5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMPS

5.4.1.1 Design Bases

In addition to the design transients in Subsection 3.9.1.1, the reactor coolant pumps are designed to:

a) Withstand 4000 cycles (pressure) of reactor coolant pump startup and stopping without exceeding code allowable stress limits. This transient is classified as a normal condition event.

b) Provide sufficient moment of inertia to reduce the flow decay through the core upon loss of pump power, ensuring that fuel design limits are not exceeded.

c) Prevent reverse rotation of the pump upon loss of pump power with the other pumps operating.

d) Operate without cooling water for periods up to three minutes without incurring seal damage.

5.4.1.2 Description

The reactor coolant is circulated by four vertical, single bottom suction, horizontal discharge centrifugal, motor driven pumps. (See Figures 5.4-1 and 5.4-2). The piping and instrumentation diagram for the reactor coolant pump is shown in Figure 5.1-3. Design parameters for the pumps are listed in Table 5.4-1. The predicted pump performance curve is shown in Figure 5.4-3.

5.4.1.2.1 Reactor Coolant Pump Assembly

The reactor coolant pump assembly consists of the following:

a) Pump Case

b) Rotating Assembly
(Containing the impeller which is welded to the shaft)

c) Mechanical Seal Cartridge Assembly

d) Motor Mount

e) Motor Assembly

5.4.1.2.1.1 Pump Case and Motor Mount

The reactor coolant pump motor is connected to and supported by the pump case through the motor mount. There are two openings on opposite sides of the motor mounts that provide access for assembly of the flanged rigid coupling between the motor and pump and for seal cartridge replacement. The pump case includes the seal heat exchanger, which cools the seal cartridge assembly.

5.4.1.2.1.2 Rotating Assembly

The pump rotating assembly consists of the impeller, a hydrostatic radial bearing, and the rotating elements of the seal cartridge assembly. The hydrostatic radial bearing maintains radial alignment of the shaft and is located just above the impeller. The upper radial bearings and an axial thrust bearing are located in the motor.

5.4.1.2.1.3 Mechanical Seal Cartridge Assembly

→(EC-6256, R302; EC-18520, R304)

The seal cartridge consists of four multiface type mechanical seals; three full-pressure seals mounted in series, and a fourth low pressure seal designed to withstand system operating pressure when the pump is not operating. A controlled bleedoff flow through the seals is used to cool the seals and to equalize the pressure drop across each seal. The seals are capable of operation without cooling water for up to ten minutes without incurring damage. The controlled bleedoff flow is collected in the volume control tank of the Chemical and Volume Control System (CVCS). Leakage past the vapor seal is routed to the Containment sump via the Floor Drain System.

←(EC-6256, R302; EC-18520, R304)

The seal cartridge concept reduces the time required for seal maintenance, thereby lowering personnel radiation exposure. The seal cartridge can be removed without draining the pump case. Details of the seal are shown in Figure 5.4-2.

5.4.1.2.1.4 Motor Assembly

The motor assembly includes the following:

a) Air Cooler

b) Motor Bearing Lubrication

c) Oil Lift Pumps

d) Motor Shaft

e) Upper and Lower Radial Guide Bearings

f) Axial Thrust Bearing

g) Flywheel

h) Anti-Reverse Rotation Device

i) Motor

Cooling water for all motor heat exchangers is supplied from the Component Cooling Water System. Heat exchangers supplied include the upper bearing lube oil cooler, lower bearing oil cooler and motor air cooler.

→(EC-8458, R307)
The oil lubrication system consists of two high pressure oil lift pumps with attached low pressure oil pumps, a lube oil cooler, motor lower bearing oil cooler, and slinger assembly all within the motor housing. During normal pump operation, the coil slinger maintains oil flow through the motor upper bearings. During pump startups and shutdowns, both the selected high pressure oil lift pump and its associated low pressure oil pumps are manually started to maintain sufficient oil pressure to the bearing and anti-reverse rotation device. If the oil pumps have not been started manually when the motor is stopped, the selected oil pump starts automatically, on a reactor coolant pump breaker trip. After the reactor coolant pump is at rated speed or has stopped, the operator must stop the lube oil pumps manually (see Subsection 5.4.1.5.2.3).
←(EC-8458, R307)

The motor bearing support system consists of a Kingsbury double acting thrust bearing and two radial bearings located above and below the motor rotor.

→(EC-8458, R307)
To improve pump coastdown characteristics in order to meet system requirements during a loss of pump power condition, the flywheel and motor-pump rotating assembly has a total moment of inertia of 106,500 lbm-ft^2.
←(EC-8458, R307)

The reactor coolant pumps are designed with an anti-reverse rotation device (see Figure 5.4-4) to prevent reverse rotation of the pump. This device prevents reverse rotation after the pump stops from normal speed (1183 rpm) while the other pumps continue to operate.

The pump assembly is sized for continuous operation while pumping primary coolant with a specific gravity of 1.0. The motor is designed for 500 heatup cycles, during which the horsepower demand decreases from 9700 to 7800 over a period of approximately seven hours. The motors are also designed to start and accelerate to speed under full load at 80 percent of rated voltage. The pump and motor assembly is designed to operate in a radiation environment of 10.5×10^6 rads (integrated) over a period of 40 years with nominal maintenance. Section 12.3 contains a further discussion of radiological considerations for these pumps.

5.4.1.3 Evaluation

→(EC-38245, R307)
The reactor coolant pumps are sized to deliver flow that equals or exceeds the design flow rate used 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 were performed to evaluate reactor coolant pump performance during the post core load hot functional testing to verify adequate flow.
←(EC-38245, R307)
→(DRN 03-2059, R14; EC-8458, R307)
The actual nominal RCS flowrate is approximately equal to 110% of the original design value of 99,000 gpm per pump. This results in a mass flowrate of approximately 163.5 Mlb/hr under normal operating conditions.
←(DRN 03-2059, R14; EC-8458, R307)
→(EC-30077, R 305; EC-38245, R307)
Leakage from the pump past the pump shaft is controlled by the shaft seal assembly. Coolant entering the seal chambers is cooled and collected in closed systems so that reactor coolant leakage to containment is essentially zero. The normal vapor stage leakoff is routed to the Containment Sump. In the event of a seal malfunction, instrumentation in the form of pressure transducers, a flow meter, and a temperature detector are provided to alert the operator to a potential problem.
←(EC-30077, R 305; EC-38245, R307)
→(DRN 05-150, R14)
Component cooling water to the reactor coolant pumps is not required to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR50.67. Low cooling flow to each pump is alarmed
←(DRN 05-150, R14)

in the main control room. Upon sounding of four alarms showing that flow is not available to all pumps, the operator trips the reactor and then trips the pumps allowing the system to be cooled down by natural circulation flow.

→(DRN 00-1059, R11-A; EC-38245, R307)

The reactor coolant pumps, by design and field experience, are not susceptible to a loss-of-coolant accident (LOCA) due to seal failure resulting from loss of seal cooling water. The pumps are equipped with four series-arranged face seals, all of which are designed for 2500 psid. The ΔP across any one of the three main seals during normal operation is 750 psi. The loss of any single seal would result in a ΔP of approximately 1100 psi. A seal leakage chamber structurally designed for 2500 psia is provided to collect controlled seal leakage and conduct it to a closed system. The fourth face seal is provided as an integral part of the seal leakage chamber to prevent liquid or gaseous leakage from escaping to the atmosphere. The seal is designed to operate normally against a backpressure of 25 to 250 psia and is capable of holding against 2500 psia in the static condition and during coastdown following failure of the three series-arranged main seals. When holding against 2500 psia in the static condition, seal leakage should not exceed the normal operating seal leakage.

←(DRN 00-1059, R11-A)

Four reactor coolant pumps with seals of similar design have been operated for up to 40 minutes with no component cooling water flow. While there was some increase in controlled seal leakage (to the closed system), the mechanical seals were subsequently dismantled and refurbished without finding major damage. Therefore, a loss of component cooling water flow to the reactor coolant pumps for up to 40 minutes is not expected to result in a complete seal failure.

←(EC-38245, R307)

The component cooling water supply to the RCP's is isolated upon a Containment Spray Actuation Signal (CSAS). The RCP's are then stopped, resulting in no requirement for component cooling water. The CSAS is provided by a high-high containment pressure signal. Such a signal can be produced by only two accidents: (1) a LOCA, or (2) a major steam line break inside containment. In either case, the Plant Protective System trips the reactor and the operators trip the pumps.

In the event of a break in the reactor coolant pump suction piping, a high reverse flow through the pump is prevented by the anti-reverse rotation device, as described in Subsection 5.4.1.2.1.4. In the event of a discharge line break, increased flow through the pump tends to accelerate the pump impeller and flywheel in the forward direction. A detailed evaluation of this incident relating to the integrity of the flywheel is presented in Subsection 5.4.1.4.

5.4.1.4 Reactor Coolant Pump Flywheel Integrity

The design, fabrication and testing of the reactor coolant pump flywheel in relation to recommendations of Regulatory Guide 1.14 (formerly Safety Guide 14) and the results of an analysis of pump and flywheel performance during a LOCA are discussed in this section.

5.4.1.4.1 Regulatory Guide 1.14, Rev. 00 (10/27/71)

The following design conditions and material specifications for the flywheels are consistent with the recommendations of Regulatory Guide 1.14, Rev. 00, "Reactor Coolant Pump Flywheel Integrity".

5.4.1.4.1.1 Flywheel Material Specifications

The material used to manufacture the flywheel is pressure vessel quality, prepared by the vacuum melt and degassing process, ASTM-A-543 Grade B, Class I steel plate. Pump flywheel material is prepared and tested such that:

a) The nil-ductility transition temperature of the flywheel is no greater than +10°F, as obtained from drop-weight tests performed in accordance with specification ASTM-E-208.

b) The Cv upper shelf energy level, in the Wr direction, as obtained per ASTM-A-370, is no less than 50 ft lb. A minimum of three Cv specimens are tested from each plate.

→(DRN 00-1059, R11-A; 05-1392, R14-A)

c) The minimum fracture toughness of the material at the normal operating temperature of the flywheel is equivalent to a dynamic stress intensity factor (K_{IC} dynamic) and is equal to or greater than 100 Ksi $\sqrt{in.}$ Compliance is demonstrated by one of three methods listed in NRC Regulatory Guide 1.14, Rev. 00, paragraph C.1.c.

←(DRN 00-1059, R11-A; 05-1392, R14-A)

d) Each finished flywheel is subjected to a 100 percent volumetric, ultrasonic examination from the flat surface using ASME Section III, Class I criteria. This inspection is performed on the flywheel after final machining and overspeed test.

e) After flame cutting, at least 1/2 in. of stock is left on the outer and bore radii for machining to final dimensions.

f) Finished machined bores, keyways and drilled holes in the flywheel are subjected to a magnetic particle or liquid penetrant examination before final assembly.

→(DRN 03-2059, R14)

5.4.1.4.1.2 Flywheel Design Criteria

The flywheels are designed to withstand normal operating conditions, anticipated transients, and the original design basis LOCA[1] combined with the safe shutdown earthquake. Thus the following criteria are met:

←(DRN 03-2059, R14)

a) The combined primary stresses at normal operating speed do not exceed one-third of the minimum specified yield strength or one-third of the measured yield strength in the weak direction of the material.

b) The combined primary stresses at design overspeed do not exceed two thirds of the minimum specified yield strength or two-thirds of the measured yield strength in the weak direction of the material. Design overspeed is defined as 125 percent of normal operating speed. See Subsection 5.4.1.4.2 for further discussion of pump overspeed following a discharge line break.

Each flywheel is tested for one minute at the design overspeed of 1500 rpm.

The flywheel is accessible for 100 percent in-place volumetric ultrasonic inspection. The motor assembly is designed to allow such inspection with a minimum of motor disassembly.

→(DRN 03-2059, R14)

[1] The effects of the original design basis LOCA (i.e., MCLBs, which have been eliminated via LBB, see Section 3.6.3) envelop the effects of the next limiting set of pipe breaks (BLPBs) on RCS flywheel design.

←(DRN 03-2059, R14)

5.4.1.4.2 Additional Data and Analysis

→(DRN 03-2059, R14)

As mentioned previously, a LOCA in the pump discharge leg of the reactor coolant piping tends to accelerate the pump in the forward direction during blowdown. A maximum speed of 1585 rpm has been calculated using the CE FLASH computer program, assuming a mechanistic break[1] located at the pump discharge with a simultaneous loss of electrical power to the pump motor. This is about 134 percent of normal operating speed. Even at pump speeds greater than 1585 rpm, no other failures of sufficient magnitude to increase the consequences of the initiating LOCA can be postulated except by assuming failure of the flywheel. Stress calculations have demonstrated that even at speeds up to 270 percent of normal operating speed with a sharp edged crack located to the base of the keyway, failure cannot be predicted. It is therefore concluded that adequate margin has been provided to preclude the development of a flywheel rupture by reactor coolant pump overspeed.

←(DRN 03-2059, R14)

5.4.1.5 Reactor Coolant Pump Instrumentation

The reactor coolant pumps and motors are equipped with the instrumentation necessary for proper operation and to warn of incipient failures. A description of the major channels follows. See Figure 5.1-3. Measurement channels are typical for each reactor coolant pump.

5.4.1.5.1 Temperature

5.4.1.5.1.1 Motor Stator Temperature

Each reactor coolant pump motor is provided with six thermocouples embedded in the stator windings. Indication of stator temperature is provided by the plant monitoring computer. During initial pump testing, the highest reading thermocouple will be selected for this temperature measurement channel. Should stator temperature exceed a predetermined limit, a high temperature alarm will be annunciated by the plant monitoring computer. High temperature is detrimental to motor winding insulation life, and may be caused by high ambient temperature, reduction in the cooling airflow to the stator, or inadequate time delay between successive starts of the motor.

5.4.1.5.1.2 Motor Bearing Temperatures

High temperature alarms for the oil lubricated bearings are annunciated by the plant monitoring computer at the main control board. These include:

a) Motor upper guide bearing

b) Upper and lower thrust bearings

c) Motor lower guide bearing

High temperature is indicative of bearing or oil supply problems. The motor thrust bearing temperature is continuously monitored at the main control board. Temperatures of the motor guide bearings are fed to the plant monitoring computer.

→(DRN 03-2059, R14)

[1] The effects of the original design basis LOCA (i.e., MCLBs, which have been eliminated via LBB, see Section 3.6.3) envelop the effects of the next limiting set of pipe breaks (BLPBs) on RCS flywheel design.

←(DRN 03-2059, R14)

5.4.1.5.1.3 Pump Controlled Bleedoff Temperature

The temperature of the controlled bleedoff flow is displayed in the main control room. An alarm signal is provided at the main control panel should the controlled bleedoff temperature exceed a high limit. A high temperature condition is an indication that the seal assembly or seal water cooler is not operating properly.

5.4.1.5.1.4 Lube Oil Cooler Inlet Temperature

→

The inlet temperature of the lube oil cooler is alarmed by the plant monitoring computer.

←

5.4.1.5.1.5 Lube Oil Cooler Outlet Temperature

→

The outlet temperature of the lube oil cooler is alarmed by the plant monitoring computer.

←

5.4.1.5.1.6 Anti-Reverse Rotation Device Bearing Temperature

→

Indication of the anti-reverse rotation device temperature is provided by the plant monitoring computer. A high temperature condition in the anti-reverse rotation device is alarmed by the plant monitoring computer and is an indication of lubrication problems.

←

5.4.1.5.1.7 Seal Water Cooler Component Cooling Water Outlet Temperature

The temperature of component cooling water is monitored at the seal water cooler outlet of each Reactor Coolant Pump (1A, 1B, 2A, 2B) with respective temperature elements. The temperature parameter is displayed in the main control room.

Two independent high temperature setpoints are used to annunciate an alarm in the main control room and to initiate an automatic isolation of the affected seal cooler by closing the component coolant water inlet and outlet valves. The high temperature alarm is an indication that leakage from the primary loop to the component coolant water has developed inside the seal water cooler or that the cooling water flow has decreased.

To countermand an isolation of seal coolers on a spurious signal, a control switch is provided in the main control room to allow the operator to override the high temperature signal and to reopen the component coolant water inlet and outlet valves to purge the seal water cooler for a preset time span. In that case, if the high temperature signal persists, the seal water cooler will be again automatically isolated at the end of the preset time span.

5.4.1.5.2 Pressure

5.4.1.5.2.1 Pump Seal Pressures

→

The middle, upper and vapor seal cavities in each pump are provided with pressure elements that generate a signal proportional to the pressure within the cavity. The pressure in the seals is displayed in the main control room and are monitored by the plant monitoring computer. High and low pressure alarms for the middle and upper seal cavities and a high pressure alarm for the vapor seal cavity are annunciated by the plant monitoring computer.

←

Abnormally high or low pressure is an indication of seal malfunction.

5.4.1.5.2.2 Lube Oil Cooler Pressure

A common low pressure alarm at the inlet to the lube oil cooler and anti-reverse rotation device is annunciated in the main control room.

5.4.1.5.2.3 High Pressure Oil Lift Pump Discharge Pressures

Pressure switches at each high pressure oil lift pump discharge actuate an alarm on low pressure in the main control room. In the event of a failure of one of the oil lift pumps, the second oil lift pump must be started. A separate measurement channel provides a control signal to the respective reactor coolant pump circuit, which prevents the starting of the reactor coolant pump if insufficient oil lift pressure exists. Another separate measurement channel provides local indication of pressure in the oil lift pump discharge header.

5.4.1.5.2.4 High Pressure Oil Lift Pump and Low Pressure Oil Pump Filters

→
Filters are used during motor start-up and shutdown at the suction and discharges of the high pressure oil lift and low pressure oil pumps. A high differential pressure across the filter indicates a clogged filter and is alarmed by the plant monitoring computer.
←

5.4.1.5.2.5 Reactor Coolant Pump Differential Pressure

Two independent differential pressure transmitters are provided on each reactor coolant pump. The differential pressure signal is indicated in the main control room. A calibration curve is used to relate pump differential pressure to pump flow.

5.4.1.5.2.6 Lube Oil Pressure to Anti-Rotation Device

Local pressure gages are provided in the oil line to the anti-rotation device for testing purposes only.

5.4.1.5.2.7 Reverse Rotation Indicator Switch

→
A differential pressure switch in the lube oil system mounted near the upper bearing bracket provides an indication that the reactor coolant pump motor is turning in the reverse direction. This switch alarms via the plant monitoring computer.
←

5.4.1.5.2.8 Seal Water Cooler Component Cooling Water Outlet Pressure

The component cooling water pressure is monitored at each of the reactor coolant pumps (RCP 1A, 1B, 2A, and 2B) seal water outlet. A high pressure signal annunciates an alarm in the main control room.

5.4.1.5.3 Flow

5.4.1.5.3.1 Pump Controlled Bleedoff Flow

A flowmeter is used to measure the controlled bleedoff flow from the bleedoff seal cavity to the CVCS. This instrument provides an indication of the flow rate and annunciates high and low flow alarms via the plant monitoring computer. Abnormal flow is an indication of improper seal performance.

5.4.1.5.3.2 Motor Circulating Oil System Flow

A lube oil flow switch is provided at the inlet to the lube oil cooler. Should the lube oil flow to the cooler fall below a predetermined setpoint, a low flow alarm is actuated via the plant monitoring computer.

5.4.1.5.3.3 Motor Low Pressure Oil Pump Flow

The lube oil flow to the anti-reverse rotation device is sensed by a lube oil flow switch. This switch actuates a low flow alarm via the plant monitoring computer when flow drops below the predetermined setpoint.

5.4.1.5.3.4 Motor High Pressure Oil Lift Pump Flow

Flow in the discharge header of the two high pressure oil lift pumps is sensed by a lube oil flow switch. This switch actuates a low flow alarm via the plant monitoring computer when flow drops below the predetermined setpoint.

5.4.1.5.4 Level

A level sensing transmitter in each oil reservoir transmits a signal for level indication and high and low alarms via the plant monitoring computer.

5.4.1.5.5 Vibration

Motor vibration is sensed by a vibration switch attached to the pump motor casing. Excessive vibration is alarmed by the plant monitoring computer.

Review of Generic Letter 89-15 prompted the addition of new RC pump and motor monitoring. Though not required by that letter, additional "monitoring" devices have been added to correlate data to provide input and trending information on the status of motor, shaft, bearings and seals. The addition consists of x and y direction radial proximity probes and one axial thrust probe at the motor thrust bearing, x and y direction radial proximity probes at the bottom of the motor and x and y direction radial proximity probes at the pump shaft. The output of these devices does not interact with any plant controls, but provides additional "monitoring" capability.

Review of Generic Letter 89-15 also prompted the addition of two (2) keyphasors at the RC pump coupling spacer. These provide additional monitoring of shaft rotative speed and vibration phase angle. These devices do not interact with any plant controls, but provide additional "monitoring" capability.

5.4.1.5.6 Speed

5.4.1.5.6.3 Reactor Coolant Pump Speed Sensors

Reactor coolant pump shaft speed is transmitted to the Core Operating Limit Supervisory System (COLSS) and the Plant Protective System (PPS) for continuous monitoring and protection. See Sections 7.2 and 7.7 for a further description.

5.4.1.6 Tests and Inspections

→(DRN 00-1059, R11-A)

The reactor coolant pump pressure boundary is nondestructively inspected as required by the ASME Code, Section III (see Table 5.2-1) for Class 1 components. The pump casing inspections include complete radiography and liquid penetrant testing. The pump casing, cover and seal cooling heat exchanger are subjected to a hydrostatic test at the manufacturing facility. The pump is hydrostatically pressure tested along with the RCS. In-service inspection will be performed during plant life in accordance with the ASME Code, Section XI as discussed in Subsection 5.2.4.

←(DRN 00-1059, R11-A)

All rotating parts of the pump are statically and dynamically balanced in two planes. Where possible, balancing is done for the entire assembly.

The pump assembly is performance tested in the vendor's shop in accordance with the Standards of the Hydraulic Institute to verify hydraulic performance, as well as the ability of the pumps to function as required by the specifications. The vibration levels are monitored during this test. Evidence of the pumps operating near a critical speed would be noted as excessive vibration.

Full scale seal testing is performed at rated pressure, temperature, water chemistry, and speed to demonstrate the capability of the seals to satisfactorily perform their design function.

5.4.2 STEAM GENERATORS

5.4.2.1 Design Basis

→(DRN 03-2059, R14; 06-1060, R15; EC-8458, R307)

Replacement Steam Generators, implemented in Cycle 18, are designed to transfer 3735 MWt from the RCS to the secondary system with 0% tube plugging, producing approximately 16.6 x 10^6 lbm/hr of 837.2 psia saturated steam. At 10% tube plugging, the RSGs produce approximately 16.6 x 10^6 lbm/hr of 818.0 psia saturated steam. Moisture separators and steam driers in the shell side of the steam

←(DRN 03-2059, R14; 06-1060, R15; EC-8458, R307)

➔(DRN 03-2059, R14; EC-8458, R307)
generator limit the moisture content of the steam to 0.10% during normal operation at full power. The steam generator design parameters are listed in Table 5.4-2. The steam generators, including the tubes, are designed for the RCS transients listed in Subsection 3.9.1.1 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 are capable of sustaining the following additional design transients without exceeding code allowable stress limits:
🡐(DRN 03-2059, R14)

a) Ten 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 1375 psia psi (test condition).

b) Secondary side tube leak test

Cycles	Shell Side Pressure (PSIG)	Shell Side Temperature (°F)
400	200	60-250
200	400	60-250
120	600	60-250
80	840	60-250

c) 5.700 cycles of adding feedwater to the steam generators through the main feedwater nozzle when in hot standby conditions (normal operations). The following conditions and cycles are based on actual operating conditions:

Cycles	Feedwater temp. (°F)	Flow change (gpm)	Rate for flow change
600	40	50 to 1100	1 second
948	40	0 to 450	1 second
1302	40	0 to 50	70 seconds
600	70	50 to 1100	1 second
948	70	0 to 450	1 second
1302	70	0 to 50	70 seconds

➔(DRN 00-1059, R11-A)
d) 600 cycles of feeding during a three hour low power transient.

Cycles	Power change (%)	Feedwater temp change (°F)	Flow change (lb/hr)
300	0 to 15	40 to 260	25,000 to 1.04×10^6
300	15 to 0	260 to 40	1.04×10^6 to 25,000

🡐(DRN 00-1059, R11-A)
e) Four thousand pressure transients of 113 psi across the primary divider plate in either direction caused by starting and stopping reactor coolant pumps (normal condition).

f) Twenty cycles of a Loss of Main Feedwater Transient.

g) One cycle for a Design Bases Earthquake transient with a Steam Line Break.

h) One cycle for a Design Bases Earthquake transient with a Feedwater Line Break.
🡐(EC-8458, R307)
The operating pressure and temperature limits for the steam generator primary side were determined in accordance with the ASME Code, Section III, Appendix G.
➔(EC-8458, R307)
The method of fastening tubes to the tube sheet conform 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. Eight tube support plates of the flat-contact broached trifoil tube hole design are used in the RSG. The broached tube support plate is designed to reduce the tube-to-tube support plate crevice area while providing for maximum steam/water flow in the open areas adjacent to the tube. Flat tube contact geometry in the RSG provides additional dryout margin.
🡐(EC-8458, R307)

→(EC-8458, R307)
The steam generator is 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 severance at the steam nozzle.
←(EC-8458, R307)

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

Onsite cleaning and cleanliness control procedures for the steam generator were consistent with the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants", (3/16/73) and ANSI N45.2-1-1973, "Cleaning of Fluid Systems and Associated Components For Nuclear Power Plants".

5.4.2.2 Description

→(DRN 03-2059, R14; EC-8458, R307)
The steam generator is illustrated in Figure 5.4-5. Moisture-separating equipment in the shell side of the steam generators limits moisture content of the exit steam to a maximum of 0.10%. Manways and handholes are provided for access to the steam generator internals.
←(DRN 03-2059, R14; EC-8458, R307)

Reactor coolant enters at the bottom of each steam generator through the single inlet nozzle, flows upward 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.

→(DRN 02-674, R12; 03-130, R14; 06-290, R14-B; EC-8458, R307)
Feedwater enters the steam generator through the feedwater nozzle where it is distributed via a feedwater distribution ring. The feedwater ring is constructed with eccentric reducer fittings and perforated discharge nozzles which discharge at the top of the feedring to avoid regions where steam pockets can be formed. The feedwater ring is located above the elevation of the feed nozzle to minimize the time required to fill the feed nozzle during a cold water addition transient.
←(DRN 02-674, R12; 03-130, R14; 06-290, R14-B; EC-8458, R307)

The downcomer in the steam generator is an annular passage formed by the inner surface of the steam generator shell and the cylindrical shell that encloses the vertical U-tubes. Upon exiting from the bottom of the downcomer, the secondary flow is directed upward over the vertical U-tubes. Heat transferred from the primary side converts a portion of the secondary flow into steam.

→(EC-8458, R307)
Upon leaving the vertical U-tube heat transfer surface, the steam-water mixture enters the centrifugal-type separators. These impart a centrifugal motion to the mixture and separate the water particles from the steam. The water exits from the perforated separator housing and combines with the feedwater to repeat the cycle. Final drying of the steam is accomplished by passage of the steam through the single-tier banked dryers. Drain pipes located at the ends of each secondary separator dryer bank carry captured water downward into the recirculation pool. The pressure drop from the steam generator feedwater nozzles to the steam outlet nozzle is approximately 31.6 psi when operating at full power. The steam generator supports are described in Subsection 5.4.14. Secondary side overpressure protection is provided by 12 spring-loaded ASME Code safety valves mounted on the main steam lines as described in Subsection 5.4.13.

The RSGs have a 3.00 in NPS blowdown nozzle connected to the steam generator secondary water through a series of holes in the tubesheet secondary surface that intersect an internal blowdown passage.
←(EC-8458, R307)

→(EC-8458, R307)
The RSGs contain an integral flow limiting device, consisting of seven 8.53 inch I.D. venturis installed in the steam outlet nozzle. This flow restrictor reduces energy release to containment and loads on the steam generator internals during a postulated steamline break.
←(EC-8458, R307)

5.4.2.3 Evaluation

5.4.2.3.1 Steam Generator Tubes

5.4.2.3.1.1 Chemistry Compatibility

→(EC-8458, R307)
The steam generators, tubed with Alloy 690 TT, 0.75 in. OD by 0.044 for Rows 1&2 and 0.043 for Rows 3 to 138 wall tubing, incorporates a general corrosion allowance that will provide for reliable operation over the plant design lifetime.
←(EC-8458, R307)

Localized corrosion has led to steam generator tube leakage in some operating plants. Examination of tube defects that have resulted in leakage has shown that two mechanisms are primarily responsible. These localized corrosion mechanisms are referred to as (1) stress assisted caustic cracking, and (2) wastage or beavering. Both of these types of corrosion have been related to steam generators that have operated on phosphate chemistry. The caustic stress corrosion type of failure is precluded by controlling bulk water chemistry to the specification limits shown in Subsection 10.3.5. Removal of solids from the secondary side of the steam generator is discussed in Subsection 10.4.8.

Localized wastage has been eliminated by removing phosphates from the chemistry control program.

Volatile chemistry (discussed in Subsection 10.3.5) has been successfully used in all C-E steam generators that have gone into operation since 1972.

5.4.2.3.1.2 Mechanical Considerations

Because the reactor coolant pumps have a rotational speed of 1180 rpm, the imposition of exciting frequencies of 19 to 20 Hz and 95 to 100 Hz was considered in the design. The low frequency range is defined as a mechanical vibration resulting from the transmission of a mechanical impulse at the frequency of pump rotation. The upper frequency range is defined as a sinusoidal pressure vibration of $\pm$ six psi in the reactor coolant piping that contains the pump. The pressure variation results from the impeller vanes interacting with the cut-water vane at the volute outlet during each revolution of the impeller.

5.4.2.3.1.3 Hydraulic Stability of Feedwater System

Pressure pulses have been observed in some feedwater lines which were initiated by steam-water interaction causing ripple formation at a steam-water interface in the feedwater piping. This resulted in the formation of a water slug which isolated the steam in the feedwater line. As the isolated steam bubble condensed, the decreasing pressure would accelerate the water slug in the direction of the void. The kinetic energy in the slug would increase until the steam bubble collapsed. When the water slug impacted with the water filling the upstream side of the pipe, pressure pulses were generated. With a design incorporating only a small amount of entrapped steam, the steam bubble will collapse before the water slug can gain significant kinetic energy, and the intensity of the pressure pulses is reduced to negligible levels.

→(EC-18652, R304)
In Waterford 3, the potential for a significant waterhammer transient in the Feedwater System is reduced by the construction of the feedwater ring, described in Subsection 5.4.2.2. Additional assurance that the feedwater ring will remain full of water is provided by automatic initiation of the Emergency Feedwater System (see Section 7.3). In addition, a downward sloping elbow at the steam generator feedwater nozzle connects to the feedwater piping. This design minimizes the drainable volume of the feedwater piping. Hence, when the feedwater ring and nozzle are drained and steam enters this region, the surface area of subcooled water exposed to saturated steam is minimized. As a result, only a small amount of steam can be trapped in the elbow and any water slug picked up during steam-water interaction is incapable of gaining significant kinetic energy before the steam bubble is collapsed.
←(EC-18652, R304)
The effectiveness of the similar modifications has already been demonstrated at Indian Point 2, Trojan and St. Lucie I. Tests at a European plant (Doel I in Belgium) have also confirmed the effectiveness of a short horizontal run in reducing waterhammer to negligible levels.

→(EC-18652, R304)
The feedring is subject to damage from pressure transients that can occur when refilling a steam generator whose water level has fallen below the feedring. Removal of the thermal liner o-ring and clamp on both steam generators and the acceptance of missing vent piping on Steam Generator #1 allows the feedring to drain after a loss of feedwater. These changes were evaluated on the basis of the Steam Generators remaining capable of performing their safety function with a damaged feedring.
←(EC-18652, R304)

5.4.2.3.1.4 Tube Wall Thinning

→(EC-8458, R307)
Tube wall thinning acceptance criteria is specified in the Technical Specifications. The Replacement Steam Generator tube structural integrity analysis is provided in WCAP-17263-P (Reference 1).
←(EC-8458, R307)

5.4.2.3.2 Potential Effects of Tube Rupture

→(DRN 01-1283, R12)
The steam generator tube rupture accident is a penetration of the barrier between the RCS 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 Steam Bypass System. Non-condensible radioactive gases in the condenser are removed by the Main Condenser Evacuation System and discharged to the atmosphere. Analysis of a steam generator tube rupture accident, assuming complete severance of a tube, is presented in Section 15.6.
←(DRN 01-1283, R12)
Experience with nuclear steam generators indicates that the probability of complete severance of a tube is remote. The material used to fabricate the vertical U-tube is a Ni-Cr-Fe alloy. 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 Subsection 5.2.5.

5.4.2.3.3 Composition of Secondary Fluid

→(DRN 00-1059, R11-A; 02-218, R11-A)
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 11.1. Activities are based on operations with 0.1 percent failed fuel cladding, a total primary-to-secondary leakage of 100 lbm./day, and 60 gpm blowdown
←(DRN 00-1059, R11-A; 02-218, R11-A)

→(DRN 00-1059, R11-A; 02-218, R11-A)
rate to the Steam Generator Blowdown System. An evaluation of the shell side radioactivity is presented in Section 11.2. Limits for radioactivity levels in the secondary side of the steam generators and the bases for these limits are provided in Technical Specifications, Chapter 16.
←(DRN 00-1059, R11-A; 02-218, R11-A)

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 Subsection 10.3.5.

5.4.2.4 <u>Tests and Inspections</u>

5.4.2.4.1 Fabrication Tests and Inspections

The steam generator is tested in accordance with ASME Boiler and Pressure Vessel Code, Section III. The nondestructive tests, some of which are not required by the code, performed during fabrication are given in Table 5.4-3.

During design and fabrication of the steam generator, additional operations beyond the requirements of the ASME Boiler and Pressure Vessel Code, Section III, were performed by the vendor. These included ultrasonic testing for defects in tube sheet clad and ultrasonic testing of weld clad for bond integrity.

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

Steam generator performance is further verified during the initial startup tests. Provisions for onsite cleaning and cleanliness control are described in Subsection 5.2.3.

In-service Inspection will be performed during plant life in accordance with the ASME Code, Section XI as discussed in Subsection 5.2.4.

→(EC-8458, R307)
Inservice inspection of steam generator tubing will comply with the Technical Specifications.

5.4.2 REFERENCES

1. WCAP-17263-P, Revision 0, November 2010, Regulatory Guide 1.121 Analysis and Structural Integrity Performance Criterion Applications for the Waterford Unit 3 Model Delta 110 Replacement Steam Generators for a NSSS Power of 1869.6 MWt/SG.

←(EC-8458, R307)

5.4.3 REACTOR COOLANT PIPING

5.4.3.1 <u>Design Basis</u>

→(DRN 06-546, R15)
The reactor coolant loop piping is designed and analyzed for normal operation including all transients discussed in Subsection 3.9.1. Loading combinations and stress criteria associated with faulted conditions are presented in Subsection 3.9.3.1. In addition, certain nozzles are subjected to local transients that are included in the design analysis of the areas affected. Thermal sleeves are installed in the
←(DRN 06-546, R15)

surge nozzle, safety injection nozzles, and charging nozzle to accommodate these additional transients. Principal parameters are listed in Table 5.4-4. The ASME code and addenda to which the piping is designed is specified in Subsection 5.2.1.

In addition to being specified as seismic Category I, the following additional vibratory requirement is specified in the engineering specification. The various piping assemblies are designed so that no damage to the equipment is caused by the frequency ranges of 19 to 20 Hz and 95 to 100 Hz. The reasons for selecting these frequencies are the same as described in Subsection 5.4.2.3.1.2 for the steam generators. Additional presentations relating to seismic and dynamic analysis and criteria for the reactor coolant piping are contained in Subsections 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. ID pipe between the reactor vessel outlet nozzle and steam generator inlet nozzle, two 30 in. ID pipes from the steam generator's two outlet nozzles to the reactor coolant pumps suction nozzle, and two 30 in. ID pipes from the pumps discharge nozzle to the reactor vessel inlet nozzles. These pipes are referred to as leg 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. schedule 160 pipe between the pressurizer and the hot leg in Loop 1, and the spray line, a 4 in. schedule 120 pipe at the pressurizer end reduced to a 3 in. schedule 160 pipe between the pressurizer and the cold legs of Loops 1A and 1B.

To minimize the possiblity of stress corrosion cracking, the reactor coolant piping is fabricated from SA 516 GR 70 base material mill clad with type 304L stainless steel. Lines such as the surge lines, spray lines, and other small lines are totally made of stainless steel. Nozzles are shop fabricated with safe ends to preclude dissimilar-metal field welds.

→(EC-19087, R305)

Where stainless steel or Ni-Cr-Fe nozzle or safe end material is used, the safe ends are welded to the assembly after final stress relief to prevent furnace sensitization. Other precautions used in the shop and during field assembly of the piping are described in Subsection 5.2.3. Pressurizer and hot leg nozzle welds that contain NI-CR-Fe materials susceptible to primary water stress corrosion cracking (PWSCC) have received structural weld overlays using Alloy 52M weld material (EC1830).

←(EC-19087, R305)

→(DRN 03-1268, R13)

The 42 in. and 30 in. pipe diameters are selected to obtain coolant velocities that 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 maximum allowable pressure of the RCS is not exceeded. The spray line is sized to ensure a minimum spray flow of 375 gpm with both spray valves open. Pressurizer parameters are listed in Table 5.4-6.

←(DRN 03-1268, R13)

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. To reduce thermal shock, thermal sleeves are provided for all nozzles two in. or greater in diameter where fluid enters the main piping from an auxiliary system during normal operation.

Flow restricting orifices (7/32 in. diameter by one in. long) are provided in the nozzles for the RCS sampling lines, the reactor coolant pipe hot leg pressure measurement nozzles, the pressurizer level and pressure instrument lines, and the reactor coolant pump differential pressure instrument lines to limit flow in the event of a break downstream of a nozzle.

→(DRN 04-292, R13; 05-892, R14)

←(DRN 04-292, R13; 05-892, R14)

5.4.3.3 Evaluation

It is demonstrated by analysis that the reactor coolant piping is adequate for all normal operating and transient conditions of Subsection 3.9.1.1. In addition, the fully assembled RCS is subjected to the required hydrostatic tests. Fracture toughness of the reactor coolant piping is discussed in Subsection 5.2.3.

During the design phase, every effort is made to displace the frequency of the RCS piping from driving frequencies of concern by proper location of piping spring characteristics. The dynamic effects of system operation are also considered and piping restraints sized accordingly.

A discussion of the radiological considerations for the reactor coolant piping is provided in Section 12.3.

5.4.3.4 Tests and Inspections

Prior to and during fabrication of the reactor coolant piping, nondestructive testing, based on the requirements of the ASME Code (see Table 5.2-1) is applied. Table 5.4-5 summarizes the component inspection program during fabrication and construction. Tests for RCS integrity following normal opening, modification, or repair are specified in the Technical Specifications. To ascertain the integrity of the piping during plant life, necessary in-service inspections required by Section XI of the ASME Boiler and Pressure Vessel are performed where required on the reactor coolant piping. To facilitate such inspections, longitudinal weld seams have been oriented at the 90 and 270 degree locations where feasible. Removable insulation is installed to assure access to the welds. Inservice inspection of the RCS piping is further discussed in Subsection 5.2.4.

5.4.4 MAIN STEAM LINE FLOW RESTRICTIONS

→(EC-8458, R307)

Main steam line flow restrictors are not required in the main steam line between the steam generator and the main steam isolation valves. A flow venturi (24 in. throat diameter) acts as a flow restriction for main steam line breaks downstream of the venturi. Refer to Figure 10.2-4 for location of the flow venturi.
Note: See Section 5.4.2.2 for a description of integral flow restrictors.

←(EC-8458, R307)

5.4.5 MAIN STEAM LINE ISOLATION SYSTEM

The Main Steam Line Isolation System is discussed in Sections 7.3 and 10.3.

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

This Subsection is not applicable to Waterford 3.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

See Subsection 9.3.6.

5.4.8 REACTOR WATER CLEANUP SYSTEM

This Subsection is not applicable to Waterford 3.

5.4.9 MAINSTEAM LINE AND FEEDWATER PIPING

The Main Steam System piping is discussed in Subsection 10.3.6; Feedwater System piping is described in Subsection 10.4.7.

5.4.10 PRESSURIZER

5.4.10.1 Design Bases

→(DRN 06-546, R15)

The pressurizer is designed and analyzed for the transients specified in Subsection 3.9.1.1 and the following additional requirements. During heatup and cooldown of the plant, the allowable rate of temperature change for the pressurizer is increased to 200°F/hr.

←(DRN 06-546, R15)

The pressurizer is designed to:

a) Maintain RCS operating pressure

b) Compensate for changes in coolant volume during load changes

c) Contain sufficient volume to prevent draining the pressurizer as a result of a reactor trip

d) Limit the water volume to minimize the energy release during LOCA

e) Prevent uncovering of the heaters by the out-surge of water following load decreases; 10 percent step decrease and five percent per minute ramp decrease

f) Provide sufficient volume to accept the reactor coolant insurge resulting from a loss of load without the water level reaching the safety valve nozzles

g) Provide sufficient volume to yield acceptable pressure response to normal system volume changes during load change transients

→(DRN 00-1059, R11-A)

h) Achieve a total coolant volume change and associated charging and letdown flows which are as small as practical and are compatible with the capacities of the volume control tank, charging pumps and letdown control valves during load-following transients

←(DRN 00-1059, R11-A)

i) Ensure that the minimum pressure observed during transients is above the setpoint of the safety injection actuation signal, and that the maximum pressure is below the high-pressure trip.

The heater capacity is selected to provide an adequate pressurizer heatup rate during plant startup.

In addition to being specified as seismic Category I, the following additional vibratory requirement is specified in the engineering specification. The pressurizer vessel, including the heaters, baffles, and supports shall be designed such that no damage to the equipment is caused by the frequency ranges of 19-20 Hz and 95-100 Hz. The lower frequency is defined as a mechanical vibration. The design basis for the higher frequency consists of a pressure pulse of five psi which diminishes internally within the vessel.

5.4.10.2 System Description

The pressurizer is shown in Figure 5.4-6 and the design parameters are given in Table 5.4-6

The pressurizer is a cylindrical carbon steel vessel with stainless steel clad internal surfaces. A spray nozzle on the top head is used in conjunction with heaters in the bottom head to provide level and pressure control. Overpressure protection is provided by two safety valves. The pressurizer is supported by a cylindrical skirt welded to the bottom head.

The pressurizer is designed and fabricated in accordance with the ASME Code requirements listed in Table 5.2-1. The interior surface of the cylindrical shell and upper head is clad with weld deposited stainless steel. The lower head is clad with a Ni-Cr-Fe alloy to facilitate welding of the Ni-Cr-Fe alloy heater sleeves to the shell. A stainless steel safe end is provided on the pressurizer nozzles, after vessel final stress relief, to facilitate field welds to the stainless steel piping.

The total volume of the pressurizer is established by consideration of the factors given in Subsection 5.4.10.1. To account for these factors and to provide adequate margin at all power levels, the water volume in the pressurizer is programmed as a function of average coolant temperature as shown in Figure 5.4-7 in conjunction with Figure 5.4-8. High or low water level error signals result in the control actions shown in Figure 5.4-9.

Pressure is maintained by controlling the temperature of the saturated liquid volume in the pressurizer. At full load conditions, slightly more than one-half of the pressurizer volume is occupied by saturated water, and the remainder by saturated steam. In order to maintain the programmed pressure, the corresponding saturation temperature must be maintained. To maintain this temperature, heaters are energized to compensate for heat losses through the vessel and to raise the continuous subcooled pressurizer spray flow to the saturation temperature.

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 5.4-8. 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 Coolant 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 additional charging pumps in the Chemical and Volume Control System (CVCS) are automatically started to add coolant to the system and restore pressurizer level.

When load 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 power dependent function. A high level error signal, produced by an insurge, causes the letdown control valves to open, releasing coolant to the CVCS and 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.

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.

Approximately one-fifth of the heaters are connected to proportional controllers that adjust the heat input as required to compensate for steady-state losses and to maintain the desired steam pressure in the pressurizer. The remaining backup heaters are connected to on-off controllers. These heaters, normally deenergized, are turned on by either a low-pressurizer pressure signal or 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. An interlock prevents operation of the backup heaters in the event of concurrent high level error and high-pressurizer pressure signals. A low-low pressurizer level signal deenergizes all heaters to protect the heaters should they become uncovered.

→(LBDCR 15-028, R308A)

In order to determine the pressurizer heater capacity required to maintain natural circulation in the hot standby condition after a loss of offsite power, it was conservatively assumed that the ambient heat loss rate through the pressurizer was 400,000 BTU/hr. The measured heat loss from startup testing was only 356,000 BTU/hr. With an assumed 400,000 BTU/hr heat loss and a safety valve leakage of up to 0.5 gpm, single phase natural circulation can be maintained at hot standby conditions with a 50°F subcooled margin indefinitely by energizing 150kW of heater capacity thirty minutes after the loss of offsite power. Loss of subcooling, however, does not imply loss of natural circulation. The natural circulation cooldown analysis (refer to FSAR Section 9.3.6.3.3.1), performed to comply with Branch Technical Position 5-4, Design Requirements of the Residual Heat Removal System, does not credit the operation of any pressurizer heaters. Therefore, the operator action to energize the Pressurizer Heaters is not a time critical operator action)

←(LBDCR 15-028, R308A)

→(DRN 00-1673)

A redundant group of pressurizer proportional heaters and three redundant groups of backup heaters have been made available to be placed manually on the emergency diesel generator after a loss of offsite power. Each bank of heaters has access to only one Class 1E division power supply.

←(DRN 00-1673)

→(LBDCR 15-028, R308A)

Part of the closing circuitry to the breakers that provide power to the 480V non-safety switchgear buses 3A32 and 3B32 (that power the Pressurizer Heaters) share a specific common circuit breaker, CVCEBKR014AB-13. CVCEBKR014AB-13 powers the interlock 52z relay, SSDEREL2348-D (SSDEREL2398-D). The interlock 52z relay checks for completion of load stripping on the respective 480V non-safety switchgear buses 3A32 (3B32) at the onset of a Loss of Offsite Power. If the load stripping is complete, the interlock 52z relay closes a contact in the closing circuitry to the breakers that provide power to the 480V non-safety switchgear buses 3A32 and 3B32 to allow the breakers to close automatically when the sequencer load block contact in the closing circuitry is closed.

Alternatively, if the specific common circuit breaker, CVCEBKR014AB-13, is Open, then the breakers that provide that power to the 480V non-safety switchgear buses 3A32 and 3B32 will not close automatically at the onset of a Loss of Offsite Power. To close the breakers that power each Pressurizer Heater electrical switchgear 3A32 (3B32), local manual action in the respective train Switchgear room is necessary.

Reenergization of the necessary heaters from the emergency onsite power can be accomplished manually from the control room. At the onset of a Loss of Offsite Power concurrent with the specific common circuit breaker, CVCEBKR014AB-13, being Open, the reenergization of the 480V non-safety switchgear buses 3A32 and 3B32 (that power the Pressurizer Heaters) will require action to be performed outside of the Control Room. To close the breakers that power the 480V nonsafety switchgear buses 3A32 and 3B32, local manual operator action in the respective train Switchgear room is necessary. Once each 32 switchgear bus is reenergized, the necessary Pressurizer Heaters powered from that bus can be reenergized from the Control Room.

The natural circulation cooldown analysis (refer to FSAR Section 9.3.6.3.3.1), performed to comply with Branch Technical Position 5-4, Design Requirements of the Residual Heat Removal System, does not credit the operation of any pressurizer heaters. Therefore, the operator action to close the breakers that power each Pressurizer Heater electrical switchgear 3A32 (3B32), located outside of the control room, is not a time critical operator action. Procedures ensure that the addition of these loads after a loss of offsite power will not exceed the rating of the emergency diesel generator. The heaters are powered from the 480V non-safety switchgear buses 3A32 and 3B32. The safety-related Class 1E breakers providing power to these buses from the 4.16kV ESF buses (3A3-S and 3B3-S) will trip upon LOOP or SIAS. In this manner, the Class 1E interfaces for main power and control power to the pressurizer heaters are protected by safety-grade circuit breakers. This scheme also ensures that in case of an SIAS the non-Class 1E pressurizer heaters are shed from their emergency power sources via Class 1E circuit breakers.

←(LBDCR 15-028, R308A)

Pressurizer spray is supplied from each of the reactor coolant pump cold legs in loop one to the pressurizer spray nozzle. Automatic spray control valves control the amount of spray as a function of pressurizer pressure; both of the spray control valves function in response to the signal from the controller. These components are sized to use the differential pressure between the pump discharge and the pressurizer to pass the amount of spray required to maintain the pressurizer steam pressure during normal load following transients. A small continuous flow is maintained through the spray lines at all times to keep the spray lines and the surge line warm to reduce thermal shock during plant transients. This continuous flow also serves to keep the chemistry and boric acid concentration of the pressurizer water the same as that of the coolant in the heat transfer loops.

An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow depressurization and cooling if the reactor coolant pumps are shut down. The capability to depressurize using auxiliary spray with one charging isolation valve failed open was demonstrated by testing. With a charging to cold leg isolation valve open, some of the charging pump flow is diverted from the auxiliary sprayflow path to the cold legs. With two charging pumps running (88 gpm) an auxiliary spray flow rate of 37 gpm was achieved. This resulted in a depressurization rate of 24 psi/min. This is sufficient to depressurize the RCS during any design basis accident where auxiliary spray is required

In the event of an abnormal transient that causes a sustained increase in pressurizer pressure at a rate exceeding the control capacity of the spray, a high-pressurizer pressure reactor trip will be initiated.

5.4.10.3 Safety Evaluation

It is shown by analysis made in accordance with the requirements of the ASME Code, Section III that the pressurizer is adequate for all normal operating and transient conditions expected during the life of the plant. Following fabrication, the pressurizer was hydrostatically and nondestructively tested in accordance with the ASME Code, Section III.

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

Overpressure protection of the RCS is provided by two ASME Code spring loaded safety valves.

A discussion of the radiological considerations for the pressurizer is provided in Section 12.3.

5.4.10.4 Inspection Testing and Requirements

Prior to and during fabrication of the pressurizer, nondestructive testing is performed in accordance with the requirements of the ASME Code Section III. The pressurizer inspection program is summarized in Table 5.4-7.

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

5.4.11 QUENCH TANK (PRESSURIZER RELIEF TANK)

5.4.11.1 Design Basis

The quench tank is designed to receive and condense the normal discharges from the primary (pressurizer) safety valves and to prevent the discharge from being released to containment.

→(DRN 03-2059; 05-316, R14; EC-8458, R307)

The tank is sized to receive and condense a total steam release of 1232 lbm. The maximum normal discharge that the quench tank must withstand occurs during the loss of condenser vacuum event (described in Section 15.2.1), which is approximately 1200 lbm.

←(DRN 03-2059; 05-316, R14; EC-8458, R307)

The quench tank is mounted on structural steel framing supported from the secondary shield wall.

5.4.11.2 System Description

The quench tank, shown in Figure 5.4-10, is an austenitic stainless steel vessel suitable for prolonged contact with borated, demineralized water. Nozzles are provided for the safety valve discharge line, vents, drains, instrumentation, makeup water, nitrogen addition, and the rupture disc. The tank prevents the steam released from the primary (pressurizer) safety valves from being released to the containment atmosphere. The steam is discharged under water by the sparger and condensed. Demineralized water is manually added from the Reactor Auxiliary Building to cool the tank water after a steam discharge. A rupture disc venting to the containment atmosphere is provided for overpressure protection. Noncondensible gases within the quench tank are vented to the containment vent header through an air-operated valve.

The tank is designed and fabricated in accordance with the ASME Code, Section VIII. The design parameters am given in Table 5.4-8.

The sparger, spray header, nozzles and rupture disc fittings are stainless steel.

5.4.11.3 Safety Evaluation

The quench tank and associated blowdown system are sized such that the maximum safety valve backpressure, 500 psig, is not reached during any anticipated transient steam release. The quench tank rupture disc has a relief capacity greater than the combined relief capacities of the primary (pressurizer) safety valves.

The quench tank is classified as non-nuclear safety class as defined in Section 3.2. The failure of any of these components in no way compromises the integrity of the RCS pressure boundary or safe shutdown of the plant, nor does it in any way jeopardize the safety of the public.

5.4.11.4 Instrumentation Requirements

The quench tank is equipped with level, pressure, and temperature instrumentation.

A brief explanation of each instrument is made below.

The location of each of the sensors and functional requirements of the instruments are indicated on Figure 5.1-3.

5.4.11.4.1 Temperature

The temperature measurement channel consists of a precision resistance temperature detector (RTD), a temperature transmitter and associated indicators and alarms. The resistance of the RTD varies as a function of the temperature of the RTD environment. This change in RTD resistance is sensed by the temperature transmitter and converted to a change in a low dc current, which is used as a signal to the remote temperature indicators and alarms.

Quench tank temperature indication and high temperature alarm are provided in the main control room. A high temperature alarm may be indicative of primary safety valve leakage to the tank.

5.4.11.4.2 Level

Level instrumentation consists of a level transmitter and associated indicating and alarming equipment. The level transmitter measures the pressure difference between a reference column of water and the vessel water level. This pressure difference is converted to a small dc current which is proportional to the level of water in the vessel. This dc current output is used as a signal to the remote level indicators and alarms.

Quench tank level indication and high and low level alarms are provided in the main control room. A low level alarm is indicative of the sparger being uncovered or of insufficient tank fluid volume to quench the design basis accident steam release. A high level alarm alerts the operator of insufficient volume in the quench tank to accept the pressurizer steam release without becoming overpressurized and causing the rupture disc to burst.

5.4.11.4.3 Pressure

A pressure measurement instrument consists of an electric force balance pressure transmitter and associated indicators and alarms. The transmitter produces dc current output that is proportional to the pressure sensed by the instrument. This dc current output is used as a signal to the remote pressure indicators and alarms.

Quench tank pressure indication and high pressure alarm are provided in the main control room. The high pressure alarm alerts the operator to the situation prior to the bursting of the rupture disc.

5.4.11.5 Inspection and Testing Requirements

The quench, tank is designed to handle the design basis steam releases as described in Subsection 5.4.11.2. However, should the rupture disc burst, the primary coolant released to the containment would be minimal when compared to the LOCA for which the engineered safety features are designed to accommodate. Therefore, no inspection or testing requirements are imposed on the quench tank.

Since, the entire system is located within the containment structure, the quench tank must be vented to containment atmosphere, during the containment leakage test, to avoid collapse of the tank.

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 the applicable ASME Code, Section III, or the Draft ASME Code for Pumps and Valves, Class I requirements and must withstand the effects of the system design transients (see Subsection 3.9.1.1) plus other transients associated with the valves location or service requirements. The valves meet seismic Category I requirements. Backseats are specified on manual and motor-operated gate and globe valves to further minimize potential leakage. Functional requirements for each valve are detailed in the individual valve specifications.

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

5.4.12.2 Design Description

→(DRN 99-811)

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 also constructed of stainless steel to eliminate corrosion problems. Valve packing glands have provisions to adjust packing compression to reduce or eliminate leakage. These features keep uncontrolled leakage from these valves at essentially zero.

←(DRN 99-811)

5.4.12.3 Design Evaluation

All valves within the reactor coolant pressure boundary are stress analyzed in accordance with the applicable ASME Code, taking into consideration cyclic loadings. Refer to Section 3.9.

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 individual valve specifications and the applicable ASME Code. Valves greater than 1 inch inner diameter are dimensionally checked, including measurements to determine wall thickness.

In-service inspection will be performed during the plant life in accordance with ASME Code Section XI, as discussed in Subsection 5.2.4.

5.4.13 SAFETY AND RELIEF VALVES

5.4.13.1 Design Basis

There are no power-operated relief valves in the RCS. The primary safety valves on the pressurizer are designed to protect the system, as required by the ASME Code, Section III.

The design basis for establishing the relieving capacity of the primary safety valves is presented in Appendix 5.2A. For the postulated transients presented in Chapter 15, the results demonstrate that the pressurizer will not go "solid" and that the 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 Main Steam System.

5.4.13.2 Description

The RCS has two safety valves to provide overpressure protection. The safety valve is illustrated in Figure 5.4-11. The design parameters are given in Table 5.4-9. They are direct acting, spring-loaded safety valves meeting ASME Code requirements. They have an enclosed bonnet and have a balanced bellows for superimposed backpressure. The safety valves pass sufficient pressurizer steam to limit the RCS pressure to 2750 psia (110 percent of design) 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 primary 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 tine up to the inlet of the turbine stop valve is provided by the secondary safety valves.

5.4.13.3 Evaluation

Overpressure protection is discussed in Subsection 5.2.2. The ASME Code Report on Overpressure Protection, is included as Appendix 5.2A.
→(DRN 00-1059; 02-88)
←(DRN 00-1059; 02-88)

→(DRN 00-1059, R11-A; 02-88, R11-A)

5.4.13.4 Tests and Inspections

←(DRN 00-1059, R11-A; 02-88, R11-A)

The valves are inspected during fabrication in accordance with ASME Code, Section III requirements. The inlet is hydrostatically tested and the outlet is pneumatically tested. Seat leakage is checked by a steam test at 93 percent of set pressure. Set pressures are adjusted using steam.

In-service inspection will be performed during plant life in accordance with the ASME Code Section XI as discussed in Subsection 5.2.4.

5.4.14 COMPONENTS SUPPORT

→(DRN 03-2059, R14)

5.4.14.1 Design Bases

The criteria applied in the design of the RCS supports are that the specific function of the supported equipment be achieved during all normal earthquake, and pipe break conditions. Specifically, the supports are designed in accordance with the design loading combinations which are applied in the design of ASME Code Class I supports. These design loading combinations are categorized with respect to their plant operating conditions, which are identified as normal, upset, emergency and faulted (as defined in the ASME Code, Section III, for Class I components). The following design loading combinations are applied:

←(DRN 03-2059, R14)

a) Loading Combination 1 (Upset)

The concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of the operational basis earthquake (OBE).

b) Loading Combination 2 (Emergency)

The concurrent loadings associated with the plant emergency condition.

c) Loading Combination 3 (Faulted)

→(DRN 03-2059, R14)

The combined loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with postulated pipe ruptures as discussed in Subsections 3.6.2, 3.6.3, and 6.2.1.2. This loading combination is assumed to occur at 100 percent power steady state operation.

←(DRN 03-2059, R14)

For supports that are an integral part of RCS components, the design stress limits of the parent component are used as follows:

a) The design stress limits applied in conjunction with loading combinations 1 and 2 are in accordance with the rules of ASME Code Section III, Paragraph NB-3223 for upset conditions and Paragraph NB-3224 for emergency conditions, respectively.

Nonintegral supports are designed and constructed to the stress limits of ASME Section III, Subsection NF. The design stress limits used are as follows:

a) The design stress limits applied in conjunction with loading combinations 1 and 2 for plate type supports and linear type supports are in accordance with the rules of Subsubarticles NF-3220 and NF-3230, respectively.

For these supports the design stress limits applied in conjunction with loading combination 3 are dependent upon the type of system or subsystem analyses used to establish design loadings. Elastic system or subsystem analysis used to determine the design loadings produced by loading combination 3 and the associated design stress limits applied in the design of supports are in accordance with the rules of the ASME Code, Section III, Appendix F, Subsubarticle F-1323, or plastic instability load criteria are applied in accordance with paragraphs F-1370 (d) and F-1370 (e) of Appendix F, including the effects of the resulting plastic deformation.

5.4.14.2 <u>Description</u>

Figure 5.4-12 illustrates the RCS support points. A description of each supported component follows.

5.4.14.2.1 Reactor Vessel Supports

The reactor vessel integral supports consist of four pads welded to the underside of the vessel inlet nozzles. Vertically, the pads rest on lubricated bearing plates, and contain studs that act as holddown devices for the vessel. Horizontal keyways interface with the pads. The arrangement of the vessel supports allows radial growth of the reactor vessel due to thermal expansion while maintaining it centered. The supports are designed to accept normal loads and seismic and pipe rupture accident loads.

Reactor vessel supports are shown in Figure 5.4-13 and Figure 3.8-34.

5.4.14.2.2 Steam Generator Supports

→(DRN 00-1059, R11-A; 03-2059, R14)
The steam generator weight 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 RCS. Two embedded keys mate with keyways within the sliding base to guide the movement of the steam generator during expansion and contraction of the RCS and, together with anchor bolts, prevent excessive movement of the bottom of the steam generator during seismic events and following a pipe break.
←(DRN 00-1059, R11-A; 03-2059, R14)

A system of keys and snubbers located on the steam drum guide the top of the steam generator during expansion and contraction of the RCS and provide support during seismic events and following a primary side or secondary side pipe break.

→(EC-8438, R307)
The mechanical/structural loads associated with the dynamic effects of a LOCA in the RCS hot leg and cold leg piping have been eliminated with the application of Leak-Before-Break (LBB) (Reference Section 3.6.3). Based on removal of the RCS dynamic pipe break loads, the shim plate pack (stop) on the reactor side of the keyway (underneath the RCS hot leg) was permanently removed from the SG-1 and SG-2 sliding base. Additionally, the SG-1 and SG-2 sliding base supports were modified to remove the shim plate from the perimeter of the SG support skirt flange. These modifications were performed as part of the changeout of the steam generators.
←(EC-8438, R307)

Steam generator supports are shown in Figure 5.4-14.

5.4.14.2.3 Reactor Coolant Pump Supports

→(DRN 03-2059, R14)

Each reactor coolant pump is provided with four vertical support columns, four horizontal support columns, and one horizontal snubber. The rigid structural columns provide support for the pumps during normal operation, earthquake conditions, and any postulated pipe break in either the pump suction or discharge line[2]. An illustration of the pump supports is shown in Figure 5.4-15.

←(DRN 03-2059, R14)

For the case of a pipe break in the pump discharge line, three structural stops are, provided to limit the pump motion. Pipe stop structures that limit pipe motion are also provided to prevent overloading of the pump support columns due to a pipe rupture at either the steam generator or reactor vessel nozzles.

5.4.14.2.4 Pressurizer Supports

→(DRN 03-2059, R14)

The pressurizer is supported by a cylindrical skirt welded to the bottom of the pressurizer and bolted to the support structure. The skirt is designed to withstand dead weight and normal operating loads as well as the loads due to earthquakes and postulated pipe break. An illustration of the pressurizer supports is shown in Figure 5.4-16.

5.4.14.3 Evaluation

The structural integrity of the reactor coolant component supports is confirmed by analyses, using the design basis presented in Subsection 5.4.14.1. The analyses employ dead weight, thermal, seismic and pipe break loadings, combined in the various load combinations. Dead weight and thermal loads are determined by static analysis of the RCS. The method of determining seismic loadings is described in Subsection 3.7.2. RCS response to postulated branch line pipe breaks (BLPBs) for extended power uprate to 3716 MWt is determined using non-linear time history analysis of a full three-dimensional ANSYS model of the RCS. The effects of BLPBs on the RCS component supports include the effects of pipe tension release, jet impingement, RV blowdown, SG subcompartment and component-to-support gaps. Subsequent evaluations demonstrate the adequacy of the RCS supports under extended power uprate conditions to 3716 MWt, per the acceptance criteria provided in Section 3.9.3.

Structures are provided to mate with the component supports to restrain and support RCS components. The loads at the support/structure interface locations are examined under normal, OBE, SSE, and pipe break conditions (see Section 3.8). Seismic and accident loads are determined by the same methods referred to in the first paragraph of this section, taking into account the structural characteristics at the support/structure interfaces. The design of each support is compatible with the design radiation levels given in Section 3.11.

←(DRN 03-2059, R14)

5.4.14.4 Testing and Inspection

Tests were conducted on materials similar to that being used for the reactor vessel and steam generator sliding supports to demonstrate that the maximum static coefficient of friction does not exceed 0.15 at a design loading of 5000 psi. Tests on sliding supports and the steam generator basis are in accordance with ASME Section III. In addition, all sliding supports are 100 percent liquid penetrant inspected at the vendor's shop. The steam generator base is ultrasonically and magnetic particle inspected.

The specifications for the steam generator snubbers require that they be tested in the vendor's shop at the rated load capacity in both tension and compression. The piston creep during these tests must comply with specification limits. Tests are also specified for initiation of snubber action in both tension and compression.

→(DRN 03-2059, R14)

[2]These pipe breaks, which have been eliminated via LBB (Section 3.6.3) and which have been replaced by postulated BLPBs for extended power uprate to 3716 MWt, continue to provide the enveloping design loads on the RCP supports.

←(DRN 03-2059, R14)

Supports integral to the RCS components receive quality assurance inspections in accordance with the ASME Code, Section III, during fabrication.

During preoperational testing of the RCS, the support displacements will be monitored for agreement with calculated displacements and/or clearances. Subsequent examinations of supports of RCS components will be in acccordance with the ASME Code, Section XI.

5.4.15 REACTOR COOLANT GAS VENTING SYSTEM

5.4.15.1 Design Bases

The Reactor Coolant Gas Venting System (RCGVS) is designed to allow for remote venting of non-condensible gases, which may collect in the RCS, via the reactor vessel head vent or pressurizer steam space vent during post-accident situations. The system may be used for normal RCS venting when required during plant outages. This system has been designed to meet the requirements of NUREG-0737.

The design criteria for the RCGVS are as follows:

a) The system permits remote (control room) venting of the reactor vessel head or the pressurizer.

b) The system is designed for a single active failure with active components powered from their respective redundant emergency power sources. The system has parallel vent paths with valves powered from alternate power sources. A single active failure in the power and control portion of the vent system will not prevent isolation of the entire vent system when required.

c) The system is designed primarily to vent noncondensible gas.

d) The vent flow capability is based upon the following considerations:

   1) The vent rate is sufficient to vent 4800 standard cubic feet of hydrogen per minute.

   2) Coolant liquid loss through the vent will not exceed the makeup capacity of one charging pump in the event of a Safety Class 2 pipe break or inadvertent valve operation, thus limiting leakage to less than the LOCA definition of 10CFR50.

   3) The vent mass rate will not result in heat loss from the RCS in excess of the normal pressurizer heater capacity.

e) Vent paths are provided to the quench tank which allows for cooling of gases and condensing water vapor by releasing the vented gases-below the water level in the tank. The vent paths are safety grade and, being a part of the RCS, meet the same qualifications as the existing RCS.

f) A method of leakage detection is provided to identify and ensure that leakage in the system is identifiable. This allows continued power operations at leak rates greater than 1 gpm but less than 10 gpm (refer to plant Technical Specifications).

g) The solenoid operated valves are powered from safety grade 120V ac power supplies. Power is removed from the fail closed valves by utilizing key-locked control switches to minimize the possibility of inadvertent operation during normal operation.

h) Valve position indication (open-closed) is provided in the control room for all remotely operated valves.

i) The system is designed so that each vent valve may be tested for operability during plant operation. Testing can be performed in accordance with Subsection IWV of Section XI of the ASME code for Category B valves.

See Figure 5.4-17 for flow diagram arrangement.

5.4.15.2 Description/Principal Modes of Operation

5.4.15.2.1 Startup

Venting of the RCS prior to plant startup can be accomplished using the RCGVS. The reactor vessel can be vented by opening either 2RC-E2559A or 2RC-E2560B and the pressurizer can be vented by opening either 2RC-E2557A or 2RC-E2558B. Venting is routed to the quench tank. Air flow can then be routed to the Waste Management System via the containment vent header.

5.4.15.2.2 Normal Operation

This system is not intended for use during normal power operation and administrative controls are provided to minimize the possibility of inadvertent operation. In addition, power is removed from all valves during normal plant operation.

During normal operation, leakage detection is maintained by use of pressure instrumentation. A rise in pressure will indicate leakage past valves 2RC-E2557A, 2RC-E2558B, 2RC-E2559A or 2RC-E2560B. The ability to identify leakage above 1 gpm from the RCS may be accomplished in using either of two methods:

a) If the pressure increase is slow enough, the leakage rate can be determined by observing the rate of pressure increase per unit time.

b) The leakage would be diverted to the quench tank and the increase in quench tank level can be translated into a leakage rate.

5.4.15.2.3 Refueling Shutdown

→(EC-5000080002, R301)

a) During refueling shutdown, valves 2RC-E2557A and/or 2RC-E2558B,2RC-E2559A and/or 2RC-E2560B are opened to align the reactor vessel to the pressurizer which is vented to atmosphere. Opening 1RC-V2590 A/B (RC-10111) and 1RC-V2506 (RC-101) provides an alternate vessel vent path. Either path prevents vacuum formation during reduction of reactor vessel level.

←(EC-5000080002, R301)

b) During shutdown for refueling, valves 2RC-E2557A, 2RC-E2558B,2RC-E2559A, 2RC-E2560B, 2RC-2561A and 2RC-E2562B are to be tested per In-service Testing requirements.

5.4.15.2.4 Accident Conditions

Prior to system operation, the quantity of non-condensible gases in the RCS can be estimated. This can be accomplished through evaluation of the inadequate core cooling instrumentation that will be provided in accordance with the requirements of NUREG-0737 II.F.2 (Section 1.9.30) for the case of gas in the reactor vessel and by the response of system pressure control methods or departure from saturation conditions in the pressurizer for the case of gas in the pressurizer. Additional indirect methods involving observing pressurizer level response to a given pressure change can be used as a backup method of gas volume determination.

The vent system is aligned to vent from the reactor vessel or pressurizer to the quench tank. Small quantities of gas may be vented to the quench tank without rupturing the quench tank rupture disc. This permits gas removal from the RCS without contaminating the containment atmosphere. Gas may be discharged from the quench tank to the Waste Management System.

5.4.15.3 Evaluation

The RCGVS may be required to operate during post-accident situations to remove noncondensible gases from the RCS. To assure operability under those conditions, the components of the system required to perform venting operations have been environmentally qualified to operate under post-accident containment conditions. They are provided with emergency power and parallel valves powered from redundant power sources. Parallel valves assure a vent flow path to containment in the event of single active failure.

The RCGVS is supplied with flow limiting orifices which limit mass loss from the RCS to an amount less than the makeup capacity of a charging pump.

The components, piping and supports in the RCGVS are specified as seismic Category I. All valves have been qualified for operability during and following a seismic event.

5.4.15.4 Required Components

a) Piping and Valves

All piping and valves are constructed of austenitic stainless steel and are safety grade as required. The piping size is 1 inch Schedule 160. The six Safety Class 2 solenoid operated valves are designed to fail closed to minimize inadvertent operation. The Safety Class I orifices, one of which is included in the present RCS design, are sized to meet the specified flow requirements.

b) Instrumentation and Controls

The system is designed to be controlled remotely from the main control room. All safety-related instrumentation is powered from emergency power sources. Position indication (open/closed) is provided for all remotely operated valves and displayed in the control room. Pressure instrumentation is also provided to monitor system performance and any valve leakage. Pressure indication is located in the control room. Quench tank level is also located in the control room.

5.4.16 REFUELING WATER LEVEL INDICATING SYSTEM (RWLIS)

5.4.16.1 Design Bases

→(DRN 00-1059)

The REFUELING WATER LEVEL INDICATING SYSTEM (RWLIS) is designed to perform the following functions.

←(DRN 00-1059)

a) Monitor the water level in the Reactor Coolant System and the refueling pool during refueling operations.

b) Monitor the water level in the RCS hot leg during maintenance evolutions requiring the RCS water levels at elevations within the range of the hot leg.

c) Provide indication of the level locally in the containment building and remotely in the control room.

d) Provide an alarm in the control room when the water level in the refueling pool exceeds a predetermined level or when the water level drops below a predetermined level in the range of the hot leg.

Refer to Figure 5.4-18 for a diagram of the system showing its connection to the RCS and the major components.

The design pressure and temperature of the RWLIS are 100 psig and 250°F. With the design conditions, the system's normal operational conditions are modes 5 and 6 with the RCS vented and pressurizer level greater than 5% and less than 75% as indicated by the pressurizer cold calibrated level instrumentation. The system is isolated from the RCS during plant operations. The RWLIS can be operated with the vessel head in place.

5.4.16.2 System Description

The RWLIS is comprised of a narrow and a wide range differential pressure transmitter attached through stainless steel piping to a primary system high point near the top of the pressurizer and to the hot leg drain on the RCS (see Figure 5.4-18). The signals from these transmitters are provided to a local indicator inside containment, and to a remote indicator in the control room.

The RWLIS is designed to operate with the reactor vessel head installed or removed. With the exception of the local RWLIS indicator in containment, the system is designed for permanent installation. The RWLIS will compensate for slight positive and negative pressure variations in the RCS by utilizing a differential level instrument arrangement to compare actual RCS water level to a reference leg vented to an optimum system high point.

The low level sensing line is permanently attached to the drain line just below hot leg #1 in the vicinity of the shutdown cooling suction line.

The reference leg is permanently attached to the existing upper level tap of the pressurizer wide range level instrument.

The transmitters and the local indicator are mounted on an instrument stand outside the biological shield. The location was chosen to minimize tubing runs and to allow easy draining of the reference leg to existing RCS level drain pipe to maintain the reference leg dry during normal RWLIS operation.

The level signals are conducted to the control room panel and annunciator window. The RWLIS signals are also provided to the Plant Monitoring Computer.

5.4.16.3 Evaluation

The valves which maintain the systems' isolation during plant operation are designed Class I and Class II for the isolation at the hot leg and the pressurizer, respectively, and are part of the Reactor Coolant Pressure Boundary. These valves are discussed in Subsection 5.4.12. The remainder of the RWLIS is installed as Class III. The tube routing and instrument stands are installed as Seismic Class 1. The RWLIS is not installed as safety-related instrumentation.

5.4.16.4 Tests and Inspections

Inservice inspection of the boundary valves is performed during the plant life in accordance with ASME Section XI as discussed in Subsection 5.2.4.

Operating procedures for the RWLIS require that the system be calibrated prior to draindown to ensure accurate level indication.

5.4.17 REACTOR COOLANT SHUTDOWN LEVEL MEASUREMENT SYSTEM (RCSLMS)

5.4.17.1 Design Bases

The Reactor Coolant Shutdown Level Measurement System (RCSLMS) is designed to perform the following functions:

a) Monitor the water level in the Reactor Coolant System (RCS) during non-power operation.

b) Monitor the water level in the RCS hot leg during maintenance evolutions requiring the RCS water levels at elevations within the range of the hot leg.

c) Provide indication of the level locally in the containment building and remotely in the control room.

d) Provide an alarm in the control room when water level drops below a predetermined level in the range of the hot leg.

Refer to Figure 5.4-19 for a diagram of the system showing its connection to the RCS and the major components.

The design pressure and temperature of the RCSLMS are 100 psig and 300°F. With the design conditions, the system's normal operational conditions are modes 5 and 6 with the RCS vented and pressurizer level greater than 5% and less than 75% as indicated by the pressurizer cold calibrated level instrumentation. The system is isolated from the RCS during plant operations by removing the temporary connections.

5.4.17.2 System Description

→(DRN 01-3975)

The RCSLMS is comprised of the Mansell Level Monitoring System (level instrument) and a skid mounted sight glass. This RCSLMS is connected by stainless steel tubing to a primary point at the top of the pressurizer and to the hot legs on the RCS (see Figure 5.4-19). The sight glass provides local indication. The Level instrument is installed on a temporary basis during refueling operations. The level instrument provides indication to the existing control room annunciators and indicators. Indication also exists on the level instrument computer system that is temporarily mounted in the Control Room during refueling operations to provide for system status of the instrument and level of the RCS.

→(DRN 00-1059)

The level instrument is comprised of two channels, each containing a reference (low-pressure) transducer assembly and a high-pressure transducer assembly that provides absolute pressure measurements to the computer system. The computer system computes the level of the RCS and displays the level in the control room. The computer system has a channel selectable output of 0-10VDC to the RCSLMS process analog control cards for conditioning and transmission of the signal to the control room indicators.

←(DRN 00-1059; 01-3975)

The RCSLMS is designed to function during non-power operations, providing a redundant measurement of RCS level. The system is designed to have temporary connections which are made only during non-power operation. The system is isolated and disconnected during normal operation.

5.4.17.3 Evaluation

During power operation, the RCSLMS is disconnected, providing isolation from the RCS pressure boundary at the hot leg and pressurizer. The temporary connections are made after the reactor pressure has dropped below 300 psig in a section of piping which is Cat. 7 NNS and has valving to provide isolation for making the connections. These valves are part of Reactor Coolant Pressure Boundary and are discussed in Subsection 5.4.12. The tubing, tube supports and instrument skid are installed as Seismic Class 1. The RCSLMS is installed as non-safety-related instrumentation.

5.4.17.4 Tests and Inspections

Visual inspection of welds and hydrostatic testing of the added piping and valves shall be performed in accordance with ANSI B31.1-1977.

Operating procedures for the RCSLMS require that the system be calibrated prior to downdrain to ensure accurate level indication.

TABLE 5.4-1 (Sheet 1 of 2) Revision 14 (12/05)

REACTOR COOLANT PUMP PARAMETERS

Parameter	Value
Number of Pumps	4
Type	Vertical, controlled leakage, centrifugal
Shaft seals, type, quantity	Mechanical, 4
Byron Jackson SU seal:	
Materials, stationary face	Carbon A GR CCP-72 or US Graphite Graphitar G114
Rotating face body	ASTM-A-351 Gr CF8
Rotating face ring	Titanium carbide, Kennametal K-162 B
Byron Jackson N-9000 seal	
Materials, stationary face	Morganite CNFJ or US Graphite Graphitar G114
Rotating Face	Tungsten Carbide, Kennametal KZ-801
Atomic Energy of Canada Limited (AECL) CAN4 seal:	
Materials, stationary face	Zr, -2.5 Wt. % Nb Carbon Graphite
Rotating face body	XM-19, annealed (Nitronic 50)
Rotating face ring	Tungsten Carbide
Design pressure, psig	2485
Design temperature, °F	650
Normal operating pressure, psig	2235
→(DRN 03-2059, R14)	
Normal operating temperature, °F	543
Rated flow, gpm (@ 553°F)	99,000
←(DRN 03-2059, R14)	

TABLE 5.4-1 (Sheet 2 of 2)

REACTOR COOLANT PUMP PARAMETERS

Parameter	Value
Motor	
Voltage, volts	6600
Frequency, Hz/ϕ	60/3
Horsepower/speed, hot hp/rpm	7800/1183
Horsepower/speed, cold, hp/rpm	9700/1183
Service factor	1.0

TABLE 5.4-2 (Sheet 1 of 2) Revision 307 (07/13)

→(EC-8458, R307)

STEAM GENERATOR PARAMETERS[a][b]

←(EC-8458, R307)

Parameter	Value
Number of units	2
→(DRN 03-2059, R14; EC-8458, R307)	
Heat transfer rate, each, Btu/hr.	6.3793×10^9
←(EC-8458, R307)	
Primary side	
Design pressure/temperature (psig/°F)	2485/650
→(DRN 06-1060, R15; EC-8458, R307)	
Coolant inlet temperature, °F	602
←(DRN 06-1060, R15; EC-8458, R307)	
Coolant outlet temperature, °F	543
→(DRN 06-1060, R15; EC-8458, R307)	
Coolant flow rate, each, lb/hr	81.8×10^6
←(DRN 03-2059, R14; 06-1060, R15)	
Coolant volume at 68 °F each, ft^3	2051
←(EC-8458, R307)	
Tube size, OD, in.	3/4
→(EC-8458, R307)	
Tube thickness, nominal, in.	0.044 for Rows 1 and 2 0.043 for rows 3 to 138
←(EC-8458, R307)	
Secondary side	
Design pressure/temperature (psia/°F)	1100/560
→(DRN 03-2059, R14; 06-1060, R15; EC-8458, R307)	
Steam pressure, psia	837.2
←(DRN 06-1060, R15)	
Steam flowrate (at 0.10% moisture) each, lb/hr	8.3×10^6
←(EC-8458, R307)	
→(DRN 06-1060, R15; EC-8458, R307)	
Feedwater temperature at full power, °F	449.8
←(DRN 06-1060, R15)	
Moisture carryover, weight maximum, %	0.10
←(DRN 03-2059, R14; EC-8458, R307)	
Primary inlet nozzle, No./ID, in.	1/42

→(DRN 03-2059, R14; EC-8458, R307)

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a. Based on full power post-EPU (Extended Power Uprate) conditions with Replacement Steam Generators.

b. The parameters presented in this table are based on a feedwater temperature of 449.8°F. The best estimate feedwater temperature is 447.7°F. An evaluation was performed to assess the impact of the difference in feedwater temperatures which concluded the difference between the two values is insignificant.

←(DRN 03-2059, R14)

→(DRN 06-1060, R15)

←(DRN 06-1060, R15; EC-8458, R307)

TABLE 5.4-2 (Sheet 2 of 2) Revision 307 (07/13)

→(EC-8458, R307)

STEAM GENERATOR PARAMETERS[(a)(b)]

←(EC-8458, R307)

Parameter	Value
Primary outlet nozzle, No./ID, in.	2/30
→(EC-8458, R307)	
Steam nozzle, No./ID, in.	1/34
←(EC-8458, R307)	
Feedwater nozzles, No./size/schedule	1/18/80
→(DRN 03-2059, R14; EC-8458, R307)	
Overall heat transfer coefficient (nominal)	1436
←(EC-8458, R307)	
Btu/hr-ft^2 -°F	
Blowdown flow (gpm nominal, per SG)	165

←(DRN 03-2059, R14)

TABLE 5.4-3 Revision 307 (07/13)

STEAM GENERATOR FABRICATION TESTING

Component	Test [(a)]
Tube sheet	
Forging	UT, MT
Cladding	UT, PT
Primary head	
→(EC-8458, R307)	
Forging	UT, MT
←(EC-8458, R307)	
Cladding	UT, PT
Secondary shell and head	
→(EC-8458, R307)	
Forging	UT, MT
←(EC-8458, R307)	
Tubes	UT, ET
Nozzles (forging)	UT, MT
Studs	UT, MT
Welds	
→(EC-8458, R307)	
Shell, circumferential	RT, MT, UT ISI welds only
←(EC-8458, R307)	
Cladding	UT, PT
←(EC-8458, R307)	
Feedwater Nozzles to shell	RT, MT, UT
←(EC-8458, R307)	
Tube-to-tube sheet	PT
Instrument connections	MT
→(EC-8458, R307)	
All accessible welds - after hydrostatic test	MT
Primary side pressure nozzles	PT
Pedestal to Head	RT, MT
←(EC-8458, R307)	
Support Lugs	MT, PT

a. UT = Ultrasonic testing
MT = Magnetic-particle testing
RT = Radiographic testing
→(EC-8458, R307)
ET = Eddy-current testing
PT = Dye-penetrant testing
←(EC-8458, R307)

TABLE 5.4-4 Revision 307 (07/13)

REACTOR COOLANT PIPING PARAMETERS

Parameter	Value
Number of loops (steam generators)	2
→(DRN 03-2059, R14; EC-8458, R307)	
Design Flow per loop, lb/hr	81.8×10^6
←(DRN 03-2059, R14; EC-8458, R307)	
Pipe size	
Reactor outlet ID/wall, in.	
Pipe	42/3-3/4 w/o clad
Elbow	42/4-1/8 w/o clad
Reactor inlet, ID/wall, in.	
Pipe	30/3 w/o clad
Elbow	30/3 w/o clad
Pump suction	
Elbow	30/3 w/o clad
Pipe	30/2-1/2 w/o clad
Surge line, (nominal pipe size in./ schedule)	12/160
Sprayline (nominal pipe size in./ schedule)	4/120 3/160
Design pressure, psia	2500
Design temperature, °F (hot & cold legs)	650
(surge line)	700

TABLE 5.4-5

REACTOR COOLANT PIPING TESTS

Component	Test [a]
Fittings (castings)	RT, PT
Piping (castings)	RT, PT, or MT
Pipe and elbows	UT, MT
Carbon steel plate	UT, PT
Welds	
Circumferential	RT, PT, or MT
Nozzles to pipe run	RT, MT, UT
Instrument connections to pipe	PT or MT
Cladding	UT, PT
Safe ends to nozzles	RT, PT

a. Key
UT = Ultrasonic testing
MT = Magnetic-particle testing
PT = Dye-penetrant testing
RT = Radiographic testing

TABLE 5.4-6 (Sheet 1 of 2) Revision 309 (06/16)

PRESSURIZER PARAMETERS

Property	Parameter
Design pressure, psia	2500
Design temperature, °F	700
Normal operating pressure, psia	2250
Normal operating temperature, °F	653
→(DRN 06-904, R15)	
Internal free volume, ft^3	1519
←(DRN 06-904, R15)	
Normal (full power) operating water volume, ft^3	800
Normal steam volume full power, ft^3	700
→(DRN 00-1673, R10; 01-1361, R12; 05-892, R14, LBDCR 16-017, R309)	
Installed heater capacity, kW (Nominal)	1350
←(DRN 00-1673, R10; 01-1361, R12; 05-892, R14, LBDCR 16-017, R309)	
→(DRN 03-1268, R13)	
Spray flow, maximum, gpm	490
←(DRN 03-1268, R13)	
Spray flow, continuous, gpm	1.5
Nozzles	
Surge line (1 ea) nominal, in.	12, schedule 160
Safety valves (3), ID, in.	6, schedule 160
Spray (1) nominal, in.	4, schedule 120
→(DRN 00-1631, R10; 05-892, R14, LBDCR 15-004, R309, LBDCR 16-017, R 309)	
Heaters (30), OD, in. (F4 Plugged / F3 & D4 CAPPED)	1.25
←(DRN 00-1631, R10; 05-892, R14, LBDCR 15-004, R309, LBDCR 16-017, R309)	
Instruments,	
Level (4) nominal, in.	3/4, schedule 160
Temperature (1) nominal, in.	1, schedule 160
Pressure (2) nominal, in.	3/4, schedule 160
Dimensions	
Overall length, including skirt and spray nozzle, in.	441
Outside diameter, in.	106-1/2
Inside diameter, in.	96
Cladding thickness, in. (minimum)	1/8

TABLE 5.4-6 (Sheet 2 of 2) Revision 15 (03/07)

PRESSURIZER PARAMETERS

Property	Parameter
→(DRN 06-904, R15)	
Dry weight, including heaters, lb	203,288
Flooded weight, including heaters, lb	297,712
←(DRN 06-904, R15)	

TABLE 5.4-7

PRESSURIZER TESTS

Component	Test[a]
Heads	
Plates	UT, MT
Cladding	UT, PT
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, UT
Shell circumferential	RT, MT, UT
Cladding	UT, PT
Nozzles	RT, MT
Nozzle safe ends	RT, PT
Instrument connections	PT
Support skirt	MT
Temporary attachment after removal	MT
All welds after hydrostatic test	MT or PT
Heater assembly, end plug weld	RT, PT

a. Key:

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

TABLE 5.4-8

QUENCH TANK PARAMETERS

Property	Parameter
Design pressure, psig (internal/external)	130/15
Design temperature, °F ,	350
Normal operating pressure, psig	3
Normal operating temperature, °F	120
Minimum internal volume, gal	2400
Blanket gas	Nitrogen
Nozzles	
Pressurizer discharge (1) nominal, in.	12 Sch. 40 S
Demineralized water (1), in./rating	2/3000 lb SW Coupling
Rupture disc (1) in.	20 flanged
Drain (1), in./rating	2/3000 lb SW Coupling
Temperature instrument (1), in./rating	1/3000 lb SW Coupling
Level instrument (2), in./rating	1/3000 lb SW Coupling
Vent (1), in./rating	2/3000 lb SW Coupling
Vessel material	ASTM-SA-240 Type 304
Dimensions	
Overall length, in.	151.75
Outside diameter, in.	72
Dry weight, lb	6700
Flooded weight, lb	26,850
Code and date	ASME Section VIII Div. 1 through Summer 1970 Addenda

TABLE 5.4-9 Revision 15 (03/07)

PRIMARY SAFETY VALVE PARAMETERS

Property	Parameter
Design pressure, psia	2500
→(DRN 06-872, R15) Design temperature, °F ←(DRN 06-872, R15)	700
Fluid	Saturated steam, 2000 ppm $H_3 BO_3$, pH=5.0
Set pressure, psia	2500 +/-3
Capacity, lb/hr. at set pressure, each	460,000
Type	Spring loaded safety-balanced bellows. Enclosed bonnet.
Accumulation, percent	3
Backpressure Max buildup/max. superimposed, psig	500/130
Blowdown, percent	5
Materials	
Body	ASME SA 182, GR. F316
Disc	ASTM A637, GR. 688
Nozzle	ASME SA 182, GR347