4.1 INTRODUCTION

The Primary Coolant System is comprised of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains one steam generator, two circulating pumps, connecting pipe and instrumentation. A pressurizer is connected to one of the reactor vessel outlet pipes by means of a surge line. Pressurizer relief and safety valves are provided which discharge to a quench tank to cool and condense pressurizer discharges. All components of the Primary Coolant System are located within the containment building.

The Primary Coolant System is designed to remove heat from the reactor core and internals and transfer it to the secondary (steam generating) system by the controlled circulation of pressurized borated water which serves both as a coolant and a neutron moderator. The dissolved boron serves as a neutron absorber and is used for long-term reactivity control. The Primary Coolant System serves as a barrier to the release of radioactive material to the containment building, and is equipped with controls and safety features that assure safe conditions within the system. Design pressure is 2,500 psia, design temperature 650°F (pressurizer 700°F) and design life 40 years.

4.2 DESIGN BASIS

4.2.1 PERFORMANCE OBJECTIVES AND PARAMETERS FOR NORMAL CONDITIONS

The Primary Coolant System is designed to operate at a power level of 2,650 MWt. The present licensing limit is, however, 2,565.4 MWt core power plus 15 MWt for the primary coolant pump heat input for a total Primary Coolant System output of 2,580.4 MWt. The principal parameters for the Primary Coolant System are listed in Table 4-1. The design parameters for each of the major components are given under the individual component discussion later in this section. The Primary Coolant System is a CP Co Design Class 1 system per Section 5.2. The applicable stress and seismic criteria are given in Section 5.10. The primary system components and controls are also designed for cyclic transient conditions as listed in Subsection 4.2.2.

4.2.2 DESIGN CYCLIC LOADS

The following design cyclic transients which include conservative estimates of the operational requirements for the components listed in Table 4-2 were used in the fatigue analysis required by the applicable code:

1. 500 heatup and cooldown cycles during the system 40-year design life at a heating and cooling rate of 100°F/h. The pressurizer is designed for a cooldown rate of 200°F/h.

2. 15,000 power change cycles over the range of 10% to 100% of full load with a ramp load change of 5% of full load per minute increasing or decreasing. (The number of cycles for the normal power change has been reduced to 2,000 due to CRDM nozzle repairs during 2004 refueling outage.)

3. 15,000 cycles of 10% of full load step power changes increasing from 10% to 90% of full power and decreasing from 100% to 20% of full power. (The number of fast power changes and normal step power changes has been reduced to 2,000 due to CRDM nozzle repairs during 2004 refueling outage.)

4. 10 cycles of hydrostatic testing the primary system at 3,110 psig and at a temperature at least 60°F above the Nil Ductility Transition Temperature (NDTT) of the component having the highest NDTT. The relationship between the allowable temperature and maximum primary to secondary pressure differential is shown in Figure 6-b of Reference 43.

5. 320 cycles of leak testing at 2,485 psig and at a temperature at least 60°F greater than the NDTT of the component having the highest NDTT. Steam generator level instrumentation will indicate at least low water level for the duration of the test and the primary to secondary differential pressure will not exceed 2,100 psia.

6. 350,000 cycles of normal operating pressure variations of ± 50 psi at operating temperature.

7. 500 reactor trips from 100% power. The 500 reactor trips from 100% power are not explicitly addressed in the reactor vessel analytical report or in the associated specification. It is anticipated that the reactor trips were not addressed due to the fact that they are less limiting than other analyzed transients and would have a minimal impact on the fatigue usage factor which is well below the limit of 1.0 (Reference 13).

In addition to the above list of normal design transients, the following abnormal transients were also considered when arriving at a satisfactory usage factor as defined in the ASME Boiler and Pressure Vessel Code.

1. 200 cycles of loss of turbine load from 100% power

2. 200 cycles of total loss of reactor coolant flow when at 100% power

4.2.3 DESIGN SERVICE LIFE CONSIDERATIONS

The major Primary Coolant System components are designed considering a 40-year service life. In order to achieve this, the strict quality control assurance standards as outlined in Subsections 4.5.4 and 4.5.5 were followed.

Component design has also considered environmental protection, adherence to established operating procedures and irradiation effects on the material.

The reactor vessel is the only component of the Primary Coolant System which is exposed to a significant level of neutron irradiation. The irradiation surveillance program is outlined in Subsection 4.5.3. To compensate for any increase in the NDTT shift caused by irradiation, the Plant operating procedures for the pressure-temperature relationship during heatup and cooldown will be periodically revised to stay within the stress limits.

The design of the Primary Coolant System components allows for adequate inspection techniques to be applied over the lifetime of the Plant. All reactor internals are designed to be removable for inspection and to allow reactor vessel internal inspection. Insulation panels are removable for external inspection of selected highly stressed areas.

4.2.4 CODES ADHERED TO AND COMPONENT CLASSIFICATION

The original design, fabrication, construction, inspection, testing and classification of all reactor coolant system components are in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 edition, including all addenda through Winter 1965 (ASME B&PV Code, Section III, 1965, W65a), and the Code for Pressure Piping, ASA B31.1, 1955 (Reference 32). The replacement steam generators installed during 1990 meet ASME Code Section III 1977 edition.

The codes adhered to and component classifications are listed in Table 4-2.

Replacement parts and components will satisfy the requirements of the original plant construction code in a manner that is consistent with 10CFR50.55a, and the rules and requirements specified in ASME B&PV Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", Article IWA-4000."

4.2.5 SAFETY CONSIDERATIONS OF DESIGN PARAMETERS

Design Pressure

After establishing the normal operating pressure conditions of the Primary Coolant System, a minimum design pressure was determined which exceeds the normal operating pressure and anticipated operating transient pressure changes.

Major considerations employed in the determination of this selected minimum design pressure include: normal operating pressure, instrumentation and control response, reactor core thermal lag, coolant transport time, system pressure drop, and safety and relief valve characteristics. The design pressures for the individual reactor coolant system components are listed in their respective component description sections.

Design Temperature

The design temperature was selected to exceed the normal operating temperature and anticipated operating transient temperature changes for each primary coolant component. The design temperatures for the primary system components are listed in their respective component description sections.

Design Loads

The Primary Coolant System was designed to the criteria for load combination and stresses as defined for a CP Co Design Class 1 system in Section 5.10. These criteria assure the integrity of the Primary Coolant System to withstand the load imposed by the design basis accident simultaneously with the load imposed by the maximum seismic disturbance without loss of safety function.

4.2.6 PRIMARY COOLANT SYSTEM ASYMMETRIC LOADS

Pursuant to industry and NRC concerns for the potential effects of asymmetric loads on the Primary Coolant System components and supports, Consumers Power (in 1978) contracted with Combustion Engineering for a study to evaluate these concerns. A generic plant evaluation (see References 33 and 34) (Section 14.17.3) was completed by Combustion Engineering for Calvert Cliffs 1 and 2, Palisades, Millstone 2 and Fort Calhoun.

A further evaluation (see reference 35) was performed by Combustion Engineering to show that a flaw in the Primary Coolant System will result in a detectable leak before a large guillotine break would occur. The analysis was reviewed by the NRC in an SER dated October 27, 1989 (Reference 36). The SER concluded that, with the exception of concerns regarding seismic grid design, Palisades reactor system would withstand the effects of asymmetric LOCA loads and that the reactor could be brought to a cold shutdown condition safely.

4.3 SYSTEM DESIGN AND OPERATION

4.3.1 GENERAL DESCRIPTION

All components of the Primary Coolant System are located within the containment building. A flow diagram of the Primary Coolant System is shown in Figure 4-1. The system includes two identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains one steam generator, two circulating pumps, flow and temperature instrumentation, and connecting piping. A pressurizer is connected to one of the reactor vessel outlet pipes by means of a surge line. The pressurizer is located with its base at a higher elevation than the reactor vessel piping. This eliminates the need for a separate drain on the pressurizer and ensures that it is drained before maintenance operations. The equipment arrangement relative to its supports and the surrounding concrete is shown in Figures 1-4, 1-5, 1-8 and 1-10.

During normal operation, the four pumps circulate water through the reactor vessel where it serves as both coolant and moderator for the core. The heated water enters the two steam generators, transferring heat to the secondary (steam) system, and then returns to the pumps to repeat the cycle.

System pressure is maintained by regulating the water temperature in the pressurizer where steam and water are held in thermal equilibrium. Steam is either formed by the pressurizer heaters or condensed by the pressurizer spray to limit the pressure variations caused by contraction or expansion of the primary coolant.

Overpressure protection is provided by three ASME Code spring-loaded safety valves connected to the top of the pressurizer. Steam discharged from the valves is cooled and condensed by water in a quench tank. In the unlikely event that the discharge exceeds the capacity of the quench tank, the tank is relieved to the containment building via the quench tank rupture disc. The quench tank is located at a level lower than the primary safety and power operated relief valves. This ensures that any power-operated relief valve or primary safety valve leakage from the pressurizer, or any discharge from these valves, drains to the quench tank.

To protect against system pressure excursions during pump starts with a low temperature water solid system, additional automatic overpressure protection was provided. As a result of a detailed engineering study (see References 1 and 30), each power-operated relief valve has independent controls providing system pressure relief whenever the conditions of low temperature (430° for lower) and high pressure (see Figure 4-15) exist concurrently. Refer to Section 7.4 for further details.

The Primary Coolant System and its associated controls are designed to accommodate Plant step load changes of ± 10% of full power and ramp changes of ± 5% of full power per minute without reactor trip. The system will accept, without damage, a complete loss of load without direct reactor trip however, a turbine trip will cause a reactor trip if the reactor is operating >15% of full power.

Primary coolant leaves the containment building in small controlled quantities for sampling, primary coolant pump seal controlled bleedoff, chemistry and volume control purposes. Water which is removed from the primary system remains in a closed system until it has been processed by the radioactive waste system.

4.3.2 INTERFACES WITH OTHER SYSTEMS

To maintain the Primary Coolant System chemistry within the limits described in Subsection 4.3.13, a continuous feed and bleed operation is maintained by the Chemical and Volume Control System during normal operation. Three nozzles, one outlet and two inlet, are provided on the primary coolant piping for this operation.

An inlet nozzle is provided on each of four reactor vessel inlet pipes to allow injection of borated water into the reactor vessel by the Safety Injection System. An outlet nozzle is provided on one reactor vessel outlet pipe. During Plant cooldown, water is removed from the Primary Coolant System via this nozzle, circulated through the shutdown cooling heat exchangers by the low-pressure safety injection pumps where it is cooled and then injected back into the Primary Coolant System through the safety injection inlet nozzles.

Drains from the Primary Coolant System to the radioactive waste disposal system are provided for draining the Primary Coolant System for maintenance operations. Only cold leg 2B has a drain line and it contains normally closed loop drain valves. Downstream of the pair of loop drain valves is a spectacle flange. The spectacle flange is normally positioned to pass flow but may be blanked to prevent leakage past the loop drain valves. There is also one hot leg drain line directly connected to the primary system drain tank with two isolation valves. A connection is also provided on the quench tank for draining it to the radioactive waste disposal system following a relief valve or safety valve discharge.

Sampling system lines are provided from the primary coolant piping, the pressurizer and the quench tank to provide a means for taking periodic samples of the coolant for chemical and radiochemical analysis. The sampling lines may also be used for primary system degasification during plant shutdown.

A connection to the quench tank from the nitrogen supply system is provided to supply nitrogen for the quench tank gas blanket. A pressure regulator in the supply line maintains a constant quench tank pressure.

A connection to the quench tank spray header from the demineralized water supply is provided for adding water to the quench tank. This water cools the tank following a pressurizer relief or safety discharge and restores the tank operating level.

Component cooling water is supplied to the primary coolant pumps. Part of the water is circulated through coils to cool the motor lubricating oil system. The remainder of the water flows through the pump integral heat exchanger, where it cools the controlled bleedoff flow, and then through the thermal barrier and seal casing where it serves to cool the seal cavity. Any loss of component cooling water to the pumps is alarmed in the control room. The operator would then have ten minutes to restore flow before protective action would be required.

4.3.3 REACTOR VESSEL

The reactor vessel and top head assembly are shown in Figure 4-2. The reactor vessel and top head are designed in accordance with ASME B&PV Code, Section III, Class A, 1965, W65a. The design parameters are listed in Table 4-3.

The inner surface of the reactor vessel, which is in contact with primary coolant, is clad with 308/309 stainless steel. In the areas of internal attachments, the interior is clad with Ni-Cr-Fe alloy. The vessel closure flange is a forged ring with a machined ledge on the inside surface to support the reactor internals and the core. The flange is drilled and tapped to receive 54 7-inch diameter closure studs and is machined to provide a mating surface for the reactor vessel seal. An externally tapered transition section connects the flange to the cylindrical shell.

Extra thickness in the vessel nozzle course provides most of the reinforcement required for the nozzles. Additional reinforcement is provided for the individual nozzle attachments. A boss located around each outlet nozzle on the inside diameter of the vessel wall provides a mating surface for the internal structure which guides the outlet coolant flow. This boss and the outlet sleeve on the core support barrel are machined to a common contour to reduce core bypass leakage. A fixed hemispherical head is attached to the lower end of the shell. There are no penetrations in the lower head.

The removable upper closure head is hemispherical. The head flange is drilled to match the vessel flange stud bolt locations. The stud bolts are fitted with spherical washers located between the closure nuts and head flange to maintain stud alignment during heat flexing due to bolt up. To ensure uniform loading of the closure seal, the studs are hydraulically tensioned with a special tool and checked with an elongation gage after tensioning. In 1971 it was discovered that borated water had leaked, during vessel venting, onto the vessel flange causing stud corrosion. The studs required rework to remove the corrosion and were subsequently reanalyzed and tested for strength and integrity. Venting equipment was modified to avoid this possibility in the future.

Flange sealing is accomplished by a double-seal arrangement utilizing two silver-plated Ni-Cr-Fe alloy, self-energized O-rings. The space between and outside the two rings is monitored to allow detection of any ring leakage. The 2.76-inch ID control rod drive mechanism nozzles (Ni-Cr-Fe alloy through the head, stainless steel flange) terminate with bolted and seal-welded flanges at the upper end which are aligned on a single plane. This arrangement standardizes control rod extension shaft lengths and provides complete interchangeability of components. There are eight 4-inch nominal instrumentation nozzles of similar construction to the control rod drive mechanism nozzles. In addition to these nozzles, there is a 3/4-inch vent connection.

All of the reactor head nozzles are made of a Ni-Cr-Fe alloy, known as Alloy 600. By the 1990's Alloy 600 had become recognized as being susceptible to a cracking phenomenon identified as Primary Water Stress Corrosion Cracking (PWSCC). Three incidents of through wall leakage from Alloy 600 (one safe end and two nozzles) were experienced at Palisades on the pressurizer during the fall of 1993. Heightened awareness to the potential for PWSCC thus exists for the Alloy 600 penetrations in the entire Primary Coolant System. During the 1995 refueling outage, inspections of many of the Alloy 600 penetrations were performed to establish a reference for future PWSCC inspections. No indication of PWSCC was detected during those inspections.

During the 2004 refueling outage, volumetric inspections were performed on all of the control rod drive mechanism, incore instrumentation and the vent line nozzles as required per NRC Order EA-03-009 (Reference 52). After the inspection data was analyzed, indications of a discontinuity between the nozzle and reactor head material were identified in control rod drive nozzle penetrations #29 and #30. The repair process removed and replaced the defective areas using Alloy 690 materials. During the 2018 refueling outage, indications were found in nozzles 25, 33, and 36. The defective areas of these nozzles were also replaced with Alloy 690 materials

The core is supported from the reactor vessel flange. The control rod drive mechanisms are supported by the nozzles in the reactor vessel head.

Separate restraints are provided to absorb horizontal forces on the control rod drive mechanisms during seismic disturbances. The reactor vessel is supported on three pads welded to the underside of the coolant nozzles. This arrangement permits thermal growth of the vessel while maintaining it centered and restrained from movement resulting from seismic forces.

The reactor vessel internals are constructed with wetted parts of Stellite, Ni-Cr-Fe, stainless steel or Zircaloy. The control rod drive mechanisms' housings which act as a primary coolant boundary are stainless steel.

During a 1973 outage to inspect the reactor vessel internals as part of an investigation of core flux oscillations observed on the excore detectors, it was discovered that the core barrel had come unclamped at the vessel flange area. This allowed the core barrel and upper guide structure to move as a unit. As a result of this motion, indications of wear were found at the reactor head compensating ring shim interface, the core barrel keys and reactor vessel head and reactor vessel keyways, the core barrel hot leg coolant nozzle interface and the core barrel snubber shim pads (see Reference 2). In addition, the six remaining compensating ring shim locating bolts were broken. The cause of the unclamping was due to insufficient hold-down force on the core barrel. This insufficient force allowed wear to occur, which further reduced the force, which compounded the wear.

To eliminate the problem, the compensating ring shim was replaced with a hold-down device that provides a much greater hold-down force. The original core internals downward force was calculated to be about 36,000 pounds. The modified hold-down system provides a downward force of approximately 700,000 pounds, which is comparable to that of other pressurized water reactors.

The core barrel keys were replaced to ensure proper vessel internals alignment. The snubber shim pads were replaced. High spots on the reactor vessel ledge were removed.

4.3.4 STEAM GENERATOR

The nuclear steam supply system utilizes two steam generators shown in Figure 4-3 to transfer the heat generated in the Primary Coolant System to the secondary system. The design parameters for the steam generators are given in Table 4-4.

The steam generator is a vertical U-tube heat exchanger and is designed in accordance with ASME B&PV Code, Section III, 1977 edition. The steam generator operates with the primary coolant in the tube side and the secondary fluid in the shell side.

Primary coolant enters the steam generator through the inlet nozzle, flows through 3/4-inch OD U-tubes and leaves through two outlet nozzles. Vertical partition plates in the lower head separate the inlet and outlet plenums. The plenums are stainless steel clad, while the primary side of the tube sheet is Ni-Cr-Fe clad. The vertical U-tubes are Inconel-600. The tubes are rolled in the full depth of the tube sheet and then the tube-to-tube sheet joint is welded on the primary side.

Feedwater enters the steam generator through the feedwater nozzle where it is distributed via a feedwater distribution ring having bottom apertures which direct the flow through the downcomer. The downcomer is an annular passage formed by the inner surface of the steam generator shell and the cylindrical shell which encloses the vertical U-tubes.
Upon exiting at the bottom of the downcomer, the secondary water is directed upward over the vertical U-tubes. Heat transfer from the primary side converts a portion of the secondary water into steam.

Upon exiting from 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. Final drying of the steam is accomplished by passage of the steam through the corrugated plate dryers. The moisture content of the exiting steam is limited to a maximum of 0.25% at design flow.

The power-operated steam dump valves and steam bypass valve prevent opening of the safety valves following turbine and reactor trip from full power. The steam dump and bypass system is described in Subsection 10.2.1.

Overpressure protection for the shell side of the steam generators and the main steam line piping up to the inlet of the turbine stop valve is provided by twenty-four (24) safety valves. These valves are ASME B&PV Code spring-loaded, open bonnet, safety valves and discharge to atmosphere. Twelve safety valves are mounted on each of the main steam lines upstream of the steam line isolation valves but outside the containment. The valves are divided into three groups of four valves, each valve within a group having the same nominal opening pressure, but with staggered group opening pressures consistent with ASME B&PV Code allowances. The valves can pass a steam flow equivalent to an NSSS power level of 2,650 MWt at the nominal 1,000 psia set pressure. Parameters for the secondary safety valves are given in Table 4-5.

Table 4-5 contains a Set Point Tolerance (as-found testing) of ±3% of set pressure for the Main Steam Relief Valves. The basis for this value is documented in Technical Specification LCO 3.7.1, “Main Steam Safety Valves,” and Amendment No 116 to Provisional Operating License No DPR-20, Secondary Safety Valves (TAC No 69225). In summary, these documents allow a ±3% as-found set point tolerance for the Main Steam Safety Valves without the requirement for increasing testing scope per ASME Code Subsection IWV. However, all valves which are tested and found to be outside of ±1% of set pressure shall be restored to within the 1% criteria as required by Technical Specifications SR 3.7.1.

The steam generator shell is constructed of carbon steel. Manways and handholes are provided for access to the steam generator internals. The steam generators are mounted vertically on bearing plates to allow horizontal motion parallel to the hot leg due to thermal expansion of the primary coolant piping. Stops are provided to limit this motion in case of a coolant pipe rupture. The top of the unit is restrained from sudden lateral movement by energy absorbers mounted rigidly to the concrete shield.

In addition to the cyclic transients listed in Subsection 4.2.2, each steam generator is also designed for the following conditions such that no component will fail either by rupture or by developing deformations (elastic or plastic) that will impair the function, performance or integrity of the steam generator for further operation.

1. One cycle during which the steam on the shell side is at 900 psia and 532°F while tube (primary) side is depressurized to atmospheric pressure.

2. 2,400 cycles of transient pressure differentials of 85 psi across the primary head divider plate due to starting and stopping the primary coolant pumps.

3. 10 cycles of hydrostatic testing of the secondary side at 1,250 psia. The primary side will be pressurized such that the secondary to primary differential does not exceed 650 psid.

4. 320 cycles of leak testing of the secondary side at 1,000 psia. The primary side will be pressurized such that the pressure differential secondary to primary does not exceed 650 psid.

5. 15,000 cycles of adding 425 gpm of 32°F feedwater with the Plant in hot standby conditions.

6. 8 cycles of adding a maximum of 300 gpm of 32°F feedwater with the steam generator secondary side dry and at 600°F.

The unit is capable of withstanding these conditions for the prescribed number of cycles in addition to the prescribed operating conditions without exceeding the allowable cumulative usage factor as prescribed in ASME B&PV Code, Section III, 1977 edition.

4.3.4.1 Steam Generator Tube Degradation

The Palisades Plant experienced its first steam generator tube leak in early 1973. Eddy current examinations of the tubing detected general wastage attack in the U-bend area of tubes in the first eleven rows from the divider plate. The attack was attributed to the use of a coordinated phosphate secondary water chemistry treatment for pH control. All tubes in these first eleven rows were plugged.

In 1974, further leakage led to discovery of increased tube wastage and evidence of intergranular attack. A flushing program was performed and subsequent plant chemical control was changed to all-volatile treatment. Subsequent eddy current examinations in 1975 through 1981 showed that corrosion of the steam generator tubing had essentially ceased although minor tube denting was occurring as a result of the switchover to all-volatile treatment.

Continued examination over the next eight years revealed further IGA and other growing problems related to denting at the tube support plates. With excessive outage times and plant operation nearing the point of power limitation due to plugged tubes, the Steam Generator Replacement Project was initiated in mid 1989.

4.3.4.2 Steam Generator Replacement

Due to the tube degradation problems noted above, replacement of both steam generators was undertaken in late 1990. The replacement steam generators are designed to physically match the essential parameters of the existing steam generators and to be compatible with the performance characteristics utilized in the FSAR and the license for operation at 2,565.4 MWt. Consistent with other PCS equipment, the replacement steam generators are designed for operation at 2650 MWt should an increase in the licensed power level be pursued in the future.

Some component design changes were made to improve the replacement units:

1. Tube wall thickness was reduced slightly (.042 vs .048) to improve heat transfer. Combined with 308 less tubes, the overall steam generator heat transfer effect is unchanged.

2. Tube support design was changed from solid plate to eggcrate dividers and other features to minimize corrosion crevices and denting.

3. Blowdown capability was improved through an internal center duct and increase in blowdown nozzle size. Sampling improvements were also made.

4. Hand holes and inspection ports were added for future internals inspection. Manway sizes were also enlarged.

5. Integral flow restrictor nozzles were added to the main steam outlets to restrict blowdown flow in the event of a ruptured steam line.

4.3.5 PRIMARY COOLANT PUMPS

The primary coolant is circulated by four pumps, Figure 4-4, which are of the vertical single suction, centrifugal type. The estimated pump performance curve is shown on Figure 4-7. The discharge nozzle is horizontal and the suction nozzle is in the bottom vertical position. The original pressure containing components were designed and fabricated in accordance with the ASME B&PV Code, Section III, Class A, 1965, W65a.

The pump impeller is pinned, bolted and locked to the shaft. In early 1971 the pump suction deflector failed and was replaced with a new design with an impeller pin keeper. A close clearance thermal barrier assembly is mounted above the hydrostatic bearing. The assembly retards heat flow from the pump to the sealed cavity located above the thermal barrier. The assembly also tends to isolate the hot fluid in the pump from the cooler fluid above, and in the event of a seal failure, serves as an additional barrier to reduce leakage from the pump. Each pump is equipped with replaceable casing wear rings. A hydrostatic bearing is located in the fluid between the impeller and thermal barrier to provide shaft support. In 1971 the pump impeller was modified to include an auxiliary "piggy back" impeller to increase flow and pressure to the hydrostatic journal bearings and thereby improve the pump seal reliability. Additional shaft support is provided by bearings in the electric motor which is connected directly to the pump shaft via a rigid coupling.

The thermal barrier is located above the shaft seal assembly, which consists of four face-type mechanical seals, three full pressure seals mounted in tandem and a fourth low-pressure backup vapor seal designed to withstand operating system pressure with pumps stopped. The originally installed sealing system was a Type SU seal provided by the pump manufacturer, Byron Jackson. A replacement seal, Type N-9000, has been approved for installation. The replacement seal design maintains the three full pressure seals mounted in tandem with a low-pressure backup vapor seal. The difference in the two designs is basically in the performance of the N-9000 seal which is designed to be more stable than the Type SU. The Type SU seals are being replaced as time and manpower availability permits. The performance of both types of shaft seal systems is monitored by pressure and temperature sensing devices in the seal systems (Figure 9-7 Sht 1). A controlled bleedoff flow through the pump seals is maintained to cool the seals and to equalize the pressure drop across each seal. The controlled bleedoff flow is collected and processed by the Chemical and Volume Control System. Any leakage past the vapor seal (the last mechanical seal) is collected in the Radwaste System so that the pump leakage to the containment atmosphere is virtually zero. The seals are cooled by circulating controlled leakage through a heat exchanger mounted integrally within the pump cover assembly. To reduce Plant down-time and personnel exposure to radiation during seal maintenance, the seal system is contained in a cartridge which can be removed and replaced in two pieces. The face seals can be replaced without draining the pump casing. The seal detail is shown in Figure 4-6.

For purposes of the Station Blackout (SBO) reactor coolant inventory evaluation, a 25 gpm seal leakage rate is assumed per an agreement between NUMARC and the NRC. If the final NRC evaluation of the Generic Issue on Reactor Coolant Pump Seal Issues (GI-23) defines a higher leak rate, the reactor coolant SBO inventory will have to be reevaluated.

The pump is provided with a flywheel which will reduce the rate of flow decay upon loss of pump power. The inertia of the pump motor and flywheel is 98,000 lb-ft^2 at operating conditions. Flow coastdown characteristics are discussed in Section 14.7. The pump motor assembly includes motor bearing oil coolers, seal chamber, controls and instruments. Cooling water is provided from the Component Cooling Water System. A nonreversing mechanism consisting of freewheeling clutches having an inner and outer race and a complement of rollers and cams is also provided to prevent reverse rotation of the pump motor for Primary Coolant Pump P-50A, B & C. For P-50D which has a Westinghouse motor, the nonreversing mechanism consists of a ratchet and pawl assembly. This feature limits backflow through the pump under nonoperating conditions. The design parameters for the primary coolant pumps are given in Table 4-6.

The primary coolant pump and motor are supported by four support lugs welded to the scroll. Lubrite pads allow movement in the horizontal plane to compensate for pipe thermal growth and contraction. Vertical movement is prevented. Excess horizontal movement is prevented by additional bracing.

The pump is constructed of high alloy casting and stainless steel parts to minimize corrosion. These materials are listed in Table 4-6. The Type SU mechanical seals consist of a rotating titanium carbide ring riding over a hard carbon face. The Type N-9000 seals have a tungsten carbide ring which rides over a hard carbon face. Each of the three full pressure seals within both types of sealing systems is designed to accept the full operating system pressure but normally operates at one-third system pressure.

The pump seal consists of three pressure breakdown stages in series and one vapor stage. Each seal stage is designed to accept the full operating system pressure. The total pressure drop across the pump seal is normally divided so the vapor stage operates near Controlled Bleedoff pressure and each other stage has approximately one third of the differential between PCS and Controlled Bleedoff pressure.

The pump motor is sized for continuous operation at the flows resulting from four pump operation or partial pump operation with 0.75 specific gravity water. The motor service factor is sufficient to allow up to 250 consecutive hours of operation and an integrated total of 5,000 hours with 1.0 specific gravity water. The motors are designed to start and accelerate to speed under full load when 80% or more of their normal voltage is applied. The motors are contained within standard dripproof enclosures and are equipped with electrical insulation suitable for a zero to 100% humidity and radiation environment of 30 R/h. The pump motors are capable of limited short-term operation at decreasing frequency on turbine generator coastdown following a loss of offsite power as discussed in Subsection 8.4.3.2.

In order to support safe and stable conditions in accordance with the requirements of National Fire Protection Association (NFPA) 805, additional control capability has been added to allow the control room operators to trip the primary coolant pumps in the event that a fire renders the normal trip capability unavailable.

4.3.6 PRIMARY COOLANT PIPING

The primary coolant piping consists of lengths of 42-inch ID hot leg pipe from the reactor vessel outlet to the steam generator inlet and lengths of 30-inch ID cold leg pipe between the steam generator outlet and the pump suction nozzle and between the pump discharge and the reactor vessel inlets. The other major piece of primary coolant piping is the 12-inch, Schedule 140 surge line pipe between the pressurizer and one hot leg. Design parameters for the main primary coolant piping are given in the piping list table, Table 4-7.

The primary coolant piping is sized to limit the velocity to about 40 feet per second. The surge line is sized to limit the friction pressure drop, during the maximum surge, to 5% of the 2,500 psia system design pressure (125 psi).

The hot and cold leg pipes have no individual supports. The hot and cold legs are supported through their respective connections at the steam generator and reactor vessel nozzles with the cold legs having additional support through their connection to the reactor coolant pumps.

The primary coolant piping is of rolled bond clad plate construction, having a base metal of ASTM A 516, Grade 70, with a cladding of 304L stainless steel with a nominal thickness of 1/4 inch.

The 12-inch surge line is Type 316 stainless steel. Thermal sleeves are installed on all 2-inch or greater inlet nozzles to reduce thermal shock effects from auxiliary systems.

The 12-inch nominal diameter safety injection nozzles on the 30-inch ID cold leg pipes are constructed of carbon steel with a stainless steel clad interior. The 12-inch nominal diameter shutdown cooling nozzle on the 42-inch ID hot leg pipe is of the same construction. The remaining nozzles, all 3 inches or less in diameter, are of Ni-Cr-Fe alloy. The large diameter nozzles on the primary coolant piping all use Ni-Cr-Fe alloy safe ends for the field weld to the connecting piping. The Ni-Cr-Fe alloy used for the small nozzles and large nozzle safe ends in the piping is known as Alloy 600. By the 1990's Alloy 600 had become recognized as being susceptible to a cracking phenomenon identified as Primary Water Stress Corrosion Cracking (PWSCC). Three incidents of through wall leakage from Alloy 600 (one safe end and two nozzles) were experienced at Palisades on the pressurizer during the fall of 1993. Heightened awareness to the potential for PWSCC thus exists for the Alloy 600 penetrations in the entire Primary Coolant System. During the 1995 refueling outage, inspections of many of the Alloy 600 penetrations were performed to establish a reference for future PWSCC inspections. No indication of PWSCC was detected during those inspections.

The piping is shop fabricated and shop welded into subassemblies to the greatest extent practicable to minimize the amount of field welding. Fabrication of piping and subassemblies is done by shop personnel experienced in making large heavy wall welds. Welding procedures and operations meet the requirements of the ASME B&PV Code, Section IX, 1965, W65a. All welds are 100% radiographed and magnetic particle or liquid penetrant tested to the acceptance criteria of the ASME B&PV Code, Section III, Class A, 1965, W65a. All primary coolant piping penetrations are attached in accordance with the requirements of the ASA B31.1, Power Piping Code, 1955. Heat treatment of the piping is performed after all fittings have been assembled, nozzle bosses welded and cladding deposited. Field welds are postweld heat-treated to the requirements of the ASME B&PV Code, Section III, Class A, 1965, W65a. Cleanliness standards consistent with nuclear service are maintained during fabrication and erection.

All small piping connected to the Primary Coolant System, such as instrument lines, is Type 316 stainless steel and welded using the same specification limits as the major piping connections.

4.3.7 PRESSURIZER

The pressurizer maintains Primary Coolant System operating pressure and compensates for changes in coolant volume during load changes. The pressurizer is designed in accordance with ASME B&PV Code, Section III, Class A, 1965, W66a. Table 4-8 gives design parameters for the pressurizer. The pressurizer is shown in Figure 4-8.

Pressure is maintained by controlling the temperature of a saturated steam/water volume in the pressurizer. At full load conditions, slightly more than one-half the pressurizer volume is occupied by saturated water and the remainder by saturated steam. A portion of the pressurizer heaters is operated continuously to offset spray and heat losses, thereby maintaining the steam and water in thermal equilibrium at the saturation temperature corresponding to the desired system pressure.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the primary coolant. The average primary coolant temperature is programmed to vary as a function of load as shown in Figure 4-9. A reduction in load results in the average primary coolant temperature dropping to its programmed value for the lower power level. The resulting contraction of the coolant lowers the pressurizer water level causing the primary system pressure to fall. This loss of pressure is partially offset by flashing of pressurizer water into steam. Automatic energization of all pressurizer heaters on low primary system pressure generates steam and further limits pressure decrease. Also, if the water level in the pressurizer drops sufficiently below its set point, additional charging pumps in the Chemical and Volume Control System (see Figure 9-18) are automatically started to add coolant to the primary system and restore pressurizer level.

An increase in steam demand results in the average primary coolant temperature (T_{avg}) being increased in response to the T_{avg} program (see Section 7.5) to its value for the higher power level. The resulting expansion of primary coolant causes a surge of water into the pressurizer, compressing the steam and raising the Primary Coolant System pressure. If the pressure increase is large enough, the pressurizer spray valves open and spray relatively cold primary coolant from the primary coolant pump discharge into the steam space. This condenses some of the steam in the pressurizer limiting the pressure increase. If the coolant surge into the pressurizer causes the water level to rise sufficiently, letdown valves in the Chemical and Volume Control System (see Figure 9-18) open allowing coolant to leave the primary system, thereby decreasing the pressurizer level and consequently pressurizer pressure. The transient operation of the charging pumps and letdown orifices is dependent upon pressurizer water level error from the programmed level which is a function of power (see Figure 4-10).

Small primary pressure and coolant volume compensations are made by providing a steam volume to absorb flows into the pressurizer and water volume to match flows out of the pressurizer. The total volume of the pressurizer is determined by consideration of the following factors:

1. Sufficient water volume is necessary to prevent draining the pressurizer as a result of a reactor trip incident. The minimum pressurizer pressure during these transients must be limited so that the safety injection signal is not actuated, thereby preventing the charging pumps from pumping concentrated boric acid into the Primary Coolant System.

2. Pressurizer heaters cannot be uncovered by the outsurge following load increases, 10% step increase and 15% ramp increase.

3. The steam volume must be sufficient to yield acceptable pressure response to normal system volume changes during load change transients.

4. Water volume is minimized to reduce the energy release and resultant containment pressure during a Loss of Coolant Accident.

5. The steam volume must be sufficient to accept the primary coolant insurge resulting from loss of load without the water level reaching the safety valve or power-operated relief valve nozzles.

6. During load following transients, the total coolant volume change and associated charging and letdown flow rates should be kept as small as practical and be compatible with the capacities of the volume control tank, charging pumps and letdown orifices in the Chemical and Volume Control System.

To account for these factors and provide adequate margin at all power levels, the water level in the pressurizer is programmed as a function of average primary coolant temperature as shown in Figure 4-10. High or low water level error signals result in the actions shown on Table 4-9 and described above.

The pressurizer heaters are sized to heat up the pressurizer, when filled to the zero power level, at a rate which maintains the pressurizer pressure approximately 200 psia greater than the loop water saturation pressure as it is heated by decay heat and primary coolant pump operation. There is a total of 120 pressurizer heaters, including 108 nonvariable output type (known as backup heaters) and 12 proportional output type. A total installed capacity of 1487.5 kW is available, 150 kW in proportional heaters and 1337.5 kW in nonvariable type. (Heater kW output is nominal value at 460 volts.) The 1500 kW heater capacity is a CE standard design which applies to a NSSS with up to a 33% higher thermal rating than that of Palisades.

They are single unit sheath-type immersion heaters which 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. Ten percent of the heaters are connected to proportional controllers which were designed to adjust the heat input as required to account for steady-state losses and to maintain the desired steam pressure in the pressurizer and are always in the automatic mode for availability. The remaining backup heaters are connected to on-off controllers. These heaters were designed to be normally de-energized and turned on by a low pressurizer pressure signal or high level error signal. This latter feature was provided since reactor power increases result in an insurge of relatively cold primary coolant into the pressurizer which decreases the temperature of the water volume. The action of the Chemical and Volume Control System to restore the level then results in a pressure undershoot below the desired operating pressure. To minimize the pressure undershoot, the backup heaters were turned on early in the transient, resulting in more heating of the water before the pressure undershoot. The plant experienced difficulty in maintaining a steady system response with this arrangement, however, and now operates with the backup heaters on continuously and the proportional heaters remaining in auto. A low-low pressurizer level signal de-energizes all heaters to prevent heater burnout.

In the event of a loss of offsite power, approximately one half of the heater capacity (737.5 kW nominally) is normally connected to the ID emergency bus and can be manually controlled via a hand switch in the control room. This would provide sufficient heater capacity to establish and maintain natural circulation in a hot standby condition. Reference 7 has demonstrated that a 150 kW heater capacity is sufficient to maintain steady hot standby condition by offsetting the expected ambient heat loss. In addition, should the heaters on the 1E Bus be needed, methods and procedures have been established for manually connecting them to the IC emergency bus via a "jumper cable." The amount of time required to make this connection (less than five hours) has been evaluated to assure that a 20°F subcooling margin due to pressure decay is not exceeded (see Reference 7).

The pressurizer spray system consists of 3-inch lines running from the PCP discharges (P50B and P50C) to two 3-inch diaphragm-operated spray control valves, which then combine to a single 4-inch line connected to a single spray head inside the top of the pressurizer. The spray head is accessible through the pressurizer upper head manway. Manual isolator valves upstream and downstream of the spray valves afford isolation should it be necessary.

These components are sized to use the differential pressure between the pump discharge and the pressurizer to pass the amount of spray required to prevent the pressurizer steam pressure from opening the safety valves during normal load following transients. Use of lines from a cold leg in each of the heat transfer loops permits spray with less than four pumps operating. An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during Plant heatup or cooling if the primary coolant pumps are shut down. A small continuous flow is maintained through the spray lines when Primary Coolant Pump P50B or P50C is operating to keep the spray lines and the surge line warm, reducing thermal shock during Plant transients. This continuous flow also aids in keeping the chemistry and boric acid concentration of the pressurizer water equal to that of the coolant in the heat transfer loops. In 1975, isolation valves were provided for the pressurizer spray valves to alleviate the necessity of draining the entire Primary Coolant System when performing maintenance on these valves.

NRC Bulletin 88-11, dated December 1988, was issued to address pressurizer surge line temperature stratification concerns. The effects of thermal stratification were evaluated by the Combustion Engineering Owners Group. The Combustion Engineering Owners Group Report (see Reference 29) concluded the structural integrity of the pressurizer surge line is acceptable for the forty-year life of the Plant. The NRC issued an SER (Reference 39) on September 13, 1993 concluding that the CEOG analysis adequately demonstrates that the bounding surge line and nozzles meet ASME Code stress and fatigue requirements for the 40-year design. In References 27 and 28, CPCo provided additional information detailing completion of the required actions of Bulletin 88-11, including the requirement to update the pressurizer surge line stress and fatigue analyses.

Overpressure protection for the Primary Coolant System for abnormal pressure is provided by three ASME Code spring-loaded safety valves mounted on top of the pressurizer. These valves are piped to the quench tank. They are further described in Subsection 4.3.9.4 and Table 4-10. If an abnormal incident results in a pressure rise which exceeds the relieving capacity of the pressurizer spray, the pressurizer high pressure will trip the Reactor Protective System which will trip the reactor. The safety valves will open if the pressure continues to increase and exceeds the valve lift set point.

During periods of low temperature-water solid system operation, the Primary Coolant System and Shutdown Cooling System are protected from overpressurization by two power-operated relief valves. These valves are also piped to the quench tank. They are further described in Subsection 4.3.9.3 and Table 4-14.

The pressurizer is supported by a circumferential skirt welded around the lower head. Since the pressurizer surge line has sufficient flexibility, no provisions are made for horizontal movement and the skirt is bolted solidly to the floor. Additional supports are provided at the top of the pressurizer to control swaying during seismic loading.

The pressurizer is constructed of ASTM A 533, Grade B, Class 1 steel plate. The interior surface of the cylindrical shell and upper head is clad with Type 304 stainless steel. The lower head is clad with an Ni-Cr-Fe alloy to facilitate welding of the Ni-Cr-Fe alloy heater sleeves to the shell; stainless steel and Ni-Cr-Fe alloy safe ends are provided on the pressurizer nozzles, as required to provide for field welds to the connecting piping.

The Ni-Cr-Fe alloy used for the nozzles and safe ends on the pressurizer is known as Alloy 600. By the 1990's Alloy 600 had become recognized as being susceptible to a cracking phenomenon identified as Primary Water Stress Corrosion Cracking (PWSCC). A significant contributing factor to the initiation and propagation of PWSCC was determined to be temperature. Thus, since the pressurizer has the highest operating temperature of any location in the Primary Coolant System (PCS), a reasonable first assumption is that the Alloy 600 in its nozzles would develop PWSCC first.

Evidence of PWSCC was first noted at Palisades in September 1993 when a through-wall crack was found in the pressurizer Power Operated Relief Valve (PORV) nozzle safe end to pipe weld heat affected zone. The crack was found after the PCS leak rate increased while the plant was in hot shutdown preparing to startup following the 1993 refueling outage. The plant was returned to cold shutdown and the leak was repaired. The cracked section of field weld was removed, the safe end modified for a new weld prep, and a short new section of pipe installed to replace the material that was removed. The failed section was saved for thorough analysis of the failure mechanism. The failure mechanism was attributed to PWSCC.

A second and third failure of Alloy 600, apparently by PWSCC, occurred in October 1993 immediately following the repair of the PORV nozzle safe end. With the plant in cold shutdown but pressurized to 250 psia, leakage was noted around the pressurizer upper head temperature element (TE-0101) nozzle. The plant was depressurized to allow the leak to be repaired. The TE-0101 nozzle was inspected using an eddy current technique. The inspection identified several axially oriented cracks in the vicinity of the "J groove" weld that attaches the nozzle to the pressurizer head. This type of cracking in this style of weld has been seen in multiple other nuclear plants around the world and is caused by PWSCC. Based on this fact, the leakage from the TE-0101 nozzle was attributed to PWSCC. An inspection of the lower shell temperature element (TE-0102) also identified leakage from around the nozzle. Due to the location of TE-0102, it was not possible to remove the thermowell from the nozzle to allow eddy current examination of the nozzle; thus the leakage was assumed to be from PWSCC based on the TE-0101 nozzle failure. Both temperature element nozzles were repaired by welding a new attachment pad on the exterior of the pressurizer.

As a result of the PORV nozzle safe end failure and the metallurgical exams performed on the failed material, it was decided that replacement of the safe end with a less susceptible material was prudent. During the 1995 refueling outage, the Alloy 600 PORV nozzle safe end was replaced with a new Type 316 stainless steel safe end/spool piece using Alloy 690 for the attachment weld to eliminate PCS contact with Alloy 600 from the pressurizer PORV nozzle. The NRC issued a SER which allowed the use of forged (SB-564) Alloy 690 material (Reference 49).

Due to the previous PWSCC failures, a heightened awareness to the potential for future PWSCC exists for the Alloy 600 penetrations in the entire Primary Coolant system. During the 1995 refueling outage, inspections of many of the Alloy 600 penetrations were performed to establish a reference for future PWSCC inspections. No indication of PWSCC was detected during those inspections.

Also during the 1995 refueling outage, the Mechanical Stress Improvement Process (MSIP) was applied to three welds connecting Ni-Cr-Fe safe ends to stainless steel piping at the PCS nozzles on both ends of the pressurizer surge line and to the shutdown cooling outlet nozzle to mitigate any possible PWSCC by removing tensile stresses on the inside diameter of the piping. It consisted of using a hydraulically powered clamp ring assembly to squeeze the piping in the vicinity of the weldment, leaving a permanently deformed ring that is deformed approximately 1% in the radial direction. MSIP is accepted by the NRC for mitigating inter-granular stress corrosion cracking in BWR's.

4.3.8 QUENCH TANK

The pressurizer quench tank is designed to collect and condense the normal discharges from the pressurizer relief valves and safety valves and prevent them from being released to the containment. Parameters for the pressurizer quench tank are given in Table 4-11. Fabrication is in accordance with ASME B&PV Code, Section III, Class C, 1965, W65a.

The tank is constructed of carbon steel and the interior surface is epoxy lined. The tank normally contains demineralized water under a nitrogen overpressure. The sparger, spray header, nozzles and rupture disc fittings are stainless steel.

In the quench tank, the steam discharged from the pressurizer is discharged underwater by a sparger to enhance condensation by uniform distribution. The normal water volume of 800 ft^3 in the quench tank is sufficient to condense the total steam mass released by the relief valves during a 0 to 112% reactor power swing without primary coolant letdown or pressurizer spray. The water temperature rise in the quench tank is limited to 80°F, assuming a maximum initial water temperature of 120°F. The gas volume in the tank is sufficient to limit the maximum tank pressure after the above steam release to 50 psig, one-half of the rupture disc set point of 100 psig, assuming a maximum initial gas pressure of 10 psig. The quench tank is equipped with a demineralized water spray system to condense steam in the tank atmosphere and cool the tank water after a steam discharge into it. A drain and spray system is used to cool the tank after a discharge.

The quench tank can condense the steam discharged during a loss-of-load incident as described in Section 14.12 without exceeding the rupture disc set point assuming normal closing of the safety valves at the end of the incident. It is not designed to accept a continuous safety valve discharge. The rupture disc vents to the containment atmosphere.

4.3.9 VALVES

4.3.9.1 General Criteria

The Primary Coolant System is protected from overpressurization and depressurization through a series of valve arrangements including check valves, actuator-operated relief and control valves, manually operated control valves and spring operated safety valves. In 1979, pursuant to NUREG-0578, acoustical monitors and associated electronics were added to the power-operated relief valves and the safety relief valves to provide for positive valve position indication in the control room should they ever actuate.

1. Actuator-Operated Valves

   The position of each valve on loss of actuating signal (failure position) is selected to ensure safe operation of the system and Plant. System redundancy is considered when defining the failure position of any given valve. Valve position indication is provided at the main control panel where such information is considered necessary to ensure safe operation of the Plant.

2. Manually Operated Valves

   The pressure drops through these valves are conservatively taken as the maximum possible at the flows given except for MV-PC1056 and MV-PC1058 which are pressurizer spray needle valves.

Valves in the two categories above have backseats to limit stem leakage when in the open position. Globe valves are generally installed with flow entering the valve under the seat since this arrangement will reduce stem leakage during normal operation or when closed.

Allowable stem leakage for these valves is specified as follows:

a. When the valve is open and fully backseated, the stem leakage shall not exceed 3 cm^3/h per inch of stem diameter with system design pressure existing in the valve.

b. When the valve is closed, the stem leakage shall not exceed 10 cm^3/h per inch of stem diameter with system design pressure on the packing.

3. Check Valves

   Design parameters for Primary Coolant System check valves are given in Section 6.7. All check valves are of the totally enclosed type. Pressure losses through the check valves are conservatively taken as the maximum for a swing-type check at the flows given.

The Reactor Safety Study (RSS), WASH-1400 (Reference 6), identified in pressurized water reactors, an inter-system Loss of Coolant Accident (LOCA) which is a significant contributor to risk of core melt accidents (Event V). The design examined in the RSS contained in-series check valves isolating the high pressure Primary Coolant System (PCS) from the lower pressure safety injection piping. The scenario which leads to the Event V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the lower pressure piping which results in a LOCA that bypasses containment.

When pressure isolation is provided by two in-series check valves, and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important to safety, they should be tested periodically to ensure low probability of gross failure. Periodic examination of check valves must be undertaken to verify that each valve is seated properly and functioning as a pressure isolation device. The testing will reduce the overall risk of an inter-system LOCA. The testing may be accomplished by direct volumetric leakage measurement or by other equivalent means capable of demonstrating that leakage limits are not exceeded.

To ensure the continued integrity of selected check valves which are relied upon to preclude a potential inter-system LOCA, special requirements for periodic leak tests are specified in the Technical Specifications. The check valves required to be tested are:

HPSI Check Valves, Loop 1A:	ES-3101 and ES-3104
HPSI Check Valves, Loop 1B:	ES-3116 and ES-3119
HPSI Check Valves, Loop 2A:	ES-3131 and ES-3134
HPSI Check Valves, Loop 2B:	ES-3146 and ES-3149
LPSI Check Valves, Loop 1A:	ES-3103
LPSI Check Valves, Loop 1B:	ES-3118
LPSI Check Valves, Loop 2A:	ES-3133
LPSI Check Valves, Loop 2B:	ES-3148

4.3.9.2 Pressurizer Throttling (Spray) Control Valves

Parameters for the actuator-operated pressurizer spray valves are given in Table 4-12. Pressurizer spray reduces and limits pressure increase in the PCS during transients by spraying relatively cold water coolant from the cold legs into the pressurizer steam space. As a design basis, the flow rate established must provide acceptable pressure response during transients. The pressurizer spray line and valves are sized to allow sufficient spray to prevent high steam pressure from opening the safety valves during normal transients.

4.3.9.3 Power-Operated Relief Valves (PORV) and Block Valves

The power-operated relief valves serve two general purposes. First, they provide overpressure relieving capability during normal full pressure operation. And second, they provide overpressure protection for the Primary Coolant System during periods of low temperature, water solid system operation. The PORVs are normally isolated from the Primary Coolant System during power operation and consequently are not available for automatic pressure relief during this mode. Since no credit has been taken for the relief capacity of these valves in Chapter 14, "Safety Analysis," the Plant is permitted to operate at full pressure and temperature with the PORV isolation valves closed.

The two power-operated relief valves are designed to relieve sufficient pressurizer steam in order to protect the system from overpressurization during abnormal transients associated with low temperature-water solid system operations. Two PORVs are required per Technical Specifications for automatic system pressure relief at less than 430°F and relief pressure as shown in Figure 4-15. Parameters for these valves are given in Table 4-14.

The valves are solenoid-operated power relief valves. The two full capacity valves are located in parallel pipes which are connected to the single pressurizer relief valve nozzle on the inlet side and to the relief line piping to the quench tank on the outlet side. A motor-actuated isolation valve is provided upstream of each of the relief valves to permit isolating a valve for maintenance or in case of valve failure. Parameters for these valves are provided in Table 4-13. In response to NUREG-0578, the motor-actuated isolation valves' power was transferred to a Safety Grade Class 1E supply. Both the power-operated relief valves and the motor-actuated isolation valves are now powered from safety class power supplies. These motor-operated valves are normally closed since no Plant transient analysis takes credit for operation of the power-operated relief valves. These motor-operated valves would be used if a feed and bleed type operation was required to cool the Primary Coolant System in an emergency shutdown situation. Refer to Subsection 7.4.2.1 for further details. The Power Operated Relief Valves (PORV) meet NUREG-0737 item II.D.1. In response to Generic Letter 95-07 concern on the potential for pressure locking, specification change SC-96-023 modified the isolation (Block) valves by drilling a small hole on the upstream disc of the double disc gate valves.

Pursuant to NUREG-0737, acoustical transducers were added to these valves as an alternate means of indicating valve position. These monitors meet the same requirements as other engineered safeguard features (except redundancy). Refer to Section 7.4 for further details. There is annunciation when a valve has opened and position indication for each valve in the control room. This system supplements the original method of determining valve position through use of the temperature detector on the exhaust lines from these valves.

4.3.9.4 Spring-Actuated Primary Safety Valves

Three primary safety valves located on the pressurizer provide overpressure protection for the Primary Coolant System. They are totally enclosed, back pressure compensated, spring-loaded safety valves meeting ASME B&PV Code, Section III, Article 9, 1965, W65a, requirements. Parameters for these valves are given in Table 4-10.

Table 4-10 contains a setpoint tolerance (as-found testing) of ±3% of set pressure for the primary safety valves. The basis for this value is documented in Technical Specification LCO 3.4.10, “Pressurizer Safety Valves,” and Amendment 167 for Facility Operating Licensing, DPR-20. In summary, these documents allow a ±3% as-found setpoint tolerance for the primary safety valves without the requirement for increasing testing scope per the ASME OM Code. However, all valves which are tested and found to be outside of ±1% of set pressure shall be restored to within the 1% criteria as required by Technical Specifications SR 3.4.10.

The primary safety valves pass sufficient pressurizer steam to limit the primary system pressure to 110% of design (2,750 psia) following a complete loss of turbine generator load without simultaneous reactor trip while operating at 2,650 MWt. The reactor is assumed to trip on a high Primary Coolant System pressure signal. To determine the maximum steam flow, the only other pressure relieving system assumed operational is the secondary safety valves. Conservative values for all system parameters, delay times and core moderator coefficient are assumed. Overpressure protection is provided to the Primary Coolant System considering the effects of primary coolant pump head, flow pressure drops and elevation heads. The primary safety valves discharge through the relief line piping into the quench tank.

Pursuant to NUREG-0737, acoustical transducers were added to these valves as an alternate means of indicating valve position. These monitors meet the same requirements as other engineered safeguard features (except redundancy). There is annunciation when a valve has opened and position indication for each valve in the control room. This system supplements the original method of determining valve position through use of the temperature detector on the exhaust lines from these valves.

4.3.10 ENVIRONMENTAL PROTECTION

1. Flooding

   The containment building is of watertight construction and is inherently safe against external flooding. Refer to Subsection 5.4.1 for additional information.

2. Missiles

   The main coolant loops and the steam and feedwater piping are protected from missiles generated within the containment building. Barriers are provided where the use of radiation shielding and/or support structures for missile shielding would not be feasible for this purpose. Refer to Section 5.5 for additional information.

3. Seismic

   The NSSS is designed to withstand the load imposed by the maximum hypothetical accident plus the load imposed by the maximum seismic disturbance without loss of safety function. Refer to Section 5.7 for additional information.

4.3.11 MATERIALS EXPOSED TO COOLANT

The materials exposed to the primary coolant have shown satisfactory performance in operating reactor plants. A listing of materials is given in Table 4-15.

4.3.12 INSULATION

Carbon steel piping and equipment are insulated with a conventional material compatible with the temperature and functions involved.

A removable metal reflective-type thermal insulation is provided on the flange stud area of the reactor vessel closure head to permit access to the head studs for removal and reinstallation of the head. The thickness of insulation is such that the exterior surface temperature is not higher than 20°F above the surrounding air temperature.

Supports for the insulation consisting of carbon steel rings formed to fit the OD of the respective shells and necessary attachment brackets are provided. The heads of respective vessels have internal tapped studs appropriately spaced for attaching the insulation. All insulation support attachments are attached prior to final stress relief.

All insulation material used on stainless steel has a low soluble chloride content to minimize the possibility of chloride-induced stress corrosion of the metal.

4.3.13 SYSTEM CHEMICAL TREATMENT

Control and variation of the primary coolant chemistry is a function of the Chemical and Volume Control System. Sampling system lines are provided from the primary coolant piping to provide a means for taking periodic samples of the coolant for chemical analysis. The water chemistry is maintained as indicated in Table 4-16.

The solids content is maintained below the design level by minimizing corrosion through careful selection of materials, chemistry control and continuous purification of the letdown stream of primary coolant through filters and demineralizers. Hydrogen is maintained in the reactor coolant to chemically combine with the oxygen produced by the radiolysis of water. The primary coolant pH is controlled by the addition of lithium hydroxide (LiOH). Hydrazine may be added during initial start-up for oxygen scavenging.

Zinc (depleted in Zn-64) is added to primary coolant through the Zinc Addition System for the removal of radioactive cobalt ions from PCS piping (inner walls). Removal of the radioactive cobalt ions reduces dose to personnel from PCS piping.

All wetted surfaces in the Primary Coolant System are compatible with the above water chemistry. Specific component material is in the sections describing individual components.

4.4 SYSTEM DESIGN EVALUATION

4.4.1 DESIGN MARGIN

The Primary Coolant System is structurally designed for operation at 2,500 psia and 650°F (pressurizer 700°F). Operation of the system at 2,100 psia nominal and 600°F will result in material stresses of 85% of design values. Detailed structural analyses have been performed by the component vendors and reviewed independently by Combustion Engineering for all portions of the system. Welding materials used have physical properties superior to the materials which they join. Inspection procedures and tests specified and independently reviewed by Combustion Engineering were carried out to assure that pressure-containing components have the maximum integrity obtainable with present code-approved inspection techniques. A detailed discussion of quality control inspections is found in Chapter 15, Quality Assurance Program and Section 4.5.

The Primary Coolant System is equipped with four primary coolant pumps. In conjunction with protective instrumentation (see Chapter 7), this redundancy ensures adequate core cooling. The primary coolant pumps are connected to two 4,160-volt buses and receive power from either the main generator or from the offsite power system. Upon loss of all 4,160-volt power sources, short-term pumping power will be provided by the coastdown of the turbine generator. Under these conditions, primary flow during coastdown will be equal to or greater than that following the simultaneous loss of two pumps. The utilization of the rotating inertia of the turbine generator allows primary coolant flow to be maintained at a much higher rate after loss of power than would be the case if only the pump flywheels were used to provide coastdown flow. The utilization of the turbine generator inertia allows the pump flywheels to be designed for the two pump loss-of-flow accident. The most severe single electrical failure analyzed, however, is the loss of offsite power and no turbine coastdown. The results are within core design limits and are presented in Section 14.7. Following coastdown, natural circulation cooldown (see Reference 8) is established by releasing energy from the steam generators by steaming to atmosphere or the condenser. Makeup water is introduced from the Auxiliary Feedwater System to the steam generators. The Primary Coolant System is maintained at a sufficient pressure during this period (natural circulation cooldown) to avoid detrimental steam void formation in the system; particularly in the reactor vessel upper head region. If steam voiding does occur, procedural guidance is provided to minimize and control the voiding.

The Plant is designed to operate at reduced power with one or two pumps out of service; however, the Plant Technical Specifications requires all four pumps to be in service for continuous Plant operation.

4.4.2 PREVENTION OF BRITTLE FRACTURE

Brittle fracture will not occur if the peak stresses do not exceed the yield stress in the brittle fracture range. The establishment of temperature-pressure limitations for operation below NDT temperature +60°F is based on not exceeding yield for the peak stresses. This is accomplished by the following:

1. Performing a complete and thorough stress analysis to establish stress distribution taking into account all geometric shapes and surface stress concentrations.

2. Establishing material properties suitable for the application by adequate specification and testing during all stages of procurement and fabrication. Periodic monitoring of the material is done during its service life to determine the shift in properties due to environment.

3. Postweld heat treatment to reduce the effect of residual stresses whose magnitudes are unknown.

4. Establishing a safe limit (see Reference 9) of yield stress divided by a strength reduction factor in determining the operating pressure to allow for defects which may be undetected by the available techniques of nondestructive testing. It should be noted that the highest local stresses, other than those from such undetected defects, occur on the surface of the material. All surfaces are examined by nondestructive methods and defects and flaws so found which exceed allowable limits are repaired.

Additional areas of conservatism are the following:

1. Minimum specified values have been used for material properties rather than values from mill test reports of actual production material.

2. The increase in yield strength due to irradiation is not considered.

The establishment of operating conditions based on the stress limitations below will avoid exceeding the yield point stress in the brittle fracture range and hence avoid brittle fracture.

Stress limitations are used to establish pressure-temperature operating curves for the Plant. The pressure-temperature operating curves consider Plant heatup and cooldown in both critical and noncritical reactor conditions.

The pressure-temperature limits are established as follows:

1. A predicted reference temperature (RT_{NDT}) temperature shift for the required fast neutron ($E \geq 1$ MeV) fluence is determined (see Reference 10). Where NDT temperature data are not available to determine the RT_{NDT}, Reference 11 is employed.

2. The ASME Code (see References 12 and 54) prescribes the methodology used for obtaining the allowable loadings for any ferritic pressure-retaining materials in ASME Class 1 components. This methodology is based upon the principles of linear elastic fracture mechanics and involves a reference stress intensity factor prediction which is a lower bound of static, dynamic and crack arrest critical values. The reference stress intensity factor is a function of coolant temperature as well as temperature gradients through the reactor vessel wall. The calculated reference stress intensity factor must exceed that produced by pressure membrane stress in the vessel wall plus that produced by vessel wall thermal gradient stress. In the inequality associated with stress intensity comparison, a safety factor of 2 is applied to operating pressure membrane stress and a safety factor of 1.5 for hydrotest membrane stress.

3. The allowable pressure for a given operating temperature and heatup or cooldown rate is determined from the stress intensity equality (see Reference 11) with the appropriate safety factors included. A more detailed discussion of limits may be found in the basis discussion of the Technical Specifications LCO 3.4.3 (Reference 23).

4. Minimum primary coolant temperature for criticality is given in the Palisades Technical Specifications. This is calculated from hydrotest pressure. In addition, a minimum of 40°F temperature margin (see Reference 14) is included for all pressure-temperature curves for the reactor in the critical condition with respect to those in the noncritical condition.

The pressure-temperature limits are based upon the hypothetical (1/4) T reference flaw which is assumed to exist on either the inner or outer vessel wall surface (see References 12 and 54). The calculations yield continuous stress limitation curves.

During the 2004 Refueling Outage, inspections of the reactor head required by NRC Order EA-03-009 (Reference 52), resulted in the need to repair two of the CRD nozzles. The repair of these two nozzles required a change to the pressure-temperature cooldown curves and a corresponding amendment to the operating license. This change in the pressure-temperature limits is based on not exceeding the fracture toughness limits of the J-groove remnant materials of the repaired nozzles.

Pressurized thermal shock concerns noted in Reference 21 have been addressed by the addition of a 200°F subcooling curve to the pressure-temperature curves in the emergency operating procedures. This curve was developed per Reference 22 and has been adjusted for normal instrument inaccuracies. This curve supercedes the maximum cooldown curve whenever the PCS has experienced an uncontrolled cooldown (specifics of which are specified in the emergency operating procedures).

4.5 TESTS AND INSPECTIONS

4.5.1 GENERAL

Shop inspection and tests of all major components were performed at the vendor's plant prior to shipment. An inspection at the site was performed to assure that no damage has occurred in transit. Testing of the Primary Coolant Systems was performed at the site upon completion of the Plant construction. These tests included hydrostatic tests of primary and secondary loops. A complete visual inspection of all welds and joints was performed prior to the installation of the insulation. All field welds were radiographically and dye penetrant inspected in accordance with the requirements of the ASME B&PV Code, Section III, Class A, 1965, W65a and special erection specifications prepared by Combustion Engineering.

A hot flow test was made of the primary loop up to zero power operating pressure and temperature without the core installed. The system was checked for vibration and cleanliness. Auxiliary systems were checked for performance.

4.5.2 NIL DUCTILITY TRANSITION TEMPERATURE DETERMINATION

The reactor vessel is designed and fabricated in such a manner that significant operational limitations will not be imposed on the Primary Coolant System resulting from shifts in reactor vessel Nil Ductility Transition (NDT) temperature. The vessel material monitoring program is designed within the guidelines of ASTM E 185-66, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors." The preirradiated NDT temperature of the baseplate material was established using drop weight tests in accordance with ASTM E 208-63T and correlations were made with Charpy impact specimen tests conducted in accordance with ASTM E 23-60. This correlation, along with the Charpy impact specimens irradiated in the surveillance program, is used to monitor vessel material NDT temperatures. For the drop weight tests (performed for unirradiated materials), the test temperature is selected to bracket the NDT temperatures of the material.

The NDT temperature is determined by testing initial specimens from the plate material used in the intermediate and lower shell courses of the reactor vessel. The plate with the highest NDT temperature was selected from these initial tests and then a series of specimens from this plate were used to establish the NDT temperature. The NDT temperature is considered to be fixed when specimens 10°F above the temperature judged to be the NDT temperature exhibit a no-break performance.

For the preirradiated Charpy tests, a minimum of three specimens of each material were tested at any one temperature. Tests were performed at a sufficient number of different temperatures to establish the energy-temperature curve.

The test material used in establishing the unirradiated NDT temperature of the base metal were obtained from (1/4) T (where T is plate thickness) and/or (3/4) T locations of sections of the plate used in the intermediate and lower shell courses. The thermal history of the plate from which the specimens are taken is representative of that of the shell plating. The impact properties at these locations are considered to be representative of the material through the plate. Since the NDT temperature of the material of the plate surface is lower than at (1/4) T, it is conservative to use the properties of (1/4) T to establish the initial minimum operating temperature and as the base for the predicted minimum operating temperature after irradiation.

4.5.3 SURVEILLANCE PROGRAM

The surveillance program monitors the radiation-induced changes in the mechanical and impact properties of the pressure vessel materials. Changes in the impact properties of the material were determined by the comparison of pre- and post-irradiation Charpy impact test specimens. Changes in mechanical properties are determined by the comparison of pre- and post-irradiation data from tensile test specimens.

Three metallurgically different materials are included in the surveillance program. These are base metal, weld metal and heat-affected zone material. Base metal specimens are fabricated from sections of that plate used to form the intermediate shell course of the reactor vessel at the (1/4) T (thickness) and (3/4) T locations. Weld metal and heat-affected zone specimens are obtained by welding material cut from the (1/4) T and (3/4) T locations of the identical plate material used in the core region of the vessel. The welding and inspection procedures employed for fabrication of the pressure vessel were used in the preparation of weld material and heat-affected zone material test specimens. A complete record of the chemical analysis, fabrication history and mechanical properties of these surveillance test materials is maintained.

The test specimens are placed within corrosion-resistant capsules to prevent deterioration of the test specimens during irradiation. The design of the surveillance capsule incorporates features which minimize the temperature differentials between the test specimens and the reactor environment. The capsule size and shape is chosen to minimize neutron flux, thermal and hydraulic perturbations within the surveillance capsules. The capsule design also makes provisions for inclusion of radiation dosimeters, temperature monitors and correlation specimens.

The location of the surveillance capsule assemblies is shown on Figure 4-11. A typical surveillance capsule assembly is shown on Figure 4-12. A typical Charpy impact compartment assembly is shown on Figure 4-13. A typical tensile monitor compartment assembly is shown on Figure 4-14. The exposure locations and a summary of the specimens at each location is presented in Table 4-17.

Fission threshold detectors (U-238) are included in each surveillance capsule to measure the fast neutron flux. Threshold detectors of Ni, Ti, Fe, S and Co-free Cu are included to monitor the fast neutron spectrum.

Selection of threshold detectors was based on the recommendations of ASTM E 261-65T, "Method for Measuring Neutron Flux by Radioactive Techniques." Activation of the specimen material is also analyzed to determine the amount of exposure they received.

Temperature monitors are included to provide an indication of the highest temperature to which the surveillance specimens are exposed, but not the time-temperature history, or the variance between the time-temperature history, of different specimens. These factors, however, affect the accuracy of the estimated vessel RT_{NDT} temperature to only a small extent.

Correlation monitor specimens included in the surveillance capsules (Charpy v-notch specimens machined from ASTM standard reference material) are irradiated along with the surveillance test specimens. The standard reference material was obtained through Subcommittee II of ASTM Committee E-10 on Radioisotopes and Radiation Effects. Use of standard reference material test specimens permits correlation of the postirradiation data obtained in the course of this surveillance program with data obtained from other surveillance programs or irradiation experiments. In addition, changes in impact properties of the correlation monitors provide a cross-check on the neutron dosimetry.

The surveillance capsules are placed in the reactor at three locations. One series of capsules are placed on the outside of the core support barrel to obtain an accelerated exposure. These specimens receive the design lifetime neutron exposure in a relatively short time and provide data for predicting the RT_{NDT} temperature shift for the pressure vessel material over the design life of the vessel.

A second series of specimens are located on the inside of the pressure vessel wall. These specimens receive, at any given time, a slightly higher neutron dose than the pressure vessel. The RT_{NDT} temperature shifts resulting from the irradiation of these specimens closely approximate the RT_{NDT} temperature shift of the vessel materials and serve as a check on the data obtained from the accelerated exposure specimens.

A third series of specimens are located in a low flux region above the core. These specimens are exposed to all reactor temperature cycles but receive a very low neutron dose. Changes in the mechanical and impact properties of the vessel materials due to thermal exposure only can, therefore, be monitored on the basis of changes in properties of these specimens.

The schedule for removal of the surveillance samples is shown in Table 4-20. All surveillance capsules were inserted into their designated holders during the final reactor assembly operation. The capsules remain in the reactor until the desired fluence level has been attained by the specimens.

Test specimens removed from the surveillance capsules are tested in accordance with ASTM Standard Test Methods for Tension and Impact Testing. The data obtained from testing the irradiated specimens are compared with the unirradiated data and an assessment of the neutron embrittlement of the pressure vessel material made. This assessment of the RT_{NDT} temperature shift is based on the temperature shift in the average Charpy curves; the average curves being considered representative of the material.

The periodic analysis of the surveillance samples permits monitoring of the neutron radiation effects upon the vessel materials. If, with due allowance for uncertainties in RT_{NDT} temperature determination, the measured RT_{NDT} temperature shift turns out to be greater than predicted, then appropriate limitations would be imposed on permissible operating pressure-temperature combinations and transients to ensure that the existing reactor vessel stresses are low enough to preclude brittle fracture.

The integrated fast neutron flux to the reactor vessel has been calculated using the methods described in Subsection 3.3.2.6. Assuming an average future cycle capacity factor of 95%, future cycle flux levels comparable with Cycle 21, and an end of License Renewal period date of March 24, 2031, the maximum fast fluence the vessel wall will receive is ~3.429×10^{19} [n/cm^2] (References 3, 56, and 57.)

Pursuant to Amendment 34 to the Provisional Operating License DPR-20, capsule A-240 was removed first after 2.26 equivalent full-power years of reactor operations. The test results are contained in reference 15.

Two surveillance capsules were removed from the Palisades reactor vessel during the 1983 outage after 5.20 EFPY (see References 17 and 31). The thermal capsule, T-330, was subject to negligible neutron exposure, while W-290, the wall capsule, was subjected to an exposure closely approximating that of the vessel inner diameter. The mechanical testing of the wall capsule Charpy specimens indicated a very large RT_{NDT} shift of 290°F for the weld metal. Chemistry sampling of the test specimens indicated very high nickel content and a very high variation of nickel content across the specimen thickness. As with the case of the accelerated capsule earlier, the 536°F thermal monitor did not melt. All of these data are contained in Reference 18.

The W-290 data precipitated a review of Combustion Engineering reactor vessel fabrication records as well as Consumers Power Company core physics records. In addition, the 536°F thermal monitor was tested to its melting temperature. As a result of these reviews and tests, it was concluded that:

1. The reactor surveillance weld material was not the MIL-B4 modified wire, heat 27204, as reported to the staff on May 23, 1978, but a RACO wire used with a nickel addition wire.

2. Changes in the core power distribution over the last three fuel cycles had resulted in greater neutron fluence at the vessel wall and the fluences computed by Westinghouse were accurate.

3. The thermal monitors, expected to melt at 536°F, melt at a significantly higher temperature.

Surveillance capsule W-110 was removed from the reactor vessel after 9.95 EFPY. The test results are described in Reference 19. As in prior capsules, the shift in weld metal RT_{NDT} was quite large and the thermal monitors did not melt.

The Palisades reactor vessel surveillance program was designed in accordance with the requirements of ASTM E185-66. If designed today, the surveillance program would include different material than was selected in 1968. Because the material in the surveillance program is not the material that would have been selected if it had been designed in accordance with a more recent edition of the standard, the condition of the reactor vessel is estimated from unirradiated material properties adjusted by generic correlations based on material type, the copper and nickel content of the material, and the best estimate fluence. The limiting reactor vessel beltline material pertaining to the fracture toughness requirements of 10CFR50 Appendix G is plate D-3804-1 located in the lower shell. This plate is projected to drop below the Upper Shelf Energy (USE) lower limit of 50 ft-lb in December 2016 (Reference 55).

As required by 10 CFR 50 Appendix G, an equivalent margins analysis was submitted to document that the materials with USE values that drop below the USE lower limit of 50 f-lb throughout the remaining life of the reactor vessel have margins of safety against fracture equivalent to those required by Appendix G of Section XI of ASME Code (Reference 60). This equivalent margins analysis was subsequently approved by the NRC (Reference 61).

During the fall of 1994, CPCo performed material properties tests and chemistry analyses of newly acquired samples of weld material that had been fabricated using the same procedures and weld wire heat number as the limiting weld in the reactor vessel. These material samples were acquired from the shells of the steam generators that had been removed from service at Palisades. These tests and analyses indicated that the degree of

embrittlement of the Palisades reactor vessel could be higher than previously calculated. An updated fluence evaluation was performed. Analyses performed in accordance with the PTS rule and accepted by the NRC indicated that the reactor vessel would satisfy the requirements of the PTS rule until approximately 2003 (Reference 45).

At the beginning of Cycle 12, two supplemental surveillance capsules, designated SA-60-1 and SA-240-1, were installed in the capsule holders located on the core support barrel. See Table 4-17 for a listing of the capsule material and test specimens. These capsules contain weld specimens fabricated using similar materials and procedures as those used to fabricate the welds in the limiting portion of the reactor vessel. The capsules also contain standard reference material fabricated from the same plate as the standard reference material included in most reactor vessel surveillance programs. Temperature and flux monitors were also included. Most of the impact specimens in these capsules were modified to increase the number of specimens that could be installed.

Supplemental surveillance capsule SA-60-1 was removed at EOC 13. Testing of the subject materials was performed to determine the irradiated Charpy 30 ft-lb transition temperatures (T_{30}) and upper shelf energies for comparison with testing previously performed on similar materials in the unirradiated condition. Testing is described in DeVan, “Test Results of Capsule SA-60-1 Consumers Energy Palisades Nuclear Plant Reactor Vessel Material Surveillance Program,” BAW-2341, Revision 2, May 2001.

Supplemental surveillance capsule SA-240-1 was removed at EOC 14. Test results are described in DeVan, “Test Results of Capsule SA-240-1 Consumers Energy Palisades Nuclear Plant Reactor Vessel Material Surveillance Program,” BAW-2398, May 2001.

The limiting PTS screening criterion date calculated in Reference 45 utilized inputs that were updated as documented in WCAP-15353, Revision 0, (Reference 3) and WCAP-15353 Supplement 1-NP (Reference 56). Reference 57 contains the updated 10 CFR 50.61 PTS evaluation for the reactor vessel. The evaluation determined that the date when the most limiting reactor vessel material was projected to reach the 10 CFR 50.61 PTS screening criterion limit was April 2017. The NRC concluded that the PTS screening criteria would not be reached until April 2017 (Reference 58). A subsequent reactor vessel fluence evaluation reflecting updated actual reactor operation determined that the 10 CFR 50.61 PTS screening criterion would not be reached until August 2017 rather than April 2017 (Reference 60). The NRC concluded that the revision to August 2017 was acceptable (Reference 61).

In 2014, a license amendment request to implement 10 CFR 50.61a was submitted to the NRC (Reference 62). The alternate PTS evaluation accompanying the submittal concluded that the reactor vessel materials remain below the 10 CFR 50.61a screening criteria through the end of

licensed life (42.1 effective full power years) (Reference 63). The NRC approved the license amendment request in 2015 (Reference 64).

Capsule W-100 was removed at EOC 16. Test results are described in BWXT Report, "Analysis of Capsule W-100 from the Nuclear Management Company Palisades Reactor Vessel Material Surveillance Program," February 2004 (Reference 51).

4.5.4 NONDESTRUCTIVE TESTS

Prior to and during fabrication of the reactor vessel, nondestructive tests based upon the ASME B&PV Code, Section III, 1965, W65a, were performed on all welds, forgings and plates as follows:

All full penetration pressure containing welds were 100% radiographed to the standards of the ASME B&PV Code, Section III, Subparagraph N-624.8, 1965, W65a. Other pressure containing welds such as used for the attachment of mechanism housings, vents and instrument housings to the reactor vessel head were inspected by liquid penetrant tests of the root pass, each 1/2 inch of weld deposit and the final surface.

All forgings were inspected by ultrasonic testing, using longitudinal beam techniques. In addition, ring forgings were tested using shear wave techniques. Rejection under longitudinal beam inspection, with calibration so that the first back reflection is at least 75% of screen height, was based on interpretation of indications causing complete loss of back reflection. Rejection under shear wave inspection was based on indications, exceeding in amplitude the indication from a calibration notch whose depth is 3% of the forging thickness, not exceeding 3/8 inch with a length of 1 inch.

All forgings were also subjected to magnetic particle examination. Rejection was based on relevant indications of:

1. Any cracks and linear indications

2. Rounded indications with dimensions greater than 3/16 inch

Plates were ultrasonically tested using longitudinal and shear wave ultrasonic testing techniques. Rejection under longitudinal beam testing performed in accordance with ASME B&PV Code, Section II, SA-435, 1965, with calibration so that the first back reflection is at least 50% of screen height, was based on defects causing complete loss of back reflection. Any defect which showed a total loss of back reflection which could not be contained within a circle whose diameter is the greater of 3 inches or one-half the plate thickness was unacceptable. Two or more defects smaller than described above which caused a complete loss of back reflection were unacceptable unless separated by a minimum distance equal to the greatest diameter of the larger defect unless the defects were contained within the area described above. For shear wave testing, the maximum permissible flaw was one which

did not exceed that from a calibrated notch having a depth of 3% of the plate thickness and 1 inch long.

Nondestructive testing of the vessel was performed during several stages of fabrication with strict quality control in critical areas such as constant calibration of test instruments, metallurgical inspection of all weld rod and wire, and strict adherence to the nondestructive testing requirements of the ASME B&PV Code, Section III, Class A, 1965, W65a.

The detection of flaws in irregular geometries was facilitated because most nondestructive testing of the materials was completed while the material is in its simplest form. Nondestructive inspection during fabrication was scheduled so that full penetration welds are capable of being radiographed to the extent required by ASME B&PV Code, Section III, Class A, 1965, W65a.

Each of the vessel studs received two ultrasonic tests and one magnetic particle inspection during the manufacturing process.

The first ultrasonic test was a radial longitudinal beam inspection and the standard for rejection was 100% loss of back reflection or an indication which reduced the adjusted back reflection by greater than 20%. The second ultrasonic test was a radial inspection using the angle beam technique with the rejection standard the same as for forgings. The use of these techniques insured that only such materials were accepted that have flaws no greater than 1/2 inch and no observable cracks or sharply defined linear defects.

Magnetic particle inspection was performed on the finished studs. Axially aligned defects whose depths are greater than thread depth and nonaxial defects were unacceptable.

The vessel closure contains 54 studs, 7 inches in diameter with 8 threads per inch. The stud material is ASTM A 540-65, Grade B24, with a minimum yield strength of 130,000 psi. The tensile stress in each stud when elongated for operational conditions is approximately 40 ksi. Calculations show that 32 uniformly distributed studs can fail before the closure will separate at design pressure. However, 16 uniformly distributed broken studs or 4 adjacent broken studs will cause O-ring leakage. Failure of at least 16 adjacent studs is necessary before the closure will fail by "zippering" open.

The vessel studs are stressed as they are elongated by the stud tensioners during the initial installation of the vessel head and at each refueling. The amount of elongation versus hydraulic pressure on the tensioner will be compared with previous readings to detect any significant changes in the elongation properties of the studs. Studs which yield questionable data during the head installation, or receive damage to the threads, will be replaced before returning the vessel to pressure operations.

Table 4-18 summarizes the inspection program by component.

4.5.5 ADDITIONAL TESTS

During design and fabrication of the reactor vessel, a number of operations over and above the requirements of the ASME B&PV Code, Section III, Class A, 1965, W65a, were performed by the vendor. Table 4-19 summarizes the additional tests by component.

During the design of the reactor vessel, detailed calculations were performed to assure that the final product would have adequate design margins. The design adequacy was established by stress concentration factors which have been obtained through the use of photoelastic models for areas which are not amenable to calculation. A detailed fatigue analysis of the vessel for all design conditions has been performed. In addition, Combustion Engineering has performed test programs for the determination, solution and verification of analytical solutions to thermal stress problems. Also, fracture mechanics and brittle fracture evaluations have been performed.

All material used in the reactor vessel was carefully selected and precautions were taken by the vessel fabricator to ensure that all material specifications were adhered to. To assure compliance, the quality control staff of Combustion Engineering reviewed the mill test reports and the fabricator's testing procedures.

All welding methods, materials, techniques and inspections comply with ASME B&PV Code, Sections III and IX, 1965, W65a. Before fabrication was begun, detailed qualified welding procedures including methods of joint preparation, together with certified procedure qualification test reports, were prepared. Also, prior to fabrication, certified performance qualification tests were obtained for each welder and welding operator. Quality control was exercised for all welding rod and wire by subjection to a complete and thorough testing program in order to insure maximum quality of welded joints.

During the manufacture of the reactor vessel, quality control by the vendor, in addition to and in areas not covered by the ASME B&PV Code, Section III, 1965, W65a, included: Preparation of detailed purchase specifications which included cooling rates for test samples; requiring vacuum degassing for all ferritic plates and forgings; specification of fabrication instructions for plates and forgings to provide control of material prior to receipt and during fabrication; use of written instructions and manufacturing procedures which enabled continual review based on past and current manufacturing experiences; performance of chemical analysis of welding electrodes, welding wire and materials for automatic welding, thereby providing continuous control over welding materials; the determination of NDT temperature through use of drop weight testing methods and test programs on fabrication of plates up to 15 inches thick to provide information about material properties as thickness increases. Shear wave and longitudinal wave ultrasonic testing was performed on 100% of all plate material.

Cladding for the reactor vessel was a continuous integral surface of corrosion-resistant material, 1/4-inch nominal thickness. The detailed procedure used; ie, type of weld rod, welding position, speed of welding, nondestructive testing requirements, etc, was in compliance with the ASME B&PV Code, Sections III and IX, 1965, W65a.

Combustion Engineering has checked the cladding on completed reactor vessels and such tests have not shown the need for 100% ultrasonic testing for weld deposited cladding after fabrication. The clad surface is ultrasonically inspected transverse to the direction of welding for lack of bond at intervals of 12 inches or 1.4 times the base metal thickness, whichever is less.

Upon completion of all postweld heat treatments, the reactor vessel was hydrostatically tested, after which all weld surfaces, including those of welds used to repair material, were magnetic particle inspected in accordance with ASME B&PV Code, Section III, Paragraph N-618, 1965, W65a.

Quality control by the licensee was also carried out during the manufacture of the vessel by a resident inspector. This work included independent review of all radiographs, magnetic particle tests, ultrasonic tests and dye penetrant tests conducted during the manufacture of the vessel. A review of material certifications, and vendor manufacturing and testing procedures was also conducted. This review included all manufacturers' records such as heat treat logs, personnel qualification files and deviation files.

4.5.6 INSERVICE INSPECTION

Provision was made in the design to permit inservice inspection as may be required. The location of the more highly stressed portions of the reactor vessel was identified. These areas are equipped with removable insulation and portions may be inspected at various intervals, utilizing appropriate nondestructive testing techniques. In addition, the inside of the reactor vessel and the internals may be subjected to routine visual inspection during refueling outages. An inspection of accessible areas of the reactor vessel and internals, with a television camera or other suitable means, may be accomplished at any time when the reactor core is completely unloaded. The design permits all vessel internals except the flow skirt to be removed so that a complete internal vessel visual inspection would be possible. During refueling outages, the reactor vessel head and the closure sealing surfaces may be visually inspected. The internal parts of the vessel which are visible, including the cladding and components, may also be visually checked, as well as the accessible external surface of the vessel, nozzles and the vessel studs.

A combination of ultrasonic, dye penetrant, magnetic particle and visual inspections will be used to conduct the inspections. The planned inspection program takes into account the mechanisms which may lead to failure in the Primary Coolant System. Emphasis has been placed on the expected high stress areas as determined by a design evaluation and experience.

The major premises of this inspection program are:

1. Selected areas of expected maximum stress will be inspected at intervals in accordance with ASME B&PV Code Section XI, except as adjusted by NRC-approved code cases and relief requests. These inspections will serve to indicate potential problems before significant flaws develop there or at other areas.

2. If flaw initiation or growth is detected in one of the selected areas, all comparable areas in the primary system will be inspected.

3. Regardless of the results of inspection of the selected areas, all major discontinuity areas in the primary system will be inspected within a ten-year period, except as adjusted by NRC-approved code cases and relief requests.

4. A surveillance program will determine the shift in reference temperature (RT_{NDT}) of the vessel in the core region due to irradiation. The vessel will not be fully pressurized below the RT_{NDT}.

Refer to Section 6.9 for a detailed description of the inservice inspection program. A summary of the inspection program is as follows:

1. Areas of expected maximum stress selected for periodic inspections are:

    a. The flange-to-shell weld of the vessel

    b. The flange-to-torus weld of the vessel head

    c. The primary coolant outlet nozzle-to-shell welds and nozzle-shell radii on the vessel ID

    d. The dissimilar welds between the primary coolant piping and pumps

    e. Longitudinal and circumferential welds in the primary coolant piping

    f. Branch piping connections to the primary coolant piping

    g. The tube sheet-to-head weld of the steam generators

h. The nozzle-to-head welds in the lower head of the steam generators

i. The internal support stand welds in the lower head of the steam generators

j. The support stand to steam generator head

2. Areas to be inspected within a ten-year period, except as adjusted by NRC-approved code cases and relief requests, are:

   a. The nozzle-to-shell welds of primary coolant nozzles not inspected on a more frequent basis

   b. The longitudinal and circumferential welds in the core region of the reactor pressure vessel

   c. The outlet nozzle-to-shell weld of the pressurizer

   d. The longitudinal weld in the pressurizer

Ultrasonic inspection of components provides indications from discontinuities, impedance mismatches (such as a junction between Inconel weld metal and carbon steel) and from changes in component geometry. Baseline data to assist in interpretation of future inspection results will be acquired from a preservice inspection and pertinent shop data.

If indications of defect initiation or growth are noted, the program will be reviewed and sufficient inspections performed to determine that defects are not being initiated or propagated in other areas of the pressure vessel or components.

The bases for the above inspection points and the frequency of inspection are the result of a review of design drawings, the test results available from the PVRC vessel test program conducted at Southwest Research Institute, the present knowledge available on the mechanics of failure of such systems, ASME Section XI Code requirements, and NRC-approved code cases and relief requests. They are also based on the fact that the component fabricator for this Plant has a history of successful vessel fabrication in accordance with the practices of the ASME B&PV Code and more restrictive self-imposed specifications. Code manufacturing procedures and inspection techniques precluded the initial presence of large flaws in the vessel. Therefore, it is believed that the most likely location of a failure would be at a point of expected maximum stress concentration and not at some random location.

Thus, high stress locations are selected for monitoring of initiation of flaws. Furthermore, with baseline ultrasonic readings obtained on the pressure vessel and other inspection points in the reactor primary system, added assurance is attained that no significant flaws exist in the pressure boundary components of the Primary Coolant System.

Additional inservice inspection requirements have been established in Technical Specifications to address augmented steam generator tube inspection and inspection of primary coolant pump flywheels.

During the 2004 Refueling Outage, inspections of the reactor head required by NRC Order EA-03-009 (Reference 52) resulted in the need to repair two of the CRD nozzles. The NRC Order required that, if repairs are necessary, visual and volumetric inspections be completed during each subsequent refueling outage. Examinations during the 2018 refueling outage identified three additional CRD nozzles requiring repair.

Palisades was granted relief in a NRC Safety Evaluation dated February 11, 2009 (Reference 59) to extend the third inservice inspection interval for reactor vessel weld examinations until December 12, 2015.

4.5.7 NDTT OF OTHER PRIMARY SYSTEM COMPONENTS

The impact properties of all carbon steel and alloy steel materials which form a part of the pressure boundary of the Primary Coolant System were determined in accordance with the requirements of the ASME B&PV Code, Section III, Paragraph N-330, 1965, W65a. The materials were required to pass the acceptance test noted in Paragraph N-330 at 40°F, although it was an objective that the materials meet this requirement at 10°F. The operating stress limits for these materials in the Primary Coolant System other than the reactor vessel will be the same as those for the reactor vessel. Shortly after Plant start-up, the integrated neutron flux will result in the reactor vessel being the controlling component.

4.5.8 NONDESTRUCTIVE TESTS OF OTHER PRIMARY SYSTEM COMPONENTS

Prior to and during fabrication of the original components of the Primary Coolant System, nondestructive testing based upon the requirements of the ASME B&PV Code, Section III, Class A, 1965, W65a, was used to determine the acceptance criteria for various size flaws. The requirements for the Class A vessels are the same as the reactor vessel. Vessels designated as Class C were fabricated to the standards of the ASME B&PV Code, Section III, Article 21, 1965, W65a. Requirements for replacement parts and components are as specified in Section 4.2.1.

4.6 OPERATING LIMITATIONS

The minimum and maximum allowable Primary Coolant System pressures versus temperature are shown in References 23 and 24. The maximum differential temperature of 350°F between the pressurizer and hot leg and minimum suction pressure of 250 psia for the primary coolant pump were considered in applying these limitations.

4.7 PRIMARY COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION

4.7.1 LEAK DETECTION

Small leaks from the Primary Coolant System can be detected by one or a combination of the following systems:

1. Containment Atmosphere Relative Humidity - Each of 4 humidity detectors (1 in each steam generator compartment, 1 at the refueling floor near the reactor refueling cavity and 1 near the 590-foot elevation) is capable of detecting a change of humidity of 10% which would result from approximately 150 gallons of primary water leakage.

2. Containment Sump Level - Containment sump water level indication is provided in the main control room by level indicators which can be used to detect primary coolant system leakage. One level indicator actuates a high-high-water level alarm at 12 inches.

3. Containment Area Radiation - One radiation monitor, sensing from the discharge of all operating containment air coolers, is capable of detecting a 100 cm^3/min leak in 45 minutes based on 1% failed fuel.

4. Reactor Vessel Flange Leak Off - The inner seal leakage goes to a closed drain line and leakage will be detected by a pressure alarm set at 1,500 psig which will be activated by a steam leakage from the reactor of approximately 130 in^3. The outer seal liquid leakage is collected and drained to a closed drain line and will be detected by action of a level switch set at 120 inches which will result from a liquid accumulation of approximately 35 in^3.

5. Steam Generator Tube Leakage - Radiation detectors are provided to monitor the liquid effluent from the blowdown tank and gas effluents from the air ejector. The monitors have a sensitivity of 4×10^{-6} $\mu Ci/cm^3$ and can be set to alarm at 1.0×10^{-5} $\mu Ci/cm^3$ depending on normal background. The expected background will require that the alarm point be set higher than 1.0×10^{-5} $\mu Ci/cm^3$ but will be well below the activity released by a 5 gpm primary to secondary tube leak with 1% failed fuel.

6. A leak between the Component Cooling Water System and the Primary Coolant System via the primary coolant pump seals can be detected by a high component cooling water system surge tank level alarm and a high component cooling water system radiation alarm.

7. Each control rod drive mechanism face seal is equipped with a leak off which is piped to the floor drains leading to containment sump. Each leak off contains a thermocouple which will activate an alarm should above-normal temperatures occur.

8. The safety and power-operated relief valves may be a potential source of contained leakage. Seat leakage of these valves drains to the pressurizer quench tank and excess leakage would be detected by temperature monitors located in the valve discharge piping. Large amounts of seat leakage would also be detected by increases in level and temperature in the pressurizer quench tank. In 1979, pursuant to NUREG-0578, acoustical monitors were added to these valves to provide positive position indication in the control room.

9. Small leaks may also be determined by comparing charging pump and letdown flow rates and observing changes in pressurizer level.

4.7.2 OPERATOR ACTION FOLLOWING LEAK DETECTION

In the event a small leak is indicated in the Primary Coolant System, immediate steps will be initiated to identify the source and isolate the leak if possible.

The initial operator action following an indication of a leak in the Primary Coolant System is to check pressurizer level and the Chemical and Volume Control system response. The next step is to attempt to determine the leak rate. This may be done by comparing charging and letdown flows and observing pressurizer level.

Technical Specifications address limits for operating with identified and unidentified PCS leakage. If the leakage rate exceeds the ability of the Chemical and Volume Control System to maintain pressurizer level, the reactor is manually tripped.

4.8 PRIMARY COOLANT GAS VENT SYSTEM

The Primary Coolant Gas Vent System (PCGVS) is designed to vent steam or noncondensible gases from the reactor vessel head and pressurizer areas of the Primary Coolant System. This is done to assure core cooling during natural circulation is not inhibited. This system was installed pursuant to NUREG-0737, Topic II.B.1.

The system, see Figure 4-1, consists of a flow-limiting orifice on both the reactor vessel vent and pressurizer vent lines, solenoid valves, a pressure transmitter for pressure indication, and connecting piping. Refer to Section 7.4 for description of valve control features.

The orifices are placed as close to the vessels as possible to limit the possibility of an uncontrolled Loss of Coolant Accident (LOCA). They are sized such that they would limit mass loss from a line break to less than the makeup capability of a single charging pump in order to maintain pressurizer level control.

The entire PCGVS is designed for Seismic Category I. The primary coolant pressure boundary within the PCGVS, up to and including the second solenoid valve, is Safety Class 2 (Safety Class 1 upstream of the flow-limiting orifices). The piping was designed, fabricated, installed, and tested to ASME B&PV Code, Section III, Subsection NC, 1974, S76a. Supports were designed, fabricated, installed and tested in accordance with Subsection NF, ASME B&PV Code, Section III, 1974, S76a. The entire PCGVS was analyzed using the ADLPIPE Computer Code, Revision 3C. The PCGVS, up to and including the second normally closed solenoid valve, will be maintained as Quality Group A (Class 1) per Reg Guide 1.26 and ASME B&PV Code, Section XI.

The PCGVS piping is AISI Type 304 or Type 316 stainless steel. The PCGVS solenoid valve bodies are AISI Type 316 stainless steel. The design pressure and temperature for the reactor vent line is 2,500 psia at 650°F. The balance of the PCGVS, up to and including the second solenoid valve, is designed to 2,500 psia at 700°F. The pressure/temperature values used for the PCGVS were chosen based on the design temperature and pressure for the reactor vessel, pressurizer and primary coolant loop.

The method for determination of the presence of voids and the actions to be taken for their venting, are described in Plant Emergency Operating Procedures (EOP) and Abnormal Operating Procedures (AOP). Refer to Section 7.4 for a description of the subcooled margin monitor.

The primary vent path for large volumes of noncondensible gases is directed into the open area of containment where adequate mixing with the containment atmosphere is assured. The PCGVS exhausts the noncondensible gases through a 3-inch diameter, open-ended pipe. The discharge is directed straight up toward the containment dome.

There are no safety grade components directly above the PCGVS in containment which could be adversely affected by the action of the PCGVS.

The secondary vent path is to the quench tank (T-73). This vent path is intended for use for discharging small volumes of gases. The use of the quench tank for discharging large volumes, with the attendant failure of the rupture disc, would lead to less severe environmental conditions than would result from a continuous operation of the safety valves on the pressurizer.