SYSTEM 80+TM

STANDARD DESIGN

CESSAR DESIGN
CERTIFICATION

Volume 5

COMBUSTION ENGINEERING

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5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

The reactor is a pressurized water reactor (PWR) with two coolant loops. The reactor coolant system (RCS) circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary system. The steam generators provide the interface between the reactor coolant (primary) system and the main steam (secondary) system. The steam generators are vertical U-tube heat exchangers with an integral economizer in which heat is transferred from the reactor coolant to the main steam system. Reactor coolant is prevented from mixing with the secondary steam by the steam generator tubes and the steam generator tube sheet, making the RCS a closed system thus forming a barrier to the release of radioactive materials from the core of the reactor to the secondary system and containment building. | I

The arrangement of the RCS is shown in Figures 5.1.3-1 and 5.1.3-2. The major components of the system are the reactor vessel; two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps; a pressurizer connected to one of the reactor vessel hot legs; and associated piping. All components are located inside the containment building.

Table 5.1.1-1 shows the principal pressures, temperatures, and flowrates of the RCS under normal steady-state, full-power operating conditions. Instrumentation provided for operation and control of the system is described in Chapter 7.

System pressure is controlled by the pressurizer, where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the pressurizer, or is condensed by the pressurizer spray to limit pressure variations caused by contraction or expansion of the reactor coolant.

The average temperature of the reactor coolant varies with power level and the fluid expands or contracts, changing the pressurizer water level.

The charging pumps and letdown control valves in the chemical and volume control system (CVCS) are used to maintain a programmed pressurizer water level. A continuous but variable letdown purification flow is maintained to keep the RCS chemistry within prescribed limits. A charging nozzle and a letdown nozzle are provided on the reactor coolant piping for this operation. The charging flow is also used to alter the boron concentration or correct the chemical content of the reactor coolant.

Other reactor coolant system penetrations are the pressurizer surge line in one hot leg; the four direct vessel injection nozzles for the safety injection system; two return nozzles to the shutdown cooling system, one in each hot leg; two pressurizer spray nozzles; vent and drain connections; and sample and instrument connections.

Overpressure protection for the reactor coolant pressure boundary is provided by four spring-loaded ASME Code safety valves connected to the top of the pressurizer. These valves discharge to the in-containment refueling water storage tank, where the steam is released under water to be condensed and cooled. If the steam discharge exceeds the capacity of the in-containment refueling water storage tank, it is vented to the containment atmosphere.

Overpressure protection for the secondary side of the steam generators is provided by spring-loaded ASME Code safety valves located in the main steam system upstream of the steam line isolation valves.

Components and piping in the RCS are insulated with a material compatible with the temperatures involved to reduce heat losses and protect personnel from high temperatures.

Principal parameters of the RCS are listed in Table 5.1.1-2. Table 5.1.1-3 lists RCS volumes.

Shielding requirements of the surrounding structures are described in Section 12.3. Reactor coolant system shielding permits limited personnel access to the containment building during power operation. The reactor vessel sits in a primary shield well. This and other shielding reduces the dose rate within the containment and outside the shield wall during full power operation to acceptable levels.

5.1.1 SCHEMATIC FLOW DIAGRAM

The principal pressures, temperatures, and flow rates at major components are listed in Table 5.1.1-1. These parameters are referenced to Figure 5.1.2-1, the piping and instrument diagram, by numbered locations. Instrumentation provided for operation and control of the RCS is described in Chapter 7 and is indicated on Figure 5.1.2-1.

TABLE 5.1.1-1
PROCESS DATA POINT TABULATION*

<u>Parameter</u>	<u>Pressurizer</u>	<u>Steam Generator 1 Midpoint</u>	<u>Pump 1-B Outlet</u>	<u>Reactor Vessel Midpoint</u>	<u>Pump 1-A Outlet</u>	<u>Steam Generator 2 Midpoint</u>	<u>Pump 2-A Outlet</u>	<u>Pump 2-B Outlet</u>	
Data Point Fig. 5.1.2-1	1	2	3	4	5	6	7	8	
Pressure, psia	2250	2240	2330	2300	2330	2240	2330	2330	E
Temperature, °F	652.7	586.5	558.0	588.8	558.0	586.5	558.0	558.0	I
Mass Flow Rate, lbm/hr	N/A**	83x10 ⁶	41.5x10 ⁶	166x10 ⁶	41.5x10 ⁶	83x10 ⁶	41.5x10 ⁶	41.5x10 ⁶	
Volumetric Flow Rate, gpm	N/A**	233,200	111,400	468,700	111,400	233,200	111,400	111,400	E

* For steady-state, 100% power conditions.

** Not applicable.

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TABLE 5.1.1-2DESIGN PARAMETERS OF REACTOR COOLANT SYSTEM

Design Thermal Power, Mwt (Including Net Heat Addition from Pumps)	3817	
Thermal Power, Btu/hr (Developed by the RCS)	1.303×10^{10}	
Design Pressure, psia	2500	
Design Temperature (Except Pressurizer), °F	650	
Pressurizer Design Temperature, °F	700	I
Coolant Flow Rate, lb/hr	165.8×10^6	E
Cold Leg Temperature, Operating, °F	558	B
Average Temperature, Operating, °F	587	
Hot Leg Temperature, Operating, °F	615	
Normal Operating Pressure, psia	2250	
System Water Volume, Ft ³ (Without Pressurizer)	13,100	E
Pressurizer Water Volume, Ft ³ (Full Power)	1200	
Pressurizer Steam Volume, Ft ³ (Full Power)	1200	B

TABLE 5.1.1-3
REACTOR COOLANT SYSTEM VOLUMES

<u>Component</u>	<u>Volume (ft³)</u>	
Reactor Vessel	5829.9	D
Steam Generators	2,800 each	B
Reactor Coolant Pumps	134 each	
Pressurizer	2400	B
Piping	.	
Hot Leg	135.3 each	
Cold Leg	214.1 each	
Surge Line (nominal)	43.6	

5.1.2 PIPING AND INSTRUMENT DIAGRAM

Figure 5.1.2-1 is the piping and instrument diagram of the RCS. The entire system is located within the containment. Fluid systems which are connected to the reactor coolant system and which are included within the limits of the reactor coolant pressure boundary, as defined in ANSI 51.1-1983 and 10 CFR 50.2(v), are identified and the appropriate piping and instrument diagrams in other sections are referenced. Figure 5.1.2-2 is the piping and instrument diagram for the reactor coolant pumps. ^B

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9. TEMPERATURE SENSORS T-113, T-114, T-115, T-123, T-124, T-125, T-133, T-134 AND T-135 PROVIDE SIGNALS TO THE PLCS FOR DENSITY COMPENSATION OF THE PRESSURIZER LEVEL INSTRUMENT REFERENCE LEGS

8. HANDSWITCHES AT RSP FOR PRESSURIZER BACKUP HEATERS ARE PROVIDED FOR ONLY THOSE HEATERS CAPABLE OF BEING ALIGNED TO THE IE POWER SUPPLY

7. * DENOTES SELECTED CONTROL SYSTEM INPUT, P-101A, P-101B, P-101C, P-101D, OR VALIDATED PRESSURIZER PRESSURE SIGNAL FOR THE PPCS, AND L-110A, L-110B, L-103, OR VALIDATED PRESSURIZER LEVEL SIGNAL FOR THE PLCS

6. FOR ALL MECHANICAL AND PLANT ARRANGEMENT INFORMATION, SEE THE RCS GENERAL ARRANGEMENT DWG. E-ALWR-321-001

5. ALL FLOW RESTRICTING ORIFICES ARE PROVIDED AS PART OF THE NOZZLE

4. THE PRESSURIZER LIQUID SAMPLE LINE IS TO TAP INTO THE SURGE LINE AS CLOSE AS POSSIBLE TO THE PRESSURIZER

3. PRESSURIZER SAFETY VALVES PROVIDED WITH CLOSED/NOT CLOSED (I.E., OPENING) POSITION INDICATION. ALARMS ON VALVE NOT FULLY CLOSED (INFC)

2. THIS DRAWING SERVES TO INDICATE INSTRUMENT FUNCTIONS ONLY. SPECIFIC DETAILS OF INSTRUMENT HARDWARE AND PROCESSING ARE SHOWN ON THE 100 SERIES MCBDS

1. FOR SYMBOLS AND ABBREVIATIONS SEE REF. DWG. E-ALWR-310-100

E-ALWR-310-112

00

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SYSTEM 80 +

PRESSURIZER & SAFETY DEPRESSURIZATION SYSTEM
PIPING AND INSTRUMENTATION DIAGRAM

Figure

5.1.2-3

5.1.3 ARRANGEMENT DRAWINGS

I

Reactor coolant system plan and elevation drawings are provided as Figures 5.1.3-1 and 5.1.3-2.

Major components of the RCS are usually surrounded by concrete structures, which provide support plus shielding and missile protection. Elevation drawings, illustrating principal dimensions of the RCS in relationship to the surrounding building structures, are presented in the figures of Section 1.2.

I

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) throughout the facility's design lifetime. The RCPB is defined in accordance with ANSI/ANS 51.1-1983. Included are all pressure containing components such as pressure vessels, piping, pumps, and valves which are: |_B

- A. Part of the Reactor Coolant System, or
- B. Connected to the Reactor Coolant System, up to and including the following:
 - 1. The outermost containment isolation valve in piping which penetrates the containment;
 - 2. The second of two valves normally closed during reactor operation in piping which does not penetrate the containment.

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10 CFR 50.55a

The codes and component classifications are listed in Table 5.2-1 and are in accordance with the provisions of 10 CFR 50.55a.

5.2.1.2 Applicable Code Cases

Reactor Coolant Pressure Boundary components are fabricated in accordance with the ASME Code, Section III. |_I

Unless otherwise stated in the site-specific SAR, Combustion Engineering, Inc., will comply with Regulatory Guides 1.84, 1.85 and 1.147 in determining suitable ASME Code Cases. Code Cases not included in the Regulatory Guides may be used with specific authorization from the Commission under 10 CFR 50.55a. |_D |_B

5.2.2 OVERPRESSURE PROTECTION

5.2.2.1 Design Bases

Appendix 5A presents the design bases for sizing the overpressurization protection system. The loss of load transient which is used to size the primary safety valves is not intended to be used as a design transient for any other NSSS equipment.

5.2.2.2 Design Evaluation

Section 15.2 of Chapter 15 provides the functional design evaluation of the overpressurization protection system. In this analysis, the adequacy of the overpressure protection system to maintain secondary and primary operating pressures within 110% of design is clearly demonstrated for the loss of load analysis. The analytical model used in the analysis is discussed in Chapter 15.

Chapter 15 lists the assumptions used in the turbine trip analysis. These assumptions are chosen so that they tend to maximize the required pressure relieving capacity of the primary and secondary valves. The analysis demonstrates that sufficient relieving capacity has been provided so that when acting in conjunction with the reactor protective system the safety valves will prevent the pressure from exceeding 110% of the design pressure. |E

5.2.2.3 Piping and Instrumentation Diagrams

The piping and instrumentation diagram showing the primary safety valves and their discharge lines is given in Figure 5.1.2-1. The piping and instrumentation diagram showing the reactor drain tank is given in Figure 9.3.4-1, Sheet 2. The secondary safety valves are discussed in Section 10.3. |E

5.2.2.4 Equipment and Component Description

The primary safety valves are direct acting, spring loaded, stainless steel valves with enclosed bonnets. These valves are mounted on the top of the pressurizer. For further description of these valves, refer to Section 5.4.13. A schematic drawing of the primary safety valve is given in Figure 5.4.13-1. Valve parameters are given in Table 5.4-13.

Primary safety valve operation is characterized by a sharp pop at the set pressure. This sharp opening is produced by two stages of reaction working together to produce a continuous action. The initial lift is produced when the steam pressure under the disc exceeds the spring force. The escaping steam reacts against the upper guide ring and pushes the disc up to a high lift. The reaction of the deflected steam against the underside of the disc lifts it higher on an accumulation of pressure. The valve reaches a lift in excess of full bore lift within an accumulation of 3 percent above the set pressure.

As the system pressure drops, the valve disc settles to a moderate lift, and closes sharply with a blowdown of approximately 18.5 percent of the set pressure.

NOZZLE IDENTIFICATION				
CODE LETTER	FUNCTION	PIPE SIZE	END PREPARATION	QTY
A	PRESSURIZER SURGE	12" SCH 160	BUTT WELD	1
B	SHUTDOWN COOLING OUTLET	16" SCH 160	BUTT WELD	2
C	SAFETY INJECTION AND SHUTDOWN COOLING INLET	10" SCH 160	BUTT WELD	4
D	PRESSURIZER SPRAY	3" SCH 160	BUTT WELD	2
E	CHARGING INLET	2" SCH 160	BUTT WELD	1
F	LET DOWN & DRAIN	2" SCH 160	BUTT WELD	1
G	DRAIN	2" SCH 160	BUTT WELD	3
H	TEMPERATURE MEASUREMENT	1" SCH 160	SOC WELD	23
K	SAMPLING	3/4" SCH 160	SOC WELD	4
L	FEEDWATER	DOWNCOMER 6" SCH 80 ECONOMIZER 14" SCH 80	BUTT WELD	2 4

NOZZLE IDENTIFICATION				
CODE LETTER	FUNCTION	PIPE SIZE	END PREPARATION	QTY
M	STEAM OUTLET	28" I.D.	BUTT WELD	4
N	SAFETY VALVE INLET	6" SCH 160	BUTT WELD	4
O	PRESSURIZER LEVEL	3/4" SCH 160	SOC WELD	4
P	PRESSURIZER PRESSURE	3/4" SCH 160	SOC WELD	2
Q	PRESSURIZER SPRAY	4" SCH 160	BUTT WELD	1
R	PRESSURIZER TEMPERATURE	1" SCH 160	SOC WELD	1
S	MANWAY	24" I.D.		9
T	RCP SUCTION/DISCHARGE PRESSURE	3/4" SCH 160	SOC WELD	8
U	S/G PRESSURE/LEVEL MEASUREMENT	3/4" SCH 160	SOC WELD	24
V	BOTTOM BLOWDOWN	6" SCH 80	BUTT WELD	4
W	PRESSURE TEST	3/4" SCH 160	SOC WELD	2
X	S/G PRESSURE DROP TAPS	3/4" SCH 160	SOC WELD	16
Y	PRIMARY SIDE OUTLET	30" I.D.	BUTT WELD	4
Z	PRIMARY SIDE INLET	48" I.D.	BUTT WELD	2
AA	RECIRCULATION	6" SCH 80	BUTT WELD	1

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SYSTEM 80 +™

REACTOR COOLANT SYSTEM ARRANGEMENT-
PLAN 3817 MW(t)

Figure

5.1.3-1

THIS DRAWING TO BE USED FOR INFORMATION
AND GUIDANCE ONLY. REFERENCE DRAWINGS
SHALL BE USED FOR DETAILED DIMENSIONS

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December 30, 1988

Figure

5.1.3-2

REACTOR COOLANT SYSTEM ARRANGEMENT -
ELEVATION 3817 MW(t)

SYSTEM 80™

Each primary safety valve inlet line is designed to pass 630,000 lbm/hr with a maximum pressure drop of 50 psi from the pressurizer to the safety valve. This pressure drop of 50 psi is for piping and nozzle losses.

Each primary safety valve discharge line is designed to pass a maximum steam flow of 630,000 lbm/hr from the safety valve to one of the main header lines in the Steam Relief System (Section 6.8.2.2.2). The safety valve discharge flow analysis includes the effects of steam flow and Rapid Depressurization Function flow which result in two-phase flow in the Steam Relief System. The safety valve discharge lines collect into two headers which are routed from the pressurizer cubicle to the In-containment Refueling Water Storage Tank (IRWST). See Section 6.8 for a description of the IRWST. Each Steam Relief System discharge line will pass the maximum steam flow from two safety valves, 1,260,000 lbm/hr (2 valves time 630,000 lbm/hr).

5.2.2.4.1 Transients

The primary safety valves will be designed to withstand the following transients without failure or malfunction:

- A. 650°F to 375°F in 50 seconds and return to 650°F in 2000 seconds for 8 cycles (Loss of secondary pressure).
- B. 100°F to 400°F and return to 100°F at a rate of 100°F/hr with concurrent pressure changes from 400 psig to 2250 psig and returning to 400 psig in step changes for 300 cycles. (Plant leak test).
- C. $\pm 10^\circ\text{F}$ step change from 653°F for 1.5×10^6 cycles. (Plant loading and unloading, $\pm 10\%$ step load, normal plant variation).
- D. 75°F to 653°F and return to 75°F at a rate of 200°F/hr with pressures at saturation levels for 750 cycles. (Plant heat up and cool down).

Note: Heat up and cool down are separate transients, each beginning at steady state conditions.

- E. Pressurize to 1.5 times set pressure at 100°F-200°F for 15 cycles plus number of hydros conducted prior to valve shipment. (Hydrostatic test).
- F. 720 cycles from closed to full open to closed. (Turbine Trip).

5.2.2.4.2 Environment

The primary safety valves are designed to operate in the following environmental conditions.

5.2.2.4.2.1 Normal Environment

- A. 122°F maximum.
- B. Relative humidity of 95% at 60°F to 80°F.
- C. Fixed moisture content equivalent to 95% relative humidity at 80°F, up to 122°F.

5.2.2.4.2.2 Main Steam Line Break (One Occurrence)

350°F maximum Superheated steam/air mixture for 12 minutes followed by saturated steam/air mixture.

5.2.2.4.3 Main Steam Safety Valves

The main steam safety valves are direct acting, spring loaded, carbon steel valves. The valves are mounted on each of the main steam lines upstream of the steam line isolation valves, and outside containment. A schematic drawing of the main steam safety valves is given in Figure 5.4.13-2. The valve parameters are given in Table 5.4.13-2. For a description of overpressure protection equipment and components for the main steam system refer to Section 10.3.2.

5.2.2.4.3.1 Main Steam Safety Valve Operation

The operation of these valves is similar to the primary safety valves, Section 5.2.2.4.2.

5.2.2.4.3.2 Transients

The main steam safety valves will be designed to withstand the following transients without failure or malfunction.

- A. 565°F to 75°F in 60 seconds for 8 cycles (loss of secondary pressure).
- B. Pressure changes from 0 psig to 1375 psig, at a temperature range of between 100°F to 200°F for 300 cycles (secondary side leak test).

- C. $\pm 10^{\circ}\text{F}$ step change from 553°F , 1.5×10^6 cycles (normal plant variations).
- D. 75°F to 565°F and return to 75°F at a rate of 100°F/hr with pressures at saturation levels for 750 cycles (plant heatup and cool down).

Note: Heat up and cool down are separate transients, each beginning at steady state conditions.

- E. Pressurize to 1.5 times set pressure at $100^{\circ}\text{F} - 200^{\circ}\text{F}$ for 15 cycles plus number of hydros conducted prior to valve shipment (hydrostatic test).
- F. 720 opening and closing cycles to full stem movement (turbine trip).

5.2.2.4.3.3 Environment

The main steam safety valves are designed to operate in the environmental conditions outlined below.

5.2.2.4.3.3.1 Normal Environment

- A. 104°F maximum.
- B. Relative humidity 95% at 60°F to 80°F .
- C. Fixed moisture content equivalent to 95% relative humidity at 80°F , up to 104°F .

5.2.2.4.3.3.2 Main Steam Line Break (One Occurrence)

- A. 330°F maximum for 3 minutes.
- B. Relative humidity of 100%.

5.2.2.4.4 Safety Injection System Relief Valves SI-169 and SI-469

These relief valves are direct acting, spring loaded, stainless steel valves with enclosed bonnets. The design parameters of these valves are:

set pressure	2485 psig
rated flow	15 gpm
water chemistry	0 - 4 weight percent boric acid

throat area .023 in²

design temperature 650°F

5.2.2.4.4.1 Valve Operation

As the set pressure is reached, the disc raises off the nozzle seat. This lift continues until the valve is fully open at 10 percent accumulation. The lift decreases as pressure drops until the seat and disc contacts and seals closed. The valve is fully closed at a maximum of 10% below set pressure (10% blowdown).

5.2.2.4.4.2 Transients

These relief valves will be designed to withstand the following transients without failure or malfunction.

- A. 60°F to 400°F in 5 seconds, 400°F to 60°F in 15 minutes for 83 cycles.
- B. 60°F to 350°F in 15 minutes, 350°F to 60°F in 2.9 hours for 750 cycles.
- C. 120°F to 60°F in 5 seconds, 60°F to 120°F in 15 minutes for 990 cycles.

5.2.2.4.4.3 Environment

These relief valves are designed to operate in the following environmental conditions.

- A. 122°F maximum.
- B. 95% relative humidity at 60°F to 80°F.
- C. Fixed moisture content equivalent to 95% RH at temperatures above 80°F.

5.2.2.4.4.4 Material Specifications

Material specifications for the primary safety valves are given in Table 5.4.13-1.

Material specifications for the main steam safety valves are given in Table 5.4.13-2.

Typical materials used for these relief valves are:

Body	ASME SA 351 GR. CF 8M
Disc	Stellite No. 6B (or equivalent)
Nozzle	ASME SA 479 Type 316 with Stellite Seat
Inlet Stud	ASME SA 193 GR. B6

5.2.2.5 Mounting of Pressure-Relief Devices

5.2.2.5.1 Location of Pressure Relief Devices

The location of the pressurizer safety valves is shown in Figure 5.1.2-3. They discharge to the In-Containment Refueling Water Storage Tank (IRWST). Figure 10.1-2 shows the location of the main steam safety valves.

5.2.2.5.2 Design Bases for Mounting of RCPB Pressure Relief Devices

The RCPB pressure relief devices are mounted and installed as follows:

- A. Each discharge pipe is supported to transfer transient discharge load to the adjacent structure, in order to stay within the allowable pressurizer nozzle loads.
- B. There is no rigid restraint in the vertical directions. The piping system moves up or down with the pressurizer during heat up or cool down. Weight of the piping system is carried by spring supports.
- C. Each safety valve discharge piping is supported so that forces and moments during operating plant conditions (normal, upset, emergency and faulted) will not jeopardize the integrity of the valves, the inlet lines to the valves, and the nozzles on the pressurizer.
- D. Pipe breaks are postulated in the high energy piping between the pressurizer nozzle and the valve flange joint. Pipe breaks are not postulated in the safety valve discharge piping (i.e., piping from the valve discharge flange to the IRWST). This piping is not considered to be high energy piping because it operates less than 2% of normal plant

operating time. Discharge piping is classified as moderate energy piping. Through-wall cracks are not postulated since this piping is adequately separated from other safety-related active systems subject to impairment by such failures.

Dynamic analysis for seismic and valve discharge loadings are performed to verify the design of the support configuration.

5.2.2.5.3 Pressurizer Safety Valve Analysis - Loading Criteria and Methods of Analysis

5.2.2.5.3.1 Loading Conditions

Loading combinations for upstream Class 1 piping is shown in Table 3.9-11. Loading for downstream Class 2 piping is shown in Table 3.9-13.

5.2.2.5.3.2 Pressure

Pressure loading is in accordance with the ASME B&PV Code. The design pressure is 2,485 psig for Class 1 piping and 700 psig for Class 2 piping.

5.2.2.5.3.3 Weight

A weight analysis is performed on the complete piping system. Hydrotest conditions are also considered.

5.2.2.5.3.4 Seismic

The structural response due to the loadings is analyzed as discussed in Section 3.7.2.

5.2.2.5.3.5 Thrust

5.2.2.5.3.5.1 Hydraulic Thrust Forces

The pressurizer safety valve and connecting piping are subjected to hydraulic thrust forces which result from changes in direction of the fluid, change in flow rate, and from differing fluid states. The discharge piping is shared with the Safety Depressurization System which, at different times, may be flowing vapor, liquid, or mixed vapor and liquid. The piping and valve supporting system is designed to withstand these thrust loads. The safety valve inlet piping is not equipped with a loop seal.

5.2.2.5.3.5.2 Structural Analysis of Thrust

The dynamic structural solution for the thrust loading is obtained using a modified-predictor-corrector-integration technique and normal mode theory.

A subprogram is used which predicts the transient flows resulting from actuation of a safety valve. It also predicts the resulting piping loads used as dynamic forcing functions for structural design of discharge piping and its supporting components. The computation is based on finite difference solutions by the method of effluent characteristics. The computed transient pressure, velocity, and density are then used to calculate loads on bends and pipe runs.

5.2.2.5.4 Main Steam Safety Valve Analysis**5.2.2.5.4.1 Valve Forces and Reactor Load Paths**

Two conditions were considered in the stress analysis of the safety valve installations, namely, the dynamic effects of the safety valve opening and the steady-state flow condition reached after the valve has opened and is exhausting into the stack. During the valve opening time, dynamic forces due to the hydraulic transients in the valve are compensated by the reaction forces of the header supports via the header. When the valve has opened, and the steam is exhausting into the stack, the discharge thrust is compensated by the reaction forces of the stack support structure.

5.2.2.5.4.2 Loading Conditions

See Table 3.9-13 for loading combinations for piping upstream of the main steam safety valves.

5.2.2.6 Applicable Codes and Classification

The applicable codes and classifications for the overpressurization protection system are contained in Table 3.2-1. The applicable codes and classification for the secondary safety valves are identified in Section 10.3.2.

5.2.2.7 Process Instrumentation

Process instrumentation for the overpressurization protection equipment that is associated with the Reactor Coolant System is shown in Figure 5.1.2-1 and described in Chapter 7. Instrumentation associated with pressurizer relief discharge is described in Section 5.4.11.

5.2.2.8 System Reliability

Reliability of the main steam system reliefs is discussed in the interface Chapter 10. The primary safety valves are passive, spring actuated mechanisms, and cannot fail closed if setpoint pressure is exceeded. The operational reliability of the primary safety valves is assured by:

- A. Stringent compliance with ASME III and XI Code for safety valves.
- B. Conservative design criteria.
- C. Selection of a vendor with proven experience and expertise.
- D. Accounting for thermal cycling during valve operation.
- E. Technical Specifications.

5.2.2.9 Testing and Inspection

Testing and inspection of the primary and secondary valves is governed by ASME Section XI.

5.2.2.10 Overpressure Protection During Low Temperature Conditions

Overpressure protection of the RCS during low-temperature conditions is provided by the relief valves located in the Shutdown Cooling System (SCS) suction lines. Section 5.4.7 provides a description of the SCS. The SCS is schematically shown on the SCS Flow Diagram (Figure 5.4.7-3, Sheets 1 and 2), the RCS P&ID (Figure 5.1.2-1) and on the Safety Injection System (SIS) P&ID (Figures 6.3.2-1A and 6.3.2-1B). The electrical schematic for the SCS isolation valves is provided in the SIS P&ID (Figure 6.3.2-1B). The SCS relief valves are shown on Figure 6.3.2-1B and described in Section 5.4.7.2.2.

Alignment of the SCS relief valve to the RCS is specified by plant procedures to ensure RCS overpressure protection for all temperatures below the LTOP temperature, T_{LTOP} . The P-T Limit curves are shown in Figures 5.3-5a and 5.3-5b. For temperatures above the LTOP temperature, overpressure protection is provided by the pressurizer safety valves described in Section 5.2.2.4 or administrative procedures.

5.2.2.10.1 Design Criteria

A discussion follows of the criteria considered in the design of the overpressure mitigating system to provide low temperature overpressure protection (LTOP) for the RCS.

5.2.2.10.1.1 Credit for Operator Action

No credit is taken for operator action for 10 minutes after the operator is made aware that a transient is in progress. It is assumed that the operator is aware of the transient when it commences. | D

5.2.2.10.1.2 Single Failure

In the LTOP mode, each SCS relief valve is designed to protect the reactor vessel given a single failure in addition to the failure that initiated the pressure transient. The event initiating the pressure transient is considered to result from either an operator error or equipment malfunction. The SCS relief valve system is independent of a loss of offsite power. Each SCS relief valve is a self actuating spring-loaded liquid relief valve which does not require control circuitry. The valve opens when the RCS pressure exceeds its setpoint.

The redundant SCS suction line trains between the RCS and SCS relief valves meet the single failure criteria as described in Section 5.4.7.1.2 and Table 5.4.7-3. No single failure of an isolation valve or its associated interlock will prevent one relief valve from performing its intended function.

5.2.2.10.1.3 Testability

Periodic testing of the SCS isolation valves is defined in the Technical Specification, Section 16.3/4.5.2.

5.2.2.10.1.4 Seismic Design and IEEE Standard 279 Criteria | B

The SCS suction line relief valves, isolation valves, associated interlocks, and instrumentation are designed to Seismic Category I requirements as discussed in Sections 3.2.1, 5.4.7.2.4 and Table 3.2-1. The interlocks and instrumentation associated with the SCS suction isolation valves satisfy the appropriate portions of IEEE Standards 279, 308 and 603 criteria as discussed in Sections 5.4.7, 7.6.1, 7.6.1.1.1, 7.6.2.1.1 and 7.6.2.2.1 and Table 7.6-1. | D
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5.2.2.10.2 Design and Analysis

In demonstrating that the SCS relief valves meet the criteria listed in Section 5.2.2.10.1, the following additional information is provided.

5.2.2.10.2.1 Limiting Transients

Transients during the low temperature operating mode are more severe when the RCS is operated in the water-solid condition. Addition of mass or energy to an isolated water-solid system produces increased system pressure. The severity of the pressure transients depends upon the rate and total quantity of mass or energy addition. The choice of the limiting LTOP transients is based on evaluations of potential transients for System 80 plants and their applicability to the System 80+ plant. The most limiting transients initiated by a single operator error or equipment failure are:

- A. An inadvertent safety injection actuation (mass addition).
- B. A reactor coolant pump start when a positive steam generator to reactor vessel ΔT exists (energy addition).

The most limiting transients are determined by conservative analyses which maximize mass and energy additions to the RCS. In addition, the RCS is assumed to be in a water-solid condition at the time of the transient; such a condition has been noticed to exist infrequently during plant operation since the operator is instructed to avoid water-solid conditions whenever possible.

Table 5.2-4 shows the results of the inadvertent safety injection actuation transient analysis for a water-solid RCS, when the RCS is in the LTOP mode. The mass addition due to the simultaneous operation of four safety injection pumps and one charging pump was considered, along with the simultaneous addition of energy from decay heat and the pressurizer heaters.

Figure 5.2-1 shows the result of the transient analysis of reactor coolant pump start when a steam generator to reactor vessel ΔT of 100°F exists. This ΔT is the maximum allowed by technical specification during the LTOP mode. In addition to considering the energy addition to the RCS from the steam generator secondary side, energy addition from decay heat, the reactor coolant pump and all pressurizer heaters were also included. In this analysis the steam generators were assumed to be filled to the zero power, normal water level. For conservatism, the secondary water, both around and above the U-tubes, was assumed to be thermally mixed in order to maximize the energy input to the primary side. This assumption is conservative since as a result of the temperature distribution within the steam generator during the transient, the water inventory above the tubes is practically isolated thermally from the heat transfer region. Therefore the heat transfer rate, and thus the primary side pressure, is not sensitive to the secondary side water level as long as the tubes are covered.

On the basis of experience, the ΔT value of 100°F used in the analysis is much larger than any ΔT that might be expected during plant operation. This maximum allowable ΔT of 100°F will prevent pressurizer pressure from exceeding the minimum P-T limit allowed for the lowest system temperature during the LTOP mode of operation. (See Figures 5.3-5a and 5.3-5b). During RCS cooldown using the Shutdown Cooling System, coolant circulating with the reactor coolant pumps operating serves to cool the steam generator to keep the temperature difference between the reactor vessel and the steam generator minimal. Procedures for System 80+ have directed the operator to maintain the ΔT below approximately 20°F.

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LTOP transients have not been analyzed for the simultaneous startup of more than one reactor coolant pump (RCP). Such operation is procedurally precluded since the operator starts only one RCP at a time and a second RCP is not started until system pressure is stabilized. Additionally, there is an LTOP transient alarm that should indicate that a pressure transient is occurring. Accordingly, the second RCP would not be started.

The operator cannot start an RCP if the ΔT exceeds 100°F. However, as mentioned above, administrative procedures for System 80 have ensured that the ΔT is maintained below approximately 20°F. With similar administrative controls on System 80+, ΔT margins will be even greater than for System 80.

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The results of the analyses provided in Figure 5.2-1 and Table 5.2-4 show that the use of either SCS relief valve will provide sufficient pressure relief capacity to mitigate the most limiting LTOP events identified above.

5.2.2.10.2.2 Provision for Overpressure Protection

During heatup, the RCS pressure is maintained below the LTOP pressure until the RCS cold-leg temperature exceeds the LTOP disable temperature. During cooldown, the RCS pressure is maintained below the LTOP pressure once the RCS cold-leg temperature reaches the LTOP enable temperature.

An LTOP enable temperature is defined in Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," to Standard Review Plan Section 5.2.2, "Overpressure Protection," issued November 1988 as Revision 2. The definition is based on measuring the degree of protection provided by the low temperature overpressure protection system (LTOP System) against violations of the P-T Limits in terms of the RT_{NDT} of the reactor vessel beltline material at either the 1/4t or 3/4t location,

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depending upon which PT Limit curve is most limiting. The LTOP enable temperature in Branch Technical Position RSB 5-2 is defined as "the water temperature corresponding to a metal temperature of at least $RT_{NDT} + 90^{\circ}\text{F}$ at the reactor vessel beltline location (either 1/4t or 3/4t) that is controlling in the Appendix G [to Section III of the ASME Code] limit calculations."

Figures 5.3-5a and 5.3-5b represent the P-T Limits at the end-of-(plant) life of 60 years. At this time in plant life, the calculated maximum RT_{NDT} for the reactor vessel beltline is 79°F for the 1/4t location, and 69°F for the 3/4t location. Given the implicit assumptions in the P-T Limit analysis described in Section 5.3.2.1.1.D.1, the 3/4t location is the controlling beltline P-T Limit for heatup, and the 1/4t location is controlling for cooldown. Therefore, in accordance with the definition of the LTOP enable temperature given in Branch Technical Position RSB 5-2, T_{LTOP} for heatup is $79^{\circ}\text{F} + 90^{\circ}\text{F} = 169^{\circ}\text{F}$, and T_{LTOP} for cooldown is $69^{\circ}\text{F} + 90^{\circ}\text{F} = 159^{\circ}\text{F}$.

Figure 5.3-5a shows that, for heatup, a temperature of 309°F is required before the pressurizer safety valves can be used for overpressure protection without violating any P-T Limits. Figure 5.3-5b shows that, for cooldown, the safety valves cannot be used below a temperature of 225°F . For the temperature interval 140°F above T_{LTOP} for heatup, as defined by Branch Technical Position RSB 5-2, and for the 66°F above T_{LTOP} for cooldown, administrative controls would be required to protect some P-T Limits, particularly the design limit of 100°F/hr . Rather than relying on administrative controls during these temperature intervals, the T_{LTOP} enable and disable temperatures for System 80+ are defined by the intersection of the controlling P-T Limit (100°F/hr) curve and the pressurizer safety valve setpoint.

During heatup, if the SCS suction isolation valves (SI-651 and -653 or SI-654 and -652) are open and RCS pressure exceeds the LTOP pressure, an alarm will notify the operator that a pressurization transient is occurring during low temperature conditions. Either SCS relief valve will terminate inadvertent pressure transients occurring while RCS temperature is below the LTOP disable temperature. For temperatures above the LTOP disable temperature, that corresponding to the intersection of the design P-T limit curve for heatup with the pressurizer safety valve setpoint, overpressure protection is provided by the pressurizer safety valves.

During cooldown, whenever the RCS cold leg temperature is below the LTOP enable temperature, that corresponding to the intersection of the design P-T Limit curve for cooldown with pressurizer safety valve setpoint, the SCS relief valves provide the necessary overpressure protection. If the SCS is not aligned to the RCS before the cold-leg temperature is decreased below the LTOP enable temperature, an alarm will notify the operator to open the SCS suction isolation valves. However, the SCS cannot be aligned to the RCS until the RCS pressure is below the LTOP enable pressure.

The LTOP conditions described above are within the SCS operating range. Technical Specification Section 16.3/4.4.8.3 requires the SCS suction line isolation valves to be open when operating in the LTOP mode. Also, this Technical Specification ensures that appropriate action is taken if one or more SCS relief valves are out of service during the LTOP mode of operation.

Either SCS relief valve will provide sufficient relief capacity to prevent any pressure transient from exceeding the isolation interlock setpoint (see Figure 5.2-1 and Table 5.2-4).

5.2.2.10.2.3 Equipment Parameters

The SCS relief valves are spring-loaded liquid relief valves with sufficient capacity to mitigate the most limiting over-pressurization event. Pertinent valve parameters are as follows:

Parameter

Nominal Setpoint 550 psia*

Accumulation 10%

Capacity 4000 (@10% acc) gpm

Since each SCS relief valve is a self actuating spring-loaded liquid relief valve, control circuitry is not required. The valve will open when RCS pressure exceeds its setpoint.

The SCS relief valves are sized, based on an inadvertent safety injection actuation signal (SIAS) with full pressurizer heaters operating from a water-solid condition. The analysis assumes simultaneous operation of four SIS pumps and one charging pump with letdown isolated. The resulting flow capacity requirement for water is 3934 gpm. The analysis in Section 5.2.2.10.2.1 assumed that either SCS relief valve relieved water at this rate.

* Pressure measured at the valve inlet.

The design relief capacity of each of two SCS relief valves (shown in P&ID Figure 6.3.2-1B) as supplied by the valve manufacturer meets the minimum required relief capacity of 4000 gpm which contains sufficient margin in relieving capacity for even the worst transient. The SCS relief valves are Safety Class 2, designed to Section III of the ASME Code.

5.2.2.10.2.4 Administrative Controls

Administrative controls necessary to implement the LTOP provisions are limited to those controls necessary to open the SCS isolation valves.

During cooldown, when the temperature of the RCS is above that corresponding to the intersection of the controlling P-T Limit and the pressurizer safety valve setpoint, overpressure protection is provided by the pressurizer safety valves, and no administrative procedural controls are necessary. Before entering the low temperature region for which LTOP is necessary, RCS pressure is decreased to below the maximum pressure required for LTOP. The LTOP pressure is less than the maximum pressure allowable for SCS operation. Once the SCS is aligned, no further specific administrative procedural controls are needed to ensure proper overpressure protection. The SCS will remain aligned whenever the RCS is at low temperatures and the reactor vessel head is secured or until an adequate vent has been established. As designated in Table 7.5-2, indication of SCS isolation valve position is provided.

During heatup, the SCS isolation valves remain open at least until the LTOP enable temperature. Once the RCS temperature has reached that temperature corresponding to the intersection of the controlling P-T Limit and the pressurizer safety valve setpoint, overpressure protection is provided by the pressurizer safety valves. The SCS can be isolated and no further administrative procedural controls are necessary.

5.2.2.11 Pressurized Thermal Shock

The System 80+ reactor vessel meets the requirements of 10 CFR 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events." The calculated RT_{PTS} is 109°F which satisfies the screening criteria in 10 CFR 50.61(b)(2).

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specification

A list of specifications for the principal ferritic materials, austenitic stainless steels, bolting and weld materials, which are part of the reactor coolant pressure boundary is given in Table 5.2-2.

Studies have shown that the irradiation induced mechanical property changes of SA-533B and SA-508 materials can depend significantly upon the amount of residual elements present in the compositions, namely; copper, nickel, phosphorous, and vanadium. It has also been found that residual sulfur affects the initial toughness of SA-533B and SA-508 materials. Specific controls are placed on the residual chemistry of reactor vessel materials and the as-deposited welds used to join these materials to limit the maximum predicted increase in the reference temperature (RT_{NDT}, which is discussed in Sections 5.3.1.6 and 5.3.2.1.1) and to limit the extent of the reactor vessel beltline. The beltline is defined by Appendix G of 10 CFR 50.

Materials used in the reactor vessel beltline and the as-deposited welds contain no greater than the following low percentages of residual elements:

Copper (in welds)	0.03		
Copper (in forgings)	0.06	Phosphorous	0.015
Nickel (in forgings)	1.00	Sulfur	0.015
Nickel (in welds)	0.10	Vanadium	0.030

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Reactor Coolant Chemistry

Controlled water chemistry is maintained within the RCS. Control of the reactor coolant chemistry is the function of the CVCS which is described in Section 9.3.4. Water chemistry limits applicable to the RCS are given in Section 9.3.4.

5.2.3.2.2 Materials of Construction Compatibility with Reactor Coolant

The materials of construction used in the RCPB which are in contact with reactor coolant are designated by an "a" in Table 5.2-2. These materials have been selected to minimize corrosion and have previously demonstrated satisfactory performance in other existing operating reactor plants.

Metallic materials in contact with reactor coolant shall be restricted in cobalt content to as low a level as practical for all stainless steel or nickel base alloy components with a large wetted surface area. Cobalt based alloys shall be avoided except in cases where no proven alternative exists.

5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

The possibility of leakage of reactor coolant onto the reactor vessel head causing corrosion of the pressure boundary has been investigated by C-E.

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Tests have shown that reactor coolant system leakage onto surfaces of the reactor coolant pressure boundary will not affect the integrity of the pressure boundary.

The insulation supplied by C-E is of the stainless steel reflective type, which minimizes insulation contamination in the event of a chemical solution spillage. In local areas around stainless steel and the nickel based alloy nozzles in the reactor vessel head, small sections of non-metallic insulation are used. However, the quantity of leachable halogens will be limited in accordance with Regulatory Guide 1.36.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

5.2.3.3.1.1 NSSS Components

Fracture toughness requirements for Reactor Coolant Pressure Boundary components are established in accordance with the ASME Boiler and Pressure Vessel Code, Section III. Fracture toughness testing of base, weld and heat affected zone materials will be conducted in accordance with the ASME Code. Data from these tests will be available after the required testing has been performed and may be examined upon request at the appropriate manufacturing facility.

C-E complies with 10 CFR Part 50 Appendix G, "Fracture Toughness Requirements" as enacted May 1983.

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Consideration is given to the effects of irradiation on material toughness properties in the core beltline region of the reactor vessel to assure adequate fracture toughness for the service lifetime of the vessel. Refer to Section 5.3.1.6 for discussion concerning prediction of irradiation effects and the material surveillance program. In addition, C-E complies with the guidance of Regulatory Guide 1.2, "Thermal Shock to Reactor Pressure Vessels."

Testing and measuring equipment for fracture toughness tests for the reactor vessel, steam generators, pressurizer, piping and reactor coolant pumps are calibrated in accordance with Subarticle NB2360 of the ASME Code, Section III.

5.2.3.3.2 Control of Welding**5.2.3.3.2.1 Avoidance of Cold Cracking**

C-E complies with the recommendations of Regulatory Guide 1.50, Control of Preheat Temperature for Welding of Low Alloy Steel, May 1973, as discussed below.

Paragraph C.1.b implies that the qualification materials are an infinite heat sink that would instantaneously dissipate the heat input from the welding process. The qualification procedure consists of starting the welding at the minimum preheat temperature. Welding is continued until the maximum interpass temperature is reached. At this time, the test material is permitted to cool to the minimum preheat temperature and the welding is restarted. Preheat temperatures utilized for low alloy steel are in accordance with Section III of the ASME Code. The maximum interpass temperature utilized is 500°F.

The paragraph C.2 recommendation is considered an unnecessary extension of procedures which apply to low-alloy steel welds, meeting ASME Code Sections III and IX requirements. The recommendations of Regulatory Guide 1.50 are met by complying with paragraph C.4. The soundness of all welds is verified by ASME Code acceptable examination procedures.

With regard to Regulatory Guide 1.43, major components are fabricated with corrosion resistant cladding on internal surfaces exposed to reactor coolant. The major portion of the material protected by cladding from exposure to reactor coolant is SA-533B Class 1 or SA-508, Class 2 or 3. Cladding of SA-508, Class 2 forging material is performed using low-heat-input welding processes controlled to minimize heating of the base metal. Low-heat-input welding processes are not known to induce underclad cracking. D

5.2.3.3.2.2 Regulatory Guide 1.34

Regulatory Guide 1.34 recommends controls to be applied during welding using the electroslag process. The electroslag process is not used in the fabrication of any RCPB components. Therefore, the recommendations of this guide are not applicable.

5.2.3.3.2.3 Regulatory Guide 1.71

C-E complies with the recommendations of Regulatory Guide 1.71 except for the differences indicated below.

Performance qualifications for personnel welding under conditions of limited accessibility are conducted and maintained in accordance with the requirements of ASME B&PV Code Sections III and IX. A requalification is required when (1) any of the essential variables of Section IX is changed, or (2) when authorized personnel have reason to question the ability of the welder to satisfactorily comply with the applicable requirements. Production welding is monitored for compliance with the procedure parameters, and welding qualifications are certified in accordance with Sections III and IX. Further assurance of acceptable welds of limited accessibility is afforded by the welding supervisor assigning only the most highly skilled personnel to these tasks. Finally, weld quality, regardless of accessibility, is verified by the performance of the required non-destructive examinations.

5.2.3.3.3 Non-Destructive Examination of Tubular Products

The non-destructive examination requirements imposed by C-E for tubular products are those specified by Section III of the ASME code. | D

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

5.2.3.4.1 Avoidance of Stress Corrosion Cracking

5.2.3.4.1.1 Avoidance of Sensitization

5.2.3.4.1.1.1 NSSS Components

Fabrication of RCPB components is consistent with the recommendations of Regulatory Guide 1.44 as described in items A through D, below, except for the criterion used to demonstrate freedom from sensitization. The ASTM A 708 Strauss Test is used in lieu of the ASTM A 262 Practice E, Modified Strauss Test, to demonstrate freedom from sensitization in fabricated, unstabilized, stainless steel.

A. Solution Heat Treatment Requirements

All raw austenitic stainless steel material, both wrought and cast, used in the fabrication of the major NSSS components in the RCPB, is supplied in the annealed condition as specified by the pertinent ASTM or ASME Code: 1900-2050°F for 1/2 to 1 hour per inch of thickness and water quenched to below 700°F. The time at temperature is determined by the size and type of component.

Solution heat treatment is not performed on completed or partially-fabricated components. Rather, the extent of chromium carbide precipitation is controlled during all stages of fabrication as described below.

B. Material Inspection Program

Extensive testing on stainless steel mockups, fabricated using production techniques, has been conducted to determine the effect of various welding procedures on the susceptibility of unstabilized 300 series stainless steels to sensitization-induced intergranular corrosion. Only those procedures and/or practices demonstrated not to produce a sensitized structure are used in the fabrication of RCPB components. The ASTM standard A 708 (Strauss Test) is the criterion used to determine susceptibility to intergranular corrosion. This test has shown excellent correlation with a form of localized corrosion peculiar to sensitized stainless steels. As such, ASTM A 708 is utilized as a go/no-go standard for acceptability.

As a result of the above tests, a relationship was established between the carbon content of 304 stainless steel and weld heat input. This relationship is used to avoid weld heat-affected-zone sensitization as described below.

C. Unstabilized Austenitic Stainless Steel

The unstabilized grades of austenitic stainless steels with carbon content of more than 0.03% used for components of the RCPB are 304 and 316. These materials are furnished in the solution annealed condition. Exposure of completed or partially-fabricated components to temperatures ranging from 800°F to 1500°F is prohibited.

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

CF8M	Cast stainless steel (delta ferrite
CF8	5FN to 33FN)
308, 309	Singly and combined stainless steel
312, 316	weld filler metals (delta ferrite
	controlled to 5FN-20FN deposited)

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

D. Avoidance of Sensitization

Exposure of unstabilized austenitic Type 300 series stainless steels to temperatures ranging from 800 to 1500°F will result in carbide precipitation. The degree of carbide precipitation, or sensitization, depends on the temperature, the time at that temperature, and the carbon content. Severe sensitization is defined as a continuous grain boundary chromium-iron carbide network. This condition induces susceptibility to intergranular corrosion in oxygenated aqueous environments, as well as those containing fluorides. Such a metallurgical structure will rapidly fail the Strauss Test ASTM A 708. Discontinuous precipitates (i.e., an intermittent grain boundary carbide network) are not susceptible to intergranular corrosion in a PWR environment.

Weld heat affected zone sensitized austenitic stainless steels (which will fail the Strauss Test, ASTM A 708) are avoided by careful control of:

- Weld heat input to less than 60 kJ/in
- Interpass temperature to 350°F maximum
- Carbon content

Homogeneous or localized heat treatment in the temperature range 800-1500°F is prohibited for unstabilized austenitic stainless steel with a carbon content greater than 0.03% used in components of the RCPB. When stainless steel safe ends are required on component nozzles, fabrication techniques and sequencing require that the stainless steel piece be welded to the component after final stress relief. This is accomplished by welding an Inconel overlay on the end of the nozzle. Following final stress relief of the component, the stainless steel safe end is welded to the Inconel overlay, using Inconel weld filler metal.

**5.2.3.4.1.2 Avoidance of Contamination Causing Stress
Corrosion Cracking****5.2.3.4.1.2.1 NSSS Components**

Specific requirements for cleanliness and contamination protection are included in the equipment specifications for components fabricated with austenitic stainless steel. The provisions described below indicate the type of procedures utilized for NSSS components to provide contamination control during fabrication, shipment, and storage as required by Regulatory Guide 1.37.

Contamination of austenitic stainless steels of the 300 type by compounds which can alter the physical or metallurgical structure and/or properties of the material is avoided during all stages of fabrication. Painting of 300 series stainless steels is prohibited. Grinding is accomplished with resin or rubber-bounded aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic alloys. Outside storage of partially-fabricated components is avoided and in most cases prohibited. Exceptions are made for certain components provided they are dry, completely covered with a waterproof material, and kept above ground.

Internal surfaces of completed components are cleaned to produce an item which is clean to the extent that grit, scale, corrosion products, grease oil, wax, gum, adhered or embedded dust or extraneous materials are not visible to the unaided eye. Cleaning is effected by either solvents (acetone or isopropyl alcohol) or inhibited water (hydrazine). Water will conform to the following requirements:

Halides

Chloride (ppm)	0.60
Fluoride (ppm)	0.40
Conductivity (μ mhos/cm)	5.0
pH	6.0-8.0
Visual clarity	No turbidity, oil, or sediment

To prevent halide-induced intergranular corrosion which could occur in aqueous environment with significant quantities of dissolved oxygen, flushing water is inhibited via additions of hydrazine. Results of tests have proven these inhibitors to be completely effective. Operational chemistry specifications restrict concentrations of halide and oxygen, both prerequisites of intergranular attacks (refer to Section 9.3.4).

**5.2.3.4.1.3 Characteristics and Mechanical Properties of
Cold-Worked Austenitic Stainless Steels for
RCPB Components**

Cold-worked austenitic stainless steel is not utilized for components of the RCPB.

5.2.3.4.2 Control of Welding

5.2.3.4.2.1 Avoidance of Hot Cracking

A. NSSS Components

1. Regulatory Guide 1.31

In order to preclude microfissuring in austenitic stainless steel welds, RCPB components are consistent with the recommendations of Regulatory Guide 1.31 as follows:

The delta ferrite content of each lot and/or heat of weld filler metal used for welding of austenitic stainless steel code components shall be determined for each process to be used in production. Delta ferrite determinations for consumable inserts, electrodes, rod or wire filler metal used with the gas tungsten arc welding process, and deposits made with the plasma arc welding process may be determined by either of the alternative methods of magnetic measurement or chemical analysis described in Section III of the ASME Code.

Delta ferrite verification should be made for all other processes by tests using the magnetic measurement method on undiluted weld deposits described by Section III of the ASME Code. The average ferrite content shall meet the acceptance limits of 5FN to 20FN for weld rod or filler metal.

2. Regulatory Guide 1.34

Regulatory Guide 1.34 is discussed in Section 5.2.3.3.2.2.

3. Regulatory Guide 1.71

Regulatory Guide 1.71 is discussed in Section 5.2.3.3.2.3.

5.2.4 IN-SERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY

An In-service Inspection (ISI) program will be provided for the examination of the Reactor Coolant Pressure Boundary (RCPB) components and supports defined as Code Class 1. The program will reflect the principles and intent embodied in the ASME Boiler and Pressure Vessel Code, Section XI. Specific Code Editions and addenda required by 10 CFR 50.55(a) are referenced in the Pressure Inspection (PSI) and ISI programs. The purpose of the in-service inspection program is to periodically monitor the systems or components requiring In-service Inspection in order to identify and to repair those indications which do not meet acceptance standards.

The ISI Program consists of three subprograms as follows:

- A. The Component Inspection Program, which includes piping system welds, hangers, supports, internal inspection of pump and valve bodies and bolting.
- B. The Pump and Valve In-service Test (IST) Program, which requires operability testing of selected pumps and valves, and
- C. The Hydrostatic Test Program, which requires flow testing and hydrostatic testing of systems.

5.2.4.1 System Boundary Subject to Inspection

The reactor pressure vessel, pressurizer, primary side of the steam generator and associated piping, pumps, valves, bolting and component supports are subjected to inspection.

5.2.4.2 Arrangement of Systems and Components to Provide Accessibility

The layout and arrangement of the plant provides adequate working space and access for inspection of specific areas of Code Class 1 components of the RCPB in accordance with IWA-1500. The Code Class 1 components of RCPB subject to inspection are those components defined by ASME Code Section III, Division 1.

Listed below are the provisions for access for examination of the RCPB:

A. Reactor Vessel and Closure Head

1. From Inside the Vessel:

All internals of the reactor vessel (which is an open structure offering insignificant impediment to access) are removable making the entire inner surface of the vessel including the beltline welds, and the weld zones of the internal load-carrying structure attachments, available for the required surface and volumetric inspections. Provisions are made in the plant design to allow for the removal and storage of all vessel internals (except the flow skirt) during in-service inspection. Ultrasonic testing of the reactor vessel welds will be in accordance with Regulatory Guide 1.150, Rev. 1.

2. From Outside the Vessel:

The bottom head of the reactor pressure vessel is manually examined from the outside surface, therefore, an access tunnel is provided to allow personnel into the area below the bottom head. Insulation is provided by removable panels over the bottom head weld seams. For interim inspections of the vessel nozzle-to-shell welds and inner radii of the outlet nozzles are accessible from inside the pressure vessel without removal of the vessel internals by using remote automated equipment. Remote inspection devices to be used in periodic ISI will be used for Preservice Baseline Inspection Program to demonstrate feasibility.

3. Closure Head

The closure head is available for inspection whenever it is removed, and its removal makes available the vessel closure flange, the flange-to-shell weld, closure stud holes and ligaments, and the closure studs and nuts. Each control element drive mechanism is removable as a unit through a closure at the top of its housing. Since many of the reactor vessel closure head examinations are done from the underside of the head, the head laydown area provides access for a man to work under this component using manual techniques.

B. Reactor Coolant Piping

Biological shielding around the reactor coolant piping in the area of the reactor vessel is designed to afford access to the circumferential and longitudinal welds, as well as the transition piece-to-nozzle welds. The volumetric examinations are performed using ultrasonic techniques.

All reactor coolant piping, as well as major components, excluding the reactor vessel, are provided with removable insulation in the areas of all welds and adjacent base metal requiring examination.

The primary coolant piping has access at each side of the welds to facilitate manual examination of the welds.

C. Steam Generators

Sufficient space is provided within the stay cylinder to permit inspection of the welds. A 12 in. diameter access opening in the steam generator support skirt is provided. The insulation in this area is removable to the extent of the full size of the access opening.

The steam generators have removable insulation and access at welds requiring examination. Manways are provided for those inspections which must be made internally on the steam generator.

D. Pressurizer

The pressurizer has sufficient clearance around the shell weld seams for manual ultrasonic examination of these welds. The insulation is removable at each weld and access is provided for ultrasonic and visual examinations in the area of the bottom head and its nozzle penetrations of the pressurizer. A manway is provided for those inspections which must be made internally on the pressurizer.

E. Reactor Coolant Pumps

The reactor coolant pumps require inside visual examination. To allow this, provisions are made in the design for removal and storage of the RCP motors and the disassembly of the reactor coolant pumps. Access is provided to the motor fly wheels for ultrasonic examination.

F. Other Components

All other components, including portions of the steam generators, reactor coolant pumps, pressurizer and primary piping, are accessible for manual examination from the outside surface.

General provisions are made for removable insulation, removable shielding, installation of handling machinery, adequate personnel and equipment access space and lay down space for all temporarily removed or serviced components. Storage space for the removable insulation panels is also provided. Working room for a man is provided adjacent to each weld in order to examine all piping system welds manually.

5.2.4.3 Examination Categories and Methods

Examinations will include liquid penetrant or magnetic particle techniques when surface examination is specified, ultrasonic or radiographic techniques when volumetric examination is specified, and visual inspection techniques will be used to determine surface condition of components and for evidence of leakage. Specific techniques, procedures and equipment, including any special techniques or equipment will be in accordance with the requirements of IWA-2200 of ASME Section XI and will be defined in the in-service inspection program.

The data from the various baseline examinations, collected in accordance with the above procedures, will be entered into a comprehensive report tabulating all the results. The report will describe the scope of the inspection, the procedures utilized, the equipment utilized and names and qualifications of the operators and all the examination results including all instrument calibration criteria in sufficient detail to ensure repeatability for each examination.

The categories and requirements appropriate for each examination area follow the categories and requirements specified in Tables IWB-2500-1 of ASME Section XI. An in-service inspection program which discusses the examination categories will be provided in accordance with IWB-2000.

5.2.4.4 Inspection Intervals

The IWA-2000 examination program for the 120 month inspection interval will be defined in the ISI plan. The ISI Plan for all Code Class 1 systems and components will be in accordance to the

ASME Code Section XI edition in effect per 10 CFR 50.55a 12 months prior to the issuance of an operating license (initial interval). Subsequent 120-month inspection intervals throughout the service life of the facility will comply, where practical with those requirements in the editions of the Code and addenda in effect per 10 CFR 50.55a 12 months prior to the start of each inspection interval.

5.2.4.5 Evaluation of Examination Results

Evaluation of examination results for Class 1 components will be conducted in accordance with Articles IWA-3000 and IWB-3000 of ASME Section XI. I

Unacceptable indications will be repaired in accordance with the requirements of Articles IWA-4000 and IWB-4000 of ASME Section XI. Criteria for establishing need for repair or replacement shall be per IWB-3000.

5.2.4.6 System Leakage and Hydrostatic Tests

The hydrostatic test for the reactor pressure vessel and reactor coolant pressure boundary will be conducted in accordance with the requirements of Articles IWA-5000 and IWB-5000 of ASME Section XI. System leak tests will be conducted prior to startup following each reactor refueling outage in accordance with Paragraphs IWB-5210 and IWB-5221. A system hydrostatic test will be conducted at or near the end of each inspection interval in accordance with IWB-5222. Examinations performed during these tests will be conducted without the removal of insulation. Technical Specifications requirements on operating limits during heatup, cooldown, and system hydrostatic pressure testing shall be employed for these tests.

5.2.5 REACTOR COOLANT PRESSURE BOUNDARY (RCPB) LEAKAGE DETECTION SYSTEMS

Means for the detection of leakage from the Reactor Coolant Pressure Boundary are provided to alert operators to the existence of leakage above acceptable limits, which may indicate an unsafe condition for the facility. The leakage detection systems are sufficiently diverse and sensitive to meet the criteria of Regulatory Guide 1.45 for leaks from identified and unidentified sources. I

5.2.5.1 Leakage Detection Methods

5.2.5.1.1 Unidentified Leakage

The methods employed to detect unidentified leakage are presented in the following sections.

5.2.5.1.1.1 Inventory Method

Leakage from the Reactor Coolant System (RCS) can be determined by net level changes in the pressurizer and in the volume control tank since the RCS and the Chemical and Volume Control System (CVCS) are closed systems. Since letdown flow and the reactor coolant pump seal controlled bleedoff flow are collected and recycled back into the RCS by the CVCS, the net inventory in the RCS and CVCS under normal operating conditions will be constant. Transient changes in letdown flow rate or RCS inventory can be accommodated by changes in the volume control tank level. Makeup flow rate provides a means of detecting leakage from the RCS through measurement of the net amount of makeup flow to the system. The net makeup to the system under no-leakage steady state conditions should be zero. The makeup flow rates and the integrated makeup flow from the Reactor Makeup System are continuously monitored and recorded. Analysis of the integrated makeup flow recorders over a period of steady state operation can provide detection of abnormal leakage. An increasing trend in the amount of makeup required will indicate an abnormal leak which is increasing in rate. Leaks occurring suddenly will be indicated by a step increase in the amount of makeup which does not decrease as would be the case for a purely transient condition.

5.2.5.1.1.2 Sump Level & Flow Method

Leakages are routed to the holdup volume sumps, or reactor cavity sump. Sump levels are monitored in the control room and alarmed on high level. Leakage rate is calculated utilizing the change of sump level. Additionally, the amount of leakage is calculated utilizing the frequency and lengths of time the sump pumps operate in combination with the known flow rate of the pumps. This system addresses Regulatory Guide 1.45 Position 2, 3(1).

5.2.5.1.1.3 Containment Air Particulate and Gaseous Radioactivity Monitoring

The monitor takes continuous containment air samples and measures the particulate and gaseous beta activity. High alarms are displayed in the control room. Leakage of reactor coolant has the effect of increasing radioactive particles and gas in the

containment, thus an increase in activity above base line is a direct indication of leakage. Conversion to leakage rate is discussed in Section 5.2.5.2.4.1. Radiological monitoring is discussed in more detail in Section 11.5.

The particulate monitoring system is capable of functioning when subjected to an SSE.

5.2.5.1.1.4 Acoustic Leak Monitoring System (ALMS)

The ALMS is fully described in Section 7.7.1.6.2. In summary, it is a system which monitors changes in acoustic levels above a normal background with no leakage. There are 9 locations where sensors are located (see Table 7.7-3). The locations were selected based on the criticality of the component or region, and in some cases (such as manways) where leakage has been experienced in the industry. The indication is qualitative; it alerts operators to investigate and compare with other leakage monitoring methods.

5.2.5.1.2 Identified Leakage

The amount of identified leakage from the Reactor Coolant System can be determined by adding up the amounts from all identified paths described below. Indicators and alarms associated with all of the identified leakage paths are provided in the control room.

5.2.5.1.2.1 Valves Located on the Reactor Coolant System

The primary safety valves and safety depressurization valves, located at the top of the pressurizer are routed to the In-Containment Refueling Water Storage Tank (IRWST). Each valve is monitored for seat leakage by an in-line Resistance-Temperature Detector (RTD).

Positive indication of valve leakage will be provided in the control room. Monitoring will be provided by an Acoustic Leak Monitoring System (ALMS) consisting of an accelerometer (acoustic sensor) mounted downstream of each valve. A plant annunciator alarm will be provided to alarm if the valve is not fully closed. The ALMS is part of the NSSS Integrity Monitoring System (see Section 7.7.1.6.2 for a detailed description).

5.2.5.1.2.2 Reactor Coolant Pump Seals

Instrumentation is provided to detect abnormal seal leakage. The reactor coolant pumps are equipped with two-stage seals plus a vapor or backup seal as described in Section 5.4.1.2. During normal operation, the reactor coolant system pressure is

decreased through the two seals to controlled bleedoff pressure. The vapor or backup seal prevents leakage to the containment atmosphere and operates at sufficient pressure to direct the controlled bleedoff to the volume control tank. The vapor or backup seal is designed to withstand full reactor coolant system pressure in the event the two primary seals fail.

The vapor seal pressure indicator would show a decrease in pressure upon a significant leak into the containment across the vapor seal, and an increased level in the reactor drain tank would be indicated. Seal leakage through the tubes of the reactor coolant pump seal coolers to the Component Cooling Water System (CCWS) would be indicated by increased temperature of the component cooling water return line from the reactor coolant pumps, by increased level in component cooling water surge tanks, and by increased radiation monitor readings in the CCWS.

5.2.5.1.2.3 Valves

Valves inside the containment which isolate the RCS from connecting systems during normal operation are equipped with stem leakoffs at the stem packing. These leakoffs are connected by piping to a collection tank. Valves in the charging and letdown system inside the containment are equipped with stem leakoffs where shown on Figure 9.3.4-1.

5.2.5.1.2.4 Reactor Vessel Closure Seals

The leakoff between the two metal O-rings which seal the RV flange is equipped with a pressure indicator which is alarmed. Leakage is collected in the RDT.

5.2.5.1.3 Leakage Through Steam Generator Tubes or Tubesheet

An increase in radioactivity indicated by condenser air removal system monitors, and blowdown system monitors will reveal reactor coolant leakage through steam generators tubes to the secondary side. Routine analysis of steam generator secondary water samples will also indicate increasing leakage of reactor coolant into the secondary system.

5.2.5.1.4 Leakage to Auxiliary Systems

Chapter 11 describes the design basis for process monitors used in all potentially contaminated auxiliary systems and their sensitivity.

5.2.5.2 Leakage Instrumentation in the Control Room**5.2.5.2.1 Safety Valve**

Indication of safety valve leakage is provided by the Acoustic Leak Monitoring System (ALMS). The signal from the accelerometer sensor is monitored continuously, along with the RMS value of the signal computed. These signals are compared to a pre-established baseline value and alarmed in the main control room via the Nuplex 80+ Data Processing System (DPS) and Discrete Indication and Alarm System (DIAS) displays (See Section 7.7).

**5.2.5.2.2 Primary Indicators of Reactor Coolant
Unidentified Leakage**

Positive control room indication of leakage from the RCS to the reactor building is provided by indicators and alarms. The primary indications of reactor coolant leakage are:

A. Containment atmosphere radiation indicators and alarms

Alarms are initiated by abnormally high air particulate and radioactive gas levels in the containment atmosphere.

B. Pumped flow from containment holdup volume and reactor cavity sump

C. Containment holdup volume and cavity sump level

D. Acoustic leakage monitoring

5.2.5.2.3 Other Indicators of Reactor Coolant Leakage

Other control room instrumentation that indicates significant reactor coolant leakage includes:

- o Pressurizer level
- o Containment pressure
- o Main condenser air removal system radiation monitor
- o Steam generator blowdown radiation monitor
- o Containment temperature
- o Containment humidity
- o Reactor drain tank level
- o Reactor drain tank pressure
- o Safety injection tank level
- o Safety injection tank pressure
- o Safety injection line pressure
- o Safety injection header pressure

5.2.5.2.4 Leakage Detection Conversion to Leakage Equivalent

The procedures for converting leakage detected by each detection method to a common leakage equivalent are indicated below.

5.2.5.2.4.1 Containment Radioactive Gas and Air Particulate Monitoring

Upon actuation of a high activity alarm, the operator follows these steps:

- A. Records particulate and gaseous activity at equal time intervals.
- B. Determines the activity increases during these intervals.
- C. Estimates RCS activity level based on previous operating history.
- D. Estimates the coolant leakage rate from the radioactivity increases. This is accomplished by comparing the measured increases to a curve showing abnormal primary coolant leakage as a function of particulate and gaseous activity variation.

Estimated leakage and measured particulate and gaseous activities then are plotted to test the consistency of the data and to estimate the time elapsed since the beginning of the leak.

As an alternative to these manual operations, these data may be entered into a computer and the results automatically computed.

Nonconstant activity increases mean either:

- o Containment atmosphere activity is approaching steady state
- o The abnormal leakage flow is not constant.

5.2.5.2.4.2 Leakage to Containment Sumps

A computer is programmed to calculate leakage rates from the rate of change of sump level. Digital readouts are available to the operator.

A computer is programmed to calculate leakage rates utilizing the frequency and interval of time the sump pumps operate in combination with the known pumping rate of the pumps. Digital readouts are available to the operator.

5.2.5.3 Maximum Allowable Total Leakage

The maximum allowable total identified leakage will be as stated in the Technical Specifications, Chapter 16.

5.2.5.4 Intersystem Leakage**5.2.5.4.1 Safety Injection System**

The Safety Injection System (SIS) is isolated from the RCS during normal operation. Leakage from the RCS to the SIS under normal operation would be detected by SIS pressure, and alarmed when the pressure approaches the SIS design pressure. If design pressure is reached the safety valves will open and the water will flow to the equipment drain tank, where the change in level will quantify the leak.

Leakage into the Safety Injection Tanks (SITs) by the two check valves isolating the tanks from the RCS is detected by an increase in pressure between the check valves. This pressure is indicated and alarmed in the control room. Secondly, the SIT level would increase. The leakage rate would be computed from the rate of change of level.

5.2.5.4.2 Pressurizer Relief Valve

Pressurizer relief valve flow is addressed in Section 5.2.5.2.1.

5.2.5.4.3 Steam Generator Leakage

The detection of leakage across the steam generator boundary between the primary and secondary side is discussed in Section 5.2.5.1.3. Leakage across this boundary would be quantified, after the indication of radioactivity in the condenser, by monitoring the pressurizer level, along with the RCS makeup water requirements.

5.2.5.4.4 Shutdown Cooling System

The Shutdown Cooling System (SCS) is a closed system without an accumulator (tank). Leakage from the RCS to SCS under normal operation, when the system is isolated from the RCS, would be detected by SCS pressure, and alarmed at a pressure less than the design pressure. If design pressure is reached the safety valves will open and the water will flow to the Equipment Drain Tank (EDT), where the change in level will be utilized to quantify the leak.

5.2.5.4.5 Component Cooling Water System (CCWS)

The CCWS cools the Reactor Coolant Pumps (RCPs), the SCS heat exchanger, the letdown heat exchanger, and containment spray pump and SCS miniflow heat exchangers. The CCWS surge tank level, CCS radiation monitors and CCWS water temperature will all reflect a leakage of RCS water to the CCWS. The change in surge tank level will be utilized to quantify the leakage.

5.2.5.5 Sensitivity and Response Time

For unidentified leakage, the leakage detecting system is designed to detect a minimum of 1 gpm in no greater time than one hour. The first warning, within minutes, will come from the containment atmospheric radiation monitors, and the operators will respond as outlined in Section 5.2.5.2.4.1. Further, they will be alerted to monitor sump levels, and will respond as outlined in Section 5.2.5.2.4.2. In the mean time, other parameters as listed in Section 5.2.5.2.3 will be monitored and correlated with the radiation monitor and sump level changes.

5.2.5.6 Operability Testing and Calibration During Plant Operation

The verification of operability begins after completion of subsystems construction with a verification that all drains go to the correct collecting destination. This is followed by the Initial Test Programs of Chapter 14, specifically tests 14.2.12.1.45; 14.2.12.1.46; and 14.2.12.1.49. All instrumentation critical in determining RCS leakage is designed and located to be calibrated while the plant is operating.

TABLE 5.2-1**REACTOR COOLANT SYSTEM PRESSURE BOUNDARY CODE REQUIREMENTS**

<u>Components</u>	<u>Codes and Classes</u>
Reactor Vessel, Steam Generators (primary side), Pressurizer	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.
Reactor Coolant Pump (structural portions necessary to assure the integrity of the reactor coolant pressure boundary)	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.
Reactor Coolant Pump Auxiliaries	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 3. Lube oil system designed for Seismic Category I requirements.
Pressurizer Spray and Safety Valves	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.
Piping and Valves	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.
Steam Generators (Secondary Side)	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 2.
Control Element Drive Mechanisms	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1.

NOTES: Code addenda requirements for System 80+ plants will comply with the requirements of 10 CFR 50.55a.

Codes listed above are construction codes. In addition, all these components are designed and constructed to meet the test and inspection requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection.

Code requirements for Safety Depressurization System valves, which meet the definition of the Reactor Coolant Pressure Boundary, are given in Section 6.7.

TABLE 5.2-2

(Sheet 1 of 5)

REACTOR COOLANT SYSTEM MATERIALS

<u>Component</u>	<u>Material Specification</u>	
Reactor Vessel		
Forgings	SA-508 Class 2 and 3	I
Cladding	Weld deposited austenitic stainless steel with 5FN-18FN delta ferrite or NiCrFe alloy (equivalent to SB-168)	D
Nozzle Safe Ends	SA-508 Class 1	I
Reactor vessel head CEDM Nozzles	SB-166	
Vessel internals ^(a)	Austenitic Stainless Steel and NiCrFe alloy	
Fuel cladding ^(a)	Zircaloy-4	
Instrument nozzles	SB-166	
Control element drive mechanism housings		
Lower	Type 403 stainless steel according to Code Case N-4-11 with end fittings to be SB-166 and/or SA-182 Type 348 stainless steel	D
Upper	SA-479 and SA-213 Type 316 stainless steel with end fitting of SA 479 Type 316 and vent valve seal of Type 316 and vent valve seal of Type 440 stainless steel seat	
Closure head bolts	SA-540 B24 or B23	
Pressurizer		
Shell Cladding ^(a)	SA-533 Grade A or B Class 1 or SA-508 Class 3 Weld deposited austenitic stainless steel with 5 FN-18FN delta ferrite or NiCrFe alloy (equivalent to SB-166)	D

TABLE 5.2-2 (Cont'd)**(Sheet 2 of 5)****REACTOR COOLANT SYSTEM MATERIALS**

Component	Material Specification	
Forged nozzles	SA-508 Class 1, 2 or 3	D
Instrument nozzles	SB-166	
Surge and safety valve nozzle safe ends	SA-182	
Studs and nuts	SA-540-B24 or B23	
Steam generator		
Primary Head	SA-533 Grade B, Class 1 or SA-508 Class 3	I
Primary Nozzles	SA-508 Class 2 or 3	
Primary head cladding ^(a)	Weld deposited austenitic stainless steel with 5FN-18FN delta ferrite	D
Tubesheet	SA-508 Class 2 or 3	
Tubesheet stay	SA-508 Class 2 or 3	
Tubesheet cladding ^(a)	Weld deposited NiCrFe alloy (equivalent to SB-168)	
Tube ^(a)	NiCrFe Alloy 690 (SB-163)	
Tube supports ^(a)	A-176, Type 409 (ASTM)	
Secondary shell	SA-533 Grade A or B, Class 1, or SA-508, Class 3	I
Secondary head	SA-516 Grade 70 or SA-508, Class 1A	
Secondary nozzles	SA-508 Class 1, 2 or 3	
Secondary nozzle safe ends	SA-508 Class 1A	
Secondary instrument nozzles	SA-106 Grade B	
Secondary studs and nuts	SA-540 Grade B24, or SA-193 Grade B7	

TABLE 5.2-2 (Cont'd)**(Sheet 3 of 5)****REACTOR COOLANT SYSTEM MATERIALS**

<u>Component</u>	<u>Material Specification</u>	
Primary studs and nuts	SA-564, Type 630H1100 or SB-637, No. 771B	I
Reactor Coolant Pumps		
Casing ^(a)	SA-508 Class 2 or 3 or austenitic stainless steel	B
Cladding	Weld deposited austenitic stainless steel with 5FN-18FN delta ferrite	D
Internals	SA-487 CA6NM, SA 336 Grade F8 or austenitic stainless steel	B
Reactor Coolant Piping		
Pipe (30 in. and 42 in.)	SA-516 Grade 70	
Cladding ^(a)	Weld deposited austenitic stainless steel with 5FN-18FN delta ferrite	
Piping nozzles and safe ends		
Nozzle forgings	SA-105, SA-541 Class 1, 2 or 3, or SB-166	D
Nozzle safe ends	SA-182 or SB-166	
Valves	SA-351 CF8M or SA-182	

TABLE 5.2-2 (Cont'd)**(Sheet 4 of 5)****REACTOR COOLANT SYSTEM MATERIALS****WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS**

Material Specification	Base Material	Weld Material
1. SA-533 Gr. B C1.1	SA-533 Gr. B C1.1	a. SFA 5.5, ^(b) E-8018-C3, E-8018-G b. MIL-E-18193, B-4
2. SA-508 C1.2	SA-533 Gr. B C1.1	a. SFA 5.5, E-8018-C3, E-8018-G b. MIL-E-18193, B-4
3. SA-508 C1.1	SA-508 C1.2	a. SFA 5.5, E-8018-C3, E-8018-G
4. SA-516 Gr. 70	SA-516 Gr. 70	a. SFA 5.1, E-7018
5. SA-182 F1	SA-516 Gr. 70	a. SFA 5.1, E-7018
6. SA-105 Gr. 11		a. SFA 5.14, ERNiCr-3
7. SA-182 F1	SA-351 CF8M	a. SFA 5.11, ENiCrFe-3
8. SA-105 Gr. 11	SA-182 F316	a. SFA 5.14, ERNiCr-3
9. SB-166	SA-182 F316	a. Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3
10. SB-167	SA-182 F304	a. Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3
11. SA-516 Gr. 70	SA-351 CF8M	a. SFA 5.1, E-7018 b. MIL-E-18193, B-4
12. SA-182 F1	SA-182 F316	a. SFA 5.1, E-7018
13. SB-166	SA-533 Gr. B C1.1	a. SFA 5.14, ERNiCr-3

TABLE 5.2-2 (Cont'd)

(Sheet 5 of 5)

REACTOR COOLANT SYSTEM MATERIALS**WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS**

<u>Material Specification</u>	<u>Base material</u>	<u>Weld Material</u>
14. SA-182	SB-167	a. SFA 5.14, ERNiCr-3
15. SA-516 Gr. 70	SA-508 Cl. 2	a. SFA 5.5, ^(b) E-8016-C3
16. Austenitic stainless steel cladding		a. SFA 5.9, ER-308 SFA 5.9, ER-309 SFA 5.9, ER-312
17. Inconel	Inconel	a. ENiCrFe-3 ERNiCr-3
18. SA-508 Cl. 3	SA-508 Cl. 3	a. SFA 5.5, ^(b) E-8018-C3, E-8018-G b. MIL-E-18193, B-4
19. SA-508 Cl. 3	SA-533 Gr. B Cl. 1	a. SFA 5.5, E-8018-C3, E-8018-G b. MIL-E-18193, B-4
20. SA-508 Cl. 3	SA-508 Cl. 2	a. SFA 5.5, E-8018-C3, E-8018-G
21. SA-508 Cl. 3	SA-516 Gr. 70	a. SFA 5.5, ^(b) E-8018-C3

Notes: a. Materials exposed to reactor coolant.

b. Special weld wire with low residual elements of copper, nickel and phosphorous as specified for the reactor vessel core beltline region.

TABLE 5.2-3

CODE CASE INTERPRETATIONS

1.	N-4-11	Requirements for Special Type 403 Modified Forgings and Bars
2.	N-71-14	Additional Materials for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated by Welding, Section III, Division 1

D

Note: Code Case Interpretations applied to the reactor coolant pressure boundary components are in accordance with Regulatory Guides 1.84 and 1.85.

TABLE 5.2-4

RESULTS OF THE INADVERTENT SAFETY INJECTION ACTUATION
TRANSIENT ANALYSIS^a
(FOR A WATER-SOLID RCS)

<u>Number of Charging Pumps</u>	<u>Number of SIS Pumps</u>	<u>Number of SCS Relief Valves Operational</u>	<u>Maximum RCS Pressure (psia)</u>	<u>Maximum Relief Flowrate (gpm)</u>
1	1	1	567	1260
1	2	1	580	2162
1	3	1	592	3055
1	4	1	608	3934

NOTE: a. Including the considerations of energy addition from decay heat and all pressurizer heaters.

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Amendment E
December 30, 1988

SYSTEM 80+TM	SYSTEM 80+ RCP START TRANSIENT	Figure 5.2-1
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5.3 REACTOR VESSEL

5.3.1 REACTOR VESSEL MATERIALS

5.3.1.1 Material Specifications

The principle ferritic materials used in the reactor vessel are listed in Table 5.2-2. These materials are in accordance with the ASME Boiler and Pressure Vessel Code, Section III.

5.3.1.2 Special Process Used for Manufacturing and Fabrication

The reactor vessel is fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III. No special manufacturing methods that could compromise the integrity of the vessel are used.

The reactor vessel is a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical upper closure head. The construction is basically that of forged rings, forged hemispherical heads, forged flanges on the closure head and vessel, and forged nozzles. The internal surfaces that are in contact with the reactor coolant are clad with austenitic stainless steel. B

The reactor vessel consists basically of a vessel flange, three shell sections (upper, intermediate and lower) and a bottom head. The vessel flange is a forged ring with a machined ledge on the inside surface to support the core support barrel, which in turn supports the reactor internals and the core. The flange is drilled and tapped to receive the closure studs and is machined to provide a mating surface for the reactor vessel closure seals. Each shell consists of one 360 degree forged ring. The bottom head is constructed of a single hemispherical forging. The three shell sections, the bottom head forging and vessel flange forging are joined together by welding, along with four inlet nozzle forgings and two outlet nozzle forgings to form a complete vessel assembly. B

The closure head is fabricated separately since it is joined to the reactor vessel by bolting. The closure head consists of a head flange and a dome. The head flange is a forged ring. The flange is drilled to match the vessel flange stud hole locations, and the lower surface of the flange is machined to provide a mating surface for the vessel closure seals. The dome is constructed of a single hemispherical forging. The dome and flange are welded together to form the closure head, and the control element drive mechanism (CEDM) nozzles are welded into the head to complete the assembly. B

5.3.1.3 Special Methods for Nondestructive Examination

Prior to, during, and after fabrication of the reactor vessel, nondestructive tests based upon Section III of the ASME Boiler and Pressure Vessel Code are performed on all welds and forgings as indicated. The nondestructive examination requirements including calibration methods, instrumentation, sensitivity, reproducibility of data, and acceptance standards are in accordance with requirements of the ASME B&PV Code, Section III. (See Table 5.2-1). These methods, procedures and requirements are compatible with Section XI of the ASME Code so that results of preservice inspections can be correlated with in-service inspections. Strict quality control is maintained in critical areas such as calibration of test instruments.

All full-penetration, pressure-containing welds are 100% radiographed to the standards of Section III of the ASME Boiler and Pressure Vessel Code. Weld preparation areas, back-chip areas, and final weld surfaces are magnetic-particle or dye-penetrant examined. Other pressure-containing welds, such as used for the attachments of nonferrous nickel-chromium-iron mechanism housings, vents, and instrument housings to the reactor vessel and head, are inspected by liquid-penetrant tests of the root pass, the lesser of one-half of the thickness or each 1/2-inch of weld deposit, and the final surface. Additionally, the base metal weld preparation area is magnetic-particle examined prior to overlay with nickel-chromium-iron weld metal.

All forgings are inspected by ultrasonic testing, using longitudinal beam techniques. In addition, ring forgings are tested using shear wave techniques.

All carbon-steel and low alloy forgings and ferritic welds are also subjected to magnetic-particle examination after stress relief.

All vessel bolting material receives ultrasonic and magnetic-particle examination during the manufacturing process.

The bolting material receives a straight-beam, radial-scan, ultrasonic examination with a search unit not exceeding 1 square-inch area. All hollow material receives a second ultrasonic examination using angle-beam, radial scan with a search unit not exceeding 1 square inch in area. A reference specimen of the same composition and thickness containing a notch (located on the inside surface) 1 inch in length and a depth of 3% of nominal section thickness, or 3/8-inch, whichever is less, is used for calibration. Use of these techniques ensures that no materials that have unacceptable flaws, observable cracks, or sharply defined linear defects are used.

The magnetic-particle inspection is performed both before and after threading of the studs.

Upon completion of all post-weld heat treatments, the reactor vessel is hydrostatically tested, and all accessible ferritic weld surfaces, including those used to repair material, are magnetic-particle inspected in accordance with Section III of the ASME Boiler and Pressure Vessel Code. | D

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Special controls for ferritic and austenitic stainless steels are as follows:

- A. Regulatory Guide 1.31, Control of Ferrite Content in Stainless Steel Weld Metal, is addressed in Section 5.2.3.4. | D
- B. Regulatory Guide 1.34, Control of Electroslag Weld Properties, is addressed in Section 5.2.3.3.
- C. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants", is addressed in Section 5.2.3.4.1.2.1. | I
- D. Regulatory Guide 1.43, Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components, is addressed in Section 5.2.3.3.
- E. Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel, is addressed in Section 5.2.3.4.
- F. Regulatory Guide, 1.50, Control of Preheat Temperature for Welding of Low-Alloy Steel, is addressed in Section 5.2.3.3.
- G. Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility, is addressed in Section 5.2.3.3.
- H. Regulatory Guide 1.99, Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials, is addressed in Section 5.3.1.6.7. | B

5.3.1.5 Fracture Toughness

In accordance with 10 CFR 50 Appendix G, Paragraph IV A, the reactor vessel beltline materials have minimum upper-shelf energy, as determined from Charpy V-notch tests on unirradiated specimens in accordance with Paragraphs NB-2322.2(a) of the ASME Code, of 75 ft-lbs. Charpy impact tests will be performed on | D

transversely (weak direction) oriented specimens from the beltline forgings to establish RT_{NDT} as required by 10 CFR 50, Appendix G.

5.3.1.6 Reactor Vessel Material Surveillance Program

The irradiation surveillance program for System 80+ will be conducted to assess the neutron-induced changes in the RT_{NDT} (reference temperature) and the mechanical properties of the reactor vessel materials. Changes in the impact and mechanical properties of the material will be evaluated by the comparison of pre- and post-irradiation test results. The capsules containing the surveillance test specimens used for monitoring the neutron-induced property changes of the reactor vessel materials will be irradiated under conditions which represent, as closely as practical, the irradiation conditions of the reactor vessel.

ASTM E-185-82, Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels, and 10 CFR 50 Appendix H, Reactor Vessel Material Surveillance Program Requirements, present criteria for monitoring changes in the toughness properties of reactor vessel beltline materials through surveillance programs. The System 80+ reactor vessel surveillance program adheres to all of the requirements of ASTM E-185-82 and satisfies the intent of 10 CFR 50, Appendix H. This surveillance program also satisfies the ASME Code, Section XI.

5.3.1.6.1 Test Material Selection

Materials selected for the surveillance program are those judged most likely to be controlling with regard to radiation embrittlement according to the recommendations of Regulatory Guide 1.99, Revision 2.

Surveillance test materials are prepared from the actual materials used in fabricating the beltline region of the reactor pressure vessel. The test materials are processed so that they are representative of the materials in the completed reactor vessel. Specimens are prepared from three metallurgically different materials, including base metal, weld metal and heat-affected zone (HAZ) material.

In addition, material is included from a standard heat of ASME SA-533 Grade B Class 1 manganese-molybdenum-nickel steel made available by the USNRC sponsored Heavy Section Steel Technology (HSST) Program. This standard reference material (SRM) is used as a monitor for Charpy impact tests, permitting comparisons among the irradiation data from operating power reactors and experimental reactors. Compilation of data generated from post-irradiation tests of these correlation monitors will be carried out by the HSST program.

Base metal test material is from a section of the shell course forging selected from the beltline of the reactor vessel. Selection shall be based on an evaluation of initial toughness (characterized by an index temperature such as RT_{NDT}) and the estimated effect of chemical composition and neutron fluence on the toughness during reactor operation. Normally, the forging with the highest adjusted index temperature (initial value plus predicted radiation induced increase in index temperature) at end-of-life shall be selected as the surveillance base metal test material. In certain circumstances, however, selection of two separate shell course forgings, from both intermediate and lower shell courses, may be warranted if both should be monitored for their effect on the pressure-temperature operating limits during the design lifetime of the reactor vessel.

Weld region test material, representative of the controlling reactor pressure vessel weld, is produced by welding together sections of forgings from the beltline of the reactor vessel. The HAZ test material is manufactured from a section of the same forgings used for base metal surveillance test material. The weld metal test material is produced from the same heat of weld wire or rod and lot of flux used in the beltline of the reactor vessel. Welding parameters duplicate those used for the beltline welds. | E

Representative stock (archive material) to provide two additional sets of test specimens for each material shall be provided with full documentation and identification.

5.3.1.6.2 Test Specimens

5.3.1.6.2.1 Type and Quantity

Drop weight, standard and precracked Charpy impact, tensile test, and compact tension fracture toughness specimens are provided for unirradiated tests. Drop weight tests will be conducted in accordance with ASTM E-208. Charpy impact tests will be conducted in accordance with ASTM E-23. Tensile tests are conducted in accordance with ASTM E-8 and E-21. Correlation of drop weight and Charpy impact tests to establish RT_{NDT} will be made in accordance with NB2300 of the ASME Code, Section III. Charpy impact and tensile test specimens are provided for post-irradiation tests. | E

The total quantity of specimens furnished for carrying out the overall requirements of this program is presented in Table 5.3-1.

For surveillance programs including two base metal forgings, the total number of specimens shown in Table 5.3-1 will be

distributed between the two forgings, but there will be a sufficient quantity of specimens from each forging to meet the requirements of ASTM E-185-82. | B

For baseline testing, the base metal specimens shown in Table 5.3-2, with the exception of longitudinal base metal drop weights, will be divided equally for both forgings. For irradiation encapsulation, the base metal specimens shown in Table 5.3-3 will be divided equally for both forgings. Three capsule assemblies will contain specimens from one base metal forging and three capsule assemblies will contain specimens from the other base metal forging.

5.3.1.6.2.2 Baseline Specimens

The type and quantity of test specimens provided for establishing the properties of the unirradiated reactor vessel materials are presented in Table 5.3-2. The data from tests of these specimens provide the basis for determining the neutron-induced property changes of the reactor vessel materials.

Twelve drop weight test specimens each of base metal weld metal and HAZ material are provided for establishing the NDTT of the unirradiated surveillance materials. These data form the basis for RT_{NDT} determination. RT_{NDT} is the reference temperature from which subsequent neutron-induced changes are determined.

Twenty-four standard Charpy test specimens each of base metal (longitudinal and transverse), weld metal, and HAZ material are provided. This quantity exceeds the minimum number of test specimens recommended by ASTM E-185 for developing a Charpy impact energy transition curve and is intended to provide a sufficient number of data points for establishing accurate Charpy impact energy transition temperatures for these materials. These data, together with the drop weight NDTT, are used to establish an RT_{NDT} for each material. | E

Twelve precracked Charpy impact test specimens each of base metal (longitudinal and transverse) and weld metal material are provided in addition to the standard Charpy impact specimens. This quantity is sufficient to determine fracture toughness properties (critical stress intensity factors under dynamic loading) over the range extending from linear elastic to elastic-plastic fracture. | E

Twelve tensile test specimens each of base metal (longitudinal and transverse), weld metal and HAZ materials are provided. This quantity also exceeds the minimum number of test specimens recommended by ASTM E-185 and is intended to permit a sufficient number of tests for accurately establishing the tensile | D

properties for these materials at a minimum of three test temperatures (e.g., ambient, operating and design).

Eight compact tension 1t test specimens and four 1/2t compact tension test specimens each of base metal (transverse) and weld metal material are provided to augment the fracture toughness data determined from the precracked Charpy tests. This quantity of specimens is sufficient to determine fracture toughness properties over the range extending from linear elastic to elastic-plastic fracture behavior.

5.3.1.6.2.3 Irradiated Specimens

Tensile test, standard and precracked Charpy impact and 1/2t compact tension test specimens are used for determining changes in the strength and static and dynamic toughness properties of the materials due to neutron irradiation. A total of 333 standard Charpy impact, 81 precracked Charpy impact, 60 1/2t compact tension and 36 tensile test specimens are provided. The type and quantity of tests specimens provided for establishing the properties of irradiated materials over the lifetime of the vessel are presented in Table 5.3-3. Compact tension specimens provided for capsule irradiation are precracked prior to insertion to reduce the time required for post irradiation testing. The types and quantity of specimens provided exceed the minimum requirements of ASTM E-185-82.

5.3.1.6.3 Surveillance Capsules

The surveillance test specimens are placed in corrosion resistant capsule assemblies for protection from the primary coolant during irradiation. The capsules also serve to physically locate the test specimens in selected positions within the reactor vessel and to facilitate the removal of a desired quantity of test specimens when a specified radiation exposure has been attained. Six surveillance capsule assemblies are provided for the reactor vessel. Three of the capsule assemblies contain precracked Charpy test specimens (PC_v Capsule) and three capsule assemblies contain 1/2t compact tension specimens (CT capsule). The type and quantity of specimens contained in each type of capsule assembly is presented in Table 5.3-4.

A typical capsule assembly, illustrated in Figure 5.3-1, consists of a series of three specimen compartments, connected by wedge couplings, and a lock assembly. Each compartment enclosure of the capsule assembly is internally supported by the surveillance specimens and is externally pressure tested to 3125 psi during final fabrication. The wedge couplings also serve as end caps for the specimen compartments and position the compartments within the capsule holders which are attached to the reactor vessel cladding. The lock assemblies fix the locations of the

capsules within the holders by exerting axial forces on the wedge coupling assemblies; this causes the wedges to exert horizontal forces against the sides of the holders preventing relative motion. The lock assemblies also serve as a point of attachment for the tooling used to remove the capsules from the reactor.

Each capsule assembly is made up of two Charpy, flux, compact tension compartment assemblies and one temperature, flux, tensile, Charpy compartment assembly. Each capsule compartment is assigned a unique identification so that a complete record of test specimen and flux/temperature monitor (tensile-monitor) compartment. Each capsule compartment is assigned a unique identification so that a complete record of test specimen location within each compartment can be maintained.

5.3.1.6.3.1 Charpy, Flux and Compact Tension Compartment Assembly

This assembly (Figure 5.3-2) contains 15 impact test specimens and a set of five flux spectrum monitors in the top section. The bottom section contains either 18 precracked Charpy test specimens or ten 1/2t compact tension specimens. The Charpy test specimens are arranged vertically in 1 x 3 arrays and are oriented with the notch toward the reactor core. The 1/2t compact tension specimens are oriented so that the opening of the crack starter notch is facing the top of the compartment. This orientation will result in a neutron flux gradient parallel to the crack front. The temperature differential between the specimens and the reactor coolant is minimized by using spacers between the specimens and the compartment and by sealing both sections of the assembly in an atmosphere of helium. This quantity of specimens provides an adequate number of data points for establishing a Charpy impact energy transition curve for a given irradiated material. Comparison of the unirradiated and irradiated Charpy impact energy transition curves permits determination of the RT_{NDT} changes due to irradiation for the various materials.

5.3.1.6.3.2 Temperature, Flux, Tensile and Charpy Compartment Assembly

This assembly (Figure 5.3-3) contains 3 base metal (transverse) tensile test specimens and 12 impact test specimens in the top section. The tension specimens are placed in a housing machined to fit the compartment. Split spacers are placed around the gage length of the specimens to minimize the temperature differential between the specimen gage length and the coolant. The impact specimens are arranged vertically in 1 x 3 arrays and are oriented with the notch toward the reactor core. Spacers are utilized between the test specimens and the compartment. The

bottom section contains a set of nine flux spectrum monitors, a set of temperature monitors, 9 SRM (Standard Reference Material) impact test specimens and 3 weld metal tension test specimens. Both compartment sections are sealed within an atmosphere of helium.

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5.3.1.6.4 Neutron Irradiation and Temperature Exposure

The changes in the RT_{NDT} of the reactor vessel materials are derived from specimens irradiated to various fluence levels and in different neutron energy spectra. In order to permit accurate predictions of the RT_{NDT} of the vessel materials, complete information on the neutron flux energy spectra, and the irradiation temperature of the encapsulated specimens must be available.

5.3.1.6.4.1 Flux Measurements

Fast neutron flux measurements are obtained by insertion of threshold detectors into each of the six irradiation capsules. Such detectors are particularly suited for the proposed application, because their effective threshold energies lie in the range of interest (0.5 to 15 MeV).

These neutron threshold detectors and the thermal neutron detectors, presented in Table 5.3-5, can be used to monitor the thermal and fast neutron spectra incident on the test specimens. These detectors possess reasonably long half-lives and activation cross sections covering the desired neutron energy range.

Three sets of flux spectrum monitors are included in each capsule assembly. Each detector is placed inside a sheath which identifies the material and facilitates handling. Cadmium covers are used for those materials (e.g., uranium, nickel, copper, cobalt and neptunium) which have competing neutron capture activities.

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The flux monitors are placed in holes drilled in stainless steel housings as shown in Figure 5.3-3 at three axial locations in each capsule assembly (Figure 5.3-1) to provide an axial profile of the level of fluence which the specimens attain.

In addition to these detectors, the program also includes correlation monitors (Charpy impact test specimens made from a reference heat of ASME SA-533 Grade B Class 1, manganese-molybdenum-nickel steel) which are irradiated along with the specimens made from reactor vessel materials. The changes in impact properties of the standard reference material (SRM) provide a cross-check on the dosimetry in any given

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surveillance program. These changes also provide data for correlating the results from this surveillance program with the results from experimental irradiations and other reactor surveillance programs using specimens of the same reference material.

5.3.1.6.4.2 Temperature Estimates

Because the changes in mechanical and impact properties of irradiated specimens are highly dependent on the irradiation temperature, it is necessary to have knowledge of specimens as well as the pressure vessel. During irradiation, instrumented capsules are not practical for a surveillance program extending over the design lifetime of a power reactor. The maximum temperature of the irradiated specimens can be estimated with reasonable accuracy by including in the capsule assemblies small pieces of low melting point alloys or pure metals. The compositions of candidate materials with melting points in the operating range of power reactors are listed in Table 5.3-6. The monitors are selected to bracket the operating temperature of the reactor vessel.

The temperature monitors consist of a helix of low melting alloy wire inside a sealed quartz tube. A stainless steel weight is provided to destroy the integrity of the wire when the melting point of the alloy is reached. The compositions and therefore the melting temperatures of the temperature monitors are differentiated by the physical lengths of the quartz tubes which contain the alloy wires.

A set of temperature monitors is included in each capsule assembly. The temperature monitors are placed in holes drilled in stainless steel housings as shown in Figure 5.3-3 and are also placed in three axial locations in each capsule assembly (Figure 5.3-1) to provide an axial profile of the maximum temperature to which the specimens are exposed.

5.3.1.6.5 Irradiation Locations

The test specimens are enclosed within six capsule assemblies the axial positions of which are bisected by the midplane of the core. A summary of the specimens contained in each of these capsule assemblies is presented in Table 5.3-4.

The test specimens contained in the capsule assemblies are used for monitoring the neutron-induced property changes of the reactor vessel materials. These capsules, therefore, are positioned near the inside wall of the reactor vessel so that the irradiation conditions (fluence, flux spectrum, temperature) of the test specimens resemble as closely as possible the irradiation conditions of the reactor vessel. The neutron

fluence of the test specimens is expected to be approximately 1.15 times higher than that seen by the adjacent vessel wall, so the measured changes in properties of the surveillance materials will be able to predict the radiation induced changes in the reactor vessel beltline materials. | E
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The capsule assemblies are placed in capsule holders positioned circumferentially about the core at locations which include the regions of maximum flux. Figure 5.3-4 presents the typical exposure locations for the capsule assemblies.

All capsule assemblies are inserted into their respective capsule holders during the final reactor assembly operation. The design also permits the remote installation of replacement capsule assemblies. The capsule holders are welded to the vessel cladding on the inside surface, and the welds are subject to inspection according to the requirements for permanent structural attachments as given in the ASME Code, Sections III and XI. | I

5.3.1.6.6 Withdrawal Schedule

The capsule assemblies remain within their holders until the specimens contained therein have been exposed to predetermined levels of fast neutron fluence. At that time, the capsule assembly is removed and the surveillance materials are evaluated. The capsule assembly removal schedule and the associated target fluence are presented in Table 5.3-7. | D

The target fluence levels for the surveillance capsules are determined at the azimuthal locations for the recommended withdrawal schedule of ASTM E-185-82, extended to a design life of 60 years (48 EFPY). The fluence values in Table 5.3-7 are accurate within +30%, -30%. The uncertainty is composed of errors in the calculational method and errors in the combined radial and axial power distribution. | E

Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule. The two standby capsules are provided in the event they are needed to monitor the effect of a major core change or annealing of the vessel, or to provide supplemental toughness data for evaluating a flaw in the beltline. | E

5.3.1.6.7 Irradiation Effects Prediction Basis

Irradiation induced RT_{NDT} shift and reduction of upper shelf energy are predicted based on Regulatory Guide 1.99 Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." Predicted changes in RT_{NDT} and upper shelf energy are used to select the surveillance materials | D

(Section 5.3.1.6.1) and to formulate the initial heatup and cooldown limit curves for plant operation. Once actual post-irradiation surveillance data become available for each reactor vessel, these data will be used to adjust plant operating limit curves.

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5.3.1.7 Reactor Vessel Fasteners

The bolting material for the reactor vessel closure head is fabricated from SA 540, B23 or B24, Class III material. This material conforms to the requirements of 10 CFR 50, Appendix G and the intent of Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." Nondestructive examination will be performed according to Subarticle NB-2580 of Section III of the ASME Code, during the manufacturing process. I

C-E specifies the use of a manganese phosphate coating on threads of studs, nuts and washers to improve anti-galling properties and resistance to corrosion. In addition, Super Moly lubricant (containing molybdenum disulfide) is specified to be added to threads and bearing surfaces at installation to further enhance anti-galling properties. Laboratory testing and field experience to date have shown no evidence of deleterious breakdown of either phosphate coating or lubricant.

5.3.2 PRESSURE-TEMPERATURE LIMITS

All components in the Reactor Coolant System (RCS) are designed to withstand the effects of cyclic loads due to RCS temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. E

The design number of cycles for heatup and cooldown is based on a rate of 100°F/hr. During unit startup and shutdown, the rate of temperature change is limited to less than 100°F/hr by administrative procedure. I

The maximum allowable RCS pressure at the corresponding minimum allowable temperature is based upon the stress limitations for brittle fracture. These limitations are derived using linear elastic fracture mechanics (LEFM) principles, the procedures prescribed by Appendix G to Section III of the ASME Code, "Protection Against Nonductile Fracture," Appendix G to 10 CFR 50, "Fracture Toughness Requirements," NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," and the procedures recommended by WRC Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials." Compliance with Appendix H to 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements," is discussed in Section 5.3.1.6. E

5.3.2.1 P-T Limit Curves

5.3.2.1.1 Material Properties

Pressure-temperature limitations (P-T Limits) are determined, in part, using material property test data for Reactor Coolant Pressure Boundary materials, as required by Appendix G to Section III of the ASME Code. Based on considerations of existing material property test data, an initial RT_{NDT} for the reactor vessel beltline material of -20°F is assumed. The initial RT_{NDT} for the remaining material of the Reactor Coolant System of $+10^{\circ}\text{F}$ is also assumed.

As a result of fast neutron irradiation in the region of the core, the RT_{NDT} of irradiated material will increase with operation. The techniques used to analytically and experimentally predict the integrated fast neutron ($E \geq 1$ Mev) fluxes of the reactor vessel are described in Sections 5.3.1.4 and 5.3.1.6. Analytical predictions of the shift in RT_{NDT} reactor vessel beltline materials are based on the procedures described in NRC Regulatory Guide 1.99. Based on these procedures, the maximum calculated shift (without any margin) in the RT_{NDT} at 1/4t over a 60-year design life is 49°F . The total predicted shift (calculated with a conservative margin of 50°F for the beltline forging) is 99°F . The corresponding RT_{NDT} shift value for the beltline girth seam weld is 88°F (38°F calculated shift plus a conservative margin of 50°F).

The measured shift in RT_{NDT} for a sample is applied to the adjacent section of the reactor vessel for later stages in plant life because the measured neutron spectra and flux at the samples and reactor vessel inside radius should be nearly identical. The measured shift in RT_{NDT} for a sample is adjusted for the difference in calculated flux magnitudes between the sample and the point of interest in the reactor vessel wall. The maximum exposure to the reactor vessel will be obtained from the measured sample exposure by application of the calculated azimuthal neutron flux variation. The neutron fluence and the actual shift in RT_{NDT} will be established periodically during plant operation by testing the reactor vessel surveillance material samples which are irradiated in holders secured to the inside wall of the reactor vessel as described in Section 5.3.1.6 and shown in Figure 5.3-4. Since the surveillance materials are irradiated very close to the vessel inside surface, the samples when tested will demonstrate the accuracy of the RT_{NDT} shift prediction as well as the effect of any additional long-term aging phenomenon. The pressure-temperature limits will then be adjusted periodically, if necessary, to stay within allowable stress limits during normal operation.

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Figures 5.3-5a and 5.3-5b are typical examples of the intended pressure-temperature limitations determined in accordance with Appendix G to 10 CFR 50 for normal operation of the Reactor Coolant System.

The P-T Limit curves are based on following:

A. Minimum Boltup Temperature

The minimum boltup temperature is principally governed by Appendix G to Section III of the ASME Code. The ASME Code requires that when the flange and adjacent shell region are stressed by the full bolt preload and by pressure not exceeding 20% of the preoperational system hydrostatic test pressure, the minimum metal temperature in the stressed region must be at least the initial RT_{NDT} plus any effects of irradiation. From the assumption that the initial RT_{NDT} for all regions of the RCS other than the beltline is $+10^{\circ}\text{F}$, and that the flange and adjacent shell regions are not subjected to significant irradiation, the minimum boltup temperature per this requirement is $+10^{\circ}\text{F}$. Based on considerations of typical refueling water temperatures, however, the minimum boltup temperature is taken to be 60°F .

B. Maximum Pressure Below Lowest Service Temperature

The maximum allowable pressure below the lowest service temperature is defined by paragraph NB-2332, Section III of the ASME Code, to be 20% of the preoperational hydrostatic test pressure. This test pressure is 125% of the design pressure (2500 psia), or 3125 psia, and 20% of this value is 625 psia. Therefore, the maximum pressure below the lowest service temperature is 625 psia (actual fluid pressure at the inside surface of the beltline).

C. Lowest Service Temperature

The lowest service temperature (LST) is defined by paragraph NB-2332, Section III of the ASME Code, to be the minimum allowable temperature at pressures above 20% of the preoperational hydrostatic test pressure. This value is defined to be no lower than $RT_{NDT} + 100^{\circ}\text{F}$, where the RT_{NDT} is considered to be for the most limiting component in the RCS other than the beltline. From the assumption that the initial RT_{NDT} for all regions of the RCS other than the beltline is $+10^{\circ}\text{F}$, the LST per this requirement is $10^{\circ}\text{F} + 100^{\circ}\text{F} = +110^{\circ}\text{F}$.

D. Operation, Heatup and Cooldown Curves

1. Reactor Vessel Beltline

P-T Limits for the reactor vessel beltline are examined for heatup and cooldown conditions. For the heatup analysis, both the 1/4t and 3/4t locations are examined. For heatup, the thermal stresses, σ_t , are compressive at the inside surface and tensile at the outside surface. The membrane stresses due to pressure, σ_m , are always tensile, but more so at the inside than outside surface. As a result, the total stress will always be greater at the outside than the inside surface. However, the maximum allowable stresses, taking into account irradiation effects, will decrease more at the inside than outside surface because the effects of irradiation are more pronounced there. Which surface stresses will approach this maximum first is not always clear and is a function of several variables. Therefore, both locations are examined for the heatup transient.

For the cooldown analysis, only the 1/4t location needs to be examined. During cooldown, the thermal stresses are tensile at the inside surface and compressive at the outside. Again, membrane stresses due to pressure are always positive. As a result, the total stress will always be greater at the inside than the outside surface. Since the maximum allowable stresses, taking into account irradiation effects, will decrease more at the inside than outside surface, it is clear that the total stress at the inside surface will approach the maximum allowable before those at the outside surface. Therefore, only the 1/4t location needs to be examined for the cooldown transient.

Heatup and cooldown rates from 0°F/hr ("Isothermal" pressurization) to the design limit of 100°F/hr at 10 °F/hr intervals are examined in determining the allowable heatup and cooldown rates as a function of temperature to meet low-temperature overpressure protection (LTOP) requirements for the Reactor Coolant system. LTOP considerations are discussed in Section 5.2.2.10.

For the beltline analysis, during normal operations, the following condition is maintained:

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$$K_{IR} = 2K_{Im} + K_{It}$$

where, K_{IR} = Reference stress intensity factor specified by Figure G-2210-1 in Appendix G to Section III of the ASME Code.

K_{Im} = Stress intensity factor for membrane stress due to pressure.

$= \sigma_m M_m$, where M_m = Membrane correction factor defined in Figure G-2214-1 in Appendix G to Section III of the ASME Code,

$$\sigma_m = \frac{Pr}{t}, \text{ where } P = \text{internal reactor vessel pressure, psia,}$$

$$r = \text{inside reactor vessel radius, in.,}$$

$$t = \text{reactor vessel wall thickness, in.}$$

K_{It} = Stress intensity factor for thermal stress.

$= \Delta T_w M_t$, where M_t = Thermal correction factor defined in Figure G-2214-2 in Appendix G to Section III of the ASME Code,

ΔT_w = Maximum steady-state temperature differential across the reactor vessel wall, °F.

The right side of the above equation ($2K_{Im} + K_{It}$) is calculated for various pressures and then set equal to the left side (K_{IR}). The 2 represents a safety factor required by Appendix G to Section III of the ASME Code. The minimum allowable temperature corresponding to a given pressure can then be calculated from this equation. The resulting combination represents a (maximum) pressure-(minimum) temperature limit which must not be exceeded to ensure nonbrittle material behavior. A set of such pressure-temperature coordinates define a P-T limit curve for a specific heatup or cooldown rate, at a particular point in plant life. Figures 5.3-5a and 5.3-5b illustrate representative P-T Limits for the end-of-(plant) life-60 years, or 48 EFPY's. Figure 5.3-6 illustrates the corresponding allowable heatup and cooldown rates.

The pressure used in computing K_{Im} and K_{It} , and the corresponding computed temperature are for the crack tip. Correction factors for pressure and temperature are taken into account when expressing P-T Limits graphically so as to show the limits in terms of indicated pressurizer pressure and indicated RCS temperature. This is done for P-T Limit Curve presentation in Technical Specifications. The correction factors take into account the effects of instrument error, pressure differentials due to flow in the Reactor Coolant System, static pressure differentials due to elevation differences, and temperature differentials due to thermal gradients in the reactor vessel wall. The P-T Limits shown in Figures 5.3-5a and 5.3-5b, however, are in terms of actual fluid conditions at the reactor vessel inside wall without any correction factors applied.

K_{IR} is a function of the temperature, T , and the adjusted reference temperature, ART , of the material at the crack-tip. The analytical expression for K_{IR} is

$$K_{IR} = 26.78 + 1.233 e^{[0.145(T - ART + 160)]}$$

as specified by Appendix G to Section III of the ASME Code. The calculation of the ART is in accordance with the procedure described in NRC Regulatory Guide 1.99. The ART is a function of the initial RT_{NDT} of the material, (Initial) RT_{NDT} , the shift in RT_{NDT} due to irradiation over a period of time, ΔRT_{NDT} , and a safety margin. The equation for the ART is

$$ART = (Initial)RT_{NDT} + \Delta RT_{NDT} + \text{Margin.}$$

The shift, ΔRT_{NDT} , is calculated from

$$\Delta RT_{NDT} = (CF) f^{(0.28 - 0.10 \log f)}$$

where CF is a chemistry factor and f is the neutron fluence at the point of interest in the reactor vessel wall. The fluence f is calculated from the surface fluence, f_{surface} , as

$$f = f_{\text{surface}} (e^{-0.24x})$$

where x is the depth into the reactor vessel wall.

The margin is calculated based on

$$\text{Margin} = 2 \sqrt{\sigma^2_I + \sigma^2_{\Delta}}$$

E

where σ_I is the standard deviation characteristic of the (Initial)RT_{NDT} data, and σ_A is the standard deviation for ΔRT_{NDT} data as described in Regulatory Position C.1.1 of NRC Regulatory Guide 1.99. σ_T is assigned a value of zero because measured properties will be used to establish RT_{NDT}, not estimated values. σ_A for forgings is 17°F per Regulatory Guide 1.99, but the full margin (2σ) for prediction purposes is assigned the conservative value of 50°F to cover unanticipated long-term aging phenomena. Once confirmatory data are available from the reactor vessel surveillance program, the margin will be reduced in subsequent shift calculations consistent with Regulatory Guidelines.

2. Reactor Vessel Flange

PT Limits for the reactor vessel flange are examined for heatup and cooldown in accordance with the procedures specified by G-2220 of Appendix G to Section III of the ASME Code. For flange analysis, during normal operations, the following condition is maintained:

$$K_{IR} = 2K_{I\text{primary}} + K_{I\text{secondary}}$$

where, $K_{I\text{primary}} = K_{m\text{pressure}} + K_{m\text{boltup}} + K_{b\text{boltup}}$

$$K_{I\text{secondary}} = K_{b\text{pressure}} + K_t$$

and $K_{m\text{pressure}} = \text{Stress intensity for membrane stress due to pressure,}$

$$= \sigma_{m\text{pressure}} M_m,$$

$$K_{m\text{boltup}} = \text{Stress intensity factor for membrane stress due to boltup,}$$

$$= \sigma_{mboltup} M_m,$$

$$K_{b\text{boltup}} = \text{Stress intensity factor for bending stress due to boltup,}$$

$$= \sigma_{bboltup} M_b,$$

$$\sigma_{m\text{pressure}} = \text{Membrane (hoop) stress due to pressure, psi,}$$

$$\sigma_{m\text{boltup}} = \text{Membrane stress due to boltup, psi,}$$

$$\sigma_{b\text{boltup}} = \text{Bending stress due to boltup, psi,}$$

M_m = Membrane correction factor, $\sqrt{\text{in.}}$,
 $= 1.1 M_K \sqrt{\pi} \sqrt{a/Q}$

M_b = Bending correction factor, $\sqrt{\text{in.}}$,
 $= M_B \sqrt{\pi} \sqrt{a/Q}$

where, M_K and M_B = correction factors defined by Figure A3-1 and A3-2 in Appendix 3 to WRC Bulletin 175.

a = crack depth, in.

Q = the flaw shape factor defined in Appendix 3 to WRC Bulletin 175.

$K_{b_{\text{pressure}}}$ = Stress intensity factor for bending stress due to pressure,

$$= \sigma_{b_{\text{pressure}}} M_b$$

$\sigma_{b_{\text{pressure}}}$ = Secondary bending stress due to pressure, psi.

K_t = Stress intensity factor for thermal stress,

$$= \frac{\alpha E}{(1-\mu)} (\Delta T)$$

where, α = Coefficient of thermal expansion, in/in,

E = Modulus of Elasticity, psi,

μ = Poisson's ratio,

ΔT = Temperature differential causing the stress, °F.

The left side of the above equation (K_{IR}) is calculated for various temperatures. The right side ($2K_{I_{\text{primary}}} + K_{I_{\text{secondary}}}$) is calculated as a function of pressure. The maximum allowable pressure corresponding to a given temperature can then be calculated from this equation. The resulting (maximum) pressure-(minimum) temperature coordinates define the flange PT Limit for a specific heatup or cooldown, at a particular point in plant life. In no case is the flange minimum temperature limit allowed to be less than the flange $RT_{NDT} + 120$ °F during normal operation (when the pressure exceeds 20% of the preoperational system hydrostatic pressure), by

$RT_{NDT} + 90^{\circ}F$ during hydrostatic pressure tests and leak tests, or by $RT_{NDT} + 60^{\circ}F$ when the core is critical, in accordance with Appendix G to 10 CFR 50.

E. In-service Inspection and Hydrostatic Test Curves

P-T Limits for in-service inspection and hydrostatic test are developed in the same manner as for the reactor vessel beltline described in D.1 above. The exception is that a safety factor of 1.5 is applied to the stress intensity factor for membrane stress due to pressure, K_{Im} , rather than 2, as allowed by G-2400 in Appendix G to Section III of the ASME Code. E

F. Core Critical Curves

The core critical curve is intended to provide additional margins of safety during core operation. The limit is defined as $40^{\circ}F$ above the minimum allowable temperature for heatup or cooldown nor lower than the minimum temperature allowable for inservice hydrostatic pressure test, in accordance with paragraph IV.A.3 in Appendix G to 10 CFR 50.

5.3.2.2 Operating Procedures

Pressure-temperature limitations and additional information are described in the Technical Specifications. The pressure-temperature limit curves provided have been prepared in accordance with Appendix G, ASME Code, Section III. Maintenance of Reactor Coolant System pressure and temperature within these prescribed limits ensures that the integrity of the reactor coolant pressure boundary is maintained.

5.3.3 REACTOR VESSEL INTEGRITY

C-E designs reactor pressure vessels and specifies corresponding fabrication methods. C-E has been involved in reactor vessel design and fabrication since the late 1950s, and this proven expertise is reflected in the System 80+ reactor vessel and the satisfactory performance of large numbers of reactor vessels in operating plants. | E | D

Vessel integrity is ensured because proven fabrication techniques are employed and because well characterized steels, which exhibit uniform properties and consistent behavior, are used. The characterization of these materials was established through industrial and governmental studies which examined the prefabrication material properties through to irradiated service operation. In-service inspection and material surveillance programs are also conducted during the service life of the vessel, which further ensure that vessel integrity is maintained.

5.3.3.1 Design

Applicable design codes are found in Table 5.2-1. A schematic of the reactor vessel is shown in Figure 5.3-5. Additional information on design may be found in Section 5.3.1.2. | D

The design permits all required inspections to be performed, and does not preclude access to areas requiring in-service inspection.

5.3.3.2 Materials of Construction

The materials used in the construction of the reactor vessel, as listed in Table 5.2-2, are in accordance with Section III of the ASME Boiler and Pressure Vessel Code.

5.3.3.3 Fabrication Methods

Fabrication of the reactor vessel is described in Section 5.3.1.2. Fabrication processes used in construction of the reactor vessel comply with Sections III and IX of the ASME B&PV Code.

5.3.3.4 Inspection Requirements

Inspection requirements of ASME Code, Section III are discussed in Section 5.3.1.3.

5.3.3.5 Shipment

The requirements of Regulatory Guide 1.38 are followed in the packaging and shipment of the reactor vessel. | I

The reactor vessels are prepared to be shipped by barge or rail to the site, while mounted on the shipping skid used for installation. The vessels are protected by closing all openings (including the top of the vessel) with wooden shipping covers. The closure heads are shipped with separate skids and covers. Vessel surfaces and covers are sprayed with a strippable coating for protection against corrosion during shipping and installation. Prior to the welding of interconnecting piping and installation of insulation, the temporary protective coating is removed by peeling.

5.3.3.6 Operating Conditions

Refer to Sections 3.9 and 4.4 for information on design transients and operating conditions, respectively.

5.3.3.7 In-service Surveillance

The reactor vessel surveillance program is described in detail in Section 5.3.1.6. It is designed on the basis of 10 CFR 50, Appendix H and ASTM E-185-82. Standard reference material to corroborate the post-irradiation surveillance data and precracked Charpy impact specimens to enable determination of fracture toughness properties before and after irradiation are included in the program. Standard Charpy specimens in excess of the number required by E-185-82 are included for key materials to increase the accuracy in defining post-irradiation index temperatures. When combined with the use of highly radiation resistant materials in the beltline of the reactor vessel, this surveillance program provides maximum assurance, consistent with commercial requirements, of the integrity of the reactor pressure vessel in terms of strength and fracture resistance.

TABLE 5.3-1
TOTAL QUANTITY OF SPECIMENS

<u>Type of Specimen</u>	<u>Orientation</u>	<u>Quantity</u>				<u>Totals</u>	
		<u>Base Metal</u>	<u>Weld Metal</u>	<u>HAZ</u>	<u>SRM*</u>		
Drop Weight	Transverse	12	12	12	-	36	
Standard Charpy	Longitudinal	51	-	-	69	120	
	Transverse	114	114	96	-	324	
Precracked Charpy	Longitudinal	39	-	-	-	39	D
	Transverse	39	39	-	-	78	
Compact Tension	1t Transverse	8	8	-	-	16	
	1/2t Transverse	34	34	-	-	68	
Tensile Test	Longitudinal	12	-	-	-	12	E
	Transverse	30	30	-	-	60	
	Total	339	237	108	69	753	D

* Standard Reference Material characterized by Heavy Section Steel Technology (HSST) Program

TABLE 5.3-2

TYPE AND QUANTITY OF SPECIMENS FOR BASELINE TESTS

<u>Type of Specimen</u>	<u>Orientation</u>	<u>Quantity</u>				<u>Totals</u>	
		<u>Base Metal</u>	<u>Weld Metal</u>	<u>HAZ</u>	<u>SRM*</u>		
Drop Weight	Transverse	12	12	12	-	36	
Standard Charpy	Longitudinal	24	-	-	15	39	D
	Transverse	24	24	24	-	72	
Precracked Charpy	Longitudinal	12	-	-	-	12	
	Transverse	12	12	-	-	24	
Tensile Test	Longitudinal	12	-	-	-	12	E
	Transverse	12	12	-	-	24	
Compact Tension	1t Transverse	8	8	-	-	16	D
	1/2t Transverse	4	4	-	-	8	
Total		120	72	36	15	243	

* Standard Reference Material

TABLE 5.3-3

TYPE AND QUANTITY OF SPECIMENS FOR IRRADIATION
EXPOSURE AND IRRADIATED TESTS

<u>Type of Specimen</u>	<u>Orientation</u>	<u>Quantity</u>				<u>Totals</u>	
		<u>Base Metal</u>	<u>Weld Metal</u>	<u>HAZ</u>	<u>SRM*</u>		
Standard Charpy	Longitudinal	27	-	-	54	81	
	Transverse	90	90	72	-	252	
Precracked Charpy	Longitudinal	27	-	-	-	27	D
	Transverse	27	27	-	-	54	
Tensile Test	Transverse	18	18	-	-	36	E
Compact Tension 1/2t	Transverse	30	30	-	-	60	
	Total	219	165	72	54	460	D

* Standard Reference Material

TABLE 5.3-4
TYPE AND QUANTITY OF SPECIMENS CONTAINED IN
EACH IRRADIATION CAPSULE ASSEMBLY

<u>Material</u>	<u>PC_V Capsule Assembly</u>			<u>CT Capsule Assembly</u>		
	<u>Standard Charpy</u>	<u>Precracked Charpy</u>	<u>Tension</u>	<u>Standard Charpy</u>	<u>1/2t Compact Tension</u>	<u>Tension</u>
Base Metal (Longitudinal)	9	9	-	-	-	-
Base Metal (Transverse)	15	9	3	15	10	3
Weld Metal	15	9	3	15	10	3
Heat-Affected (HAZ)	12	-	-	12	-	-
Standard Reference Material (SRM)	9	-	-	9	-	-
	<u>60</u>	<u>27</u>	<u>6</u>	<u>51</u>	<u>20</u>	<u>6</u>

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D

TABLE 5.3-5

CANDIDATE MATERIALS FOR NEUTRON THRESHOLD DETECTORS

<u>Material</u>	<u>Reaction</u>	<u>Threshold Energy (MeV)</u>	<u>Half-Life</u>	
Niobium	Nb ⁹³ (n,n) Nb ^{93m}	0.03	16 years	B
Neptunium	Np ²³⁷ (n,f) Cs ¹³⁷	0.5	30.2 years	
Uranium	U ²³⁸ (n,f) Cs ¹³⁷	0.7	30.2 years	D
Iron	Fe ⁵⁴ (n,p) Mn ⁵⁴	4.0	314 days	
Nickel	Ni ⁵⁸ (n,p) Co ⁵⁸	5.0	71 days	
Copper	Cu ⁶³ (n,α) Co ⁶⁰	7.0	5.3 years	
Titanium	Ti ⁴⁶ (n,p) Sc ⁴⁶	8.0	84 days	
Cobalt	Co ⁵⁹ (n,γ) Co ⁶⁰	Thermal	5.3 years	

TABLE 5.3-6

COMPOSITION AND MELTING POINTS OF
CANDIDATE MATERIALS FOR TEMPERATURE MONITORS

<u>Composition</u> <u>(wt%)</u>	<u>Melting Temperature</u> <u>(°F)</u>
80 Au, 20 Sn	536
90.0 Pb, 5.0 Sn, 5.0 Ag	558
97.5 Pb, 2.5 Ag	580
97.5 Pb, 0.75 Sn, 1.75 Ag	590

TABLE 5.3-7CAPSULE ASSEMBLY REMOVAL SCHEDULE

<u>Capsule</u>	<u>Azimuthal Location</u>	<u>Removal Time</u>	<u>Target Fluence (n/cm²)</u>
1	38°	EOL	-----
2	43°	Standby	-----
3	137°	5-8 EFPY	9.0x10 ¹⁸
4	142°	Standby	-----
5	230°	22-26 EFPY	3.6x10 ¹⁹
6	310°	38-42 EFPY	6.0x10 ¹⁹

E

NOTE: Schedule may be modified to coincide with those refueling outages or schedule shutdowns most closely approximating the withdrawal schedule.

Amendment E
December 30, 1988

SYSTEM 80+™

TYPICAL SURVEILLANCE CAPSULE ASSEMBLY

Figure

5.3-1

Amendment E
December 30, 1988

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CHARPY, FLUX AND COMPACT TENSION
COMPARTMENT ASSEMBLY

Figure

5.3-2

Amendment E
December 30, 1888

SYSTEM 80+™

TEMPERATURE, FLUX, TENSILE AND
CHARPY COMPARTMENT ASSEMBLY

Figure

5.3-3

Amendment E
December 30, 1988

SYSTEM 80+™

LOCATIONS OF
SURVEILLANCE CAPSULE ASSEMBLIES

Figure

5.3-4

Amendment E
December 30, 1988

SYSTEM 80+ PT LIMIT CURVES
EOL (60 YEARS) - HEATUP

Figure

5.3-5a

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December 30, 1988

SYSTEM 80+TM

SYSTEM 80+ PT LIMIT CURVES
EOL (60 YEARS) - COOLDOWN

Figure

5.3-5b

SYSTEM 80+™

SYSTEM 80+
ALLOWABLE HEATUP & COOLDOWN RATES

Amendment E
December 30, 1988

Figure
5.3-6

NOZZLE SCHEDULE	
SERVICE	
COOLANT INLET	4
COOLANT OUTLET	2
CEDM NOZZLES	97
INSTRUMENTATION	61
VENT	1
SEAL LEAK MONITOR	1

10" DVI NOZZLE
(8.5" I.D.)

30" ID
INLET
NOZZLE

9 1/16"
CLADDING 1/8" MIN

CEDM NOZZLE

CLOSURE
STUD

42" ID
OUTLET
NOZZLE

EFFECTIVE
CORE
DIAMETER

48'

CORE STOP

SNUBBER

FLOW BAFFLE

INSTRUMENT
NOZZLES

Amendment E
December 30, 1988

SYSTEM 80+™

REACTOR VESSEL

Figure
5.3-7

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMPS

The reactor coolant pumps provide sufficient forced circulation flow through the Reactor Coolant System to assure adequate heat removal from the reactor core during power operation. A low limit on reactor coolant pump flow rate (i.e., design flow) is established to assure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded. Design flow is derived on the basis of the thermal-hydraulic considerations presented in Section 5.2.

The reactor coolant pump and motor assembly in conjunction with the flywheel, provide sufficient coastdown flow following loss of power to the pumps to assure adequate core cooling.

The reactor coolant pump pressure boundary is designed for the transients given in Section 3.9 so that the ASME Code Section III allowable stress limits are not exceeded for the specified number of cycles. Stress criteria concerning earthquake and pipe rupture conditions are presented in Section 3.9.3.

The design overspeed of the reactor coolant pump is 125 percent of normal speed.

5.4.1.1 Pump Flywheel Integrity

A. The material used to manufacture the flywheel of the reactor coolant pump motor will be produced by a commercially acceptable process that minimizes flaws, such as the vacuum melt and degassing process. This provides adequate fracture toughness properties under reactor operating conditions. The acceptance criteria for flywheel design will be compatible with the safety philosophy of the Pressure Vessel Research Committee (PVRC) of the Welding Research Council (WRC) primary coolant pressure boundary criteria as appropriate considering the inherent design and functional requirement differences between the pressure boundary and the flywheel. | D

1. The reference nil-ductility transition temperature (RT_{NDT}) of the material, as determined per ASME Code NB-2331(a), will be no greater than 10°F. | D
2. The Charpy V-notch (Cv) upper shelf energy level, in the "weak" (Wr) direction, as obtained per ASTM-A-370 will be no less than 50 ft-lb. A minimum of three Cv specimens will be tested from each plate or forging.

3. The minimum fracture toughness of the material at the normal operating temperature of the flywheel will be equivalent to a dynamic stress Intensity factor (K_{IC} dynamic) of at least 100 ksi/ $\sqrt{\text{in}}$. Compliance will be demonstrated by either of the following:
 - a. Testing of the actual material of the flywheel to establish the K_{IC} (dynamic) value at the normal operating temperature, or
 - b. Use of a lower bound fracture toughness curve obtained from tests on the same type of material. The curve will be translated along the temperature coordinate until the K_{IC} (dynamic) value of 45 ksi/ $\sqrt{\text{in}}$. is indicated at the NDT of the material, as obtained from drop-weight tests.
 4. Each finished flywheel will be subjected to a 100 percent volumetric ultrasonic inspection from the flat surface per ASME BPVC Section III.

This inspection will be performed on the flywheel after final machining and the overspeed test.
 5. If the flywheel is flame cut, at least 1/2 inch of stock will be left on the outer and bore radii, for machining to final dimensions.
 6. The flywheel will be subjected to a magnetic particle or liquid-penetrant examination per "Section III" before final assembly. The inspection will be performed on finished machined bores, keyways, and on both flat surfaces to a radial distance of 8 inches minimum beyond the final largest machined bore diameter but not including small drilled holes. There will be no stress concentrations such as stamp marks, center punch marks, or drilled or tapped holes within 8 inches of the edge of the largest flywheel bore.
- B. The flywheels will be designed to withstand normal operating conditions, anticipated transients, and the largest mechanistic pipe break size remaining after application of leak before break as described in Section 3.6, combined with the Safe Shutdown Earthquake.

The following criteria will be satisfied:

1. The combined stress, both centrifugal and interference, at normal operating speed will not exceed one-third of the minimum specified yield strength or 1/3 of the measured yield strength in the weak direction of the

material if appropriate tensile tests have been performed on the actual material of the flywheel. | I

2. The design speed of the flywheel will be 125 percent of normal operating speed.

The lowest of the critical speeds of the flywheel will be at least 10% above the highest anticipated overspeed of the pump. The highest anticipated overspeed is predicted for the largest break size remaining after application of leak before break as described in Section 3.6. | D

3. The combined centrifugal and interference stresses at the design speed will be limited to two-thirds of the minimum specified yield strength or $2/3$ of the measured yield strength in the weak direction if appropriate tensile tests have been performed on the actual material of the flywheel. Design speed is defined as 125 percent of normal operating speed. | I

4. The motor and pump shaft or bearings and coupling will withstand any combination of normal operating loads or anticipated transients, and the largest remaining pipe break after application of leak before break as described in Section 3.6, combined with the Safe Earthquake Shutdown. | D

Each flywheel will be tested at design speed, 125 percent of normal operating speed, as defined in B.2 above.

The flywheel will be accessible for 100 percent in-place volumetric ultrasonic inspection. The flywheel-motor assembly is designed to allow such inspection with a minimum of motor disassembly. The in-service inspection program will include ultrasonic examinations of the areas of high stress concentration at the bore and keyway at about 3 $1/3$ year intervals, during the refueling or maintenance shutdown coinciding with the in-service inspection schedule as required by the ASME Code, Section XI. Removal of the flywheel is not required. | I

Liquid penetrant examination or magnetic particle methods and 100% volumetric examination by ultrasonic methods will be conducted at about ten-year intervals during the plant shutdown coinciding with the in-service inspection schedule as required by the ASME Code, Section XI. | I

5.4.1.2 **Description**

Table 5.4.1-1 lists the principal parameters of the reactor coolant pumps and Figure 5.4.1-1 depicts the arrangement of the pump and motor. Reactor coolant pump supports are discussed in Section 5.4.14. The pump piping and instrument diagram is given in Figure 5.1.2-2.

The four reactor coolant pumps are vertical, single stage bottom suction, horizontal discharge, motor-driven centrifugal pumps. The pump impeller is keyed and locked to its shaft. Pump shaft alignment is maintained by a water lubricated radial bearing within the pump and by radial and thrust bearings located in the motor stand. The pump and motor shafts are directly connected by a coupling.

The shaft seal assembly consists of two face-type, mechanical seals in series, with controlled leakage bypass to provide the same pressure differential across each seal. The seal assembly is designed for 2500 psi differential and to reduce the leakage pressure from Reactor Coolant System pressure to the volume

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control tank pressure. A third, face-type, low-pressure vapor seal at the top is designed to withstand system operating pressure when the pumps are not operating. The leakage past the second pressure seal and the controlled leakage is piped to the volume control tank in the Chemical and Volume Control System. Leakage past the low-pressure vapor seal is collected and piped to the reactor drain tank.

The temperature of the water in the seal assembly is maintained within acceptable limits by a water-cooled heat exchanger. Water is also injected into the seal area from an external seal injection system. The performance of the shaft seal system is monitored by pressure and temperature sensing devices in the seal system. The seal assembly can be replaced without draining the pump casing or removing the shaft.

The seal assemblies are designed to limit seal leakage plus controlled bypass flow to approximately the values given below:

Seal leakage plus controlled bypass flow, per pump:

All seals functioning (normal)	3.9 gpm
One seal functioning (abnormal)	5.4 gpm

The motor is sized for continuous operation at the flows resulting from four-pump or one-pump operation with 1.0 to 0.74 specific gravity water. The motors are designed to start and accelerate to speed under full load with a drop to 80 percent of normal rated voltage at the motor terminals.

Each motor is provided with an anti-reverse rotation device. The device is designed to prevent impeller rotation in the reverse direction due to each of the following conditions: motor starting torque, if the motor was incorrectly wired for reverse rotation; and, reactor coolant flow through the pump in the reverse direction due to the largest remaining pipe break after application of leak before break as described in Section 3.6, which could result in reverse flow through the pump.

5.4.1.3 Evaluation

The reactor coolant pumps are sized to deliver flow that equals or exceeds the design flow rate utilized in the thermal hydraulic analysis of the Reactor Coolant System. Analysis of steady-state and anticipated transients is performed assuming the minimum design flow rate. Tests are performed to evaluate reactor coolant pump performance during the post-core load hot functional testing to verify adequate flow.

Leakage from the pump via the pump shaft is controlled by the shaft seal assembly. Reactor coolant entering the seal chambers

is cooled and collected in closed systems to prevent reactor coolant leakage to containment. Instrumentation is provided to monitor seal operation.

The design speed of the flywheel is 125 percent of normal speed. An overspeed test of each flywheel at the design speed is performed prior to assembly. Refer to pump flywheel integrity Section 5.4.1.1.

In the event of a break which is not eliminated by leak before break which would result in reverse flow through the pump, the anti-reverse rotation device prevents impeller rotation in the reverse direction. In the event of a break which is not eliminated by leak before break and which could result in increased flow through the pump tending to accelerate the pump impeller, the highest predicted pipe break induced overspeed is less than the lowest critical speed of the flywheel. | D

The pump and motor oil lubricated bearings are lubricated by internal oil systems. Each bearing assembly has its own internal oil system consisting of either an oil bath or force-feed type system. During normal operation, no external pumps will be required because pumping action is accomplished by internal pumping devices. Lubricating oil is cooled by cooling coils submerged in the oil sumps. Both sumps and cooling coils are internal to the motor structural frame and are designed for Seismic Category I operation, and use the intent of the ASME Boiler and Pressure Vessel Code, Section III, Class 3 as a guide for design and construction. This is established within the Combustion Engineering Topical Report, CENPD-201-A, which demonstrates the reactor coolant pump performance during a loss of component cooling water incident. Although the pump-motor assembly operation is not considered necessary for plant safety, this design minimizes the direct effects of seismic events on the reactor coolant pump and motor assembly oil lubricating systems so that adequate coast down characteristics are not detrimentally affected.

Bearing metal temperatures, oil flow and/or pressure, oil levels, cooling water flow and/or pressure are continuously monitored and alarmed in the control room.

In the unlikely event that component cooling water to the reactor coolant pump and motor oil lubricating systems is not available or that an oil leak occurs during operation, the operator is alerted as soon as cooling water to the oil system is lost and has a time period of at least 30 minutes in which to reduce power, if necessary, and isolate cooling water and shut the reactor coolant pump motor assembly down to prevent bearing seizure.

This time limit is established in the Topical Report CENPD-201-A. Combustion Engineering will perform a test to verify the analysis in Appendix A. The component cooling water will be secured to the pump lubrication oil sumps and data taken so as to demonstrate the heat up rate of the oil sump up to a maximum sump temperature of 200°F. A test securing cooling water to the pump seals will also be performed to demonstrate the seals operability as detailed in Combustion Engineering Topical Report CENPD-201-A. In the remote possibility of a simultaneous loss of component cooling water to all reactor coolant pump motor assemblies, 30 minutes is adequate to secure the plant and maintain the normal coast down capabilities of the reactor coolant pump motor assemblies.

During a loss of component cooling water event, it is unlikely that a shaft seizure due to bearing failure will occur for the following reasons:

- A. The design is such that the heat generated in the bearing normally carried away by the cooling water, is transferred by alternate paths. The lube oil sump baths surrounding the bearings, the stagnant cooling water remaining in the heat exchanger coils, and the bearing and sump assembly metal masses, all act as heat sinks. Also conduction down the pump shaft and outer sump shell radiation help to reduce the temperature rise.
- B. The rotation of the bearing assemblies insures adequate oil flow and mixing of heated oil to insure the heat transfer as described in A.
- C. In the event that the oil temperature rises such that the viscosity degrades significantly, the design of the thrust bearing continues to produce a hydrodynamic film so as to preclude metal to metal contact.
- D. Operation and test experience has demonstrated that the reactor coolant pump motor assembly will operate without cooling water to the lubricating oil system for at least the calculated 30 minute time period.

Should an oil leak occur, redundant instrumentation will alert the operator to shut down the reactor coolant pump motor assembly and thereby avoid bearing damage.

In the event of an oil leak, the separation of lubrication systems would limit the problem to a single reactor coolant pump.

The loss of the oil in the bearing oil reservoir would not result in bearing seizure for the following reasons:

- A. Temperature and oil level monitors will provide appropriate indication of an abnormal condition.
- B. The vibration monitoring device furnished on the pump will respond to bearing degradation and allow the operator to shutdown the pump.
- C. If the above protective measures fail, the high torque produced by the motor will cause a slow breakdown of the bearings and not a rapid shaft seizure. Industry experience indicates that the babbitt bearing surfaces wear away and the bearing pads and sleeves will be badly worn but the shaft will continue to rotate.

If the extremely remote possibility of bearing seizure occurs while the reactor coolant pump motor assembly is in operation, adequate flow to the core is available from the other reactor coolant pump motor assemblies as demonstrated by the one pump loss of flow study.

Figure 5.1.2-2 shows a separate oil lift system which is required for start-up of the pump assembly. The oil lift system furnishes high pressure oil to the pump assembly thrust bearings, thereby lifting the rotor and reducing bearing friction during pump start-up. Interlocking devices are furnished which prevent pump start-up until oil lift flow is established. The oil lift system is automatically shutdown when the pump reaches full speed. Since oil lift is not required during normal operation, an oil leak in this system will not cause a bearing failure.

5.4.1.4 Tests and Inspections

The reactor coolant pump pressure boundary is nondestructively inspected as required by ASME Section III for Class 1 components. The pump casing inspections include complete radiography and liquid penetrant or ultrasonic testing. The pump receives a hydrostatic pressure test in the vendor's shop and with the Reactor Coolant System. Inservice inspection of the pump pressure boundary will be performed during plant life in accordance with ASME Section XI.

The pump assembly is performance tested in the vendor's shop over at least the normal operating range in accordance with the Standards of the Hydraulic Institute. The tests also demonstrate ability of the pumps to function under the various operating conditions specified. Tests commonly performed are hot and cold

performance and stop-start cycling. Special testing will also be performed on one pump. Such testing will include loss of cooling and/or seal injection water. Vibrations are monitored at several places on the pump during shop testing.

In addition to meeting an absolute criterion for vibration amplitude, the test results are examined for evidence of critical speed problems.

The pump motors undergo a "routine" test in accordance with NEMA MG-1. This test also confirms that the motors are within their vibration limits. At least one motor is tested by being used as the driver for the pump assemblies, during the pump manufacturer's shop testing.

The following testing may also be performed where significant seal experience is lacking to develop confidence in the sealing system:

- o Seal materials testing for suitability in reactor coolant environment.
- o Long term testing of an entire seal assembly.

To the greatest extent practicable, all conditions of operation within the reactor coolant pump will be duplicated.

Reactor coolant pump flywheel inspections and testing are described in Section 5.4.1.1.

B

TABLE 5.4.1-1

REACTOR COOLANT PUMP PARAMETERS

Number of Units	4	
Type	Vertical, single stage centrifugal	
Design Total Dynamic Head, ft*	365	
Design Flow, gpm	111,400	
Design Pressure, psia	2500	
Design Temperature, F	650	
Normal Operating Pressure, psia	2250	
Normal Operating Temperature, F*	558	E
NPSH Required (at design flow), ft*	220	
Suction Temperature, F*	557	E
Water Volume, each, ft ³	134	
Weight (including motor), dry, lbs.	279,000	
Shaft Seals	Mechanical Face Seals	
Pump Speed, rpm*	1190	
Motor Synchronous Speed, rpm	1200	
Motor Type	AC Induction	
Horsepower, hot*	9000	
cold	12,000	
Rated Brake Horsepower	12,000	
Voltage	13,200	
Phase	3	
Frequency	60 Hz	
Insulation Class	F	
Starting Current, at 100% Voltage, amps	3,000	

*Parameters are related to four-pump, full power operating conditions.

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March 31, 1988

<p>REACTOR COOLANT PUMP</p>	<p>Figure 5.4.1-1</p>
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5.4.2 STEAM GENERATORS

5.4.2.1 Design Bases

The two steam generators are designed to transfer 3817 MWt from the RCS to the secondary system, producing approximately 17.12×10^6 lb/hr of 1000 psia saturated steam, when provided with 450°F feedwater. Moisture separators and steam driers in the shell side of the steam generator limit the moisture content of the steam to maximum 0.25 wt%* during normal operation at full power. The steam generator design parameters are listed in Table 5.4.2-1. The steam generators, including the tubes, are designed for the RCS transients listed in Section 3.9.1 so that the code allowable stress limits are not exceeded for the specified number of cycles. All transients have been established based on conservative assumptions of operating conditions in consideration of supportive system design capabilities. The steam generators will be capable of sustaining the following additional design transients without exceeding code allowable stress limits:

- A. Fifteen secondary side hydrostatic tests with secondary side pressurized to 1-1/4 times the design pressure and the primary side pressurized so that the tube differential pressure does not exceed 820 psid (test condition);
- B. Three hundred secondary side leak tests with the secondary side pressurized from 820 psia to design pressure, with the primary side pressurized so that tube differential pressure (secondary to primary) does not exceed 820 psid (test condition);
- C. Ten thousand cycles of adding 40°F feedwater at 800 gpm to the steam generator through the downcomer feedwater nozzle when at hot standby conditions (normal condition);
- D. Seven hundred and fifty cycles of adding 40°F feedwater at 800 gpm to the steam generator through the downcomer feedwater nozzle during loading conditions (normal condition);
- E. Seven hundred and fifty cycles of adding 100°F feedwater at 800 gpm to the steam generator through the downcomer feedwater nozzle during loading conditions (normal condition);

* However, steam dryers are capable of providing 99.9% quality steam.

- F. Seven cycles of adding 40°F feedwater at 800 gpm to the steam generator through the downcomer feedwater nozzles during a steam line break. This provides for one steam line break incident with the emergency feedwater cycled a maximum of seven times (faulted condition); D
- G. Four hundred and twenty cycles of adding 40°F feedwater at 850 gpm to the steam generator through the downcomer feedwater nozzles with the flow initiated 30 seconds after a loss of normal feedwater. This provides for 60 loss of normal feedwater incidents with the emergency feedwater cycled a maximum of seven times (upset condition); B
- H. Six thousand pressure transients of 85 psi across the primary divider plate in either direction caused by starting and stopping reactor coolant pumps (normal condition). B

The steam generator was designed to ensure that critical vibration frequencies are well out of the range expected during normal operation and during abnormal conditions. The tubing and tubing supports are designed and fabricated with considerations given to both secondary side flow induced vibration and reactor coolant pump induced vibrations. In addition, the steam generator assemblies are designed to withstand the blowdown forces resulting from steam and feedwater line breaks (see Appendix 5B and 5C). The two accidents are not considered simultaneously. I

The steam generator tubes are Ni-Cr-Fe alloy 690, 3/4-inch OD, with 0.042-inch nominal wall thickness.

A steam generator tube rupture incident is a penetration of the barrier between the reactor coolant system and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would mix with water in the shell side of the affected steam generator. This radioactivity would be transported by steam to the turbine and then to the condenser, or directly to the condenser via the Turbine Bypass System. Noncondensable radioactive gases in the condenser are removed by the main condenser's evacuation system and discharged to the plant ventilation system.

Experience with nuclear steam generators indicates that the probability of complete severance of a tube is remote. A double-ended rupture has never occurred in a steam generator of this design. The more probable modes of failure, which result in smaller penetrations, are those involving the occurrence of

pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet. Detection and control of steam generator tube leakage is described in Section 5.2.

The concentration of radioactivity in the secondary side of the steam generators is dependent upon the concentration of radionuclides in the reactor coolant, the primary-to-secondary leak rate, and the rate of steam generator blowdown. The expected specific activities in the secondary side of the steam generators during periods of normal operation are given in Section 11.2.

The recirculation water within the steam generators will contain volatile additives necessary for proper chemistry control. These and other chemistry considerations of the main steam system are discussed in Section 10.3.5.

5.4.2.2 Description

The steam generator is illustrated in Figure 5.4.2-1. Moisture-separating equipment in the shell side of the steam generators limits moisture content of the exit steam. Manways and handholes are provided for access to the steam generator internals. Reactor coolant enters at the bottom of each steam generator through the single inlet nozzle, flows through the U-tubes, and leaves through the two outlet nozzles. A vertical divider plate separates the inlet and outlet plenums in the lower head.

The steam generator with integral economizer is in most respects similar to earlier U-tube recirculating steam generators. The basic difference is that instead of introducing feedwater only through a sparger ring to mix with the recirculating water flow in the downcomer channel, feedwater is also introduced into a separate, but integral section of the steam generator. A semi-cylindrical section of the tube bundle, at the cold leg or exit end of the U-tubes, is separated from the remainder of the tube bundle by vertical divider plates. Feedwater is introduced directly into this section and pre-heated before discharge into the evaporator section.

The lower portion of the evaporator section and the downcomer channel occupy only one-half of the steam generator cross-section. The effect of this non-symmetry is considered in calculation of recirculation ratio, internal flow considerations, and in design of tube support structures.

The steam-water mixture leaving the vertical U-tube heat transfer surface enters the separators which impart a centrifugal motion to the mixture and separate the water particles from the steam. The water exits from the perforated separator housing and recirculates through the downcomer channel to repeat the cycle. Final drying of the steam is accomplished by passage of the steam through corrugated plate dryers.

A recirculation system allows the circulation of water through the steam generator during wet lay-up and the addition of chemical cleaning agents. The recirculation system consists of a distribution ring located above the tube bundle below the normal water level with connecting piping to the blowdown system. This piping requires a nozzle penetration through the pressure boundary (shell). Suction is taken at the blowdown nozzle and recirculated through the distribution ring. The recirculating header can effect a rapid changeover of the steam generator inventory should a chemical intrusion occur requiring rapid removal of impurities.

Primary head draining capability is provided by channel head drains. Tubesheet drains allows secondary side draining. Such drain capability enhance access for inspection and maintenance.

The System 80+ Standard Design steam generators have 21" manways (Figure 5.4.2-1). On the primary side there is one manway for the cold leg side and another for the hot leg side. Manway locations are optimized for use of remote manipulators for inspection and maintenance. On the secondary side, two manways are provided to allow access to the separator dryer area. In addition, an internal hatch provides access to the top of the tube bundle. These openings allow inspection, which provides information on the condition of separation equipment, feedwater ring, and top of the tube bundle.

At the tubesheet elevation, two 8" handholes are included to provide access for tubesheet sludge lancing as well as for inspection of the downcomer annulus. These handholes can be used to remotely inspect for, and retrieve, loose parts. Access for eddy current testing is through the primary-side manways.

The blowdown system for System 80+ (Section 10.4.8) is designed for intermittent blowdown of up to 10% of feed flow. The system is effective in removing particulate accumulations at the tubesheet. Access for sludge lancing is also provided via the 8" handholes.

The pressure drop from the steam generator feedwater nozzles to the steam outlet nozzle including the economizer is approximately 40 psi.

The steam generator supports are described in Section 5.4.14.

5.4.2.3 Economizer Integrity

The economizer section is designed in full consideration of operating transients, startup and standby operation, and accident conditions such as loss of feedwater flow and feedwater line break. The structural design of the various parts is adequate to withstand the thermal and pressure loadings from these various conditions, consistent with the appropriate load classifications and design rules in the ASME Code, Section III, see Appendix G.

The components of the steam generator economizer section have been designed for the primary stresses which occur due to the blowdown associated with a feedline break. The divider plates, which separate the economizer region from the evaporator region of the secondary side, are supported from the vessel shell and the central cylindrical support welded to the tubesheet. This divider cylinder becomes an extension of the primary tubesheet stay cylinder, though less massive, and extends the full height of the economizer. The tube support/flow baffle plates are supported from the vessel shell, the divider cylinder and the tubesheet via an array of support rods. The support rods, which also serve as support plate spacers are solid and designed for either tensile or buckling loads. Thin plates which may collapse when subjected to differential pressure are not used. | I

5.4.2.4 Steam Generator Materials

The pressure boundary materials used in the construction of the steam generator are listed in Table 5.2-2. These materials are in accordance with the ASME Boiler and Pressure Vessel Code, Section III. Code cases used in the fabrication of the steam generator are discussed in Section 5.2.1.

The Class 1 components of the steam generator will meet the fracture toughness requirements of the ASME code. An additional discussion of fracture toughness testing is included in Section 5.2.3.

Discussion of the techniques used to maintain cleanliness during final assembly and shipment are discussed in Section 5.2.3. | I

5.4.2.4.1 Steam Generator Tubes

The method of fastening tubes to the tube sheet conforms with the requirements of Sections III and IX of the ASME Code. Tube expansion into the tube sheet is total with no voids or crevices occurring along the length of the tube in the tube sheet. After the tubing is installed in the steam generator and properly positioned, a seal weld is performed between the tube and cladding on the primary side of the tubesheet. Subsequently, expansion of the tube in the tubesheet is performed using an explosive charge ("expansion"). Tolerances on charge length and strength are specified so that the crevice between tube and tubesheet is closed through the full depth of the tubesheet and so that no crevice occurs on the secondary face of the tubesheet.

Operating experience with this joint has been trouble-free from the standpoint of secondary side corrosion at the tube-to-tubesheet interface and from the standpoint of primary side stress corrosion cracking in the expanded portion of the tube.

The "expansion" procedure creates residual stress in the transition zone between the expanded and unexpanded regions of tubing. Residual stress measurements have been performed on this transition zone by X-ray diffraction. The results verify the absence of any high residual tensile stress in the transition zone. Material specifications such as the use of thermally treated tubing, welding, and fabrication procedures preclude the need for complete-bundle stress relief after assembly.

Operating C-E steam generators have experienced the following corrosion degradation mechanisms: phosphate wastage, sulfate wastage, intergranular attack, secondary side stress corrosion cracking, and pitting and denting resulting from tube support corrosion. With respect to these phenomena, the most important design feature of the System 80+ Standard Design steam generators is the selection of tubing and tubing support materials. For the System 80+ steam generators, Ni-Cr-Fe Alloy 690 in a thermally treated (TT) condition is specified for the tubes. For the tube supports, Stainless Steel 409 material is specified (see Table 5.2-2).

Volatile chemistry (discussed in Section 10.3.5) has been successfully used to minimize corrosion in all C-E steam generators that have gone into operation since 1972. Removal of solids from the secondary side of the steam generator is discussed in Section 10.4.8.

There are three types of structures within the System 80+ Standard Design steam generators which support the tubes. These are the horizontal grid or "eggcrate", the vertical supports, and the diagonal supports, all of which are fabricated from Stainless Steel 409. One of the design considerations for these supports is prevention of dryout at support locations. With one exception, all tube supports in the System 80+ steam generator are constructed of flat strips which present a flat surface to the tube. The one exception is the flow distribution plate just above the entrance to the economizer section of the tube bundle. At this location secondary water is subcooled and, therefore, dryout will not be experienced.

The eggcrate may have three different configurations depending upon its location in the evaporator--a full circular structure, a half circular structure, or a structure bounded by the circumference and a chord. An eggcrate is composed of strips intersecting at an angle of 60 degrees and joined together at the outer and inner perimeters with a pair of square bars on top and bottom. The strips alternate between a 2-inch slotted type and a 1-inch unslotted type; both are 0.090 inches thick (see Figure 5.4.2-4). The eggcrates themselves are supported and spaced by tie rods located throughout the tube bundle and by weldment to the tube bundle shroud.

The eggcrates form an open lattice and thus minimize the potential for local dryout conditions. The number of eggcrates is selected to maintain the natural frequency of the tubes significantly higher than the exciting frequencies induced by cross flow at the fluid entrances to the bundle. Both analysis and test results have been applied to define spacing which precludes vibration induced damage (fretting and wear). In addition, careful attention is paid to localized flow path details where velocities may be higher than nominal.

The vertical supports (see Figure 5.4.2-5) are assembled concurrently with tube installation and are comprised of vertical, slotted 2-inch strips intersecting with horizontal 0.5 inch strips; both 0.90 inches thick. The assembly is bounded about the periphery by either square bars or custom shaped plates depending on the location. These vertical supports in the bend region are made of perforated strips to enhance their free flowing nature. As shown on Figure 5.4.2-3, the vertical supports are also integrated with perforated diagonal strips which provides thorough vibration support of the bend region without compromising the free flowing character of the supports.

5.4.2.5 Tests and Inspections

Prior to, during and after fabrication of the steam generator, nondestructive tests based upon Section III of the ASME Code are performed.

Initial hydrostatic tests of the primary and secondary sides of the steam generator were conducted in accordance with ASME Code, Section III. Leak tests were also performed. Following satisfactory performance of the hydrostatic tests, magnetic-particle inspections are made on all accessible welds.

Inservice inspection of the steam generator is described in Section 5.2.4.

TABLE 5.4.2-1

(Sheet 1 of 2)

STEAM GENERATOR PARAMETERS^(a)

Parameter	Value	
Number of units	2	
Heat transfer rate per SG, Btu/hr		
w/0.2% Blowdown ^(a)	6.517 x 10 ⁹	B
w/1.0% Blowdown ^(a)	6.530 x 10 ⁹	
Number of Tubes per SG	12,113	I
Average Tube Length per SG, ft	60.2	
Heat Transfer Area per SG (approximate), ft ²	146,450	
<u>Primary Side</u>		
Design pressure/temperature (psig/°F)	2485/650	
Coolant inlet temperature, °F	615	B
Coolant outlet temperature, °F	558	I
Coolant flow rate, each, lb/hr	82.9 x 10 ⁶	E
Coolant volume at 68F each, ft ³	2800	B
Tube size, OD, in.	0.75	
Tube thickness, nominal, in.	0.042	
<u>Secondary Side</u>		
Design pressure/temperature (psia/°F)	1200/570	
Steam pressure, psia	1000	B
Steam flowrate (at 0.25% moisture) per SG, lb/hr	8.56 x 10 ⁶	
Feedwater temperature at full power, °F	450	
Moisture carryover, weight maximum, %	0.25	
Primary inlet nozzle, No./ID, in.	1/42	
Primary outlet nozzle, No./ID, in.	2/30	

TABLE 5.4.2-1

(Sheet 2 of 2)

STEAM GENERATOR PARAMETERS^(a)

<u>Parameter</u>	<u>Value</u>	
Steam nozzle, No./ID, in.	2/28	
Feedwater nozzles, No./size/schedule (Economizer)	2/14/80	
Feedwater nozzles, No./size/schedule (Downcomer)	1/6/80	
Overall heat transfer coefficient		
Evaporator, Btu/hr-ft ² -°F	1371	B
Economizer, Btu/hr-ft ² -°F	593	
Liquid Inventory per SG (approximate), lbm	115,000	I
Boil Dry Time From Normal Level (approximate), min.	54	
Boil Dry Time From Low Level (approximate), min.	31	
(a) Blowdown w/12% quality.		B

NO.	SERVICE	NO. REQ'D
1	PRIMARY INLET	1
2	PRIMARY OUTLET	2
3	DOWNCOMER FEEDWATER	1
4	STEAM OUTLET	2
5	BLOWDOWN	2
6	PRIMARY MANWAY	2
7	SECONDARY MANWAY	2
8	HANDHOLE	2
9	ECONOMIZER FEEDWATER	2
10	RECIRCULATION NOZZLE	1

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SYSTEM 80+™

STEAM GENERATOR

Figure

5.4.2-1

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SYSTEM 80+™

SYSTEM 80+ STEAM GENERATOR ECONOMIZER
AND LOWER TUBE BUNDLE REGION

Figure

5.4.2-2

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SYSTEM 80+™

SYSTEM 80+ STEAM GENERATOR COLD SIDE
RECIRCULATING FLUID ENTRANCE REGION

Figure

5.4.2-3

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SYSTEM 80 TM +

STEAM GENERATOR TUBE EGGRATE SUPPORT

Figure

5.4.2-4

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SYSTEM 80TM +

STEAM GENERATOR TUBE VERTICAL SUPPORTS

Figure

5.4.2-5

5.4.3 REACTOR COOLANT PIPING

5.4.3.1 Design Basis

Applicable design codes are found in Table 5.2-1. The reactor coolant loop piping is designed and analyzed for all transients specified in Section 3.9.1. In addition, those nozzles subjected to local thermal transients, caused by fluid entering the Reactor Coolant System from an auxiliary system, are analyzed to ensure that the nozzles can accommodate the additional transients.

In addition to being specified as Seismic Category I, the piping is designed to ensure that critical vibration frequencies are well out of the range expected during normal operation and during abnormal conditions. Additional presentations relating to seismic and dynamic analysis and criteria for the reactor coolant piping is contained in Sections 3.7.2 and 3.9.2, respectively.

5.4.3.2 Description

Each of the two heat transfer loops contains five sections of pipe: one 42-in. internal diameter pipe between the reactor vessel outlet nozzle and steam generator inlet nozzle, two 30-in. internal diameter pipes from the steam generator's two outlet nozzles to the reactor coolant pumps suction nozzle, and two 30-in. internal diameter pipes from the pumps discharge nozzle to the reactor vessel inlet nozzles. These pipes are referred to as the hot leg, the suction legs, and the cold legs, respectively. The other major pieces of reactor coolant piping are the surge line, a 12-in. pipe between the pressurizer and the hot leg, and the spray line, a 4-inch pipe at the pressurizer end reduced to a 3-inch pipe and connected to two cold legs.

The 42-in. and 30-in. pipe diameter are selected to obtain coolant velocities which provide a reasonable balance between erosion-corrosion, pressure drop, and system volume. The surge line is sized to limit the frictional pressure loss through it during the maximum in-surge so that the pressure differential between the pressurizer and the heat transfer loops is no more than 5 percent of the system design pressure. The spray line sizing is discussed in Section 5.4.10.

To reduce the amount of field welding during plant fabrication, the 42-in. and 30-in. pipes are supplied in major pieces, complete with shop-installed instrumentation nozzles and connecting nozzles to the auxiliary systems. Where required, the nozzles are supplied with safe ends to facilitate field welding of the connecting piping.

Flow restricting orifices (7/32" dia. x 1" long) are provided in the nozzles for the RCS instrumentation and sampling lines to limit flow in the event of a break downstream of a nozzle.

5.4.3.3 Materials

The materials used in the fabrication of the piping are listed in Table 5.2-2. These materials are in accordance with the ASME Code, Section III. The provisions taken to control those factors that contribute to stress corrosion cracking are discussed in Section 5.2.3.

Fracture toughness of the reactor coolant piping is discussed in Section 5.2.3.

5.4.3.4 Tests and Inspections

Prior to, during and after fabrication of the reactor coolant piping, nondestructive tests based upon Section III of the ASME Code were performed. In addition, the fully assembled reactor coolant system is hydrostatically tested in accordance with the Code.

Inservice inspection of the reactor coolant system piping is discussed in Section 5.2.4.

5.4.4 MAIN STEAM LINE RESTRICTIONS

The steam generator outlet nozzles are one piece forgings with an integral venturi type flow restrictor. The venturi section of the nozzle is designed to reduce the flow area by 70%.

5.4.5 MAIN STEAM LINE ISOLATION SYSTEM

The Main Steam Line Isolation System (see Section 10.3.2 for more details) is composed of portions of the Main Steam System and the Engineered Safety Features Actuation System. Discussed here are those portions of these systems that respond to a Main Steam Isolation Signal, as defined in Section 7.3. A discussion of radiological considerations is provided in Section 12.3. | 1

5.4.5.1 Design Bases

- A. The Main Steam Line Isolation Valves are designed to isolate the steam generators and the main steam lines in the event of a main steam line rupture.

- B. The Main Steam Line Isolation Valves are designed to perform containment isolation functions for the main steam lines in the event of a design basis accident, as discussed in Section 6.2.4. In the event of a steam line break outside the containment, the isolation function serves to reduce the potential leakage of radioactivity to the environment.
- C. The Main Steam Isolation Valves are designed to isolate the main steam lines and the steam generators as required for maintenance.

5.4.5.2 System Design

5.4.5.2.1 General Description

Each of the four main steam lines is provided with a power-actuated Main Steam Isolation Valve designed to stop flow from either direction when it is tripped closed. Each valve is located outside containment and is provided with means of actuation from the Engineered Safety Features Actuation System, meeting the requirements of IEEE Standard 279.

The logic circuitry required to isolate the main steam lines is discussed in Section 7.3. The main steam system valves and arrangement are discussed in Section 10.3.2.3.2.

5.4.5.2.2 Component Description

The main steam isolation system consists of the Main Steam Isolation Valves and their associated controls and instrumentation. The Main Steam Isolation Valves are remotely operated valves designed to either fail closed or be guaranteed to close upon receipt of a Main Steam Isolation Signal. The Main Steam Isolation Valves can be monitored and controlled locally and in the control room.

5.4.5.2.3 System Operation

The Main Steam Isolation Valves are designed to isolate the main steam lines and the steam generators as required during operation and under accident conditions.

A steam line break inside containment would result in a pressure rise in the containment. Reverse flow protection is also achieved through the Main Steam Isolation Valves. To achieve reverse flow protection in the case of the main steam pipe rupture, the valve is fully closed within 5.0 seconds from receipt of the initiating signal.

The main steam line isolation system components are qualified to serve in the environment specified in Section 3.11.

5.4.5.3 Design Evaluation

Design evaluations are listed to correspond with the design bases listing.

- A. The Main Steam Isolation Valves are capable of isolating the steam generators within 5.0 seconds after receiving a signal from the Engineered Safety Features Actuation System. In the event of a steam line break, this action prevents continuous uncontrolled steam release from more than one steam generator. Protection is offered for breaks inside or outside the containment. | B
- B. The Main Steam Isolation Valves, their operators, and associated circuitry are Seismic Category I, and are protected against missiles and the effect of high-energy line breaks.

5.4.5.4 Tests and Inspections

All main steam isolation valves are designed, fabricated, tested, and installed in accordance with the codes and standards identified in Chapter 10. | I

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

This system is not applicable to a Pressurized Water Reactor.

5.4.7 SHUTDOWN COOLING SYSTEM**5.4.7.1 Design Bases****5.4.7.1.1 Summary Description**

The Shutdown Cooling System (SCS) is used in conjunction with the Main Steam and Main or Emergency Feedwater Systems (see Sections 10.1 and 10.4.9) to reduce the temperature of the Reactor Coolant System (RCS) in post shutdown periods from the hot shutdown operating temperature to the refueling temperature. The initial phase of a cooldown is accomplished by heat rejection from the Steam Generators (SG) to the condenser or atmosphere. After the reactor coolant temperature and pressure have been reduced to approximately 350°F and 400 psia, the SCS is put into operation to reduce the RCS temperature to the refueling temperature, and to maintain this temperature during refueling.

Additionally, the SCS is used in addition to the SG atmospheric steam release capability and the Emergency Feedwater System to cooldown the RCS following a small break LOCA (see Section 6.3). The SCS is also used subsequent to steam and feedwater line breaks, steam generator tube ruptures, and is used during plant startup prior to RCP restart to maintain flow through the core.

5.4.7.1.2 Functional Design Bases

The following functional design bases apply to the SCS:

- A. Two independent divisions of the SCS are provided, each with its own suction and discharge connections to the RCS.
- B. No single active failure prevents at least one complete train of the SCS from being brought on line from the control room, whether this is during normal plant cooldown or following a Design Basis Event (DBE).
- C. The design bases defined in Section 5.4.7.1.1 are met assuming the failure of a single active component during shutdown cooling or a single active or limited leakage passive failure of a component during long-term operations (i.e., >24 hours) following a DBE. Limited leakage passive failure is defined based on maximum flow through a failed valve packing or pump (e.g., SCS pump mechanical seal).
- D. The SCS is designed such that the SCS pumps and containment spray pumps are identical and functionally interchangeable. Provisions are made for locking closed those valves used in the SCS/CSS interconnection.

E. The SCS is designed for a pressure of 900 psig and a temperature of 400°F to provide greater operating flexibility. | I

F. No single failure allows the SCS to be overpressurized by the RCS. SCS components whose design pressure is less than the RCS design pressure are provided with overpressure protection (see Section 5.4.7.2.3). | C

G. The SCS reduces the RCS temperature as follows:

1. Two Train Cooldown

- a. to 140°F - within 24 hours after reactor shutdown.
- b. to 130°F - by the time reactor vessel head stud detensioning operations are started (i.e., within approximately 40 hours).
- c. to 120°F - within 96 hours after reactor shutdown.

2. One Train Cooldown

- a. to 200°F - within 24 hours after reactor shutdown in conjunction with other heat removal systems (e.g., steam generator atmospheric dump valves). | C

Typical cooldown curves are shown in Figures 5.4.7-1 and 5.4.7-2.

H. The components of the shutdown cooling system are designed in accordance with the applicable codes and classifications discussed in Section 5.4.7.2.4.

I. Materials are selected to preclude system performance degradation due to the effects of short and long term corrosion.

J. The SCS heat exchangers are sized to remove decay heat 96 hours after shutdown based upon a refueling water temperature of 120°F and a service water temperature of 95°F with an average reactor core burnup of two years. | C

K. The SCS is designed so that the SCS pumps can be tested at full-flow conditions with the reactor operating at power. |

L. The SCS is designed to transfer RCS fluid to the CVCS for purification during SCS operation. | I

- M. The SCS is designed to transfer refueling pool water back to the IRWST following refueling operations.
- N. The SCS is designed to provide a backup to the CSS for IRWST heat removal during accident conditions.
- O. The SCS is designed to provide RCS low temperature overpressurization protection.

5.4.7.2 System Design

5.4.7.2.1 System Schematic

The SCS is shown on the RCS P&ID (Figure 5.1.2-1) and on the SIS and SCS P&IDs (Figures 6.3.2-1A, 6.3.2-1B, 6.3.2-1C and 5.4.7-3). The SCS contains two heat exchangers and two pumps. One SCS pump is capable of meeting safety-grade cooldown criteria. Two SCS pumps are required to meet normal cooldown design criteria. SCS detailed design parameters are given in Table 5.4.7-1.

During initial shutdown cooling, a portion of the reactor coolant flows out the SCS nozzles located on the reactor vessel outlet (hot leg) pipes and is circulated through the SCS heat exchangers by the SCS pumps. The return to the RCS is through SIS direct vessel injection (DVI) nozzles. The pressure and temperature of the RCS system varies from 400 psia (approx) and 350°F at the initiation of shutdown cooling to atmospheric pressure and 120°F at refueling conditions. The SCS suction side pressure and temperature follow RCS conditions. The discharge side pressure is higher by an amount equal to the pump head. The temperature is lower at the shutdown cooling heat exchanger outlet.

Shutdown cooling flow is measured by flow orifices installed in each train of the SCS discharge piping. The information provided is used by the operator for flow control during SCS operation. The cooldown rate is controlled by adjusting flow through the heat exchangers with throttle valves on the discharge of each heat exchanger. The operator maintains a constant total SCS flow to the core by adjusting the heat exchanger bypass flow to compensate for changes in flow through the heat exchangers.

5.4.7.2.2 Component Description

A. Shutdown Cooling Heat Exchangers

The design temperature is based upon the temperature of the reactor coolant at the initiation of shutdown cooling plus a design tolerance. The SCS heat exchangers are used to remove core decay, RCS sensible and SCS pump heat during normal plant cooldowns after partial cooldown has been

accomplished, and core decay, RCS sensible, and SCS pump heat during safe cold shutdown conditions. The SCS heat exchangers are designed to maintain a refueling water temperature of 120°F, with a service water temperature of 95°F at 96 hours after shutdown following an assumed reactor core average burnup of two years. A conservative fouling resistance is assumed, resulting in additional margin for heat exchanger performance.

B. Instrumentation

The instrumentation and controls for the SCS are designed in accordance with the applicable portions of the IEEE Standards delineated in Section 7.1.2.

The operation of the SCS is controlled and monitored through the use of dedicated redundant instrumentation which provides the capability to monitor the cooldown rate and shutdown cooling flow to detect degradation of flow or SCS heat removal capabilities. The instrumentation provided for monitoring SCS components during normal SCS operation is discussed below, and also summarized in Section 7.5.

1. Shutdown Cooling Heat Exchanger Inlet and Return Line Temperature

The temperature of each shutdown cooling heat exchanger inlet and return line is monitored at locations both upstream and downstream of each heat exchanger. Temperatures are indicated (at a remote location) and recorded (in the control room) for each shutdown cooling heat exchanger inlet and return line. These indications are used to provide a measurement of system performance, and provide information allowing the operator to adjust the cooldown rate.

2. Shutdown Cooling Heat Exchanger Outlet Temperature

The temperature at the outlet of each heat exchanger is indicated in the control room. This instrument functions to monitor heat exchanger performance by directly measuring the outlet temperature.

3. Shutdown Cooling Pump Suction and Discharge Pressure

The pump suction and discharge pressures are indicated in the control room and function to monitor pump performance.

4. Shutdown Cooling Flow

A shutdown cooling flow indicator in each train of the SCS measures shutdown cooling flow, and indicates the flowrate in the control room. A low flow alarm is provided in the control room.

In addition, the CESSAR-DC design includes the following features to facilitate continued SCS operations during reduced RCS inventory:

- a. Two independent, highly reliable instruments are provided for RCS level measurement. These instruments function to monitor RCS level, to preclude SCS suction line vortexing and air entrainment.
- b. Two independent thermocouples are provided to measure core exit temperature, with a range optimized for SCS and refueling modes.
- c. Instruments which will monitor the state of SCS performance (such as pump suction pressure, vortexing monitoring equipment, flow instrumentation and/or pump motor current) are provided. These instruments function to sufficiently eliminate SCS pump loss events by monitoring the formation of vortexing and subsequent air entrainment.
- d. SCS suction isolation valves are not automatically closed in the event of an RCS pressurization during shutdown cooling. This precludes a loss of shutdown cooling by automatic closure of the isolation valves.
- e. The plant design provides other means of initiating alternate cooling for loss of SCS events. The plant design also ensures that a vent pathway is available to prevent pressurization (see Section 6.7).

Although the features described above do not exclusively describe SCS system instrumentation, they are provided in this section since they focus on precluding RHR system failures due to loss of the SCS pumps.

C. Piping

All SCS piping is austenitic stainless steel. All piping joints and connections are welded, except for a minimum number of flanged connections that are used to facilitate equipment maintenance or accommodate component design. SCS piping is designed to accommodate venting the SCS pumps to the RCS, if necessary, with as few high points as practical.

D. Valves

The location of valves, along with their type, type of operator, position (during the normal operating mode of the plant), type of position indication, and failure position is shown on Figures 6.3.2-1A, 6.3.2-1B, and 6.3.2-1C. | I

1. Relief Valves

Protection against the overpressurization of components within the SCS is provided by conservatively designing the system piping, appropriate valving between high pressure sources and lower pressure piping, and by relief valves. The SCS suction lines, up to and including SI-653 and SI-654, are designed for full RCS pressure. Relief valves are provided as required by applicable codes. All relief valves are of the totally enclosed, pressure tight type, with suitable provisions for gagging. | I
| C

2. Actuator Operated Throttling and Stop Valves

The failure position of each valve on loss of actuating signal or power supply is selected to ensure safe operation. System redundancy is considered when defining the failure position of any given valve. Valve position indication is provided at the main control panel, as indicated in Figures 6.3.2-1A, 6.3.2-1B and 6.3.2-1C. A momentary push button, with appropriate status control on the main control panel and/or a manual override handwheel, is provided where necessary for efficient and safe plant operation. All actuator operated valves have stem leakage controlled by a double packing with a lantern ring leakoff connection. | I
| C

3. Vent and Drain Valves

Vent and drain valves are provided to ensure that maintenance can be performed in each SCS component. | I

E. Shutdown Cooling System Pumps

The function of the SCS pumps is to provide flow through the SCS heat exchangers and reactor core for normal plant shutdown operation or for long term post-accident core cooling. During normal power operation, the SCS pumps are isolated from the RCS by redundant motor-operated valves in series. | C

The shutdown cooling and containment spray functions have been evaluated to select a single pump to serve both functions. The flow available with a single SCS pump is sufficient to both maintain an acceptable cooldown rate (75°F/hr maximum) during shutdown cooling operation and supply the CSS. NPSH available exceeds NPSH required for both pumps for all conditions under which the pumps will be operated.

SCS pump data is provided in Table 5.4.7-1. The design temperature for the SCS pumps is based upon the temperature of the reactor coolant at the initiation of shutdown cooling (350°F nominal) plus a design tolerance, resulting in a design temperature of 400°F.

The SCS pumps are vertical, single-stage centrifugal units equipped with mechanical seals backed up by a bushing, with a leakoff to collect the leakage past the seals. The seals are designed for operation with a pumped fluid temperature of 400°F. The pump motors are specified to have the capability of starting and accelerating the driven equipment, under load, to design point running speed within 5 seconds, based upon an initial voltage of 75% of the rated voltage at the motor terminals, and increasing linearly with time to 90% voltage in the first 2 seconds, and increasing to 100% voltage in the next 2 seconds.

The pumps are provided with drain and flushing connections to facilitate reduction of radiation levels before maintenance. The pressure containing parts are fabricated from stainless steel; the internals are selected for compatibility with boric acid solutions. The pumps are provided with minimum flow protection (recirculation lines) to prevent damage when starting against an isolated discharge pathway.

5.4.7.2.3 Overpressure Prevention

A. Overpressurization of the SCS by the RCS is prevented in the following ways:

1. The shutdown cooling suction isolation valves (SI-651, 652, 653, 654) are powered by four independent power supplies such that a fault in one power supply or valve will neither line up the RCS to either of the two SCS trains inadvertently nor prevent the initiation of shutdown cooling with at least one SCS train.

2. An interlock associated with the shutdown cooling suction isolation valves prevents the valves from being opened in the event that RCS pressure exceeds SCS design pressure. The interlock setpoint is calculated considering tolerances necessary for LTOP protection and plant P/T limits. The instrumentation and controls which implement this are discussed in Section 7.6. | I
3. The redundant SCS suction valves inside the containment are designed for full RCS pressure with the second valve forming the pressure boundary and safety class change. The valve operators are sized to ensure they can be opened even if a pressure buildup of fluid trapped between the valves occurs. | I
4. Alarms on SI-651, 652, 653 and 654 annunciate when the SCS suction isolation valves are not fully open (with concurrent low RCS temperature). Also, if SI-651 and 653 or SI-652 and 654 valves are open, and RCS pressure increases to the maximum pressure for SCS operation, an alarm will notify the operator that a pressurization transient is occurring during low temperature conditions. | I
5. Relief valves are provided as discussed in Section 5.4.7.2.2.
6. System piping is conservatively designed and maximum utilization of welded connections is made. | I

5.4.7.2.4 Applicable Codes and Classifications

- A. The piping and valves from the RCS, up to and including SI-653 and 654, are designed to ASME B&PVC Section III, Class 1. | I
- B. The remainder of the piping, valves, and components of the SCS, with the exception of those in Section 5.4.7.2.4(A) are designed to ASME B&PVC Section III, Class 2. | I
- D. The component cooling water side of the SCS heat exchanger is designed to ASME B&PVC Section III, Class 3.
- E. The power operated valves are designed to applicable IEEE Standards.
- F. The SCS is a Seismic Category I System.

5.4.7.2.5 System Reliability Considerations

The SCS is designed to perform its design function assuming a single failure, as described in Section 5.4.7.1.2. To assure availability of the SCS when required, redundant components and power supplies are utilized. The RCS can be brought to refueling temperature utilizing one of the two redundant SCS trains. However, with the design heat load, the cooldown would be considerably longer than the specified 96 hour time period. Since the SCS is essential for a safe (cold) shutdown of the reactor, it is a Seismic Category I system and designed to remain functional in the event of a safe shutdown earthquake.

The SCS does not utilize any pneumatically operated valves. The instrumentation, controls, and electric equipment pertaining to the SCS is designed to applicable portions of IEEE Standards 279, 308 and 603. In addition to normal offsite power sources, physically and electrically independent and redundant emergency power supply systems are provided to power safety-related components. See Chapter 8 for further information.

For long-term performance of the SCS without degradation due to corrosion, only materials compatible with the pumped fluid are used. Environmental envelopes are specified for system components to ensure acceptable performance in normal and applicable accident environments (see Section 3.11).

A limited leakage passive failure is defined as the failure of a pump seal or valve packing, whichever is greater. The maximum leakage is expected to be from a failed SCS pump seal. This leakage to the pump compartment will drain to the room sump. From there it is pumped to the waste management system. The sump pumps in each room will handle expected amounts of leakage. If leakages are greater than the sump pump capacity, the room will be isolated. In the event of a limited leakage passive failure in one train of the SCS, continued core cooling is provided by the unaffected independent SCS train. The limited leakage passive failure will be identified via appropriate leak detection provisions. Makeup of the leakage is provided by manually aligning the SIS to the IRWST, or by opening the safety injection tank isolation valves. The affected SCS train can then be isolated and core cooling continued with the other train.

5.4.7.2.6 Manual Actions

A. Plant Cooldown

Plant cooldown is the series of manual operations which bring the reactor from hot shutdown to cold shutdown.

Cooldown to approximately 350°F is accomplished by releasing steam from the secondary side of the steam generators. An inadvertent SIAS or safety injection tank discharge is precluded during a controlled cooldown by manually decreasing the SIAS setpoint, and depressurizing and isolating the safety injection tanks.

When RCS temperature and pressure decrease to 350°F and the maximum pressure for SCS operation (400 psia (approx)), the SCS is used. Operator action requires that the RCS be depressurized to below the maximum pressure for SCS operation in order to clear the permissive SCS interlock (see paragraph 5.4.7.2.3, item A.2).

During SCS operation, if SI-651 and 653 or SI-652 and 654 (SCS suction isolation valves) are open, and RCS pressure exceeds the maximum pressure for SCS operation, an alarm will notify the operator that a pressurization transient is occurring. If the SCS suction relief valves are not aligned to the RCS before cold leg temperature is reduced to below the maximum RCS cold leg temperature requiring LTOP, an alarm will notify the operator to open the SCS suction isolation valves (SI-651, 652, 653, 654). The maximum temperature requiring LTOP is based upon the evaluation of applicable RCS pressure/temperature curves (see Section 5.2.2.10.2.2).

Shutdown cooling is initiated using the SCS pumps. The SCS is warmed up and placed in operation as follows (refer to Figures 6.3.2-1A, 6.3.2-1B, 6.3.2-1C, and 5.4.7-3):

1. The SCS suction line isolation valves (SD-651*, 652, 653*, 654, 655*, and 656) are opened.
2. The SCS heat exchanger flow control valves (SI-310*, 311) are cracked open.
3. The SCS warmup line isolation valves (SI-690, 691*) are opened and the SCS pumps are started to induce recirculation flow through the SCS.
4. Once flow has been established in the SCS, the SCS discharge isolation valves (SI-600, 601*) are cracked open to allow a small amount of flow from the RCS to heat up SCS valves and piping.

5. The SCS discharge isolation valves (SI-600, 601*) are then gradually opened, while the warmup line isolation valves (SI-690, 691*) are gradually closed to maintain a constant flow. When complete, the system is in its normal operational mode. I
6. The SCS heat exchanger flow control valves (SI-310*, 311) and the SCS bypass flow control valves (SI-312*, 313) are adjusted as necessary, to maintain an RCS cooldown rate of 75°F/hour or less, until the refueling temperature of 120°F is attained. C

A maximum rate of cooldown (not to exceed 75°F/hour) is maintained by adjusting the flow rate of reactor coolant through the SCS heat exchangers, utilizing the SCS heat exchanger flow control valves (SI-310*, 311) in conjunction with the SCS bypass flow control valves (SI-312*, 313). With the shutdown cooling flow indicators, the operator maintains a total shutdown cooling flow rate by adjusting the amount of coolant which bypasses the SCS heat exchangers. I

When the system is first put into operation, the temperature difference for heat transfer across the SCS heat exchanger is large, and only a portion of the total flow from the SCS pumps is diverted through the heat exchangers. As the cooldown proceeds, the temperature differential across the heat exchanger decreases, and the flow rate through the heat exchangers is increased to maintain the maximum permissible cooldown rate. The flow to the SCS heat exchangers is increased periodically until full SCS pump flow through the heat exchangers is attained. A graph of RCS temperature vs. time after shutdown for a normal design basis cooldown is presented in Figure 5.4.7-1. C

Shutdown cooling is continued throughout the entire period of plant shutdown to maintain a refueling water temperature of 120°F or less. Whenever shutdown cooling is in operation, shutdown purification flow may be initiated through the CVCS. k

* Valves with asterisk are located in SCS Train 1. Valves not designated with an asterisk are located in SCS Train 2. I

B. Plant Heatup

Plant heatup is a series of manual operations which bring the RCS from cold shutdown to hot standby. The SCS heat exchangers are bypassed to maintain flow through the core without the heat removal effect of the heat exchangers. Flow can be initiated to the heat exchangers if necessary to control the heatup rate. When the reactor coolant pumps can be run, the SCS pumps are stopped and the system is isolated for the standby mode.

C. Abnormal Operation

1. Initiation of shutdown cooling with the most limiting single failure (loss of one shutdown cooling train) can be accomplished via plant procedures using equipment in the operable train (i.e., operating the valves with (*) for SCS train number 1 or the valves without (*) for SCS train number 2).
2. The SCS pumps can be used alternatively as containment spray pumps.
3. The SCS pumps, in conjunction with the SCS heat exchangers, can be used for IRWST cooling.

D. Design Bases Event Operations

Following certain DBEs (feedwater line break, small break LOCA, steam line break, or loss of offsite power), shutdown cooling can be initiated with RCS hot leg conditions which exceed the normal shutdown cooling initiation temperature of 350°F. However, shutdown cooling will never be initiated at conditions which exceed the design temperature of the SCS components.

5.4.7.3 Performance Evaluation

The design point of the SCS is taken at 96 hours after plant shutdown. At this point, the design basis is to maintain a 120°F refueling temperature with a service water temperature of 95°F. Two SCS heat exchangers and two SCS pumps are assumed to be in operation at the design flow. The SCS heat exchanger size is determined at this point, since it requires the greatest heat transfer area due to the relatively small ΔT between primary fluid and component cooling water. The design input heat load at 96 hours is based on decay heat at 96 hours, assuming an average reactor core burnup of two years. Additional energy input to the

RCS from two SCS pumps running at design flow rate was also included with no credit taken for component energy losses to the external environment.

E

At each time interval in the cooldown, an iterative process is utilized to analyze transient performance, whereby the heat removal is established by balancing the available heat load with the SCS heat exchanger heat removal capability. The cooldown rate is limited to a maximum of 75°F/hour throughout the cooldown. The normal two train cooldown curve is shown in Figure 5.4.7-1.

With the most limiting single active failure in the SCS, RCS temperature can be brought to 200°F within 24 hours following shutdown using one SCS pump and one SCS heat exchanger, assuming that the RCS pressure and temperature are reduced to SCS initiation conditions by other heat rejection means in 3.5 hours. The single train cooldown curve is shown in Figure 5.4.7-2.

C

The SCS is designed utilizing a philosophy of total physical separation of redundant trains such that the system can carry out its safety function assuming a single active failure during both normal and short-term post accident modes and a single active or passive failure during long-term post accident modes (i.e., time periods >24 hr) after event initiation. Total train separation assures that a single failure in one train cannot preclude the second train from accomplishing its safety functions. A Failure Modes and Effects Analysis for the SCS is presented in Table 5.4.7-3.

C

Adequate sampling capability of the SCS is ensured for all modes of SCS operation to verify boron concentration and fission product activity.

I

5.4.7.4 Preoperational Testing

Preoperational tests are conducted to verify proper operation of the SCS. The preoperational tests include calibration of instrumentation, verification of adequate cooling flow, and verification of the operability of all associated valves. In addition, a preoperational hot functional performance test is made on the installed SCS heat exchangers as part of the precore hot functional test program. See Chapter 14 for further details on these tests.

The SCS also undergoes a series of preoperational hydrostatic tests conducted in accordance with Section III of the ASME Boiler and Pressure Vessel Code.

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TABLE 5.4.7-1

(Sheet 1 of 3)

SHUTDOWN COOLING DESIGN PARAMETERS

SYSTEM DESIGN PARAMETERS

Shutdown cooling system initiation	Approximately 3.5 hours after reactor shutdown or trip	
Reactor coolant system maximum cooldown rate (at initiation of shutdown cooling), °F/hr	75	
Nominal shutdown cooling flow, gpm/HX	5000	I

TABLE 5.4.7-1 (Cont'd)

(Sheet 2 of 3)

SHUTDOWN COOLING DESIGN PARAMETERS

COMPONENT DESIGN PARAMETERS

Shutdown Cooling Heat Exchanger Data

Quantity	2	
Type	Shell and tube, horizontal U-tube	
Service transfer rate, $\text{Btu/hr-}^{\circ}\text{F-ft}^2$	350	C
Heat Transfer area, ft^2/HX	6010	E

Tube Side

Fluid	Reactor coolant	E
Design pressure, psig	900	
Design temperature, $^{\circ}\text{F}$	400	C
Material	Austenitic stainless steel	
Code	ASME Section III, Class 2	E

Shell Side

Fluid	Component cooling water	
Design pressure, psig	150	
Design temperature, $^{\circ}\text{F}$	250	
Material	Carbon steel	
Code	ASME Section III, Class 3	E

At 96 hours after shutdown:

Tube Side

Flow, million lb/hr	2.47	
Inlet temperature, $^{\circ}\text{F}$	120	C

Shell Side

Flow, million lb/hr	5.47	E
Inlet temperature, $^{\circ}\text{F}$	95	I

TABLE 5.4.7-1 (Cont'd)

(Sheet 3 of 3)

SHUTDOWN COOLING DESIGN PARAMETERS

COMPONENT DESIGN PARAMETERS (Cont'd)

Shutdown Cooling Pump Data

Quantity	2	
Type	Single Stage, vertical, centrifugal	
Safety Classification	2	
Code	ASME III, Class 2	
Design Pressure, psig	900	I
Design Temperature, °F	400	
Design Flow Rate, gpm	5,000	E
Design Head, ft	400	
Materials	Stainless steel type 304, 316 or approved alternate	C
Seals	Mechanical	E

TABLE 5.4.7-2

(Sheet 1 of 7)

SHUTDOWN COOLING SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

<u>No.</u>	<u>Name</u>	<u>Failure Mode</u>	<u>Cause</u>	<u>Symptoms and Local Effects Including Dependent Failures</u>	<u>Method of Detection</u>	<u>Inherent Compensating Provision</u>	<u>Remarks and Other Effects</u>	
1)	Shutdown Cooling Pump Suction Isolation Valve SI-106, SI-107	a) Fails Closed	Corrosion, mechanical binding, operator error	Effective loss of one shutdown cooling train cooling	Low flow indication F-302, F-305; periodic testing	Parallel redundant shutdown cooling path	Valve is normally locked open	E
		b) Fails Open	Same as 1a)	No effect on SCS operation	Periodic testing	None required		E
2)	Shutdown Cooling Pump 1, 2	a) Fails to start	Mechanical failure, electrical failure	Effective loss of one SCS train	Low flow indication F-302, F-305; periodic testing	Parallel redundant shutdown cooling path		I
								E
3)	Shutdown Cooling Pump Discharge Isolation Valve SI-578, SI-579	a) Fails Closed	Corrosion, mechanical binding, operator error	Effective loss of one shutdown cooling pump	Low flow indication F-302, F-305 valve; periodic testing	Parallel redundant shutdown cooling path	Valves are locked open; min. flow line will provide the min. flow required to protect the pump	I
		b) Fails Open	Same as 3a)	No effect on SCS operation	Periodic testing	None required		E

TABLE 5.4.7-2 (Cont'd)

(Sheet 2 of 7)

SHUTDOWN COOLING SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

<u>No.</u>	<u>Name</u>	<u>Failure Mode</u>	<u>Cause</u>	<u>Symptoms and Local Effects Including Dependent Failures</u>	<u>Method of Detection</u>	<u>Inherent Compensating Provision</u>	<u>Remarks and Other Effects</u>	
4)	Shutdown Cooling Heat Exchanger 1,2	a) Loss of Cooling	Insufficient component cooling water flow, excessive fouling	Diminished ability of subsystem to provide RCS heat removal	High temperature indication from T-302, T-305	Parallel redundant shutdown cooling path		E
5)	Shutdown Cooling Heat Exchanger Bypass Valve SI-312, SI-313	a) Fails Closed	Corrosion, mechanical binding, electrical failure	Delays use of affected SCS train	Valve position indicator; periodic testing	Parallel redundant shutdown cooling path	Same as 3a)	I E
		b) Fails Open	Mechanical failure, electrical failure	Loss of one SCS train	Valve position indicator; periodic testing	Parallel redundant shutdown cooling path		I E
6)	Shutdown Cooling Heat Exchanger Outlet Isolation/Flow Control Valve SI-310, SI-311	a) Fails Closed	Mechanical failure, electrical failure	Effective loss of one shutdown cooling path	Valve position indicator; periodic testing	Parallel redundant shutdown cooling path		I
		b) Fails Open	Mechanical failure, electrical failure	Delays use of affected SCS train	Valve position indicators; periodic testing	Parallel redundant shutdown cooling path		E

TABLE 5.4.7-2 (cont'd)

(Sheet 3 of 7)

SHUTDOWN COOLING SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

<u>No.</u>	<u>Name</u>	<u>Failure Mode</u>	<u>Cause</u>	<u>Symptoms and Local Effects Including Dependent Failures</u>	<u>Method of Detection</u>	<u>Inherent Compensating Provision</u>	<u>Remarks and Other Effects</u>	
7)	SCS Warmup Line Flow Control SI-690, SI-691	a) Fails Open	Elect. Malf., Mech. binding	Diversion of flow from discharge leg to suction leg of SCS without passing through the reactor core during shutdown cooling operations	Position indication in control room; periodic testing	Redundant shutdown cooling subsystem will not be affected		E
		b) Fails Closed	Elect. Malf., Mech. binding	Inability to gradually warm-up the shutdown cooling lines during the SCS alignment procedure	Position indication in control room, periodic testing	Same as 8a)	See 8a)	I E
8)	SCS Suction Isolation SI-655, SI-656	a) Fails Open	Elect. Malf., Mech. binding	No effect on shutdown cooling	Valve position indication in control room; periodic testing	None required	Valve is normally locked closed in control room	E I
		b) Fails Closed	Elect. Malf., Mech. binding	Inability to align one shutdown cooling subsystem for shutdown cooling	Valve position indication in control room; periodic testing	Redundant shutdown cooling subsystem		E
9)	SCS Isolation Valve SI-651, SI-653, SI-652, SI-654	a) Fails Open	Elect. Malf., Mech. binding	None	Position indication in control room; periodic testing	The redundant series valve ensures that SCS is protected from normal RCS pressure during power operation	Interlocks associated with the valves prevent overpressuriza- tion. These interlocks prevent the	I E

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December 21, 1990

TABLE 5.4.7-2 (Cont'd)

(Sheet 4 of 7)

SHUTDOWN COOLING SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

<u>No.</u>	<u>Name</u>	<u>Failure Mode</u>	<u>Cause</u>	<u>Symptoms and Local Effects Including Dependent Failures</u>	<u>Method of Detection</u>	<u>Inherent Compensating Provision</u>	<u>Remarks and Other Effects</u>	
9)	(Cont'd)						valves in the suction line of the SCS from being opened if RCS pressure exceeds a value which would result in exceeding the design pressure of the SCS (with tolerances)	E
		b) Fails Closed	Elect. Malf., Mech. binding	Effective loss of one SCS train	Position indication in control room; periodic testing			I
10)	SCS Flow Indication F-302, F-305	a) False indica- tion	Elect. Malf.	Inability to control cooldown rate in affected train. Possible isolation of functional train	Comparison with redundant indicator, with all other process instrumentation and valve position indications consistent	Redundant SCS train		E
								I
								E

TABLE 5.4.7-2 (Cont'd)

(Sheet 5 of 7)

SHUTDOWN COOLING SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

<u>No.</u>	<u>Name</u>	<u>Failure Mode</u>	<u>Cause</u>	<u>Symptoms and Local Effects Including Dependent Failures</u>	<u>Method of Detection</u>	<u>Inherent Compensating Provision</u>	<u>Remarks and Other Effects</u>	
11)	Containment Spray Pump Isolation Valve SI-110, SI-111	a) Fails Open	Corrosion, mechanical binding, operator error	Loss of one SCS train	Low temperature in SCS; periodic testing	Redundant SCS train	Valve is normally locked closed	E 1
		b) Fails Closed	Corrosion, mechanical binding	No effect on SCS operation	Periodic testing	None required		E
12)	SDCHX Inlet/Outlet Temperature Recorder T-300, T-301	a) False indica- tion	Elect. Malf.	Inability to control cooldown rate in affected train. Possible isolation of functional SCS train	Comparison with redundant indicators, with all other process instrumentation and valve position indications consistent. Periodic testing	Redundant SCS train		I E

TABLE 5.4.7-2 (Cont'd)

(Sheet 6 of 7)

SHUTDOWN COOLING SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

<u>No.</u>	<u>Name</u>	<u>Failure Mode</u>	<u>Cause</u>	<u>Symptoms and Local Effects Including Dependent Failures</u>	<u>Method of Detection</u>	<u>Inherent Compensating Provision</u>	<u>Remarks and Other Effects</u>	
13)	SCS Miniflow HX 1,2	a) Loss of Cooling	Single failure in CCWS system	Possible damage to associated SCS pump	Periodic testing	Redundant SCS train		E
14)	SIT Discharge Isolation Valve SI-614, SI-624, SI-634, SI-644	a) Fails Open	Elect. Malf., Mech. binding	Unable to isolate one SI tank from the RCS. No effect on SCS operation	Position indication in control room; periodic testing	None required	During shutdown cooling these valves are closed. However, if a LOCA occurs a SIAS will automatically open these valves. Pressure in SIT's can be reduced if necessary prior to SCS entry	I
		b) Fails Closed	Elect. Malf., Mech. binding	No effect during shutdown cooling	Valve position indications in control room; periodic testing	None required		E
15)	Shutdown Purification Isolation Valve SI-418, SI-419, SI-420, SI-421	a) Fails Open	Corrosion, Mech. binding	None during shutdown cooling	Periodic testing	Redundant isolation valves in series	Valve is normally locked closed	
		b) Fails Closed	Corrosion, Mech. binding	Inability to remove contaminants from one SCS flow path during long-term cooling	The failure to purify would be detected by periodic sampling	Redundant purification connection to other SCS subsystem		I

TABLE 5.4.7-2 (Conc'd)

(Sheet 7 of 7)

SHUTDOWN COOLING SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

<u>No.</u>	<u>Name</u>	<u>Failure Mode</u>	<u>Cause</u>	<u>Symptoms and Local Effects Including Dependent Failures</u>	<u>Method of Detection</u>	<u>Inherent Compensating Provision</u>	<u>Remarks and Other Effects</u>	
16)	SCS Test Return Isolation Valve SI-314, SI-315 SI-688, SI-693	a) Fails Open	Corrosion, mechanical binding	None	High temperature indication from T-300, T-301; periodic testing	Series isolation valves in IRWT return		E
		b) Fails Closed	Corrosion, mechanical binding	No effect on SCS operation	Periodic testing	None required		I
17)	SCS/CSS Crossover Valve SI-430, SI-431	a) Fails Open	Corrosion, mechanical binding, operator error	None	Periodic testing	CSS is normally isolated		E
		b) Fails Closed	Corrosion, mechanical binding	No effect on SCS operation	Periodic testing	None required		I
18)	PCPS Isolation Valve SI-458, SI-450, SI-455, SI-454	a) Fails Open	Mech. binding	None	Periodic testing	Redundant series valve provides isolation	Valve is normally locked closed at valve	E
		b) Fails Closed	Mech. binding	None	Periodic testing			I

Amendment I
December 21, 1990

Amendment C
June 30, 1988

SYSTEM 80+™

SHUTDOWN COOLING SYSTEM
TWO TRAIN COOLDOWN

Figure
5.4.7-1

Amendment C
June 30, 1988

SYSTEM 80+™

SHUTDOWN COOLING SYSTEM
ONE TRAIN COOLDOWN

Figure
5.4.7-2

CALMA FILE:
[.ALWRJFIG54731

Amendment I
December 21, 1990

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SHUTDOWN COOLING SYSTEM FLOW DIAGRAM
SHUTDOWN COOLING MODE

Figure
5.4.7-3

CALMA FILE:
[.ALWR]FIG54732

Amendment I
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SHUTDOWN COOLING SYSTEM FLOW DIAGRAM
SHUTDOWN COOLING MODE

Figure

5.4.7-3

5.4.8 REACTOR COOLANT CLEANUP SYSTEM

One function of the Chemical and Volume Control System (CVCS) is to provide radiological and chemical cleanup of the Reactor Coolant System. A description of the CVCS is given in Section 9.3.4. Radiological considerations are described in Chapters 11 and 12.

5.4.9 MAIN STEAM LINE AND FEEDWATER PIPING

See Chapter 10.

5.4.10 PRESSURIZER**5.4.10.1 Design Bases**

The pressurizer is designed to:

- A. Maintain RCS operating pressure such that the minimum pressure during operating transients is above the setpoint for the Safety Injection Actuation Signal and low pressure reactor trip, and such that the maximum pressure is below the high pressure reactor trip setpoint. B
- B. Meet the design transients specified in Section 3.9.1 except that the maximum allowable rate of change in pressurizer temperature during plant heatup and cooldown is 200°F/hr.
- C. Provide sufficient water volume in the pressurizer to prevent uncovering the heaters as a result of a reactor trip. B
- D. Provide sufficient water volume to prevent pressurizer heaters from being uncovered by the outsurge following step load decreases of 10% starting within the range of 100% to 25% of full-rated power or a 5% per minute ramp decrease from 100% to 15% of full-rated power. I
- E. Provide sufficient steam volume to avoid lifting the primary safety valves as a result of a loss of condenser vacuum (LOCV) event (normal control systems are operational). B
- F. Provide sufficient steam volume to allow acceptance of the insurge resulting from any loss of load transient without liquid or two-phase flow reaching the primary safety valve nozzles. B
- G. Minimize the total reactor coolant mass change and associated charging and letdown flow rates in order to reduce the quantity of wastes generated by load follow operations.

- H. Provide sufficient pressurizer heater capacity to heat up the pressurizer, filled with water at the zero power level, at a rate that ensures a pressurizer temperature (and thus pressure) which will maintain an adequate degree of subcooling of the water in the reactor coolant loop as it is heated by core decay heat and/or pump work from the reactor coolant pumps.
- I. Contain a total water volume that does not adversely affect the total mass and energy released to the containment during the maximum hypothetical accident.
- J. Ensure that, in addition to being specified as Seismic Category I, the pressurizer vessel, including heaters, baffles, and supports shall be designed such that no damage to the equipment is caused by the frequency ranges of 19-20 cps and 95-100 cps. The lower frequency is defined as for the reactor vessel. The design basis for the higher frequency consists of a pressure pulse of 5 psi which diminishes internally within the vessel.
- K. The combination of maximum heat loss from the pressurizer and the pressurizer heater capacity shall be such as to maintain the pressurizer at normal operating pressure during hot standby conditions. This capability shall be provided by redundant trains of heaters powered from off-site power and Class 1E emergency power.
- L. The total spray flows shall be sufficient to keep the pressure below the reactor trip setpoint during an insurge of water during the "Maneuvering and Load Follow" and "Loss of Load" transients. | E
- M. The pressurizer size and spray capacity shall be adequate so the pressurizer safety valves are not actuated by overpressure events initiated by normal operation transients. | E

5.4.10.2 Description

The pressurizer, as shown in Figure 5.4.10-1, is a vertically mounted, bottom supported, cylindrical pressure vessel. Replaceable direct immersion electric heaters are vertically mounted in the bottom head. The pressurizer is furnished with nozzles for the spray, surge, and safety lines, and pressure and level instrumentation. A manway is provided in the top head for access for inspection of the pressurizer internals. The pressurizer surge line is connected to one of the reactor coolant hot legs and the spray lines are connected to two of the cold legs at the reactor coolant pump discharge. Heaters are

supported inside the pressurizer to preclude damage from vibration and seismic loadings. Principal design parameters are listed in Table 5.4.10-1. B

The pressurizer and surge line are located entirely above the reactor coolant loops. The surge line is continuously rising from the hot leg nozzle to the pressurizer, thus ensuring that the line contains no water traps. The pressurizer surge line is sized and arranged to minimize the flow resistance. The surge line L/D will not exceed 210, assuming a 12-inch, Schedule 160 pipe. The L/D statement above includes the effective L/D of all piping elbows but does not include the surge line entrance and exit losses. I

The maximum allowable pressure drop through the pressurizer spray line piping only is 19 psi at a total flow rate of 375 gpm and at a water temperature of 565°F.

The pressurizer is designed and fabricated in accordance with the ASME Code listed in Table 5.2-1. The interior surface is clad with weld deposited stainless steel.

The total volume of the pressurizer is established by consideration of the factors given in Section 5.4.10.1. To account for these factors and to provide adequate margin at all power levels, the water level in the pressurizer is programmed as a function of average coolant temperature as shown in Figure 5.4.10-2, in conjunction with Figure 5.4.10-3. High or low water level error signals result in the control actions shown in Figure 5.4.10-4. The pressurizer surge line is sized to accommodate the flow rates associated with the RCS expansion and contraction due to the transients specified in Section 3.9.1.

The pressurizer maintains Reactor Coolant System operating pressure and, in conjunction with the Chemical and Volume Control System (CVCS), Section 9.3.4, compensates for changes in reactor coolant volume during load changes, heatup, and cooldown. During full-power operation, the pressurizer is about one-half full of saturated steam. Reactor Coolant System pressure may be controlled automatically or manually by maintaining the temperature of the pressurizer fluid at the saturation temperature corresponding to the desired system pressure. A small continuous spray flow is maintained to the pressurizer to avoid stratification of pressurizer boron concentration and to maintain the temperature in the surge and spray lines, thereby reducing thermal shock as the spray control valves open. An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow cooling if the reactor coolant pumps are shut down.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the reactor coolant. The average reactor coolant temperature is programmed to vary as a function of load as shown in Figure 5.4.10-3. A reduction in load is followed by a decrease in the average reactor coolant temperature to the programmed value for the lower power level. The resulting contraction of the coolant lowers the pressurizer water level, causing the reactor system pressure to decrease. This pressure reduction is partially compensated by flashing of pressurizer water into steam. All pressurizer heaters are automatically energized on low system pressure, generating steam and further limiting pressure decrease. Should the water level in the pressurizer drop sufficiently below its setpoint, the letdown control valves close to a minimum value, and the charging flow control valve(s) open in the chemical and volume control system are automatically controlled to add coolant to the system and restore pressurizer level. B

When steam demand is increased, the average reactor coolant temperature is raised in accordance with the coolant temperature program. The expanding coolant from the reactor coolant piping hot leg enters the bottom of the pressurizer through the surge line, compressing the steam and raising system pressure. The increase in pressure is moderated by the condensation of steam during compression and by the decrease in bulk temperature in the liquid phase. Should the pressure increase be large enough, the pressurizer spray valves open, spraying coolant from the reactor coolant pump discharge (cold leg) into the pressurizer steam space. The relatively cold spray water condenses some of the steam in the steam space, limiting the system pressure increase. The programmed pressurizer water level is a temperature dependent function. A high level error signal, produced by an in-surge, causes the letdown control valves to modulate open, releasing coolant to the chemical and volume control system, and the charging flow control valve is closed to a minimum value, thus restoring the pressurizer to the programmed level. Small pressure and primary coolant volume variations are accommodated by the steam volume that absorbs flow into the pressurizer and by the water volume that allows flow out of the pressurizer. B

The pressurizer heaters are single unit, direct immersion heaters that protrude vertically into the pressurizer through sleeves welded in the lower head. Each heater is internally restrained from high amplitude vibrations and can be individually removed for maintenance during plant shutdown.

A number of the heaters are connected to proportional controllers, which adjust the heat input to account for steady-state losses and to maintain the desired steam pressure in the pressurizer. The remaining heaters are connected to on-off controllers. These heaters are normally deenergized but are automatically turned on by a low pressurizer pressure signal or a high level error signal. This latter feature is provided since load increases result in an in-surge of relatively cold coolant into the pressurizer, thereby decreasing the bulk water temperature. The CVCS acts to restore level, resulting in a transient pressure below normal operating pressure. To minimize the extent of this transient, the backup heaters are energized, contributing more heat to the water. Backup heaters are deenergized in the event of concurrent high-level error and high-pressurizer pressure signals. A low-low pressurizer water level signal deenergizes all heaters before they are uncovered to prevent heater damage. The pressure control program is shown in Figure 5.4.10-5.

5.4.10.3 Evaluation

It is demonstrated by analysis in accordance with requirements for ASME Code, Section III, Class 1 vessels that the pressurizer is adequate for all normal operating and transient conditions expected during the life of the facility. Following completion of fabrication, the pressurizer is subjected to the required ASME Code, Section III hydrostatic test and post-hydrostatic test non-destructive testing.

During hot functional testing, the transient performance of the pressurizer is checked by determining its normal heat losses and maximum depressurization rate. This information is used in setting the pressure controllers.

Further assurance of the structural integrity of the pressurizer during plant life will be obtained from the inservice inspections performed in accordance with ASME Code, Section XI, and described in Section 5.2.

Overpressure protection of the Reactor Coolant System is provided by four ASME Code spring-loaded safety valves. Refer to Section 5.4.12 and 5.4.13.

5.4.10.4 Tests and Inspections

Prior to and during fabrication of the pressurizer, non-destructive testing is performed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Table 5.4.10-2 summarizes the pressurizer

inspection program, which also includes tests not required by the Code. Refer to Section 5.2.1 for inservice inspections of the pressurizer.

TABLE 5.4.10-1PRESSURIZER PARAMETERS

<u>Property</u>	<u>Parameter</u>	
Design pressure, psia	2500	
Design temperature, °F	700	
Normal operating pressure, psia	2250	
Normal operating temperature, °F	652.7	
Internal free volume, ft ³	2400	B
Normal (full power) operating water volume, ft ³	1200	
Normal (full power) steam volume, ft ³	1200	
Installed heater capacity, kW	2400	E
Heater type	Immersion	
Spray flow, minimum design capacity, gpm	375	D
Spray flow, (maximum) continuous, gpm	3	E
Nozzles		
Surge, in. (nominal)	12, schedule 160	
Spray, in. (nominal)	4, schedule 160	
Safety valves, in. (nominal)	6, schedule 160	E
Instrument		
Level, in. (nominal)	3/4, schedule 160	
Temperature, in. (nominal)	1, schedule 160	
Pressure, in. (nominal)	3/4, schedule 160	
Heater, O.D., in.	1-1/4	

TABLE 5.4.10-2PRESSURIZER TESTS

<u>Component</u>	<u>Tests</u> ^(a)
Heads	
Plates	UT, MT
Cladding	
Shell	
Plates	UT, MT
Cladding	UT, PT
Heaters	
Tubing	UT, PT
Centering of elements	RT
End Plug	UT, PT
Nozzle (Forgings)	UT, MT
Studs	UT, MT
Welds	
Shell, longitudinal	RT, MT
Shell, circumferential	RT, MT
Cladding	UT, PT
Nozzles	RT, MT
Nozzle safe ends	RT, PT
Instrument connections	PT
Support Skirt	MT, RT
Temporary attachment after removal	MT
All welds after hydrostatic test	MT or PT
Heater assembly, end plug weld	PT

(a) Key:

UT = ultrasonic testing
MT = magnetic particle testing
PT = dye-penetrant testing
RT = radiographic testing

NOZZLE SCHEDULE		
NO.	SERVICE	NO. REQ'D
1	MANWAY	1
2	SURGE	1
3	SPRAY	1
4	SAFETY VALVE	4
5	INSTRUMENT TAPS	4
6	INSTRUMENT TAPS	2
7	TEMPERATURE	1
8	HEATER	48

Amendment E
December 30, 1988

SYSTEM 80+™

TYPICAL PRESSURIZER

Figure
5.4.10-1

Amendment I
December 21, 1990

SYSTEM 80 +TM

TYPICAL PRESSURIZER LEVEL SETPOINT PROGRAM

Figure

5.4.10-2

Amendment B
March 31, 1988

SYSTEM 80+™

TYPICAL TEMPERATURE CONTROL PROGRAM

Figure
5.4.10-3

Amendment B
March 31, 1988

SYSTEM 80+™

TYPICAL PRESSURIZER LEVEL ERROR PROGRAM

Figure
5.4.10-4

Amendment B
March 31, 1988

SYSTEM 80+™

PRESSURE CONTROL PROGRAM

Figure
5.4.10-5

5.4.11 PRESSURIZER RELIEF TANK

The In-containment Refueling Water Storage Tank (IRWST) is used as the pressurizer relief tank. The design and description of this tank are given in Section 6.8.

| E
| I

5.4.12 VALVES**5.4.12.1 Design Basis**

The safety-related functions of valves within the reactor coolant pressure boundary are to act as pressure retaining vessels and leaktight barriers during normal plant operation, accidents and seismic disturbances.

These valves are designed in accordance with ASME Code, Section III, Class I requirements and must withstand the effects of the system design transients (see Section 3.9.1) plus other transients associated with each valves location or service requirements. The valves meet Seismic Category I requirements. Valve stem leakage control for reactor coolant pressure boundary valves is accomplished through the use of double packings with a leak-off connection at the lantern ring to provide for collection of leakage. Backseats are specified on manual and motor-operated gate and globe valves to further minimize potential leakage.

Materials of construction are specified to assure compatibility with the environment and contained fluids.

5.4.12.2 Design Description

All valves in the reactor coolant system are constructed primarily of stainless steel. Other materials in contact with the coolant, such as hard facing and packing, are compatible materials. Fasteners, packing, gland assemblies, and yoke fasteners are constructed of stainless steel to eliminate corrosion problems. Valves not constructed with diaphragms, or bellows, or not normally backseated, are provided with two independent sets of packing, separated by a lantern ring and leak-off connections piped to a collection system to eliminate uncontrolled leakage.

5.4.12.3 Design Evaluation

All valves within the reactor coolant pressure boundary are stress analyzed in accordance with ASME Code, Section III, Class I, taking into consideration cyclic loadings.

5.4.12.4 Tests and Inspections

The valves are hydrostatically tested and leak tested across the seats and across the packing in accordance with the ASME B&PV Code as specified in Table 5.2-1.

5.4.13 SAFETY AND RELIEF VALVES**5.4.13.1 Design Basis**

The safety valves on the pressurizer are designed to protect the primary system, as required by the ASME B&PV Code, Section III. | D

The design basis for establishing the relieving capacity of the pressurizer safety valves is presented in Appendix 5A. For the postulated transients presented in Chapter 15, the results indicate that relieving capacity of the safety valves is sufficient to provide overpressure protection in accordance with Section III of the ASME Code.

Safety valves on the steam side of each steam generator are designed to protect the steam system, as required by the ASME Code, Section III. They are conservatively sized to pass a steady flow equivalent to the maximum expected power level at the design pressure of the steam system.

5.4.13.2 Description

The RCS has four safety valves to provide overpressure protection. A typical safety valve is illustrated in Figure 5.4.13-1. The design parameters are given in Table 5.4.13-1. These valves are connected by piping to the top of the pressurizer. They are direct acting, spring-loaded safety valves meeting ASME Code requirements. They have an enclosed bonnet and have a balanced bellows to compensate for backpressure. The safety valves pass sufficient pressurizer steam to limit the Reactor Coolant System pressure to 110% of design pressure (2750 psia) following a complete loss of turbine generator load without simultaneous reactor trip. A delayed reactor trip is assumed on a high-pressurizer pressure signal. To determine maximum steam flow through the pressurizer safety valves, the main steam safety valves are assumed to be operational. Values for the system parameters, delay times, and core moderator coefficient are given in Chapter 15.

Overpressure protection for the shell side of the steam generators and the main steam line up to the inlet of the turbine stop valve is provided by the secondary safety valves.

These valves are each sized to pass a steam flow of 945,292 lb/hr at 1308 psia. This limits steam generator pressure to less than 110% of steam generator design pressure during worst case transients. The secondary safety valves consist of two banks of 10 valves with staggered set pressures. The valves are spring-loaded safety valves procured in accordance with ASME Boiler and Pressure Vessel Code, Section III (see Table 5.2-1). Parameters for the secondary safety valves are given in Table 5.4.13-2.

5.4.13.3 Evaluation

Overpressure protection is discussed in Section 5.2.2. The ASME Code report on Overpressure Protection is included as Appendix 5A.

5.4.13.4 Tests and Inspections

The valves are inspected during fabrication in accordance with ASME III Code requirements.

5.4.13.4.1 Pressurizer Safety Valves

The inlet and outlet portions of the valves are hydrostatically tested with water at the appropriate pressures required by the applicable section of the ASME Code. Set pressure and seat leakage tests can be performed with steam using a pro-rated spring. Final set pressure tests are performed with the final springs using either high pressure steam or low pressure steam with an assist device. Final seat leakage tests are performed prior to shipment with the final springs using either hot air or hot nitrogen. Valve adjustment shall be made to a valve ring setting combination selected to provide stable valve operation on the basis of the EPRI Safety Valve Test Program results.⁽¹⁾

5.4.13.4.2 Main Steam Safety Valves

The inlet portion of the valve is hydrostatically tested with water in accordance with the ASME Code. Set pressure and set leakage tests are performed using steam. Adjustment is made to provide a valve blowdown meeting the requirement specified in Table 5.4.13-2.

(1) CEN-227 "Summary Report on the Operability of Pressurizer Safety Relief Valve in C-E Designed Plants", December 1982.

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TABLE 5.4.13-1PRESSURIZER SAFETY VALVE PARAMETERS

<u>Property</u>	<u>Parameter</u>	
Design pressure, psia	2500	
Design temperature, °F	700	
Fluid	Saturated Steam, 4400 ppm boron, pH = 4.5 to 10.6	
Set pressure, psia	2500 ± 1%	
Min. capacity, lb/hr at, accumulation pressure, each	460,000	
Type	Spring loaded safety- balanced bellows, enclosed bonnet	D
Orifice area, in. ²	4.34	
Accumulation, %	3	
Backpressure		B
Max. buildup/max superimposed, psig	700 [*] /340	
Blowdown, %	18.5	
Min. blowdown pressure, psia	2040	
Typical materials		
Body	ASME SA 182, GR. F316	
Disc	ASTM B637, UNS N07750	D
Nozzle	ASME SA 182, GR. 347	

^{*} Maximum allowable back pressure.

TABLE 5.4.13-2MAIN STEAM SAFETY VALVE PARAMETERS

<u>Property</u>	<u>Parameter</u>	
Design pressure, lb/in. ² _g	1300	I
Design temperature, °F	575	
Fluid	Saturated Steam	
Set pressure, lb/in. ² _g	1185, 1220, 1245	I
Min. capacity, lb/h at accumulation pressure	19x10 ⁶ Total (20 Valves)	
Type	Spring loaded	
Orifice area, in. ²	16	
Accumulation, %	3	
Backpressure		
Max. buildup/max superimposed, lb/in. ² _g	125/0	
Approx. dry weight, lbs.	1545	
Minimum blowdown pressure, psig	1175	
Typical materials		
Body	ASME SA 105	
Disc	ASTM A565, GR. 616	
Nozzle	ASME SA 182, GR. F316

SYSTEM 80+™

PRIMARY SAFETY VALVE

Figure
5.4.13-1

5.4.14 COMPONENT SUPPORTS

5.4.14.1 Design Basis

The criteria applied in the design of the Reactor Coolant System supports are that the specific function of the supported equipment be achieved during all normal, earthquake, safety valve actuation and Branch Line Pipe Break (BLPB) conditions. (BLPB includes feedwater line breaks and all loss-of-coolant-accident conditions resulting from breaks not eliminated by leak-before-break analysis in piping to branch nozzles of the reactor coolant system.) Specifically, the supports are designed to support and restrain the Reactor Coolant System components under the combined Safe Shutdown Earthquake and Branch Line Pipe Break loadings in accordance with the stress and deflection limits of Section III, ASME Code.

B

B

5.4.14.2 Description

Figure 5.4.14-1 illustrates the Reactor Coolant System support points. A description of the supports for each supported component follows:

A. Reactor Vessel Supports

The reactor vessel is supported by four vertical columns located under the vessel inlet nozzles. These columns are designed to flex in the direction of horizontal thermal expansion and thus allow unrestrained heatup and cooldown. They also act as holddown devices for the vessel.

Horizontal keyways located alongside the upper portion of the column guide the vessel during thermal expansion and contraction of the Reactor Coolant System and maintain the vessel centerline.

Four horizontal keys are welded to the bottom vessel head. The column base plate acts as a keyway for these keys to restrain the bottom of the vessel.

The supports are designed to accept normal, seismic, and Branch Line Pipe Break loads. Irradiation effects are addressed in the fracture mechanics analysis of columns, in combination with the design basis loads, to ensure that structural integrity will be maintained.

D

Reactor vessel supports are shown in Figure 5.4.14-2.

B. Steam Generator Supports

The steam generator is supported at the bottom by a sliding base bolted to an integrally attached conical skirt. The sliding base rests on low friction bearings which allow unrestrained thermal expansion of the Reactor Coolant System. Two keyways within the sliding base mate with embedded keys to guide the movement of the steam generator during expansion and contraction of the Reactor Coolant System and limit movement of the bottom of the steam generator during seismic events and Branch Line Pipe Breaks. | D

A system of keys and snubbers located on the steam drum guide the top of the steam generator during expansion and contraction of the Reactor Coolant System and provide support during seismic events and following Branch Line Pipe Breaks. | D

Typical steam generator supports are shown in Figure 5.4.14-3.

C. Reactor Coolant Pump Supports

Each reactor coolant pump is provided with four vertical support columns, four horizontal support columns, and two horizontal snubbers. The rigid structural columns provide support for the pumps during normal operation, earthquake conditions, and Branch Line Pipe Breaks. An illustration of the pump supports is shown in Figure 5.4.14-4. | D

D. Pressurizer Supports

The pressurizer is supported by a cylindrical skirt welded to the pressurizer and bolted to the support structure. The skirt is designed to withstand deadweight and normal operating loads as well as the loads due to earthquakes, safety valve actuation, and Branch Line Pipe Breaks. Four keys welded to the upper shell provide additional restraint for earthquake, safety valve actuation, and Branch Line Pipe Break conditions. | D

5.4.14.3 Evaluation

The structural integrity of the reactor coolant system support components is ensured by quality assurance inspections in accordance with Section III of the ASME Code during fabrication. The non-integral supports are procured by individual equipment specifications which impose appropriate quality assurance requirements commensurate with the respective component's functions.

During pre-operational testing of the Reactor Coolant System, the support displacements will be monitored for concurrence with calculated displacements and/or clearances. Subsequent inspections of supports which are integral with Reactor Coolant System components will be in accordance with Section XI of the ASME Code.

Amendment B
March 31, 1988

SYSTEM 80+™

REACTOR COOLANT SYSTEM ARRANGEMENT
AND SUPPORT POINTS

Figure
5.4.14-1

SYSTEM 80+™

REACTOR VESSEL SUPPORTS

Figure
5.4.14-2

PLAN VIEW A-A

LOWER SUPPORTS

Amendment D
September 30, 1988

SYSTEM 80[™] +

STEAM GENERATOR SUPPORTS

Figure
5.4.14-3

SYSTEM 80+™

REACTOR COOLANT PUMP SUPPORTS

Figure
5.4.14-4

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APPENDIX 5A
OVERPRESSURE PROTECTION
FOR
THE SYSTEM 80+ STANDARD DESIGN

I

APPENDIX 5A
OVERPRESSURE PROTECTION
FOR
THE SYSTEM 80+ STANDARD DESIGN

ABSTRACT

This Appendix documents the adequacy of overpressure protection provided for Combustion Engineering's (C-E's) System 80+ Standard Design steam generators and reactor coolant system.

Overpressurization of the Reactor Coolant System (RCS) and steam generators is precluded by means of primary safety valves, secondary safety valves and the Reactor Protection System (RPS). Pressure relief capacity for the steam generators and RCS is conservatively sized to satisfy the overpressure requirements of the ASME Boiler and Pressure Vessel Code, Section III. The safety valves in conjunction with the RPS, are designed to provide overpressure protection for a loss-of-load incident with a delayed reactor trip.

APPENDIX 5A

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1.0 INTRODUCTION

Overpressure protection for the System 80+ Standard Design steam generators and reactor coolant system is in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III. Overpressure protection is provided via primary safety valves, secondary safety valves, and the Reactor Protection System. The worst case transient, loss-of-load, in conjunction with a delayed reactor trip, is the design basis for evaluating the adequacy of the primary safety valves. The primary safety valves, secondary safety valves, and Reactor Protection System are designed to maintain the reactor coolant system pressure below 110% of design pressure during worst case transients. The secondary safety valves are sized conservatively to pass steam flow at greater than the proposed licensed power level of 3817 Mwt. Steam generator pressure is limited to less than 110% of steam generator design pressure during worst case transients. | I

2.0 ANALYSIS

2.1 METHOD

C-E has performed a parametric study to determine the design basis incident for determining the adequacy of the capacity of the primary safety valves. The design basis incident is a loss-of-load with a delayed reactor trip. The analysis is performed using digital computer codes which accurately model the thermal, hydraulic, and nuclear performances of the reactor coolant and steam systems. The digital computer codes used in the transient analysis include models for the reactor kinetics, thermal and hydraulic performance of the Reactor Coolant System (RCS), and the thermal and hydraulic performance of the steam generators. The computer simulation includes the effects of reactor coolant pump performance, elevation heads, inertia of surge line water and the pressure drop in the surge line. Worst case steady state initial conditions and nuclear parameters are assumed for the analysis. The reactor is assumed to trip at a RCS pressure of 2464 psia while the primary safety valves are assumed to lift at a pressure of 2525 psia, which is 25 psia above the system design pressure. During the analysis, the valve capacity is varied parametrically to determine the sensitivity to peak primary pressure for this event. | I

2.2 ASSUMPTIONS

- A. At the onset of the loss-of-load transient, the reactor coolant and main steam systems are at maximum rated output plus a two percent uncertainty. By choosing the highest possible power output, the heatup rate of the primary loop is maximized along with the rate of pressurization.

- B. Moderator temperature coefficient is zero. Analytical studies supported by core data show that the moderator temperature coefficient can vary between zero and $-3.5 \times 10^{-4} \Delta\rho/^\circ\text{F}$ for various phases of core life. Therefore, a coefficient of zero is chose to maximize the power/pressure transient.
- C. Doppler coefficient of -1.3×10^{-5} delta K/K/F is used in the loss-of-load analysis. Actual operating coefficient can be expected to range from -1.5×10^{-5} at zero power to $-1.3 \times 10^{-5} \Delta\rho/^\circ\text{F}$ delta K/K/F at full power. By choosing a relatively small Doppler coefficient, the reduction in reactivity with increasing fuel temperature is minimized thereby maximizing the rate of power increase.
- D. No credit is taken for letdown, charging, pressurizer spray, turbine bypass, or feedwater addition after the turbine trip in the loss-of-load analysis. Letdown and pressurizer spray both act to reduce primary pressure. By not taking credit for these systems the rate of pressurization is increased. By not taking credit for the addition of feedwater the steam generator secondary inventory will be depleted at a faster rate. This in turn reduces the capacity of the steam generator to remove heat from the primary loop and maximizes the rate of primary pressurization.
- E. No credit is taken for CEA motion prior to the reactor trip. Normally the Reactor Power Cutback System (RPCS) would automatically effect a partial scram to rapidly reduce reactor power. The Reactor Regulating System (RRS) would also (slowly) insert CEAs to reduce reactor power.
- F. The analysis reflects consideration of plant instrumentation error and safety valve setpoint uncertainty. For example, all safety valves are assumed to open at their maximum opening pressure. This extends the period of time before energy can be removed from the system. The reactor trip setpoint errors are always assumed to act in such a manner that delays the reactor trip and results in maximum pressurization.
- G. Pressurizer pressure at the onset of the event was varied from 2200 psia to 2275 psia. This is the normal range of plant steady state operation without automatic backup heater or pressurizer spray actuation. The maximum primary pressure during the loss of load event was insensitive to the initial pressure over this pressure range.

2.2.1 SECONDARY SAFETY VALVES

The secondary safety valves are conservatively sized to pass excess steam flow. This limits steam generator pressure to less than 110% of steam generator design pressure during worst case transients. The plant's secondary safety valves consist of three banks of valves with staggered set pressures. The valves are spring-loaded type safety valves in accordance with ASME Boiler and Pressure Vessel Code, Section III.

The discharge piping serving the secondary safety valves is designed to accommodate rated relief capacity without imposing unacceptable backpressure on the safety valves. | I

Figure 5A-2 depicts the steam generator pressure transient for this worst case loss-of-load incident. As can be seen in Figure 5A-2, the steam generator pressure remains below 110 percent of design pressure during the event.

2.2.2 PRIMARY SAFETY VALVES

The in containment refueling water tank spargers, inlet and discharge piping are sized to preclude unacceptable pressure drops and backpressure which would adversely affect valve operation. Primary safety valve backpressure is limited by the design pressure of the valve bellows. These bellows prevent any accumulated backpressure from being imposed on the valve spring, thus allowing valve operation at its design setpoint rather than at its setpoint plus backpressure. | I

The design basis event for evaluating the adequacy of the primary safety valves is a loss of turbine-generator load in which the reactor is not immediately tripped. The reactor scram is initiated by the second safety-grade signal from the reactor protection system. The first safety-grade signal and all possible control grade reactor trips are ignored. No credit is taken for any pressure-reducing devices except the primary and secondary safety valves. In reality, the loss of load event would be terminated by a number of reactor trip signals generated before the second safety grade signal. These include: | I

- A. Steam generator low level trip.
- B. High pressurizer pressure trip.
- C. Manual trip.

If the second safety-grade trip were to become inoperative, other reactor trips would proceed to shut the reactor down as their setpoints are exceeded.

The loss-of-load event was parametrically analyzed with various sizes of primary safety valves. The results of the analyses are shown in Figure 5A-1 along with the minimum design safety valve capacity. The maximum primary pressure is relatively insensitive to the capacity of the primary safety valves for the range of valve capacities presented. The reactor trip and subsequent power reduction rapidly reduces the insurge into the pressurizer and has a much larger effect in mitigating the pressure excursion than the primary safety valve discharge. Further decreases in the safety valve capacity would eventually have a larger effect on the peak primary pressure.

Figures 5A-2, 5A-3 and 5A-4 present the steam generator pressure, maximum Reactor Coolant System (RCS) pressure and core power versus time for the worst case loss load event. Figures 5A-2 and 5A-3 show the maximum steam generator pressure and reactor coolant system pressures remain below 110% of design pressure during this worst case transient.

Note that the maximum RCS pressure occurs at the pump discharge while the primary safety valves and the reactor trip are actuated on pressurizer pressure.

The first and second banks of secondary safety valves open at approximately 5.6 and 7.7 seconds. The third bank of safety valves is not actuated. The secondary safety valves remove energy from the RCS and thus mitigate the pressure excursion. The primary safety valves are conservatively assumed to open at 1 percent above the normal RCS design pressure. This occurs 5.05 seconds after the initiation of the event.

In the event that a complete loss of load the protection provided by the second safety grade reactor trip, primary safety valves and secondary safety valves is sufficient to assure that the integrity of the RCS and main steam system is maintained and that the minimum DNB ratio is not less than 1.24.

2.2.3 ACCEPTABILITY OF SAFETY VALVE BLOWDOWN

2.2.3.1 Background

Full scale, full pressure prototypical testing of pressurizer safety valves was performed by EPRI in 1981 (Reference 1). The blowdown settings required to insure stable valve operation during the blowdown from the set pressure were above the 5% setting specified in the ASME Code. In order to insure that the extended blowdown would not adversely affect overpressure protection or plant operation, analyses were performed to evaluate the NSSS response. The analyses described below demonstrate that a blowdown setting, including associated uncertainties, of 18.5% is acceptable.

2.2.3.2 Results of Evaluation

An extended blowdown of the safety valves could result in swelling of the pressurizer liquid level due to flashing and possible liquid carryover through the safety valves. Since the safety valve design specification requires dry saturated steam flow conditions, it is desirable to show that these conditions are maintained during the extended blowdown. It is also desirable to verify that the RCS remains in a subcooled condition in order that steam bubble formation in the RCS is precluded.

A computer analysis was performed of the loss-of-load event with delayed reactor trip, similar to that used in safety valve capacity evaluation, except that a conservative 20% safety valve blowdown and initial conditions biased to maximize pressurizer liquid level were assumed. The purpose of this analysis was to determine the pressurizer liquid level response and the RCS subcooling under these conservative conditions. For additional conservatism, adjustments were made to the computer-calculated pressurizer level on the basis of a very conservative pressurizer model. This model assumed that the initial saturated pressurizer liquid did not mix with the cooler insurge liquid, that the initial liquid remained in equilibrium with the pressurizer steam space, and that the steam which flashed during blowdown remained dispersed in the liquid phase and caused the liquid level to swell. The adjusted pressurizer water level vs time curve showed a maximum level of 78%, Reference 2, (1874 ft³), below the safety valve nozzle elevation which is greater than 100% level, so that dry saturated steam flow to the safety valves is assured throughout the blowdown. The computer analysis also showed that adequate subcooling was maintained in the RCS during the blowdown, so that steam bubble formation is precluded.

In addition, the System 80+ Standard Design safety analyses assumed a 18.5% blowdown below nominal set pressure for the pressurizer safety valves in lieu of the 5% specified by the ASME Code. For the feedwater line break event analysis, which produces the greatest increase in pressurizer level, the increased blowdown did not result in the pressurizer liquid level reaching the safety valve nozzle elevation, thus ensuring normal safety valve operation. Further, subcooling in the RCS was maintained during the blowdown.

In summary, analysis show that adequate plant overpressure protection and RCS subcooling are ensured during a blowdown of 18.5% below nominal pressurizer safety valve set pressure.

3.0 **CONCLUSIONS**

The System 80+ Standard Design, steam generators, and reactor coolant system are protected from overpressurization in accordance with the guidelines set forth in the ASME Boiler and Pressure Vessel Code, Section III. Peak reactor coolant system and secondary system pressures are limited to 110% of design pressures during a worst case loss load event. Overpressure protection is afforded by primary safety valves, secondary safety valves, and the Reactor Protection System. | I

4.0 **REFERENCES**

1. CEN-227, "Summary Report on the Operability or Pressurizer Safety Relief Valves in C-E Designed Plants", December 1982. | I
2. Water level expressed as the percentage of the distance from the lower level nozzle to the upper level nozzle.

Amendment I
December 21, 1990

SYSTEM 80+™

OPTIMIZED SAFETY VALVE CAPACITIES

Figure

5A-1

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December 21, 199

SYSTEM 80 +

STEAM GENERATOR PRESSURE vs TIME
COMPLETE LOSS OF TURBINE GENERATOR LOAD

Figure

5A-2

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SYSTEM 80 +

MAXIMUM REACTOR COOLANT SYSTEM PRESSURE vs TIME
FOR WORST CASE LOSS OF LOAD INCIDENT

Figure
5A-3

Amendment I
December 21, 199

SYSTEM 80 +

MAXIMUM REACTOR POWER vs TIME
FOR WORST CASE LOSS OF LOAD INCIDENT

Figure

5A-4

APPENDIX 5B

STRUCTURAL EVALUATION OF STEAM LINE BREAK

FOR STEAM GENERATOR INTERNALS

APPENDIX 5B

STRUCTURAL EVALUATION OF STEAM LINE BREAK
FOR STEAM GENERATOR INTERNALS

ABSTRACT

This report documents the structural adequacy of Combustion Engineering's (C-E's) System 80+ ALWR steam generator internals to withstand a steam line rupture. | D

The steam generator separator deck, shroud and tubes are subjected to a hypothetical steam line break accident. The resulting stresses in the structures are compared to ASME Boiler and Pressure Vessel Code, Section III allowables to determine the adequacy of the structure. | D

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APPENDIX 5B

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5B-1	Peak Pressure Loads
5B-2	Comparison of SYSTEM 80 Margins to Anticipated SYSTEM 80+ ALWR Margins

D

1.0 INTRODUCTION

The hydraulic loadings on the steam generator internals resulting from the rupture of one 28.0 inch I.D. main steam line nozzle are calculated using the CEFLASH computer code to perform the hydrodynamic analysis of the steam generator blowdown during the accident. The structural analysis of the System 80+ ALWR steam generator internal parts which are significantly loaded by the Main Steam Line Break (MSLB) accident, as established by the above CEFLASH analysis, is accomplished by adjusting the stresses calculated for the System 80 steam generator by the differences in pressure loadings and size of the parts. The results of these adjustments show the expected margins of safety for the System 80+ ALWR steam generator to be smaller than the margins for the System 80 steam generator. However, the expected stress levels are still much less than the allowed stress levels defined by the ASME Code, Section III.

2.0 ANALYSIS

2.1 DISCUSSION

2.1.1 HYDRODYNAMIC ANALYSIS

The hydrodynamic analysis to establish hydraulic pressure loads on the steam generator internals is performed using the CEFLASH computer code. Details of this analysis are given in Reference 1. The analysis is done for the 15% and 0% power conditions. Pressure loadings for the MSLB accident are given in Table 5B-1. The basic assumptions for the MSLB hydrodynamic analysis are:

- A. Rupture of one 28.0 inch steam outlet nozzle.
- B. A break opening time of .0094 seconds for the rupture of the steam line.
- C. A Moody critical flow correlation with a discharge coefficient of 1.0.

2.1.2 STRUCTURAL EVALUATION

The structural evaluation of the System 80+ ALWR steam generator internals is based on the structural analysis of the System 80 steam generator internals presented in References 2 and 3. The following internal parts are considered to be significantly loaded by the MSLB accident and are evaluated.

- 1. Separator Deck
- 2. Shroud
- 3. Tubes

The basis of evaluation and comparison is the margin of allowable stress for the System 80 and System 80+ ALWR steam generators. The margin is calculated as the percentage of remaining allowable stress, or:

$$\text{Margin (\%)} = \frac{[\text{Allowable Stress} - \text{Calculated Stress}]}{\text{Allowable Stress}} \times 100$$

The allowable stress is the elastic allowable stress as defined in the ASME Code, Section III. The margins for the System 80+ ALWR steam generator are based on adjustments of the calculated System 80 stresses by the ratios of hydraulic pressure loading and part size. A comparison of the margins is given in Table 5B-2. Materials for the parts of the System 80+ ALWR generator are the same as materials for the System 80 generator, except for the tubing. The tube differential pressure is conservatively considered to be 2250 psi during the MSLB event. This is based on the operating pressure with the assumption that the secondary pressure has decayed to zero.

2.2 CONCLUSIONS

The results of the evaluation, which are given in Table 5B-2, show that the anticipated margins of the parts for the System 80+ ALWR steam generator are smaller than the calculated margins for the System 80 steam generator. However, the smallest anticipated margin for the System 80+ ALWR generator is 60.0% for the shroud. Therefore, the separator deck, tubes, and shroud are considered to be adequately designed to withstand a hypothetical Main Steam Line Break accident.

3.0 REFERENCES

1. Combustion Engineering Calculation, ST-801, Loads for Steam Line Break, July 1988.
2. System 80, CESSAR FSAR, Appendix 5B, Amendment No. 7, March 31, 1982.
3. Combustion Engineering Report No. CENC-1479, Analytical Report for Arizona Public Service Company, Palo Verde Unit No. 3 Steam Generators, August, 1981.
4. ASME Boiler and Pressure Vessel Code, Section III for Nuclear Vessels.

TABLE 5B-1

PEAK PRESSURE LOADS

0% and 15% Power

<u>Location</u>	<u>ΔP (psi)</u>	<u>Time (sec)</u>	<u>Direction</u>
Separator Deck	28.5	.10	Upward
Upper Shroud	29.9	.30	Tension
Lower Shroud	36.6	.40	Tension
Tube Bend Region	26.6	.10	Upward

D

TABLE 5B-2**COMPARISON OF SYSTEM 80 MARGINS
TO ANTICIPATED SYSTEM 80+ ALWR MARGINS**

<u>Part</u>	<u>Stress Category</u>	<u>Allowable Stress (ksi)</u>	<u>System 80 Margin (%)</u>	<u>Anticipated System 80+ ALWR Margin (%)</u>
Separator				
Deck	PM+PB	73.5	90.7	84.0
Stiffener	PM+PB	73.5	87.8	80.0
Tubes	PM	56.0	68.2	68.2
Shroud	PM	49.0	90.8	60.0

D

APPENDIX 5C

**STRUCTURAL EVALUATION OF FEEDWATER LINE BREAK
FOR STEAM GENERATOR INTERNALS**

APPENDIX 5C

STRUCTURAL EVALUATION OF FEEDWATER LINE BREAK
FOR STEAM GENERATOR INTERNALS

ABSTRACT

This report documents the structural adequacy of Combustion Engineering's (C-E's) System 80+ ALWR steam generator internals to withstand a feedwater line break.

D

The economizer divider plate, support cylinder, cold leg flow distribution plate and feedwater box are subjected to a hypothetical feedwater line break during 100% power operation. The resulting stresses in the structures are compared to ASME Boiler and Pressure Vessel Code, Section III allowables to determine their acceptance.

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5C-1	Peak Pressure Loads
5C-2	Comparison of SYSTEM 80 Margins to Anticipated SYSTEM 80+ ALWR Margins

D

1.0 INTRODUCTION

The hydraulic loadings on the steam generator internals resulting from the rupture of one 14.0 inch I.D. feedwater nozzle are calculated using the CEFLASH computer code to perform the thermohydrodynamic analysis of the accident. The structural analysis of the System 80+ ALWR steam generator internal parts which are significantly loaded by the Feedwater Line Break (FLB) accident, as established by the above CEFLASH analysis, is accomplished by adjusting the stresses calculated for the System 80 steam generator by the differences in pressure loadings and size of the parts. The results of these adjustments show that the expected margins of safety for the System 80+ ALWR steam generator are greater than the margins for the System 80 steam generator. D

2.0 ANALYSIS**2.1 DISCUSSION****2.1.1 HYDRODYNAMIC ANALYSIS**

The hydrodynamic analysis to establish hydraulic pressure loads on the steam generator internals is performed using the CEFLASH computer code. Details of this analysis are given in Reference 1. The analysis is done for the 100% power conditions. Pressure loadings for the FLB accident at 100% power are given in Table 5C-1. The basic assumptions for the FLB hydrodynamic analysis are:

- A. Rupture of one 14.0 inch feedwater inlet nozzle.
- B. A break opening time of .001 seconds based on a mechanistic analysis of slot ruptures. D
- C. A Moody critical flow correlation with a discharge coefficient of 1.0.

2.1.2 STRUCTURAL EVALUATION

The structural evaluation of the System 80+ ALWR steam generator internals is based on the structural analysis of the System 80 steam generator internals presented in References 2 and 3. The following internal parts are considered to be significantly loaded and are evaluated.

- 1. Feedwater Box
- 2. Economizer Divider Plate
- 3. Flow Distribution Plate
- 4. Support Cylinder

The basis of evaluation and comparison is the margin of allowable stress for the System 80 and System 80+ ALWR steam generators. The margin is calculated as the percentage of remaining allowable stress, or:

$$\text{Margin (\%)} = \frac{[\text{Allowable Stress} - \text{Calculated Stress}]}{\text{Allowable Stress}} \times 100$$

The allowable stress is the elastic allowable stress as defined in the ASME Code, Section III. The margins for the System 80+ ALWR steam generator are based on adjustments of the calculated System 80 stresses by the ratios of hydraulic pressure loading and part size. A comparison of the margins is given in Table 5C-2. Materials for the parts of the System 80+ ALWR generator are the same as materials for the System 80 generator.

2.2 CONCLUSIONS

The results of the evaluation, which are given in Table 5C-2, show that the anticipated margins of the parts for the System 80+ ALWR steam generator are larger than the calculated margins for the System 80 steam generator. The smallest anticipated margin for the System 80+ ALWR generator is 51% for the flow distribution plate. Thus, the feedwater box, economizer divider plate, flow distribution plate, and support cylinder are considered to be adequately designed to withstand a hypothetical Feedwater Line Break.

3.0 REFERENCES

1. Combustion Engineering Calculation, ST-801, Loads for Feedwater Line Break, July 1988.
2. System 80, CESSAR FSAR, Appendix 5C, Amendment No. 7, March 31, 1982.
3. Combustion Engineering Report No. CENC-1479, Analytical Report for Arizona Public Service Company, Palo Verde Unit No. 3 Steam Generators, August, 1981.
4. ASME Boiler and Pressure Vessel Code, Section III for Nuclear Vessels.

TABLE 5C-1**PEAK PRESSURE LOADS****100% Power**

<u>Location</u>	<u>ΔP (psi)</u>	<u>Time (sec)</u>	<u>Direction</u>
Feedwater Box	588	.015	Downward
Top of Divider Plate	15.2	.007	Toward Economizer
Bottom of Divider Plate	62.0	.007	Toward Economizer
Flow Distribution Plate	18.4	.013	Downward
Shroud Hot Leg	7.4	.055	Tension
	-6.5	.148	Compression
Shroud Cold Leg	10.0	.018	Tension
	-21.0	.050	Compression

TABLE 5C-2**COMPARISON OF SYSTEM 80 MARGINS**
TO ANTICIPATED SYSTEM 80+ ALWR MARGINS

<u>Part</u>	<u>Stress Category</u>	<u>Allowable Stress</u>	<u>System 80 Margin (%)</u>	<u>Anticipated System 80+ ALWR Margin (%)</u>
Feedwater Box	PM+PB	73.5	53.5	65.0
Economizer Divider Plate	PM+PB	73.5	43.5	56.0
Economizer Divider Plate Lug	PM+PB	58.7	15.5	83.0
Flow Distribution Plate	PM+PB	67.6	43.2	51.0
Support Cylinder	PM+PB	67.6	84.4	88.0

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