control board monitor all damper positions via limit switches on the damper linkage.
Water (35-40% ethylene glycol by volume) Chiller	Water Chiller trips	Two 100% capacity chillers, each supplied from a separate emergency bus, are provided. Secure the tripped train and place the redundant train in service.
Chilled Water Pump	Chilled Water Pump trips	Four 100% capacity pumps, two per train with each train supplied from a separate emergency bus, are provided. The diverse pump can be manually aligned and started to provide chilled water flow or the redundant train can be placed into service.
Air Handling Unit (including cooling coils)	Air Handling Unit trips	Two 100% capacity air handling units, each supplied from a separate emergency bus, are provided. The redundant train may be started to supply conditioned air to the control room.
Air Handling Unit Discharge Damper	Fails to open	Each of the two 100% capacity units is provided with its own discharge damper. When the damper fails to open, the air handling unit and the corresponding chiller and the chilled water pump will all trip. The redundant train may be started. Indicating lights on the air handling unit control panel monitor all damper positions via limit switches on the damper linkage.
	Fails to close	The damper may be manually positioned through the use of the handwheel on the actuator. Indicating lights on the air handling unit control panel monitor all damper positions via limit switches on the damper linkage.

Component	Malfunction	Comments
3-Way Temperature Control Valve (TCV)	TCV fails	Each train has its own TCV, and it is designed to fail-safe to "full flow to the air handling unit cooling coils". The failed train may be secured and the redundant train may be started.
Chiller Condenser Exhaust Fan	Exhaust fan fails	Two 100% capacity exhaust fans, each supplied from a separate emergency bus, are provided. The redundant train may be started.
Exhaust Fan Control Damper	Damper fails to close on actuation of emergency makeup air subsystem and/or a high radiation signal	Two redundant isolation dampers are configured in series. Both dampers close upon actuation of the emergency makeup air fans or generation of a remote intake high radiation signal ensuring isolation of the exhaust subsystem. Indicating lights on the main control board monitor all damper positions via limit switches on the damper actuator.
Cable Spreading Room Exhaust Fan	Exhaust Fan Belt failure and continued motor operation	Since the cable spreading room ventilation system is only placed in service using administrative controls, operator action to secure the associated cable spreading room supply fan can be performed promptly. Securing the supply fan will ensure that the Control Room Complex remains at the required differential pressure with respect to the cable spreading room.
Electrical Tunnel Exhaust Fan	Exhaust Fan Fails	Perforated plate in the cable spreading room exhaust fan inlet duct prevents the cable spreading room from over pressurizing when the cable spreading room ventilation system is not in service. This ensures that the Control Room Complex remains at the required differential pressure with respect to the cable spreading room.

TABLE 6.5-1 COMPLIANCE OF CONTAINMENT ENCLOSURE AIR CLEANING UNITS TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

Regulatory Guide Section	Applicability to this System	Comment Index	Regulatory Guide Section	Applicability to this System	Comment Index
C.1.a	Yes	Note 1	C.3.f	Yes	---
C.1.b	Yes	---	C.3.g	Yes	Note 11
C.1.c	Yes	---	C.3.h	Yes	---
C.1.d	Yes	---	C.3.i	Yes	Note 17
C.1.e	Yes	---	C.3.j	Yes	Note 12
C.2.a	Yes	Note 2	C.3.k	Yes	---
C.2.b	Yes	---	C.3.l	Yes	Note 13
C.2.c	Yes	---	C.3.m	Yes	---
C.2.d	Yes	Note 3	C.3.n	No	Note 5
C.2.e	Yes	---	C.3.o	Yes	---
C.2.f	Yes	---	C.3.p	Yes	---
C.2.g	Yes	Note 15	C.4.a	Yes	Note 14
C.2.h	Yes	Note 7	C.4.b	Yes	---
C.2.i	Yes	---	C.4.c	Yes	---
C.2.j	Yes	---	C.4.d	Yes	---
C.2.k	No	Note 4	C.4.e	Yes	---
C.2.l	Yes	---	C.5.a	Yes	Note 16
C.3.a	Yes	Note 8	C.5.b	Yes	Note 16
C.3.b	No	Note 9	C.5.c	Yes	Note 16
C.3.c	No	Note 2	C.5.d	Yes	Note 16
C.3.d	Yes	Note 6	C.6.a	Yes	Note 16
C.3.e	Yes	Note 10			

NOTES:

1. The design basis LOCA and rupture of a CRDM housing are the postulated design basis accidents.

2. Demisters will also serve as prefilters. No other prefilters are provided. Regulatory Guide 1.52, Revision 2, Section C.2, permits the use of demisters as prefilters for HEPA filters in an ESF System. The demister element consists of a 5½" thick, 0.006" diameter, 340 stainless steel mesh, and fibrous glass fill. The "dry" efficiency of the demister is approximately 45 percent when tested in accordance with NBS Dust Spot Test. This efficiency is similar to that of a prefilter. The demister will therefore serve a dual purpose, that of a demister and a prefilter; and there is no need to consider HEPA filter particulate loading without a prefilter for this application.

   Flow instrumentation is provided in common ductwork downstream of containment enclosure emergency exhaust filter fans. One channel of volumetric flow indication is indicated and alarmed at the MCB as well as indicated locally. Secondary flow indication is available using filter train differential pressure and fan status indication.

3. No significant pressure surges to this system are foreseen; thus, no special protective devices are needed.

4. There are no outdoor air intakes that could affect the operation of the system.

5. The system is located in the containment enclosure, the area served. Therefore, any leakage will eventually be re-routed through the cleanup system before being expelled to the atmosphere.

6. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. HEPA filters are designed and qualified to MIL-F-51068, MIL-F-51069 and UL-586. There is no need to withstand iodine removal sprays.

7. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Regulatory Guide 1.89 - IEEE Std. 323, Regulatory Guide 1.30 - IEEE Std. 336, Regulatory Guide 1.100 - IEEE Std. 344, Regulatory Guide 1.118 - IEEE Std. 338, Regulatory Guide 1.32 - IEEE 308.

8. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Demisters are qualified to MSAR 71-45.

NOTES:

9. Heaters are not required for this unit. The ambient design conditions for the site are 88°F dry bulb with a maximum relative humidity of 74.4 percent. Section 5.5 of ANSI 509-1976 states that "approximately 70% RH" is required upstream of the moisture separator. Only 33 percent of the total air supplied to the containment enclosure area is outside air. The remainder is recirculated by the containment enclosure cooling units as explained in Section 9.4.6. The cooling units will maintain the space temperature at or below 153°F at the outside design conditions. Therefore, since no moisture is added to the Supply Air System, the relative humidity will not exceed 50 percent; which is less than the 70 percent RH required by ANSI N509.

10. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Filter and adsorber mounting frames are constructed and designed in accordance with Section 4.3 of ERDA 76-21.

11. Equipment was purchased prior to issuance or Regulatory Guide 1.52, Rev. 2. Filter housings, floors and doors are constructed in accordance with ERDA 76-21.

12. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Tray type adsorber cells are designed, constructed and tested in accordance with AACC CS-8T.

13. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. The system fan, its mounting, and the ductwork connections are designed, constructed and tested in accordance with the requirements of Regulatory Guide 1.52, Rev. 1 and ANSI N509-1980 and ANSI N510-1980.

14. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Accessibility of components and maintenance are compatible with the intent of Regulatory Guide 1.52, Rev. 1.

15. Pertinent pressure drops (high delta-P across filter unit) are indicated, alarmed, and recorded at MCB.

16. In-place inspection and testing is performed in accordance with ANSI N510-11980.

17. Original charcoal was tested in accordance with ANSI N509-1976. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5 -2 COMPLIANCE OF FUEL STORAGE BUILDING AIR CLEANING UNITS TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

Regulatory Guide Section	Applicability to this System	Comment Index	Regulatory Guide Section	Applicability to this System	Comment Index
C.1.a	Yes	Note 1	C.3.f	Yes	---
C.1.b	Yes	---	C.3.g	Yes	Note 11
C.1.c	Yes	---	C.3.h	Yes	---
C.1.d	Yes	---	C.3.i	Yes	Note 16
C.1.e	Yes	---	C.3.j	Yes	Note 12
C.2.a	Yes	Note 2	C.3.k	Yes	---
C.2.b	Yes	---	C.3.l	Yes	Note 13
C.2.c	Yes	---	C.3.m	Yes	---
C.2.d	Yes	Note 3	C.3.n	No	Note 5
C.2.e	Yes	---	C.3.o	Yes	---
C.2.f	Yes	---	C.3.p	Yes	---
C.2.g	Yes	---	C.4.a	Yes	Note 14
C.2.h	Yes	Note 7	C.4.b	Yes	---
C.2.i	Yes	---	C.4.c	Yes	---
C.2.j	Yes	---	C.4.d	Yes	---
C.2.k	No	Note 4	C.4.e	Yes	Note 15
C.2.l	Yes	---	C.5.a	Yes	Note 15
C.3.a	Yes	Note 8	C.5.b	Yes	Note 15
C.3.b	Yes	Note 9	C.5.c	Yes	Note 15
C.3.c	No	Note 2	C.5.d	Yes	Note 15
C.3.d	Yes	Note 6	C.6.a	Yes	Note 15
C.3.e	Yes	Note 10			

NOTES:

1. A fuel handling accident is the postulated design basis accident.
2. Demisters will also serve as prefilters. No other prefilters are provided. Regulatory Guide 1.52, Revision 2, Section C.2, permits the use of demisters as prefilters for HEPA filters in an ESF System. The demister element consists of a 5½" thick, 0.006" diameter, 340 stainless steel mesh, and fibrous glass fill. The "dry" efficiency of the demister is approximately 45 percent when tested in accordance with NBS Dust Spot Test. This efficiency is similar to that of a prefilter. The demister will therefore serve a dual purpose, that of a demister and a prefilter; and there is no need to consider HEPA filter particulate loading without a prefilter for this application.
3. No significant pressure surges to this system are foreseen; thus, no special protective devices are needed.
4. There are no outdoor air intakes that could affect the operation of the system.
5. The system is located in the Fuel Storage Building, the area served. Therefore, any leakage will eventually be re -routed through the cleanup system before being expelled to the atmosphere.
6. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. HEPA filters are designed and qualified to MIL -F -51068, MIL -F -51069 and UL -586. There is no need to withstand iodine removal sprays.
7. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Regulatory Guide 1.89 -IEEE Std. 323, Regulatory Guide 1.30 - IEEE Std. 336, Regulatory Guide 1.100 - IEEE Std. 344, Regulatory Guide 1.118 - IEEE Std. 338, Regulatory Guide 1.32 - IEEE 308.
8. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Demisters are qualified to MSAR 71 -45.
9. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Heaters are constructed to industry standards and are seismically qualified.
10. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Filter and adsorber mounting frames are constructed and designed in accordance with Section 4.3 of ERDA 76 -21.
11. Equipment was purchased prior to issuance or Regulatory Guide 1.52, Rev. 2. Filter housings, floors and doors are constructed in accordance with ERDA 76 -21.
12. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Tray type adsorber cells are designed, constructed and tested in accordance with AACC CS -8T.

13. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. The system fan, its mounting, and the ductwork connections are designed, constructed and tested in accordance with the requirements of Regulatory Guide 1.52, Rev. 1 and ANSI N509 and N510 -1980.
14. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Accessibility of components and maintenance are compatible with the intent of Regulatory Guide 1.52, Rev. 1.
15. In -place inspection and testing is performed in accordance with ANSI N510 -1980.
16. Original charcoal was tested in accordance with ANSI N509 -1976. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803 -1989.

TABLE 6.5-3 COMPLIANCE OF CONTROL ROOM EMERGENCY FILTRATION SUBSYSTEM TO REGULATORY GUIDE 1.52, REV. 2. MARCH 1978

REGULATORY GUIDE SECTION	APPLICABILITY TO THIS SYSTEM	COMMENT INDEX
C.1.a	Yes	-------
C.1.b	Yes	Note 1
C.1.c	Yes	-------
C.1.d	Yes	-------
C.1.e	Yes	-------
C.2.a	Yes	Note 2
C.2.b	Yes	-------
C.2.c	Yes	-------
C.2.d	No	Note 3
C.2.e	Yes	-------
C.2.f	Yes	-------
C.2.g	Yes	Note 4
C.2.h	Yes	-------
C.2.i	Yes	-------
C.2.j	Yes	-------
C.2.k	Yes	-------
C.2.l	Yes	Note 8
C.3.a	No	Note 2
C.3.b	Yes	Note 5
C.3.c	Yes	Note 5
C.3.d	Yes	Note 5
C.3.e	Yes	Note 9, 19
C.3.f	Yes	-------
C.3.g	Yes	Note 10
C.3.h	Yes	-------
C.3.i	Yes	Note 11, 19
C.3.j	Yes	Note 12, 19
C.3.k	Yes	Note 6
C.3.l	Yes	Note 17
C.3.m	Yes	-------
C.3.n	Yes	
C.3.o	Yes	--------
C.3.p	Yes	Note 18

REGULATORY GUIDE SECTION	APPLICABILITY TO THIS SYSTEM	COMMENT INDEX
C.4.a	Yes	Note 16
C.4.b	Yes	Note 13
C.4.c	Yes	Note 14
C.4.d	Yes	Note 7
C.4.e	Yes	-------
C.5.a	Yes	Notes 7, 15
C.5.b	Yes	Notes 7, 15
C.5.c	Yes	Notes 7, 15
C.5.d	Yes	Notes 7, 15
C.6.a	Yes	Notes 7, 15

NOTE 1: The control room emergency exhaust unit is located in environmental zone CB-3 at Elevation 75' where the total of 40-year integrated radiation level is less than or equal to 1.0×10^3 rads. The other system components upstream and downstream of the unit are also located in "mild" radiation environmental zones.

NOTE 2: Demisters are not included in the filter design. The filter components for each redundant train include prefilters, an electric air heater, upstream HEPAs, an activated carbon adsorber bank, and downstream HEPAs. The piping which processes outside makeup air to the filters includes long vertical runs which will remove droplets entrained in the air stream.

NOTE 3: No accident which will cause pressure surges is postulated for the area in which the filters are located.

NOTE 4: The flow rate for each redundant filter train is indicated in the control room. High and low flow are alarmed in the control room. The pressure drop across the combined internal components for each filter train is indicated in the control room with high differential pressure generating an alarm. Pressure drops across each individual filter component are indicated locally in the control room HVAC equipment room. Additional details on system instrumentation are provided in Subsection 6.5.1.5

NOTE 5: The prefilters, air heaters, and HEPA filters were designed, constructed, and tested per ANSI N509-1980.

NOTE 6: The filter system design includes a low-flow air bleed system. A 4-inch diameter pipe with a manual isolation valve is provided to cross-connect the two redundant filter units at a point downstream of the carbon adsorber sections. During single train operation, this alignment will provide approximately 50 cfm of air flow through the carbon adsorber section of the inactive train. The configuration ensures that this low-flow cooling air is low humidity recirculation air. This satisfies the 70 percent maximum relative humidity criteria given a single active failure.

NOTE 7: System operability verification and surveillance testing will be performed in accordance with plant Technical Specifications.

NOTE 8: The atmosphere cleanup system housings and ductwork have been designed to exhibit on test a maximum, total leakage rate as defined in Section 4.12 of ANSI N509-1980. Leak tests are performed in accordance with Section 6 of ANSI N510-1980.

NOTE 9: The filter and adsorber mounting frames for Train A, except for the prefilter mounting frame, were designed and constructed prior to the issuance of this Regulatory Guide and meet the intent of ANSI N509-1980. The filter and adsorber mounting frames for Train B along with the prefilter mounting frame for Train A were designed and constructed per ANSI N509-1980.

NOTE 10: The filtration unit for Train A including floor/drains, et C., were designed and constructed prior to issuance of this Regulatory Guide and meet the intent of ANSI N509-1980. The Train B filtration unit was designed and constructed to ANSI N-509-1980.

NOTE 11: Carbon has been qualified to ANSI N509-1980*. The Train B adsorber cell is a four-inch deep bed with a minimum residence time of 0.25 seconds. The design iodine removal efficiency is consistent with that of a two-inch deep bed (95%).

NOTE 12: The adsorber cells for Train A have been designed and constructed per ANSI N509-1976 and meet the intent of ANSI N509-1980. The adsorber cells for Train A have been tested per ANSI N510-1980. The adsorber cell for Train B has been designed, constructed, and tested per ANSI N509-1980.

NOTE 13: Train B replaceable components are designed for removal from outside the filter unit.

---

* Information is historical and applies to the original charcoal installed. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

NOTE 14: Meets requirements of Section 4.11 of ANSI N509-1980.

NOTE 15: All in-place testing/inspection is per ANSI N510-1980 requirements, with acceptance criteria of ANSI N509-1980 as applicable.

NOTE 16: The system layout for Train A was performed prior to the issuance of this Regulatory Guide, however, it meets the requirements of Section 4.7 of ANSI N509-1980 and the intent of Subsection 2.3.8 of ERDA 76-21. Train B meets the requirements of ANSI N509-1980 and ERDA 76-21.

NOTE 17: The system fan motor, mountings and ductwork connections for Train A were procured prior to issuance of this Regulatory Guide. However, they are designed and constructed to meet the intent of Sections 5.7 and 5.8 of ANSI N509-1976. They are field tested/inspected per Section 8 of ANSI N510-1980, with the acceptance criteria of ANSI N509-1980. Train B has been designed, constructed, and tested per ANSI N509-1980.

NOTE 18: The system dampers were procured prior to the assurance of this Regulatory Guide. However, they are designed, constructed, and tested per the intent of Section 5.9 of ANSI N509-1976 and ANSI N510-1975. They are field tested/inspected as a part of ductwork per ANSI N510-1980 requirements, with the acceptance criteria of ANSI N509-1980.

NOTE 19: Carbon adsorber cells ordered after May 27, 2009 have been designed and constructed in accordance with Regulatory Guide 1.52, Revision 3, June 2001. Revision 3 endorses ASME AG-1-1997, "Code on Nuclear Air and Gas Treatment" and ASME N509-1989, "Nuclear Power Plant Air Cleaning Units and Components." Revision 3 also notes that ESF Atmosphere Cleanup Systems Designed to ASME N509-1989 (or earlier versions) and tested to ASME N510-1989 (or earlier versions) are considered adequate to protect public health and safety. This exception to Regulatory Guide 1.52, Revision 2, applies only to carbon adsorbent units.

TABLE 6.5-4 CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANING SYSTEM MATERIALS

Component	Parameter
1) Moisture Separator (2)	
Filtration efficiency at design air flow (2000 cfm)	99% entrained moisture
Water capacity at design air flow (2000 cfm)	3 lb/min entrained water
Case Material	Type 304 Stainless Steel
Media Material	Type 304 Stainless Steel & Fiberglass
2) HEPA Filter (4)	
Efficiency at rated flow(2000 cfm), 20% and 120% rated flow	99.97% at 0.3 microns of DOP
Dust holding capacity	4 lbs.
Case Material	409 Stainless Steel
Media Material	Fiberglass
Separator Material	None
3) Carbon Adsorber	
a. Lot Requirements (See Note 2)	Efficiency
Low Temperature	99%
Ambient Pressure	
Methyl Iodide at 95%	
RH and 25°C	
Low Temperature	99.9%
Ambient Pressure	
Elemental Iodine at 95%	
RH and 30°C	
High Temperature	99%
Ambient Pressure	
Methyl Iodide at 95%	
RH and 80°C	
Methyl Iodine in	98%
Containment at 95% RH	

Component		Parameter
	130°C, 3.7 Atm., 1 Hour	
	Load and 4 Hour Post-Sweep	
b.	Batch Requirements (See Note 2)	
	Low Temperature	97%
	Ambient Pressure	
	Methyl Iodide at 95% RH and 30°C	
	High Temperature	99%
	Ambient Pressure	
	Methyl Iodide at 95% RH and 80°C	
	Except Pre and Post Sweep at 25°C	
	Elemental	99.9% Loading
	Iodide Retention at 180°C	99.5% Retentivity
	Media	Activated Coconut Shell
	Carbon	
	Impregnating Material	KI_3
	Ignition Temperature (ASTM D3466)	330°C
	Density (ASTM D2854)	0.38g/cc (min)
	Hardness (ASTM D3802)	97%
	Mesh Size (ASTM D2862)	5% Maximum Retention on 8 90-100% thru 8 on 16 (8x12 Mesh 40-60%) (12x16 Mesh 40-60%) 5% maximum thru 16; 1% Maximum thru 18
	Depth of carbon bed	4 inches
	Total weight of carbon	804 lbs
	Carbon Bed Envelope Material	Type 304 Stainless Steel
4)	Filter Mounting Frames	Type 304 Stainless Steel
5)	Filter System Housing	Epoxy Coated Carbon Steel
6)	Ductwork	Galvanized Steel
7)	Fan	Carbon Steel

Note 1: Refer to Chapter 15 Appendix B for filter efficiencies assumed for design basis accidents.

Note 2: Testing information is historical and applies to the original charcoal installed. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5-5 FUEL STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM MATERIALS

	Component	Parameter
1)	Moisture Separator (12)	
	Filtration efficiency at design air flow (17,000 cfm)	99% entrained moisture
	Water capacity at design air flow (17,000 cfm)	24 lb/min entrained water
	Case Material	Type 304 Stainless Steel
	Media Material	Type 304 Stainless Steel & Fiberglass
2)	Medium Efficiency Filter (12)	
	Filtration efficiency at design air flow (17,000 cfm)	80%
	Dust Holding Capacity	340 grams
	Case Material	409 Stainless Steel
	Media Material	Fiberglass
	Separator Material	None
3)	HEPA Filters (24)	
	Efficiency at rated flow(17,000 cfm), 20% and 120%	99.97% at 0.3 microns of DOP rated flow
	Dust Holding Capacity	4 lbs
	Case Material	409 Stainless Steel
	Media Material	Fiberglass
	Separator Material	None
4)	Carbon Adsorber	
	a. Lot Requirements (See Note 2)	Efficiency
	Low Temperature	99%
	Ambient Pressure	

Component	Parameter
Methyl Iodide at 95% RH and 25°C	
Low Temperature	99.9%
Ambient Pressure	
Elemental Iodine at 95%	
RH and 30°C	
High Temperature	99%
Ambient Pressure	
Methyl Iodine at 95%	
RH and 80°C	
b. Batch Requirements (See Note 2)	
Low Temperature	97%
Ambient Pressure	
Methyl Iodide at 95% RH and 30°C	
High Temperature	99%
Ambient Pressure	
Methyl Iodide at 95% RH and 80°C	
Except Pre and Post Sweep at 25°C	
Elemental	99.9% Loading
Iodine Retention at 180°C	99.5% Retentivity
Media	Activated Coconut Shell
Carbon	
Impregnating Material	KI_3
Ignition Temperature (ASTM D3466)	330°C

	Component	Parameter
	Density (ASTM D2854)	0.38 g/cc (min)
	Hardness (ASTM D3802)	97%
	Mesh Size (ASTM D2862)	5% Maximum Retention on 8 90-100% thru 8 on 16 (8x12 Mesh 40-60%) (12x16 Mesh 40-60%) 5% Maximum thru 16; 1% Maximum thru 18
	Depth of carbon bed	4 inches
	Total weight of carbon	6500 lbs
	Carbon bed envelope material	Type 304 Stainless Steel
5)	Filter Mounting Frames	Type 304 Stainless Steel
6)	Filter System Housing	Epoxy Coated Carbon Steel
7)	Ductwork	Galvanized Steel
8)	Fan	Carbon Steel

Note 1: Refer to Chapter 15 Appendix B for filter efficiencies assumed for design basis accidents.

Note 2: Testing information is historical and applies to the original charcoal installed. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5-6 CONTROL ROOM EMERGENCY MAKEUP AIR AND FILTRATION SUBSYSTEM MATERIALS

TRAIN A FILTER

	Component	Parameter
1)	Medium Efficiency Filter (1)	
	Filtration Efficiency at Design Air Flow (1100 cfm)	80%
	Dust Holding Capacity	340 grams
	Case Material	409 Stainless Steel
	Media Material	Fiberglass
	Separator Material	None
2)	HEPA Filters (4)	
	Filtration Efficiency at Design Air Flow (1100 cfm), 20% and 120% Rated Flow	99.97% at 0.3 Microns of DOP
	Dust Holding Capacity	4 lbs.
	Case Material	409 Stainless Steel
	Media Material	Fiberglass
	Separator	None
3)	Carbon Adsorber (7 trays)	
	a. Lot Requirement (See Note 1)	
	Methyl Iodide, 80°C, 95% RH (ASTM D3803)	1% Penetration, Maximum
	Molecular Iodine, 30°C, 95% RH (ASTM D3803)	0.1 Penetration, Maximum
	b. Batch Requirement (See Note 1)	
	Molecular Iodine, 180°C (ASTM D3803)	99.5 Retentivity, Minimum
	Methyl Iodine, 30°C, 95% RH (ASTM D3803)	3% Penetration, Maximum
	Media Carbon	Activated Coconut Shell
	Impregnating Material	Iodine Salts & Tertiary Amines
	Ignition Temperature (ASTM D3466)	330°C Minimum
	Density (ASTM D2854)	0.38 g/cc Minimum
	Hardness (ASTM D3802)	92 Minimum

Note 1 Testing information is historical and applies to the original charcoal installed. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TRAIN A FILTER

Component	Parameter
b. Batch Requirement (See Note 1)	
Mesh Size (ASTM D2862)	Retained on #6 Sieve: 0.1% Maximum
	Retained on #8 Sieve: 5.0% Maximum
	Through #8, on #12 Sieve: 60% Maximum
	Through #12, on #16 Sieve: 40% Minimum
	Through #16 Sieve: 5.0% Maximum
	Through #18 Sieve: 1.0% Maximum
Depth of Carbon Bed	2 Inches (tray type-6 trays, 1 test tray)
Total Weight of Carbon	300 lbs. (not including test canisters)
Carbon Tray Envelope Material	Type 304 Stainless Steel
4) Filter Mounting Frames	Type 304 Stainless Steel
5) Filter System Housing	Epoxy Coated Carbon Steel
6) Ductwork/Piping	Galvanized Steel/Carbon Steel
7) Fan	Carbon Steel Housing, Aluminum Blades and Hub

TRAIN B FILTER

Same as Train A filters with the following exceptions:

1) HEPA Filters (2)	
2) Carbon Adsorber	
Depth of Carbon Bed	4 Inches
Total Weight of Carbon	390 lbs. (not including test canisters
3) Filter System Housing	Type 304 Stainless Steel
4) Fan	Carbon Steel

Note 1 Testing information is historical and applies to the original charcoal installed. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5-7 VOLUMES OF CONTAINMENT ENCLOSURE AREAS

AREA	VOLUME (ft^3)
Containment Enclosure Annulus	524,344
Electrical Penetration Areas	84,035
Mechanical Penetration Areas	70,320
RHR and SI Equipment Vaults	102,816
Containment Enclosure Equipment Area	92,568
Charging Pump Areas	12,000

TABLE 6.5-8 CONTAINMENT OPERATION FOLLOWING A DESIGN BASIS ACCIDENT

General

- Type of Structure	Concrete cylinder with hemispherical dome and welded steel liner
- Fission Product Removal System	Water/NaOH building spray
- Free Volume	2.715×10^6 ft^3
- Hydrogen Purge Mode	Containment internal recombiners with backup charcoal filtered purge to environment

Time Dependent Parameters	Anticipated	Conservative
- Leak Rate of Containment (percent of building volume per day)		
0-24 hours	0.10	0.20
24 hours-30 days	0.05	0.10
- Containment Enclosure Emergency Exhaust Filter Bypass (percent of primary containment leakage)		
0-24 hours	7.5	15.0
24 hours-30 days	3.75	7.5

TABLE 6.8-1 EMERGENCY FEEDWATER PUMP DATA

Total Number Per Unit	2
Electric Motor-Driven	1
Turbine-Driven	1
Design Flow (each)	710 gpm
Design Basis Flow (each)	≥650 gpm
Design Head	3050 ft. (1320 psi)
Feedwater Design Temperature	50-100°F
Required BHP	770
Motor Size, HP	900
Turbine Rating, HP	900

TABLE 6.8-2 EMERGENCY FEEDWATER SYSTEM FAILURE ANALYSIS

Active Failure Concurrent with a Feedwater System Pipe Break (One SG* Faulted)

COMPONENT	ACTIVE FAILURE	SYSTEM RESPONSE
Emergency Feed Pump	Pump fails to start (loss of power source)	Second pump starts and provides required flow to the intact SGs. Flow sensing elements identify and isolate faulted SG.
Flow Control Valve	One of two valves in line to faulted SG fails to close	No effect on system response. The second (redundant) valve in line closes to isolate the faulted SG.
Flow Control Valve	Valve in line to intact SG closes on spurious signal	Two pumps start; flow sensing elements identify and isolate faulted SG. Pumps provide the required flow to the two intact SGs.
Check Valve in Pump Discharge Piping	Check valve fails to open; flow from one pump blocked	Second pump provides required flow to the three intact SGs. Flow sensing elements identify and isolate faulted SG.

* SG - steam generator