4.2 SYSTEM DESIGN AND OPERATION

4.2.1 General Description

The Reactor Coolant System consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, loop piping and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system is shown in Figures 4.2-1 and 4.2-1A.

Reactor Coolant System design data are listed in Tables 4.1-1 through 4.1-8 and Table 4.1-12.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by a pressurizer spray, to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressure control system is described in Chapter 7. Spring-loaded safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

4.2.2 Components Description

4.2.2.1 Reactor Vessel

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the core, core support structures, control rods, thermal shield and other parts directly associated with the core. The reactor vessel closure head contains head adaptors. These head adaptors are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adaptors contain Acme threads for the assembly of control rod drive mechanisms (Unit 1 and Unit 2 design has eliminated the Core Exit Thermocouple (CET) ACME threads). A dedicated nozzle, near the center of the reactor head, connects to vent piping, which vents to the upper containment volume, to provide reactor vessel head venting of non-condensable gas while maintaining adequate core cooling and containment integrity (Both Unit 1 and Unit 2 have a dedicated reactor head vent nozzle). For further details see Sub-Section 4.2.2.6. The seal arrangement at the upper end of these adaptors consists of omega seal weld to the CRDM pressure housing. The upper end of the instrument adaptor consists of a mechanical sealing assembly. The vessel has inlet and outlet nozzles located in a horizontal plane just below the vessel flange but above the top of the core. Coolant enters the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom and flows up through the core to the outlet nozzles.

The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear in-core detection instrumentation. Each tube is attached to the inside of the bottom head by a partial penetration weld.

The reactor vessel is designed to provide the smallest and most economical volume required to contain the reactor core, control rods and the necessary supporting and flow-directing internals. Inlet and outlet nozzles are spaced around the vessel. Outlet nozzles are located on opposite sides of the vessel to facilitate optimum layout of the Reactor Coolant System equipment. The inlet nozzles are tapered from the coolant loop-vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The reactor vessel flange and head are sealed by two hollow metallic O-rings. Seal leakage is detected by means of two leak-off connections; one between the inner and outer ring, and one outside of the outer O-ring. Piping and associated valving are provided to direct any leakage to the reactor coolant drain tank. Leakage will be indicated by a high-temperature alarm from a detector in the leakoff line.

Ring forgings have been used in the following areas of the reactor vessel:

A. Vessel Flange

B. Eight Primary Nozzles

C. The Unit 1 and Unit 2 closure heads are one-piece forgings.

Core structural load bearing members were made from annealed type 304 stainless steel, so there is no possibility that they may become furnace sensitized[2]. The only exception is the core barrel itself, which required stress relief during manufacture at temperatures over 750°F, to minimize the possibility of severe sensitization while maintaining the necessary conditions for relieving residual fabrication stresses.

[2] The term "furnace sensitized" is interpreted as austenitic stainless steel wrought material and weld metal components which have been post weld heat treated in accordance with ASME Section III requirements, and which on the basis of its composition and thermal history would not be expected to pass ASTM-A-393.

Other pressure or strength bearing stainless steel components or parts in the reactor vessel and associated Reactor Coolant System that may become furnace sensitized [1] during the fabrication sequence include:

4.2.2.1.1 Unit 1

A. Reactor Vessel

1. Primary nozzle safe ends - Type 316 forging

B. Steam Generators

1. Primary nozzle safe ends - weld metal buttered ends - Type 316 LN forgings (Unit 1 Replacement Steam Generator). [3]

4.2.2.1.2 Unit 2

A. Reactor Vessel

1. Primary nozzle safe ends - Type 316 forging overlaid with weld prior to final post weld heat treatment.

2. Monitor Tubes.

B. Steam Generators

1. Primary nozzle safe ends - weld metal buttered ends.

Westinghouse has evaluated the use of sensitized stainless steel and reactor components in pressurized water reactors. The results of this evaluation are summarized in Reference 1, which covers the nature of sensitization, conditions leading to stress corrosion and associated problems with both sensitized and non-sensitized stainless steel. The results of extensive testing and service experience that justify the use of stainless steel in the sensitized condition for components in Westinghouse PWR Systems is presented in Reference 1.

Although Westinghouse testing and evaluation showed justification for the use of sensitized stainless steel, extra modifications were made during the fabrication of these vessels, when it was revealed that the work would not significantly affect the delivery schedules and would result in a

[3] Through an R & D program performed by Babcock & Wilcox (BWI), it was demonstrated that the primary nozzle safe end material does not sensitize when subjected to the PWHT times that the Unit 1 Replacement Steam Generator (RSG) was subjected to.

somewhat more conservative design. Also, for the Unit 2 replacement steam generator lower assemblies the use of severely sensitized stainless steels was prohibited on the primary nozzle safe ends.[2]

The cylindrical portion of the reactor vessel below the refueling seal ledge is permanently insulated with a metallic reflective-type insulation supported from the reactor coolant nozzles. This insulation consists of inner and outer sheets of stainless steel spaced 3 inches apart with multilayers of stainless steel. Removable panels of the metallic reflective insulation described above are provided for the reactor vessel head and closure region.

These panels are supported on the refueling seal ledge and vent shroud support ring. The rest of the closure head is insulated with removable panels of at least three inches of the reflective insulation described. The bottom head is also insulated with reflective insulation, which is not removable.

Schematics of the reactor vessel are shown in Figures 4.2-2 and 4.2-2A. The materials of construction are given in Table 4.2-1 and the design parameters are given on Table 4.1-3. A description of the reactor vessel internals is given in Chapter 3.

4.2.2.2 Pressurizer

The pressurizer provides a point in the Reactor Coolant System where liquid and vapor can be maintained in equilibrium under saturated conditions for control purposes.

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. Electrical heaters are installed through the bottom head of the vessel while the spray nozzle, relief and safety valve connections are located in the top head of the vessel. The heaters are removable for maintenance or replacement. A vent connection is provided on the piping ahead of the power-operated relief valves to vent non-condensible gases or steam from the pressurizer to the upper containment volume. For further details see Sub-Section 4.2.2.6.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line, which is attached to the bottom of the pressurizer, connects the pressurizer to the hot leg of one reactor coolant loop.

During an insurge, the spray system, which is fed from two cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the set-point of the power-operated relief valves. The spray valves on the pressurizer are modulating, air operated, control valves. In addition, the spray valves can be operated manually by a switch in the control room. A small continuous spray flow is provided through a manual bypass valve around the power-operated spray valves to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excessive cooling of the spray piping.

During an outsurge, flashing of water to steam and generating of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during insurges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop. A screen at the surge line nozzle and baffles in the lower section of the pressurizer prevents cold insurge water from flowing directly to the steam/water interface and it assists mixing.

The volume of the pressurizer is equal to or greater than, the minimum volume of steam, water, or total of the two, which satisfies all of the following requirements:

1. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.
2. The water volume is sufficient to prevent the heaters from being uncovered during a step-load increase of ten percent of full power.
3. The steam volume is large enough to accommodate the surge resulting from the design step load reduction of full load with reactor control and steam dump without the water level reaching the high-level reactor trip point.
4. The steam volume is large enough to prevent water relief through the safety valves following a loss of load with the high water level initiating a reactor trip.
5. The pressurizer will not empty following reactor trip and loss of load.
6. The Emergency Core Cooling Signal will not be activated during reactor trip and turbine trip.

The general configuration of the pressurizer is shown in Figure 4.2-3 and the design data are given in Table 4.1-4.

4.2.2.2.1 Pressurizer Spray

Two separate, automatically controlled spray valves with remote manual overrides are used to regulate pressurizer spray. In parallel with each spray valve is a manual throttle valve which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open, and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping to the pressurizer forms a water seal, which prevents steam buildup back to the control, valves. The spray rate from one valve is sufficient to prevent the pressurizer pressure from reaching the operating (set) point of the power relief valves during a step reduction in power level of ten percent full load.

The pressurizer spray lines and valves are large enough to provide adequate spray flow using as the driving force the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force. The line may also be used to assist in equalizing the boron concentration between the reactor coolant loops and the pressurizer.

A flow path from the Chemical and Volume Control System to the pressurizer spray line is also provided. This additional facility provides auxiliary spray to the vapor space of the pressurizer during cooldown if the reactor coolant pumps are not operating. The thermal sleeve on the pressurizer spray connection is designed to withstand the thermal stresses resulting from the introduction of cold spray water.

4.2.2.2.2 Surge Line

The surge line is sized to limit the pressure drop during the maximum anticipated surge to less than the difference between the maximum allowable pressure in the reactor vessel and the loops (at the point of highest pressure) and the pressure in the pressurizer at the maximum allowable accumulation with the safety valves discharging.

The surge line and the thermal sleeves at each end are designed to withstand the thermal stresses resulting from volume surges of relatively hotter or colder water, which may occur during operation.

4.2.2.3 Pressurizer Relief Tank

The pressurizer relief tank condenses and cools the discharge from the pressurizer safety and relief valves, as well as several smaller relief valves. The tank normally contains water and a predominantly nitrogen atmosphere; however, provision is made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and/or oxygen.

The pressurizer relief tank, by means of its connection to the Waste Disposal System, provides a means for removing any non-condensable gases from the Reactor Coolant System, which might collect in the pressurizer vessel.

Steam is discharged through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature. The tank is equipped with an internal spray and a drain, which are used to cool the tank following a discharge. The tank is protected against a discharge exceeding the design value by two rupture discs, which discharge into the reactor containment. The tank is carbon steel with a corrosion-resistant coating on the wetted surfaces. A flanged nozzle is provided on the tank for the pressurizer discharge line connection. This nozzle and the discharge piping and sparger within the vessel are austenitic stainless steel.

The tank design is based on the requirement to condense and cool a discharge of pressurizer steam equal to 110 percent of the volume above the 100%-power pressurizer water level set-point. The tank is not designed to accept a continuous discharge from the pressurizer. The volume of water in the tank is capable of absorbing the heat from the assumed discharge, with an initial temperature of 120°F and increasing to a final temperature of 210°F. If the temperature in the tank rises above 126°F during plant operation, the tank is cooled by spraying in cool water and draining out the warm mixture to the Waste Disposal System.

The spray rate is designed to cool the tank from 200°F to 120°F in approximately one hour following the design discharge of pressurizer steam. The volume of nitrogen gas in the tank is selected to limit the maximum pressure following a design discharge to 50 psig.

The rupture discs on the relief tank have a relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure is twice the calculated pressure resulting from the maximum safety valve discharge described above. The tank and rupture discs holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

Principal design parameters of the pressurizer relief tank are given in Table 4.1-4.

4.2.2.3.1 Discharge Piping

The discharge piping (from the safety and power-operated relief valves to the pressurizer relief tank) is sized to prevent back-pressure at the safety valves from exceeding 20 percent of the set-point pressure at full flow. The pressurizer safety and power relief valves discharge lines are stainless steel.

4.2.2.4 Steam Generators

The steam generators are vertical shell and U-tube heat exchangers with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. Manways are provided for access to both sides of the divided head. Feedwater enters the steam generators and is distributed through a feedwater ring located just below the moisture separators. Thermal sleeves are provided in the feedwater piping elbows at the steam generator inlet (Unit 2 only). For the Unit 1 Babcock & Wilcox Model 51R replacement steam generators, the thermal sleeves are welded to a transition ring which is then welded to the main feedwater nozzle forging on the steam drum shell side of the steam generator. Feedwater flow is out of the top of the feedwater ring through "J" tubes, down between the steam generator shell and tube bundle wrapper and into the tube bundle just above the tube sheet. The "J" tubes prevent rapid drainage of the feedwater ring due to a drop in steam generator water level and thus eliminate or reduce the possibility of water hammer in the feedwater line. Steam is generated on the shell side of the tube bundle and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

The units are primarily constructed of carbon steel. The heat transfer tubes are Inconel, the primary side of the tube sheets are clad with Inconel, and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel.

The Unit 2 and Unit 1 steam generators of this type are shown in Figures 4.2-4 and 4.2-4a through 4.2-4e respectively. Design data are given in Table 4.1-5.

Each steam generator is designed to produce 25 percent of the steam flow required at full-power operation. The internal moisture-separating equipment is designed to insure that the moisture carryover will not exceed 0.045 percent by weight for Unit 1 and 0.15 percent by weight for Unit 2 under the following conditions:

a. Steady-state operation up to 105 percent of full-load steam flow, with water at the normal operating level.

b. Loading or unloading at a rate of five percent of full power steam flow per minute in the range from 15% to 105% of full load steam flow. The Unit 1 Babcock & Wilcox Model 51R replacement steam generators have been analyzed for both the pre-Measurement Uncertainty Recapture power uprate power rating (3264 MWt) and the 3600 MWt power uprate condition for a continuous and uniform ramp power change of 5% per minute between 0% and full load. The Unit 2 Model 51 replacement steam generators have been analyzed for both the pre-Measurement Uncertainty Recapture power uprate power rating (3425 MWt) and the 3600 MWt power uprate condition for a continuous and uniform ramp power change of 5% per minute between 0% and full load, with 11,680 cycles as described above.

c. A step-load change of ten percent of full power in the range from 15% to 105% full load steam flow.

In 2000, the Unit 1 steam generators were replaced. The Babcock & Wilcox (BWI) Model 51R replacement steam generators consist of a lower replacement steam generator subassembly (RSGSA), replacement steam drum internals, and replacement feedring fabricated by BWI plus the re-used existing Model 51R steam drum pressure boundary. The procurement of the replacement steam generator subassemblies did not affect the original design basis. Where appropriate, the tables and subsections of Chapter 4 have been revised to reflect the design enhancements of the Unit 1 RSG.

During the summer of 1988, Unit 2 steam generators were replaced. This entailed the procurement of new replacement lower assemblies and refurbishment of the upper assemblies and internals (steam dome). The procurement of the replacement steam generator assemblies did not affect the original design basis. Where appropriate, the tables and subsections of Chapter 4.0 have been revised to reflect the design enhancements of the replacement steam generator lower assemblies with their refurbished upper assemblies.

The Unit 1 and Unit 2 design pressure limit for primary-to-secondary pressure differential is 1600 psi. Certain operating conditions (e.g., low full-power vessel average temperature, high steam generator tube plugging levels, and reactor coolant system pressure controlled to 2250 psia) can result in the maximum primary-to-secondary pressure gradient to exceed the 1600 psi limit during normal transients. Calculations indicate that a minimum full-power steam pressure of 679 psia is necessary such that the maximum primary-to-secondary pressure gradient remains less than or equal to 1600 psi during normal transients for either Unit 1 or Unit 2. In order to provide additional conservatism relative to the design differential pressure limit, the minimum full-power steam pressure shall be restricted to 690 psia when reactor coolant system pressure is controlled to 2250 psia.

4.2.2.5 Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage mixed flow pump, which employs a controlled leakage seal assembly. The principal design parameters for the pumps are listed in Table 4.1-6. The reactor coolant pump estimated performance and NPSH characteristics are shown in Figure 4.2-8.

Reactor coolant is drawn up through the pump impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The rotor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pump in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, a second seal which directs the controlled leakage out of the pump, and a third seal which minimizes the leakage of water out of the pump. The second and third seals drain to the reactor coolant drain tank.

The shaft seal section consists of the number 1 controlled leakage, film riding face seal, a shutdown seal (SDS) assembly and the number 2 and number 3 rubbing face seals. The seals are contained within the main flange and seal housing. The SDS is housed within the number 1 seal area and is a passive device actuated by high seal flow temperature resulting from a loss of seal injection and component cooling water (CCW) cooling to the thermal barrier cooling coil.

In the event of a loss of seal injection and CCW flow to the thermal barrier heat exchanger, reactor coolant begins to travel along the RCP shaft and displaces the cooler seal injection water. Once the temperature within the number 1 seal reaches the actuation temperature range of the SDS, the SDS will activate to limit leakage from the RCS through the RCP seal package. The loss of reactor coolant through the RCP seal package is limited when the SDS polymer ring activates (clamps down) around the number 1 seal sleeve.

A portion of the high pressure water flow from the charging pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System around the thermal barrier cooling coil and through a labyrinth seal on the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount, which leaks through the second seal, is also collected and removed from the pump.

Component cooling water is supplied to the motor bearing oil coolers and the thermal barrier-cooling coil. Should the seal injection water flow be lost or interrupted, Reactor Coolant flows across the thermal barrier-cooling coil in the reverse direction and is cooled. It then becomes the source of water to the pump radial bearing and to the pump seals.

The squirrel cage induction motor driving the pump is air-to-water cooled, and has oil lubricated thrust and radial bearings. A water-lubricated bearing provides radial support for the pump shaft.

An oil collection system is provided for each reactor coolant pump motor to minimize the fire potential from spillage. The fire protection/suppression system is described in Section 9.8.1.

A flywheel on the shaft above the motor provides additional inertia to extend flow coastdown. An inadvertent, early actuation of the SDS on the pump shaft, with the shaft still rotating, will not adversely impact RCP coastdown. Each pump contains a ratchet mechanism to prevent reverse rotation. The reactor coolant pump flywheel is shown in Figure 4.2-6.

Precautionary measures, taken to preclude missile formation from primary coolant pump components, assure that the pumps will not produce missiles under any anticipated accident condition.

Components of the reactor coolant pump motor have been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing.

The most adverse operating condition for the flywheel is visualized to be the loss-of-load situation. The following conservative design-operation conditions precluded missile production by the flywheel. The wheels were fabricated from rolled, vacuum-degassed, ASTM A-533 steel plates. Flywheel blanks were flame-cut from the plate, with allowance for exclusion of flame-affected metal. A minimum of six Charpy tests were made from each plate, three parallel and three normal to the rolling direction. The tests determined that each blank satisfied the design requirements. An NDTT less than +10°F is specified. The finished flywheels were subjected to 100% volumetric ultrasonic inspection. The finished machined bores were also subjected to magnetic particle, or liquid penetrant examination.

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 4.2-7) less than 50% of the minimum specified material yield strength at room temperature (100 to 150°F).

Bursting speed of the flywheels has been calculated on the basis of Griffith-Irwin's results (References 2, and 3), to be 3900 rpm, more than three times the operating speed.

A fracture mechanics evaluation was completed for the reactor coolant pump flywheel.

To estimate the magnitude of fatigue crack growth during plant life, an initial radial crack length of 10% of the distance through the flywheel (from the keyway to the flywheel outer radius) was conservatively assumed. The analysis assumed 6000 cycles of pump starts and stops. The existing analysis is valid for the period of extended operation associated with license renewal.

The reactor coolant pump motor bearings are of conventional design, the radial bearings are the segmented pad type, and the thrust bearings are tilting pad Kingsbury bearings. All are oil lubricated; the lower radial bearing and the thrust bearings are submerged in oil, and the upper radial bearing is oil fed from an impeller integral with the thrust runner. Low oil levels would signal an alarm in the control room. Each motor bearing contains embedded temperature detectors; therefore initiation of failure, separate from loss of oil, would be indicated and alarmed in the control room as high bearing temperature. This would alert the operator to take corrective action. Even if the bearing proceeded to failure, the low melting point Babbitt metal on the pad surfaces would ensure that no sudden seizure of the bearing would occur. In this event the motor would continue to drive, as it has sufficient reserve capacity to operate, even under such conditions. However, it would draw excessive currents and at some stage would shut down because of the high current.

It may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft would fail in torsion just below the coupling to the motor. This would constitute a loss of coolant flow in the one loop; the effect of which is analyzed in Chapter 14. Following the seizure, the motor would continue to run without any overspeed, and the flywheel would maintain its integrity, as it would still be supported on a shaft with two bearings.

There are no other credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing would be precluded by shearing of the graphitar in the bearing. Any seizure in the seals would result in a shearing of the anti-rotation pin in the seal ring. An inadvertent actuation of the shutdown seal on the shaft will not interrupt core cooling flow provided by the RCP. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions would be initially by high temperature signals from the bearing water temperature detector, excessive No. 1 seal leakoff indications, and off-scale #1 seal leakoff indications, respectively. Following these signals, pump vibration levels would be checked. These would show excessive levels, indicating some mechanical trouble.

The design specifications for the reactor coolant pumps include, as a design condition the stresses induced by a design basis earthquake. Beside evaluating the externally produced loads on the nozzles and support lugs, an analysis was made of the effect of gyroscopic reactions on the flywheel, the bearings and in the shaft, due to rotational movements of the pump about a horizontal axis, during the maximum seismic disturbance.

The pump would continue to run unaffected by such conditions. In no case does any bearing stress in the pump or motor exceed or even approach a value, which the bearing could not carry.

The design requirements of the bearings are primarily aimed at ensuring a long life with negligible wear, so as to give accurate alignment and smooth operation over long periods of time. To this end, the surface bearing stresses are held at a very low value, and even under the most severe seismic transients or accidents, do not begin to approach loads which cannot be adequately carried for short periods of time.

Because there are no established criteria for short-time stress-related failures in such bearings, it is not possible to make a meaningful quantification of such parameters as margins to failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gives assurance of the adequacy of the bearing to operate without failure.

As is generally the case with machines of this size, the shaft dimensions are predicated on avoidance of shaft critical speed conditions, rather than actual levels of stress.

There are many machines as large as, and larger than these, that are designed to run at speeds in excess of first shaft critical. However, it is considered more desirable to operate below first critical speed, and the reactor coolant pumps are designed in accordance with this philosophy. This results in a shaft design, which, even under the most severe postulated transient, gives very low values of actual stress. While it would be possible to present quantitative data of imposed operational stress relative to maximum tolerable levels, if the mode of postulated failure were clearly defined, such figures would have little significance in a meaningful assessment of the adequacy of the shaft to maintain its integrity under operational transients. However, a qualitative assessment of such factors gives assurance of the conservative stress levels experienced during these transients.

In each of these cases, where the functional requirements of the component control its dimensions, it can be seen that if these requirements are met, the stress-related failure cases are more than adequately satisfied.

It is thus considered to be beyond the bounds of reasonable credibility that any bearing or shaft failure could occur that would endanger the integrity of the pump flywheel.

4.2.2.6 Reactor Coolant System Vents

The reactor coolant system is provided with a reactor vessel head vent and a pressurizer steam space vent to remove non-condensible gas or steam from the system. The vents are designed to pass a combined capacity equal to one-half of the reactor coolant system volume in one hour at system design pressure and temperature. They are designed to mitigate a possible condition of inadequate core cooling, inadequate natural circulation or inability to depressurize the system to permit initiation of the residual heat removal system as a result of a condition causing the accumulation of non-condensible gas or steam in the reactor coolant system. The reactor vessel head vent and the pressurizer steam space vent are designed as seismic Class I with two parallel, one inch nominal pipe size flow paths. Each path contains redundant safety grade, fail-closed solenoid valves. Orifices (1/4" reactor head vent path, 3/8" pressurizer steam space vent path) are installed upstream of the solenoid operated isolation valves to limit the maximum postulated flow, in the case of a pipe break down stream of the orifices, to less than the capacity of one centrifugal charging pump. Sealed-open, hand operated valves are installed upstream of the orifices. The solenoid valves in one flow path are powered independently from the valves in the second flow path and each valve has a separate control switch. All are normally closed and have stem position indicators to provide remote indication of valve position.

Downstream of the solenoid valves, RTDs are installed to detect leakage and provide an alarm. For the reactor vessel head vent, one 1/4" orifice is installed downstream of the solenoid operated isolation valves to limit piping and support loads during venting. Both vents discharge to the upper volume of the containment in an area, which will provide adequate dilution of any combustible gas.

4.2.2.7 Reactor Coolant Piping

The reactor coolant piping and fittings, which make up the loops, are austenitic stainless steel. The reactor coolant piping is made by a centrifugal casting process. All smaller piping which comprises part of the Reactor Coolant System boundary, such as the pressurizer surge line, spray and relief line, loop drains, and connecting lines to other systems is also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. The joints and connections in piping which comprise part of the reactor coolant boundary are welded, except for the pressurizer safety valves, where flanged joints are used. Thermal sleeves are installed at points in the system where high thermal stresses could develop due to rapid changes in fluid temperature during normal operational transients. These points include:

1. Charging connections from the Chemical and Volume Control System.
2. Return lines from the Residual Heat Removal Loop (also part of the Emergency Core Cooling System).
3. Both ends of the pressurizer surge line.
4. Pressurizer spray line connection to the pressurizer.

Thermal sleeves are not provided for the remaining injection connections of the ECCS since these connections are not in normal use.

Piping connections from auxiliary systems are made above the horizontal centerline of the reactor coolant piping, with the exception of:

1. Residual heat removal pump suction, which is 45o down from the horizontal centerline. This enables the water level in the reactor coolant system to be lower in the reactor coolant pipe while continuing to operate the residual heat removal system, should this be required for maintenance.
2. Loop drain lines and the connection for temporary level measurement of water in the reactor coolant system during refueling and maintenance operation.
3. The differential pressure taps for flow measurement are downstream of the steam generators on the 90o elbow.
4. RVLIS piping connections are located at the horizontal centerline of Loops 1 and 3. A connection for the Mid-Loop instrument piping is located 60º down from the horizontal centerline of Loop 2.

Penetrations into the coolant flow path are limited to the following:

1. The spray line inlet connections extend into the cold-leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force.
2. The reactor coolant sample system taps are inserted into the main stream to obtain a representative sample of the reactor coolant.
3. The wide range temperature detectors are located in RTD wells that extend into the reactor coolant pipes.
4. Three thermowell-mounted narrow-range RTDs extend into the hot leg scoops to provide a representative hot leg temperature.
5. A thermowell-mounted narrow-range RTD extends into each reactor coolant cold leg pipe.

Principal design data for the reactor coolant piping are given in Table 4.1-7.

Piping was restrained for postulated break conditions to prevent plastic hinge formation, except for certain breaks where no damage to Class I systems or the containment liner could result which would violate the criteria discussed in Section 4.2.4.

Numerous pipe whip restraints were designed for postulated ruptures occurring within the reactor coolant boundary to limit the consequences of the postulated ruptures. The pipe whip restraints were designed for circumferential ruptures in the reactor coolant system and connecting systems at changes in direction of the piping and nozzle junctions when consequential damage from these ruptures might occur. They were also provided to limit the consequences of longitudinal ruptures having a jet force equal to that of a circumferential rupture. Longitudinal splits were postulated to occur at selected points within the reactor coolant boundary.

4.2.2.8 Valves

All valves in the reactor coolant system which are in contact with the coolant are constructed primarily of stainless steel. Other materials in contact with the coolant are special materials such as hard surfacing and packing.

Hard surfacing performed on austenitic stainless steel pressure boundary parts is controlled to minimize severe sensitization of the stainless steel. Seat rings are utilized in valve design to preclude sensitization of the reactor coolant pressure boundary wall.

Indication of valve position for the pressurizer safety and power-operated relief valves is provided by a four channel acoustic flow monitor. There are four accelerometers; one strapped to the discharge of each of the three pressurizer safety valves and one on the common discharge of the three power relief valves. Flow through any of these valves produces an acoustic energy input to the respective accelerometer and this is amplified on the assigned channel of the monitor which is located in the control room. Indication on four vertical rows of light emitting diodes represents a bar graph display of relative flow through the monitored valves.

4.2.2.8.1 Pressurizer Safety Valves

The pressurizer safety valves are totally enclosed pop-type valves. The valves are spring-loaded, self-activated and with back-pressure compensation designed to prevent system pressure from exceeding the design pressure by more than 110 percent, in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The set pressure of the valves is 2485 psig.

The 6” pipes connecting the pressurizer nozzles to their respective safety valves are shaped in the form of a loop seal. Piping is connected to the bottom of each loop seal to drain any condensate that accumulates in the loop seal. An acoustic flow monitor and a temperature indicator on each valve discharge alerts the operator to the passage of steam due to leakage or valve lifting.

4.2.2.8.2 Power Relief Valves

The pressurizer is equipped with 3 power-operated relief valves, which limit system pressure for a large power mismatch and thus lessen the likelihood of an actuation of the fixed high-pressure reactor trip. The relief valves operate automatically or by remote manual control. The original design for 3 PORVs was to provide 100% load rejection capability. Since the load rejection capability has been reduced to 50%, the third PORV is now considered an installed spare. The 50% load rejection transient is an ANS Condition 1 event, also known as a Normal Operating Transient, or Plant Condition 1. ANSI/ANS Standard 51.1 does not consider the effects of single failure for Condition 1 events. The operation of these valves also limits the undesirable operation of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the power-operated relief valves. An acoustic flow monitor and a temperature indicator on the common discharge of the relief valves alerts the operator to the passage of steam due to leakage or valve opening. Indication of valve position is also provided by limit switches on each valve.

During startup and shutdown, a manually energized safeguard circuit is in service while the reactor coolant system temperature is below 266°F for Unit 1 and 299°F for Unit 2. This allows automatic opening of that Unit's two power relief valves at ≤435 psig for low temperature overpressure protection (LTOP) of the reactor vessel. This safeguard circuit ensures that the reactor pressure remains below the ASME Section III, Appendix G "Protection Against Nonductile Failure" limits in the case of an LTOP event.

The PORVs are spring-loaded-closed, air required to open valves. Normally this air is supplied by the plant control air source. To assure operability of the valves upon a loss of control air, a backup air supply is provided for two of the PORVs. The backup air supply consists of compressed air bottles. The backup air supply contains sufficient air for the required number of PORV strokes in a ten minute period during an LTOP event.

Design parameters for the pressurizer spray control, safety, and power relief valves are given in Table 4.1-8.

4.2.2.9 Reactor Coolant System Supports

1. Steam Generator Support

   Each steam generator is supported by a structural system consisting of four vertical support columns and upper and lower lateral restraints approximately 46½ feet apart. The vertical columns have a ball joint connection at each end to accommodate both the radial growth of the steam generator itself and the radial movement of the vessel from the reactor center.

   The lower lateral support consists of an inner frame, keyed and shimmed to the four steam generator support feet to accommodate radial growth of these feet. The inner frame is surrounded by an outer frame, which is embedded in both the primary shield and crane wall concrete. The connection between the inner and outer frame consists of a series of shimmed points, which act as both guides and limit stops to allow for expansion from the center of the reactor. The lower lateral support restrains both torsional and translational movements.

   The upper lateral support consists of a ring band, which is shimmed to the steam generator at twelve locations around the circumference. Attached to this band are lugs 180o apart which are shimmed and guided to a structural framing system which is embedded in the crane wall and steam generator enclosure wall concrete. Hydraulic snubbers are also connected 180O apart on the band and tied to other embedded frames in a direction coincident with the direction of movement away from the reactor center. The upper lateral support restrains rapid translational movements in all horizontal directions.

2. Reactor Vessel Supports

   The reactor vessel is supported by four of its eight nozzles by four individual weldments embedded in the primary shield concrete. Each nozzle pad bears on a shoe that is supported by a heavy U-shaped weldment, which wraps around the shoe. The U-shaped weldment is water-cooled at the junction of the outer flange and the web by two continuous welded angles on either side of the web. The U-shaped weldment bears vertically on two shims and is restrained horizontally by a series of shims and bearing plates. These bearing plates and shims are connected to an outer weldment, which completely surrounds the U-shaped weldment and is embedded in the concrete.

   The reactor support system allows the reactor to expand radially from its vertical centerline but resists rotational motion in all orthogonal planes. The nozzle horizontal centerlines translate in the vertical direction relative to the shoes.

3. Pressurizer Support

   The pressurizer is supported on a ring girder, which is in turn supported on a concrete slab. Horizontally, the vessel is restrained at two elevations approximately 27 feet apart.

   The lower restraint consists of anchor bolts in slightly oversize holes in the ring girder. The upper restraint consists of four individual weldments embedded in concrete that allow the pressurizer to expand radially, but resist torsional and translational horizontal movements.

4. Reactor Coolant Pump Support

   Each reactor coolant pump is supported vertically by three ball joint ended columns. This structural column system resists both overturning and vertical movement while allowing for expansion from the center of reactor. Excessive torsional and horizontal translational movements are resisted by a combination of lateral thrust columns anchored into the crane wall concrete.

4.2.3 Pressure-Relieving Devices

The Reactor Coolant System is protected against overpressure by control and protective circuits such as the high-pressure trip and by relief and safety valves connected to the top head of the pressurizer. The safety valves are currently analyzed for steam discharge only. The power operated relief valves are analyzed for steam or water discharge. However, evaluations have shown that the pressurizer will not become water solid before at least 10 minutes following a spurious Safety Injection or a feedline break. The relief and safety valves discharge into the pressurizer relief tank, which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 4.2-1A, and the valve design parameters are given in Table 4.1-8. The valves are further discussed in Sub-Section 4.2.2.8.

4.2.4 Protection against Proliferation of Dynamic Effects

The Reactor Coolant System is surrounded by concrete shield walls. These walls provide shielding to permit access into certain areas of the containment building during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the containment liner plate and all essential equipment inside the containment against blowdown jet forces and pipe whip to meet the missile protection criteria of Section 1.4.1 and the following:

1. A break of a steam or feedwater pipe inside the containment must not cause a break in a steam or feedwater pipe of another loop.
2. The leak tightness of the containment liner must not be damaged by a whip or blowdown jet force of a pipe which is part of the reactor coolant pressure boundary or which is necessary to function after a LOCA.

The concrete deck over the Reactor Coolant System also provides shielding and missile damage protection.

Reactor coolant pressure boundary equipment and piping are supported and provided with restraints to resist the actions of seismic, thermal expansion and pipe rupture effects.

4.2.5 Materials of Construction

The materials used in the Reactor Coolant System are selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1.

Reactor Coolant System materials which are exposed to the coolant are corrosion-resistant. They consist of stainless steel and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 4.2-2. Reactor coolant chemistry is further discussed in Section 4.2.8.

The secondary side water chemistry is controlled to minimize corrosion and sludge buildup in the steam generators. Plant procedures list the limits for containments in the steam generators. The levels of these contaminates are normally maintained well below the limits.

The phenomena of stress-corrosion cracking and corrosion fatigue are not generally encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, stress, and time.

It is a characteristic of stress corrosion that combinations of alloy and environment, which result in cracking, are usually quite specific. Environments, which have been shown to cause stress-corrosion cracking of stainless steels, are free alkalinity in the presence of chlorides, fluorides, and free oxygen. Experience has shown that deposition of chemicals on the surface of tubes can occur in a steam-blanketed area within a steam generator. In the presence of this environment under very specific conditions, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses, which resulted from normal manufacturing procedures. The steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel alloy has excellent resistance to general and pitting-type corrosion in severe operating water conditions. Extensive operating experience with Inconel units has confirmed this conclusion.

External insulation of Reactor Coolant system components is compatible with component materials. The cylindrical shell exterior, closure flanges and bottom head of the reactor vessel are insulated with stainless steel, metallic, reflective insulation. The closure head is insulated with stainless steel, metallic, reflective insulation. Other external corrosion-resistant surfaces in the Reactor Coolant System are insulated with low or halide-free insulating material as required.

The remaining material in the reactor vessel, and other Reactor Coolant System components, meets the appropriate design code requirements and specific component function.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility which ensures a low RT_{NDT} temperature, and thereby gives assurance that the finished vessels can be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. The stress limits established for the reactor vessel are dependent upon the temperatures at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the RT_{NDT} temperature.

The effects and methods of calculating the cumulative fast neutron (E > 1 MeV) exposure of the vessel wall material is described in Section 4.5.

To evaluate the RT_{NDT} temperature shift of welds, heat affected zones and base material for the vessel; test coupons of these material types have been included in the reactor vessel surveillance program described in Sub-Section 4.5.1.3.

The methods used to measure the initial RT_{NDT} temperature of the reactor vessel base plate material are given in Sub-Section 4.5.1.3.

4.2.6 Maximum Heating and Cooling Rates

The Reactor Coolant System operating cycle used for design purposes is given in Table 4.1-10 and described in Section 4.1.5. The normal system heating and cooling rate is 60°F/hr. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level, of 55°F/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant.

The fastest cooldown rates, which result from the hypothetical case of a break of a main steam line, are discussed in Chapter 14.

Surface thermocouples on each Steam Generator above the level of the tubesheet are provided to permit a direct measurement of Steam Generator temperature, to determine that no more than a 50°F difference exists with the Reactor Coolant System cold leg temperature prior to starting a reactor coolant pump in the inactive loop with the Reactor Coolant System in the water solid condition. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer, or if the SG/RCS Delta T is less than 50°F.

4.2.7 Leakage

The existence of leakage from the Reactor Coolant System to the lower containment compartment regardless of the source of leakage is detected by one or more of the following conditions:

a. Two radiation-sensitive instruments provide the capability for detection of leakage from the Reactor Coolant System. The containment air particulate monitor is quite sensitive to low leak rates and can be used to alarm the presence of new leaks, if desired. The containment gas monitor is much less sensitive but can be used as a backup to the air particulate monitor.

b. A third instrument used in leak detection is the humidity detector. This provides a backup means of measuring overall leakage from all water and steam systems within the containment but furnishes a less sensitive measure. The humidity monitoring method provides backup to the radiation monitoring methods.

c. An increase in the amount of coolant make-up water which is required to maintain normal level in the pressurizer, or an increase in containment sump level.

4.2.7.1 Leakage Prevention

Reactor Coolant System components are manufactured to exacting specifications which exceeds normal code requirements (as outlined in Section 4.1.6). In addition, because of the welded construction of the Reactor Coolant System and the extensive non-destructive testing to which it is subjected (as outlined in Sub-Chapter 4.5), it is considered that leakage through metal surfaces or welded joints is very unlikely.

However, some leakage from the Reactor Coolant System is permitted by the reactor coolant pump seals. Also all sealed joints are potential sources of leakage even though the most appropriate sealing device is selected in each case. Thus, because of the large number of joints and the difficulty of assuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable.

4.2.7.2 Locating Leaks

Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

Methods of leak location, which can be used during plant shutdown, include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the Reactor Coolant System in the leaking fluid and then left behind by the evaporation process. Portable sonic detectors sensitive to ultrasonic frequencies provide another means for locating small leaks.

4.2.7.3 Leak Detection Methods

a. Containment Air Particulate and Containment Radiogas Monitors

The containment air particulate monitor is the most sensitive instrument of those available for detection of reactor coolant leakage into the containment. This instrument is capable of detecting particulate radioactivity in concentrations as low as 10-9 μCi/cc of containment air.

The sensitivity of the air particulate monitor to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal base line leakage into the containment. The sensitivity is greatest where base line leakage is low, as has been demonstrated by the experience of Indian Point Unit No. 1, Yankee Rowe, and Dresden Unit 1. Where containment air particulate activity is below the threshold of detectability, operation of the monitor with stationary filter paper would increase leak sensitivity to a few cubic centimeters per minute. Assuming a low background of containment air particulate radioactivity, a reactor coolant corrosion product radioactivity (Fe, Mn, Co, Cr) of 0.2 μCi/cc (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking radioactive solids into the containment air, the air particulate monitor is capable of detecting leaks as small as approximately 0.0013 gal/min (5 cc/minute) within thirty minutes after they occur. If only ten percent of the particulate activity is actually dispersed in the air, the threshold of detectable leakage is raised to approximately 0.013 gpm (50 cc/minute).

For cases where base line reactor coolant leakage falls within the detectable limits of the air particulate monitor, the instrument can be adjusted to alarm on leakage increases from two to five times the base-line value.

The containment radiogas monitor is inherently less sensitive (threshold at

10-6μCi/cc) than the containment air particulate monitor, and would function only in the event that significant reactor coolant gaseous activity exists due to fuel cladding defects. Assuming a reactor coolant gas activity of 0.3 μCi/cc, the occurrence of a leak of two to four gpm would double the background (predominantly argon-41) in less than one hour. In these circumstances this instrument would be a useful backup to the air particulate monitor.

b. Humidity Detector

The humidity detection instrumentation offers another means of detection of leakage into the containment. This instrumentation has not nearly the sensitivity of the air particulate monitor, but has the advantage of being sensitive to vapor originating from all sources, the reactor coolant, the steam, and the feedwater systems. Plots of containment air dew point variations above a base-line maximum should be sensitive to incremental leakage equivalent to 0.2 to 1.0 gpm.

c. Liquid Inventory in the Process Systems and in the Containment Sump

An increase in the amount of coolant make-up water, which is required to maintain normal level in the pressurizer, will be indicated by an increase in charging flow or change in volume control tank level. Further details of the operation of the charging system is supplied in Chapter 9.

Gross leakage will be indicated by a rise in normal containment sump level and periodic operation of containment sump pumps. A run time meter is provided to monitor the frequency of operation and running time of each containment sump pump.

4.2.8 Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of Reactor Coolant System surfaces.

Reactor coolant water chemistry specifications are listed in Table 4.2-2. Periodic analysis of the coolant chemical composition are performed to verify that the reactor coolant water quality meets these specifications. Maintenance of the water quality to minimize chemical control is maintained with the Chemical and Volume Control System and Sampling System, which are described in Chapter 9.

4.2.9 Reactor Coolant Flow Measurements

Elbow taps are used in the Reactor Coolant System as an instrument device that indicates the status of the reactor coolant flow (Reference 4). The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation:

$$\frac{\Delta P}{\Delta P_o} = \left(\frac{\omega}{\omega_o}\right)^2,$$

where:

ΔP_0 is the referenced pressure differential with the corresponding referenced flow rate ω_0 ΔP is the pressure differential with the corresponding flow rate ω.

The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within ± 10% and field results have shown the repeatability of the trip point to be within ± 1%. The analysis of the loss of flow transient presented in Sub-Chapter 14.1 assumes instrumentation error of ± 3%.

4.2.9.1 Reactor Coolant Margin To Saturation

A digital subcooling monitor is provided to display in the control room either the temperature or pressure margin available for the sub-cooled operating condition below the corresponding saturation pressure or saturation temperature.

The device selects the highest temperature reading from 8 core exit thermocouples and 8 hot and cold leg RTD's, and the lower pressure reading from two RVLIS reactor coolant wide range pressure sensors, and then calculates the corresponding saturation conditions, and displays the available margin of subcooling below saturation, in either temperature (°F) or pressure (psi).

The Plant Process Computer (PPC) may also be used to display the margin of subcooling temperature (°F) on a trend recorder in the control room. The computer uses a calculated saturation temperature derived from the lowest valid value of the RC System wide range pressure inputs, the atmospheric pressure constant and the steam tables in conjunction with one of the following:

1. Hottest of the valid Hot and Cold Leg Wide Range RCS Temperature Inputs (RTD's);
2. Hottest of the valid Incore Thermocouples;
3. Average of the valid Hot and Cold Leg Wide Range RCS Temperature Inputs (RTD's);
4. Average of the valid Incore Thermocouples.

4.2.10 Loose Parts Detection

A loose parts monitoring system is used to detect loose metallic parts impacting within the reactor coolant system. Metallic debris may appear as a result of outage work or wear of system internals. Carried through the system, such debris may damage internal primary system components. The loose parts monitoring system provides early detection of loose metallic parts to minimize damage. The system was designed to meet the intent of Regulatory Guide 1.133, Rev. 1.

The system consists of redundant accelerometers at the reactor vessel and steam generators. A metallic impact will result in minute accelerations in the reactor coolant system component material, which will be detected by the accelerometers. The system alarms when impacts above a previously established threshold occur. Bypassing of alarms based on plant conditions is controlled by plant procedures.

4.2.11 Reactor Vessel Water Level

A Reactor Vessel Level Instrumentation System (RVLIS) is provided to indicate the relative vessel water level or the relative void content of fluid in the vessel during post-accident conditions. This level indication assists the operator in recognizing conditions, which may lead to high temperatures that could damage the vessel or its internals. Level indicators and recorders are located in the control rooms.

Sensors measuring the differential pressure between the vessel head and the bottom and between the head and the hot legs provide the basis for level indication. Because flow through the vessel affects differential pressure measurement, three level indication ranges are provided by separate sensors. One range monitors water level from the vessel bottom to the head during full flow conditions in the reactor vessel. The remaining two ranges monitor the entire vessel level and partial water level (top reactor head to hot leg) at zero forced flow conditions (no reactor coolant pump operating).

The differential pressure measurements are compensated for process effects using reactor coolant system pressure and temperature measurements. They are also compensated for environmental temperature effects on the RVLIS sensing lines using temperature measurements at representative sensing line locations.

4.2.11.1 References for Section 4.2

1. “Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems WCAP 7735 (Westinghouse Class 3), July 1971.”
2. Ernest L. Robinson, “Bursting Tests of Steam-Turbine Disk Wheels”, Transactions of the A.S.M.E., July 1944.
3. “Application of the Griffith-Irwin Theory of Crack Propagation to the Bursting Behavior of Disks, Including Analytical and Experimental Studies”, by D. H. Winne and B. M. Wundt, ASME, December 1, 1957.
4. J. W. Murdock, “Performance Characteristic of Elbow Flowmeters”, Transactions of the ASME, September, l964.