5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEM

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5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEM

5.1 SUMMARY DESCRIPTION

The reactor coolant system (RCS) shown on drawings D-175037, sheet 1, D-205037, sheet 1, D-175037, sheet 2, D-205037, sheet 2, D-175037, sheet 3, and D-205037, sheet 3, consist of similar heat transfer loops connected in parallel to the reactor pressure vessel. Each loop contains a reactor coolant pump, steam generator, and associated piping and valves. In addition, the system includes a pressurizer, a pressurizer relief tank, interconnecting piping, and instrumentation necessary for operational control. All of the above components are located in the containment building.

During operation, the reactor coolant system transfers the heat generated in the core to the steam generators, where steam is produced to drive the turbine generator. Borated, demineralized water is circulated in the reactor coolant system at a flowrate and temperature consistent with achieving the reactor core thermal hydraulic performance. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The reactor coolant system pressure boundary provides a barrier against the release of radioactivity generated within the reactor, and is designed to ensure a high degree of integrity throughout the life of the plant.

Reactor coolant system pressure is controlled by the pressurizer, where water and steam are maintained in equilibrium by electrical heaters and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations caused by contraction and expansion of the reactor coolant. Spring-loaded safety valves and power-operated relief valves are mounted on the pressurizer and discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

The extent of the reactor coolant system is defined as:

A. The reactor vessel, including control rod drive mechanism housings.

B. The reactor coolant side of the steam generators.

C. Reactor coolant pumps.

D. A pressurizer attached to one of the reactor coolant loops.

E. Safety and relief valves.

F. The interconnecting piping, valves, and fittings between the principal components listed above.

G. The piping, fittings, and valves leading to connecting auxiliary or support systems up-to-and-including the second isolation valve (from the high-pressure side) on each line.

Reactor Coolant System Components

A. Reactor Vessel

The reactor vessel is cylindrical, with a welded hemispherical bottom head and a removable, flanged, and gasketed hemispherical upper head. The vessel contains the core, core supporting structures, control rods, and other parts directly associated with the core.

The vessel has inlet and outlet nozzles located in a horizontal plane just below the reactor vessel flange, but above the top of the core. Coolant enters the vessel through the inlet nozzles and flows down the core barrel vessel wall annulus, turns at the bottom, and flows up through the core to the outlet nozzles.

B. Steam Generators

The steam generators are vertical shell and U-tube evaporators with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

C. Reactor Coolant Pumps

The reactor coolant pumps are identical, single-speed, centrifugal units driven by air-cooled, three-phase induction motors. The shaft is vertical with the motor mounted above the pumps. A flywheel on the shaft above the motor provides additional inertia to extend pump coastdown. The inlet is at the bottom of the pump; discharge is on the side.

D. Piping

The reactor coolant loop piping is specified in sizes consistent with system requirements.

The hot leg inside diameter is 29 in. and the cold leg return line to the reactor vessel is 27-1/2 in. The piping between the steam generator and the pump suction is increased to 31 in. in diameter to reduce pressure drop and improve flow conditions to the pump suction.

E. Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. Electrical heaters are installed through the bottom head of the vessel while the spray nozzle, relief, and safety valve connections are located in the top head of the vessel.

F. Safety and Relief Valves

The pressurizer safety valves are of the totally enclosed pop-type. The valves are spring-loaded and self-activated, with back-pressure compensation. The power-operated relief valves limit system pressure for large power mismatch. They are operated automatically or by remote manual control. Remotely operated valves are provided to isolate the inlet to the power-operated relief valves if excessive leakage occurs.

Reactor Coolant System Performance Characteristics

Tabulations of important design and performance characteristics of the reactor coolant system are provided in table 5.1-1.

Reactor Coolant Flow

The reactor coolant flow, a major parameter in the design of the system and its components, is established with a detailed design procedure supported by operating plant performance data, by pump model tests and analyses, and by pressure-drop tests and analyses of the reactor vessel and fuel assemblies. Data from all operating plants have indicated that the actual flow has been well above the flow specified for the thermal design of the plant. By applying the design procedure described below, it is possible to specify the expected operating flow with reasonable accuracy.

Three reactor coolant flowrates are identified for the various plant design considerations. The definitions of these flows are presented in the following paragraphs, and the applications of the definitions are illustrated by the system and pump hydraulic characteristics on figure 5.1-1.

Best Estimate Flow

The best estimate flow is the most likely value for the actual plant operating condition. This flow is based on the best estimate of the reactor vessel, steam generator and piping flow resistance, and on the best estimate of the reactor coolant pump head, with no uncertainties assigned to either the system flow resistance or the pump head. System pressure losses based on best estimate flow are presented in table 5.1-1. Although the best estimate flow is the most likely value to be expected in operation, more conservative flowrates are applied in the thermal and mechanical designs.

Thermal Design Flow

Thermal design flow is the basis for the reactor core thermal performance, the steam generator thermal performance, and the nominal plant parameters used throughout the design. To provide the required margin, the thermal design flow accounts for the uncertainties in reactor vessel, steam generator and piping flow resistances, reactor coolant pump head, and the methods used to measure flowrate. The combination of these uncertainties is equivalent to increasing the best estimate reactor coolant system flow resistance by approximately 15 percent. ***[HISTORICAL][****The intersection of this conservative flow resistance with the best estimate pump curve, as shown in figure 5.1-1, established the original/plant thermal design flow. This procedure provides a flow margin for thermal design of approximately 4 percent.**]*** For this plant, changes subsequent to the original specification of thermal design flow have resulted in additional margin. The thermal design flow is confirmed when the plant performs precision RCS flow measurements at the beginning of each cycle. Tabulations of important design parameters based on the thermal design flow are provided in table 5.1-1.

Mechanical Design Flow

Mechanical design flow is the conservatively high flow used in the mechanical design of the reactor vessel internals, fuel assemblies, and other system components. ***[HISTORICAL][****To ensure that a conservatively high flow is specified, the original plant mechanical design flow was set at least 4% higher than the original best estimate flow.* ***]*** The mechanical design flow is 101,800 gpm/loop, which is 5.8% above the current best estimate flow of 97,600 gpm/loop with 0% steam generator tube plugging and thimble plugs removed after best estimate flow is adjusted to account for measured RCS flow. This best estimate flow is based on Unit 2, since it yields the minimum margin to mechanical design flow.

Pump overspeed, because of a turbine generator overspeed of 20 percent, results in a peak reactor coolant flow of 120 percent of the mechanical design flow. The overspeed condition is applicable only to operating conditions when the reactor and turbine generator are at power.

Interrelated Performance and Safety Functions

The interrelated performance and safety functions of the reactor coolant system and its major components are listed below:

A. The reactor coolant system provides sufficient heat transfer capability to transfer the heat produced during power operation and when the reactor is subcritical, including the initial phase of plant cooldown, to the steam and power conversion system.

B. The system provides sufficient heat transfer capability to transfer the heat produced during the subsequent phase of plant cooldown and cold shutdown to the residual heat removal (RHR) system.

C. The system heat removal capability under power operation and normal operational transients, including the transition from forced to natural circulation, will ensure no fuel damage within the operating bounds permitted by the reactor control and protection systems.

D. The reactor coolant system provides the water used as the core neutron moderator and reflector and as a solvent for chemical shim control.

E. The system maintains the homogeneity of soluble neutron poison concentration and rate of change of coolant temperature so that uncontrolled reactivity changes do not occur.

F. The reactor vessel is an integral part of the reactor coolant system pressure boundary and is capable of accommodating the temperatures and pressures associated with the operational transients. The reactor vessel functions to support the reactor core and control rod drive mechanisms (CRDM).

G. The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, reactor coolant volume changes are accommodated in the pressurizer via the surge line.

H. The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generators.

I. The steam generators provide high-quality steam to the turbine. The tube and tube sheet boundary are designed to prevent the transfer of activity generated within the core to the secondary system.

J. The reactor coolant system piping serves as a boundary for containing the coolant under operating temperature and pressure conditions and for limiting leakage (and activity release) to the containment atmosphere. The reactor coolant system piping contains demineralized, borated water, which is circulated at the flowrate and temperature consistent with achieving the reactor core thermal and hydraulic performance.

Interlocks on critical motor-operated valves are discussed in subsection 7.6.2 and paragraph 6.3.2.15.

5.1.1 SCHEMATIC FLOW DIAGRAM

The reactor coolant system is shown on drawings D-175037, sheet 1, D-205037, sheet 1, D-175037, sheet 2, D-205037, sheet 2, D-175037, sheet 3, and D-205037, sheet 3, and principal pressures, temperatures, flowrates, and coolant volume data under normal steady-state, full-power operating conditions are provided in table 5.1-1.

5.1.1.1 System Operation

Brief descriptions of normal, anticipated system operations are provided below. These descriptions cover plant startup, power generation, hot shutdown, cold shutdown and refueling.

5.1.1.1.1 Plant Startup

Plant startup encompasses the operations which bring the reactor plant from cold shutdown to no-load power operating temperature and pressure. Before plant startup, the reactor coolant loops and pressurizer are filled completely, by the use of the charging pumps, with water containing the cold shutdown concentration of boron. The loops are vented using either the Reactor Coolant Vacuum Refill System (RCVRS) or the dynamic venting process. The secondary side of the steam generator is filled to normal startup level with water which meets the steam plant water chemistry requirements.

If the RCVRS is used, air is removed from the RCS by a skid-mounted vacuum pump system. The RCVRS is connected to the RCS via a special connection to the pressurizer relief tank (PRT) inlet line. The RCS evacuation path includes the pressurizer surge line (while at midloop conditions), the reactor vessel head vent paths, and the pressurizer spray line (once the surge line is submerged). Transportation of the air from the hot legs to the cold legs occurs through the air gap between the internal and external hot leg reactor vessel nozzles and the core bypass flow nozzles.

Initial conditions are as follows: the RCS level is at midloop and the PRT level is below the sparging header. The vacuum pump skid suction hose is connected to the PRT inlet line connection. The RHR flow is adjusted to prevent vortexing and to ensure adequate NPSH. The air evacuation path is established by opening the reactor vessel head vent valves, the pressurizer spray valves, the PORV block valves and the PORVs.

Prior to starting the air evacuation via the RCVRS, letdown flow and charging flow are adjusted to maintain a constant VCT level with RCP seal injection in service. The RCVRS is then used to pull the air from the RCS via the connection to the PRT inlet line. The RCS is filled via one charging path while maintaining the vacuum in the RCS. Once the RCS is filled to a pressurizer level approximately equal to the steam generator tube elevation, the RCS vacuum is broken. Charging is continued until a level increase is detected in the PRT. Finally, the PORVs, pressurizer spray valves and reactor vessel head vent valves are closed. This completes the RCS filling and venting operation.

If the RCVRS is not used, the RCS is pressurized, by use of the low pressure control valve and one centrifugal charging pump, to obtain the required pressure drop across the number one seal of the reactor coolant pumps. The pumps may then be operated intermittently to assist in venting operations.

During operation of the reactor coolant pumps, one charging pump and the low pressure letdown path from the residual heat removal loop to the chemical and volume control system (CVCS) are used to maintain the reactor coolant system pressure in an appropriate range. Plant operating experience and instrument inaccuracy are used to establish a pressure range which ensures that all RCP support conditions are met and that the LTOP relief valves are not challenged during RCP start, the ensuing transient, and any subsequent operation. The fracture prevention temperature limitations of the reactor vessel impose an upper limit of approximately 450 psig. The charging pump supplies seal-injection water for the reactor coolant pump shaft seals. A nitrogen atmosphere and normal operating temperature, pressure, and water level are established in the pressurizer relief tank.

Upon completion of venting, the reactor coolant system is pressurized, the reactor coolant pumps are started, and the pressurizer heaters are energized to begin heating the reactor coolant. When the cold leg temperature reaches between 175-180°F and the pressurizer temperature increases to the saturation temperature corresponding to a saturation pressure of about 375 psig, a steam bubble is formed in the pressurizer while the reactor coolant pressure is maintained in an appropriate range. Plant operating experience and instrument inaccuracy are used to establish a pressure range which ensures that all RCP support conditions are met and that the LTOP relief valves are not challenged during RCP start, the ensuing transient, and any subsequent operation. The pressurizer liquid level is reduced until the no-load power level volume is established. During the initial heatup phase, hydrazine is added to the reactor coolant to scavenge the oxygen in the system. The heatup is not taken beyond 250°F until the oxygen level has been reduced to the specified level.

An alternative to water-solid operation to establish RCS pressure for RCP operation is the use of a pressurizer steam bubble. In this case, the RCVRS is used to remove most of the system air. Hydrazine is then added to the pressurizer via auxiliary spray to remove dissolved oxygen from the pressurizer liquid. The pressurizer heaters are actuated to establish a steam bubble to pressurize the RCS and RHR flow is reduced or bypassed to allow the RCS to heat up to 150-160°F. The combination of RCS letdown flow diversion to the recycle holdup tanks and RHR flow adjustment is used to maintain a constant pressurizer level as the RCS expands. When the pressurizer pressure reaches the appropriate range, the RCPs are started to remove the small volume of air trapped in the top of the steam generator tubes. Plant operating experience and instrument inaccuracy are used to establish a pressure range which ensures that all RCP support conditions are met and that the LTOP relief valves are not challenged during RCP start, the ensuing transient, and any subsequent operation.

The VCT is then burped as required to reduce the oxygen in the gas space. Additional hydrazine is then added by the normal charging flow path to reduce the RCS dissolved oxygen concentration within Technical Requirements Manual limits before the RCS is allowed to heat up above 250°F.

The reactor coolant pumps and pressurizer heaters are used to raise the reactor coolant temperature and pressure to normal operating levels.

As the reactor coolant temperature increases, the pressurizer heaters are manually controlled to maintain adequate suction pressure for the reactor coolant pumps. When the normal operating pressure of 2235 psig is reached, pressurizer heat and spray controls are transferred from manual to automatic control.

5.1.1.1.2 Power Generation and Hot Shutdown

Power generation includes steady-state operation, ramp changes not exceeding the rate of 5 percent of full power per minute, step changes of 10 percent of full power (not exceeding full power), and step load changes with steam dump not exceeding the design step load decrease.

During power generation, reactor coolant system pressure is maintained by the pressurizer controller at-or-near 2235 psig, while the pressurizer liquid level is controlled by the charging letdown flow control of the chemical and volume control system.

When the reactor power level is less than 15 percent, the reactor power is controlled manually. At power above 15 percent, the reactor control system controls automatically maintain an average coolant temperature, consistent with the power relationships, by control rod movement.

During the hot shutdown operations, when the reactor is subcritical, the reactor coolant system temperature is maintained by steam dump to the main condenser. This is accomplished by a controller in the steam line, operating in the pressure control mode, which is set to maintain the steam generator steam pressure. Residual heat from the core or operation of a reactor coolant pump provides heat to overcome reactor coolant system heat losses.

5.1.1.1.3 Plant Shutdown

Plant shutdown is the operation which brings the reactor plant from no-load power operating temperature and pressure to cold shutdown. Concentrated boric acid solution from the chemical and volume control system is added, as necessary, to the reactor coolant system to increase the reactor coolant boron concentration to ensure adequate shutdown margin is maintained as required by plant Technical Specifications. If the reactor coolant system is to be opened during the shutdown, the hydrogen and fission gas in the reactor coolant is reduced by degassing the coolant in the volume control tank.

Plant shutdown is accomplished in two phases; the first is by the combined use of the reactor coolant system and steam systems, and the second is by the residual heat removal system. During the first phase of shutdown, residual core and reactor coolant heat is transferred to the steam system via the steam generator. Steam from the steam generator is dumped to the main condenser. At least one reactor coolant pump is kept running to assure uniform reactor coolant system cooldown. The pressurizer heaters are de-energized and spray flow is manually controlled to cool the pressurizer while maintaining the required reactor coolant pump suction pressure.

When the reactor coolant temperature is below approximately 350°F and the pressure is in the range of 400 to 450 psig, the second phase of shutdown commences with the operation of the residual heat removal system.

When the reactor coolant temperature is below 200°F, the pressurizer steam bubble is collapsed. One reactor coolant pump (either of those in a loop containing a pressurizer spray line) remains in service as the coolant temperature approaches 160°F. One or more RCPs may remain in service after the steam bubble is collapsed to facilitate mixing of the RCS. Pressurizer cooldown is continued by initiating auxiliary spray flow from the chemical and volume control system. Plant shutdown continues until the reactor coolant temperature is 140°F or less.

5.1.1.1.4 Refueling

Before removing the reactor vessel head for refueling, the system temperature has been reduced to 140°F or less and hydrogen and fission product levels are reduced. A clear plastic tube is attached to one of the reactor coolant loops to indicate when the water has been drained below the reactor vessel head vent. Draining continues until the water level is below the reactor vessel flange. The vessel head is then raised. Upon completion of refueling, the system is refilled for plant startup.

5.1.2 PIPING AND INSTRUMENTATION DIAGRAM

A piping and instrumentation diagram of the reactor coolant system is shown on drawings D-175037, sheet 1, D-205037, sheet 1, D-175037, sheet 2, D-205037, sheet 2, D-175037, sheet 3, and D-205037, sheet 3. The diagrams show the extent of the systems located within the containment, and the points of separation between the reactor coolant system and the secondary (heat utilization) system. The isolation provided between the reactor coolant pressure boundary and connected systems is discussed in subsection 6.2.4.

5.1.3 ELEVATION DRAWING

Figures 1.2-6 and 1.2-7 are plant general arrangements which show the elevations and relative locations of the major components in the reactor coolant loop.

TABLE 5.1-1 (SHEET 1 OF 2)

SYSTEM DESIGN AND OPERATING PARAMETERS[(a)]

Plant design life (years)	40[(b)]
Nominal operating pressure (psig)	2235
Total system volume including pressurizer and surge line (ft^3)	9591
System liquid volume, including pressurizer water at maximum guaranteed power (ft^3)	8939
NSSS power (Btu/h)	9694×10^6

System Thermal and Hydraulic Data
Based on Thermal Design Flow:

	High Temp. Operating Conditions[(c)]	Low Temp. Operating Conditions[(c)]
Thermal design flow (gal/min/loop)	86,000	86,000
Total coolant flowrate (lb/h)	98.3×10^6	98.3×10^6
Reactor vessel coolant temperature at full power:		
Inlet (°F)	540.5	529.9
Outlet (°F)	614	604.5
Coolant temperature rise in vessel at full power (avg.)(°F)	73.5	74.6
Steam generator outlet temp. (°F)	540.8	530.3
Pressurizer spray rate, (max.)(gal/min)	600	600
Pressurizer heater capacity (kW)	1400	1400
Pressurizer relief tank volume (ft^3)	1300	1300
Steam press. at full power (psia)[(d)]	733	664
Steam flow at full power (lb/h)(total)	12.52×10^6	12.49×10^6
Steam generator steam temperature (°F)	508.2	497.2
Feedwater inlet temperature (°F)	446.0	446.0

Flows and Pressure Drops
Based on Best Estimate Flow:

	Unit 1/Unit 2[(d)]	Unit 1/Unit 2[(f)]
Best estimate flow (gal/min/loop)	93,800/93,200	98,400/97,600
Pump head (ft)	267/263	243/240
Reactor vessel ΔP (psi)	38.3/37.7	40.2/39.6
Steam generator ΔP (psi)	41.7/41.1	32.0/31.6
Piping ΔP, (psi)	7.7/7.5	8.5/8.4

TABLE 5.1-1 (SHEET 2 OF 2)

SYSTEM DESIGN AND OPERATING PARAMETERS[a]

---

a. Parameters are based on an average steam generator tube plugging level of 15% and are bounding for 0% tube plugging conditions.

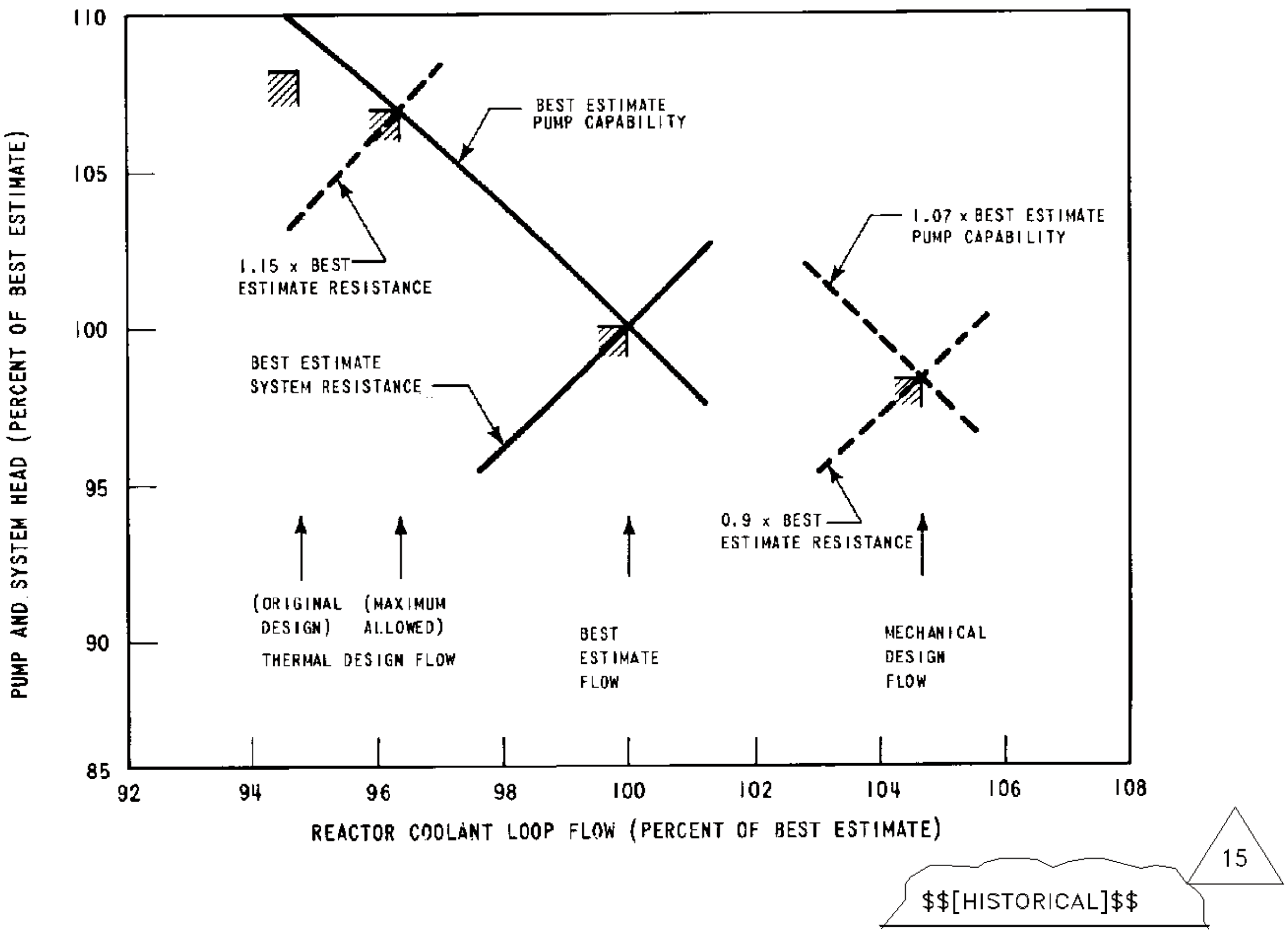
b. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 18).

c. Parameters for high temperature and low temperature operating conditions are based on T_{avg} of 577.2°F and 567.2°F, respectively.

d. Steam pressure at exit of steam generator outlet nozzle containing integral flow restrictor.

e. Minimum best estimate flow and pressure drops are based on high reactor vessel average temperature, 20% steam generator tube plugging, and thimble plugs installed.

f. Maximum best estimate flow and pressure drops are based on low reactor vessel average temperature, 0% steam generator tube plugging, and thimble plugs removed.



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NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PUMP HEAD – FLOW CHARACTERISTICS

FIGURE 5.1-1]

5.2 INTEGRITY OF THE REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant system (RCS) boundary for Westinghouse (W) pressurized-water reactors (PWR) is defined as stated in American Nuclear Society document N18.2 "Nuclear Safety Criteria for the Design of Stationary Pressurized-Water Reactor Plants", January 1972, paragraph 5.4.3.2. This definition of the RCS boundary is consistent with the definition of the reactor coolant pressure boundary (RCPB) as defined in 10 CFR 50.2, part V, as applied to codes and standards required by 10 CFR 50.55a (with appropriate footnotes), and ASME Section XI requirements for inservice inspection.

5.2.1 DESIGN OF REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

The RCS boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. The system is protected from overpressure by means of pressure-relieving devices as required by applicable codes. Materials of construction are specified to minimize corrosion and erosion and to provide a structural system boundary throughout the life of the plant. Fracture prevention measures are taken to prevent brittle fracture. Inspections in accordance with applicable codes and provisions are made for surveillance of critical areas to enable periodic assessment of the boundary integrity, as described in subsection 5.2.8.

5.2.1.1 Performance Objectives

The performance objectives of the RCS for normal operation are described in section 5.1. The performance objectives for upset and faulted conditions are given in subsection 5.2.1 above. No transient is classified as an emergency condition.

Equipment code and classification lists for the components within the reactor coolant system boundary are given in table 3.2-1.

The RCS, in conjunction with the reactor control and protection systems, is designed to maintain the reactor coolant at conditions of temperature, pressure, and flow adequate to protect the core from damage. The design requirement for safety is to prevent conditions of high power, high reactor-coolant temperature, or low reactor-coolant pressure or combinations of these which could result in a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit.

The RCS is designed to provide controlled changes in the boric acid concentration and the reactor coolant temperature. The reactor coolant is the core moderator, reflector, and solvent for the chemical shim. As a result, changes in coolant temperature or boric acid concentration affect the reactivity level in the core.

The following design bases have been selected to ensure that the uniform RCS boron concentration and temperature will be maintained:

A. Coolant flow is provided by either a reactor coolant pump or a residual heat removal (RHR) pump to ensure uniform mixing whenever the boron concentration is decreased.

B. The design arrangement of the RCS eliminates deadended sections and other areas of low coolant flow in which nonhomogeneities in coolant temperature or boron concentration could develop.

C. The RCS is designed to operate within the operating parameters, particularly the coolant temperature change limitations.

5.2.1.2 Design Parameters

The design pressure for the RCS is 2485 psig, except for the pressurizer relief line from the safety valve to the pressurizer relief tank, which is 600 psig, and the pressurizer relief tank, which is 100 psig. For components with design pressures of 2485 psig, the normal operating pressure is 2235 psig. The design temperature for the RCS is 650°F, except for the pressurizer and its surge line, which are designed for 680°F, and the pressurizer relief line from the safety valve to the pressurizer relief tank, which is designed for 600°F. The seismic loads for Farley Nuclear Plant (FNP) are given in section 3.7.

Reactor coolant system and component test pressures are discussed in paragraph 5.2.1.5.

5.2.1.3 Compliance With 10 CFR 50.55a

The components of the RCPB are designed and fabricated in accordance with the rules of 10 CFR 50, Section 50.55a, Codes and Standards, except as noted in table 5.2-1. This table lists the components, the code to which the components were designed and fabricated, the code required by Section 50.55a based on the August 1972 construction permit date, and the differences between the code requirements as designed and fabricated and as required by Section 50.55a.

All of the exceptions listed result from the issuance of a construction permit being delayed beyond June 30, 1972, because of the extensive period for environmental review of the Farley project after the safety evaluation was essentially completed. Total time from filing the construction permit application was 34 months, which was substantially beyond the period anticipated at the times that the components were purchased. Efforts were made to upgrade the components beyond the codes listed in the Preliminary Safety Analysis Report (PSAR) in order to comply with Section 50.55a; those areas in which the efforts were not completely successful are listed in table 5.2-1.

5.2.1.4 Applicable Code Cases

The ASME Code case interpretations that may have been applied to the components of the RCS boundary are tabulated in table 3.2-5.

5.2.1.5 Design Transients

The following five ASME operating conditions are considered in the design of the RCS.

A. Normal Conditions

Any condition in the course of startup, operation in the design power range, and hot standby and system shutdown other than upset, emergency, faulted, or testing condition.

B. Upset Conditions

Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients resulting from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients because of loss-of-load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition is included in the design specifications.

C. Emergency Conditions

Those deviations from normal conditions that require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence, but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events will not cause more than 25 stress cycles having an S_A value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code Section III.

D. Faulted Conditions

Those combinations of conditions associated with extremely low-probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

E. Testing Conditions

Testing conditions are those tests in addition to the hydrostatic or pneumatic tests permitted by the ASME Code Section III, including leak tests or subsequent hydrostatic tests.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The transients selected are representative of operating conditions which prudently might be considered to occur during plant operation and are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of transients that, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant. As required by Technical Specifications administrative controls, the components identified in table 5.2-2a are designed and shall be maintained within the cyclic or transient limits of table 5.2-2a. The Fatigue Monitoring Program, as described in chapter 18, subsection 18.3.2, will be used to monitor plant transients that are significant contributors to the fatigue cumulative usage factor to ensure the design limit on fatigue usage is not exceeded during the period of extended operation.

The following five transients are considered normal conditions:

A. Heatup and Cooldown

For design evaluation, the heatup and cooldown cases are represented by continuous heatup or cooldown at a rate of 100°F/h. The heatup occurs from ambient to the no-load temperature and pressure condition and the cooldown represents the reverse situation.

In actual practice, the rate of temperature change of 100°F/h will not usually be attained because of other limitations such as:

1. Criteria for prevention of nonductile failure, which establish maximum permissible temperature rates of change as a function of plant pressure and temperature.

2. Slower initial heatup rates when using pumping energy only.

3. Interruptions in the heatup and cooldown cycles because of such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry, and gas adjustments.

B. Unit Loading and Unloading

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5% min between 15% load and full load. This load swing is the maximum possible consistent with operation with automatic reactor control. The reactor coolant temperature varies with load as prescribed by the temperature control system.

C. Step Increase and Decrease of Ten Percent

The $\pm$ 10-percent step change in load demand is a control transient assumed to be a change in turbine control valve opening that might be occasioned by disturbances in the electrical network into which the plant output is tied. The reactor control system is designed to restore plant equilibrium without reactor trip, following a $\pm$ 10-percent step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15-percent and 100-percent full load, the power range for automatic reactor control. In effect, during load change conditions, the reactor control system attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint, at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step-load decrease in turbine load, the secondary-side steam pressure and temperature initially increase, since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature is ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine generator load as determined by first stage turbine pressure measurement. The pressurizer pressure also decreases from its peak pressure value and follows the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash, which reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step-load increase in turbine load, the reverse situation occurs; i.e., the secondary-side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature is raised to a value above its initial equilibrium value at the beginning of the transient.

D. Large Step Decrease in Load

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary-side steam pressure and temperature automatically initiates a secondary-side steam dump system that prevents a reactor shutdown or lifting of steam generator safety valves. Thus, when a plant is designed to accept a step decrease of 95 percent from full power, it signifies that a steam dump system provides a heat sink to accept 85 percent of the turbine load. The remaining 10 percent of the total step change is assumed by the rod control system. If a steam dump system were not provided to cope with this transient, there would be such a large mismatch between what the turbine is demanding and what the reactor is furnishing that a reactor trip and lifting of steam generator safety valves would occur.

Although Farley has been designed for a 50-percent step change, the transient for the 95-percent step-load decrease is considered since it represents a more severe condition than the lower percentages.

E. Steady-State Fluctuations

The reactor coolant average temperature, for purposes of design, is assumed to increase or decrease a maximum of 6°F in 1 min. The temperature changes are assumed to be around the programmed value of T_{avg}, ($T_{avg} \pm 3$°F). The corresponding reactor coolant average pressure is assumed to vary accordingly.

The following six transients are considered upset conditions:

A. Loss of Load Without Immediate Turbine or Reactor Trip

This transient applies to a step decrease in turbine load from full power occasioned by the loss of turbine load without immediately initiating a reactor trip and represents the most severe transient on the RCS. The reactor and turbine eventually trip as a consequence of a high pressurizer level trip initiated by the reactor trip system. Since redundant means of tripping the reactor are provided as a part of the reactor protection system, transients of this nature are not expected, but are included to ensure a conservative design.

B. Loss of Power

This transient applies to a blackout situation involving the loss of outside electrical power to the station with a reactor and turbine trip. Under these circumstances, the reactor coolant pumps are deenergized and, following the coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which, at this time, are receiving feedwater from the auxiliary feed system operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

C. Loss of Flow

This transient applies to a partial loss-of-flow accident from full power in which a reactor coolant pump is tripped out of service as a result of a loss of power to the pump. The consequences of such an accident are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal results in reactor coolant at cold leg temperature being passed through the steam generator and cooled still further. This cooler water then passes through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

D. Reactor Trip From Full Power

A reactor trip from full power may occur for a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator.

This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of secondary steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the reactor trip system causes the control rods to move into the core.

E. Inadvertent Pressurizer Auxiliary Spray Initiation

The inadvertent pressurizer auxiliary spray transient will occur if the auxiliary spray valve is opened inadvertently during normal operation of the plant. This will introduce cold water into the pressurizer with a very sharp pressure decrease as a result.

The temperature of the auxiliary spray water is dependent upon the performance of the regenerative heat exchanger. The most conservative case is that in which the letdown stream is shut off and the charging fluid enters the pressurizer unheated. Therefore, for design purposes, the temperature of the spray water is assumed to be 100°F. The spray flowrate is assumed to be 200 gal/min.

The pressure decreases rapidly to the low pressure reactor trip point. At this pressure, the pressurizer low pressure reactor trip is assumed to be actuated. This accentuates the pressure decrease until the pressure is finally limited to the hot leg saturation pressure. At 5 min the spray is stopped and all the pressurizer heaters return the pressure to 2250 psia. This transient is more severe on a two-loop plant than on a three-loop plant; e.g., a bigger and more rapid pressure decrease. Therefore, the transient for a two-loop plant is used as design basis for the FNP.

For design purposes it is assumed that no temperature changes in the RCS will occur as a result of initiation of auxiliary spray except in the pressurizer.

F. Operating Basis Earthquake (OBE)

The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transient loads resulting from fluid pressure and temperature. Their magnitude however, is considered in the design analysis for comparison with appropriate stress limits.

The following four transients are considered faulted conditions:

A. Reactor Coolant System Boundary Pipe Break

This accident involves the postulated rupture of a pipe belonging to the RCS boundary. It is conservatively assumed that the system pressure is reduced rapidly and the emergency core cooling system (ECCS) is initiated to introduce water into the RCS. The safety injection signal also will initiate a turbine and reactor trip.

The criteria for locating design basis pipe ruptures used in the design of the supports and restraints of the RCS in order to assure continued integrity of vital components and engineered safety systems is given in section 3.6.

Analyses reported in reference 1 and service experiences show that the criteria given in section 3.6 offer a practical equivalent to ensure the same degree of protection to public health and safety as postulating both longitudinal and circumferential breaks at any location. Westinghouse nuclear steam supply system (NSSS) piping and support components are designed to these criteria.

Protection criteria against dynamic effects associated with pipe breaks are covered in section 3.6.

B. Steam Line Break

For component evaluation, the following conservative conditions are considered:

1. The reactor is initially in hot, zero-power subcritical condition, assuming all rods in, except the most reactive rod, which is assumed to be stuck in its fully withdrawn position.

2. A steam line break occurs inside the containment resulting in a reactor and turbine trip.

3. After the break the reactor coolant temperature cools down to 212°F.

4. The ECCS pumps restore the reactor coolant pressure.

The above conditions result in the most severe temperature and pressure variations which the component will encounter during a steam-break accident.

The dynamic reaction forces associated with circumferential steam line breaks will be considered in the design of supports and restraints in order to assure continued integrity of vital components and engineered safety features. Protection criteria against dynamic effects associated with pipe breaks are covered in section 3.6.

C. Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and RCS pressure. Reactor trip occurs because of a safety injection signal on low pressurizer pressure. The planned procedure for recovery from this accident calls for isolation of the steam line leading from the affected steam generator (reference section 15.4). Therefore, this accident results in a transient which is no more severe than that associated with a reactor trip.

D. Safe Shutdown Earthquake (SSE)

The mechanical stress transient resulting from the safe shutdown earthquake (SSE) is considered on a component basis.

The above design conditions are given in the Equipment Specifications which are written in accordance with the ASME Code.

The design transients and the number of cycles of each that are normally used for fatigue evaluations are shown in table 5.2-2. In accordance with the ASME Boiler and Pressure Vessel Code, faulted conditions are not included in fatigue evaluations.

Prior to plant startup the following tests are carried out:

A. Turbine Roll Test

This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature and the steam generated is used to perform a turbine roll test. However, the plant cooldown during this test exceeds the 100°F/h maximum rate.

B. Hydrostatic Test Conditions

The pressure tests are outlined below:

1. Primary-Side Hydrostatic Test Before Initial Startup

The pressure tests covered by this section include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydro test is performed, prior to initial fuel loading, at a water temperature which is compatible with reactor vessel fracture prevention criteria requirements and a maximum test pressure of 3107 psig, or 1.25

times the design pressure. In this test, the primary side of the steam generator is pressurized to 3107 psig coincident with no pressurization of the secondary side. To hydrostatically test the RCS, a separate hydro-test pump is provided.

2. Secondary-Side Hydrostatic Test Before Initial Startup

   The secondary side of the steam generator is pressurized to 1360 psia, or 1.25 times the design pressure of the secondary side coincident with the primary side at 0 psig.

3. Primary-Side Leak Test

   After each time the primary system has been opened, a leak test is performed. For design purposes, the primary system pressure is assumed to be raised to 2500 psia during the test, with the system temperature above design transition temperature, while the system is checked for leaks. In actual practice, the primary system will be pressurized to < 2500 psia to prevent the pressurizer safety valves from lifting during the leak test.

   During this leak test, the secondary side of the steam generator will be pressurized so that the pressure differential across the tubesheet does not exceed 1600 psi. This is accomplished by closing off the steam lines.

Since the tests outlined under items 1 and 2 occur prior to plant startup, the number of cycles is independent of plant life.

The design loading combinations and the associated stress and deformation limits are provided in tables 5.2-3 through 5.2-7.

5.2.1.6 Identification of Active Pumps and Valves

ASME Code Class 1 active and inactive valves in the RCPB, as defined in 10 CFR 50.2, are reflected in table 5.2-8. Active valves are those in the pressure boundary whose operability through a mechanical motion is relied on to perform a safety function (as well as reactor shutdown function) during the transients or events considered in each operating condition category. Pressure boundary valves which have no required motion and must only retain their structural integrity, are not classified as active valves.

There are no active pumps in the RCPB. The reactor coolant pumps, which are the only pumps within the RCS boundary, are classified as inactive.

Every valve and pump is hydrostatically tested by the manufacturer to ASME Boiler and Pressure Vessel Code requirements to ensure the integrity of the pressure boundary parts. This test is followed by a seat leak test to MSS-SP-61 criteria to ensure that no gross deformation is caused by the hydrostatic test.

The design methods and procedures used to show that active valves listed in table 5.2-8 will operate during a faulted condition are described in section 3.9. The control and instrumentation are discussed in chapter 7.0.

5.2.1.7 Design of Active Valves

Valves required to open or close during or following any specified plant design transient condition have been designed in accordance with various codes and procedures that have been widely used by the nuclear industry. These codes and procedures are based on engineering judgment, inservice performance, and fundamental principles of engineering mechanics rather than the requirements of a detailed stress analysis. This basis has resulted in conservative designs which, in conjunction with periodic inspections, ensure that these components will function as required.

5.2.1.8 Inadvertent Operation of Valves

Those remotely-operated valves that are used in the isolation of the RCPB during normal plant operation, and are not relied on to function after an accident, are redundant. The inadvertent operation of one of these redundant valves, excluding the pressurizer power operated relief valve, does not increase the severity of any transient. Should the pressurizer power-operated relief valve inadvertently open, operator action is required to close the pressurizer power-operated relief valve block valve to ensure that the severity of any transient is not increased.

5.2.1.9 Stress and Pressure Limits

The loading combinations and associated stress or deformation limits for inactive components are provided in tables 5.2-3 through 5.2-7.

Allowable stress limits for active Code Class 1 valves are provided in paragraph 3.9.4.1. There are no active Code Class 1 pumps within the RCPB.

5.2.1.10 Stress Analysis for Structural Adequacy

The design evaluation of the RCS, including the types of analyses that are performed to ensure the performance and the structural adequacy of the RCS, is provided below in paragraph 5.2.1.10.1.

5.2.1.10.1 Design Evaluation

The RCS provides for heat transfer from the reactor to the steam generators under conditions of forced circulation flow and natural circulation flow. The heat transfer capabilities of the RCS are analyzed in chapter 15 for various transients.

During the second phase of plant cooldown and during cold shutdown and refueling, the heat exchangers of the RHR system are employed. Their capability is discussed in section 5.5.

The pumps of the RCS ensure heat transfer by forced circulation flow. Design flowrates are discussed in conjunction with the reactor coolant pump description in section 5.5.

Initial RCS tests are performed to determine the total delivery capability of the reactor coolant pumps. Thus, it is confirmed prior to plant operation that adequate circulation is provided by the RCS.

To ensure a heat sink for the reactor under conditions of natural circulation flow, the steam generators are at a higher elevation than the reactor. In the design of the steam generators, consideration is given to provide adequate tube area to ensure that the RHR rate is achieved with natural circulation flow.

Whenever the boron concentration of the RCS is reduced, plant operation will be such that good mixing is provided in order to ensure that the boron concentration is maintained uniformly throughout the RCS.

Although mixing in the pressurizer will not be achieved to the same degree, the fraction of the total RCS volume which is in the pressurizer is small. Thus, the pressurizer liquid volume is of no concern with respect to its effect on boron concentration.

Also, the design of the RCS is such that the distribution of flow around the system is not subject to the degree of variation which would be required to produce nonhomogeneities in coolant temperature or boron concentration as a result of areas of low coolant flow rate. An exception to this is the pressurizer, but for the same reasons as discussed above, it is of no concern. Operation with one reactor coolant pump inoperable is possible under certain conditions and, in this case, there would be backflow in the associated loop even though the pump itself is prevented from rotating backwards by its antirotation device. The backflow through the loop would cause departure from the normal temperature distribution around the loop, but would maintain the boron concentration in the loop the same as that in the remainder of the RCS.

The range of coolant temperature variation during normal operation is limited and the associated reactivity change is well within the capability of the rod control group movement.

For design evaluation, the heatup and cooldown transients are analyzed by using a rate of temperature change equal to 100°F/h. Over certain temperature ranges, fracture prevention criteria will impose a lower limit to heatup and cooldown rates.

Concentrated boric acid solution from the chemical and volume control system is added, as necessary, to the reactor coolant system to increase the reactor coolant boron concentration to ensure adequate shutdown margin is maintained as required by plant Technical Specifications.

Therefore, it is concluded that the temperature changes imposed on the RCS during its normal modes of operation do not cause any abnormal or unacceptable reactivity changes.

The design cycles as discussed in the preceding section are conservatively estimated for equipment design purposes and are not intended to be an accurate representation of actual transients or, for all cases, to reflect operating experience.

Certain design transients, with an associated pressure and temperature curve, have been chosen and assigned an estimated number of design cycles for the purpose of equipment design. These curves represent an envelope of pressure and temperature transients on the RCS boundary with margin in the number of design cycles chosen based on operating experience.

To illustrate this approach, the reactor trip transient can be mentioned. Four hundred design cycles are considered in this transient. One cycle of this transient would represent any operational occurrence which would result in a reactor trip. Thus, the reactor trip transient represents an envelope design approach to various operational occurrences.

This approach provides a basis for fatigue evaluation to ensure the necessary high degree of integrity for the RCS components.

System hydraulic and thermal design parameters are used as the basis for the analysis of equipment, coolant piping, and equipment support structures for normal and upset loading conditions. The analysis is performed using a static model to predict deformation and stresses in the system. Results of the analysis give six generalized force components, three bending moments and three forces. These moments and forces are resolved into stresses in the pipe in accordance with the applicable codes. Stresses in the structural supports are determined by the material and section properties assuming linear elastic small deformation theory.

In addition to the loads imposed on the system under normal and upset conditions, the design of mechanical equipment and equipment supports requires that consideration also be given to abnormal loading conditions, such as seismic and pipe rupture.

Analysis of the RCLs and support systems for seismic loads is based on a three-dimensional, multimass elastic dynamic model with nonlinear bumper and tie-rod supports. This model is coupled to a simplified reactor containment building model (reference 32). The seismic model is then subjected to time-history seismic OBE and SSE excitation (reference 33) with all the SG snubbers removed. The piping, equipment nozzle, and equipment support loads from this seismic analysis are obtained and evaluated.

The dynamic analysis employs the displacement method, lumped parameter, and stiffness matrix formulations and assumes that all components behave in a linearly elastic manner. The reduced modal analysis method and modal superposition method are used in the time-history seismic analyses. Seismic analyses are covered in detail in section 3.7.

Analysis of the RCLs and support systems for blowdown loads resulting from a loss-of-coolant accident (LOCA) is based on the time-history response of simultaneously applied blowdown forcing functions on a single broken and unbroken loop dynamic model. The forcing functions are defined at points in the system loop where changes in cross-section or direction of flow occur such that differential loads are generated during the blowdown transient. Stresses and loads are checked and compared to the corresponding allowable values.

The stresses in components resulting from normal sustained loads and the blowdown analysis are combined with the seismic analysis to determine the maximum stress for the combined loading case. This is considered a very conservative method since it is highly improbable that both maxima will occur at the same instant. These stresses are combined to determine that the RCLs and support system will not lose its intended functions under this highly improbable situation.

Protection criteria against dynamic effects associated with pipe breaks are covered in section 3.6.

For fatigue evaluations, in accordance with the ASME Boiler and Pressure Vessel Code, maximum stress intensity ranges are derived from combining the normal and upset condition transients given in paragraph 5.2.1.5. Note that there are no emergency conditions designated. The stress ranges and number of occurrences are then used in conjunction with the fatigue curves in the ASME Boiler and Pressure Vessel Code to get the associated cumulative usage factors.

The criterion presented in the ASME Boiler and Pressure Vessel Code is used for the fatigue failure analysis. The cumulative usage factor is < 1.0 and hence, the fatigue design is adequate. Metal fatigue, including the effect of environmentally assisted fatigue, was evaluated for license renewal as a TLAA in accordance with 10 CFR 54.21. The results for the period of extended operation are summarized in chapter 18, subsection 18.4.2.

The reactor vessel vendor's stress report is reviewed and approved by Westinghouse Electric Corporation. The stress report includes a summary of the stress analysis for regions of discontinuity analyzed in the vessel, a discussion of the results (including a comparison with the corresponding code limits), a statement of the assumptions used in the analyses, descriptions of the methods of analysis and computer programs used, a presentation of the actual calculations used, a listing of the input and output of the computer programs used, and a tabulation of the references cited in the report. The contents of this stress report and other Class I component stress reports are in accordance with the requirements of the ASME Boiler and Pressure Vessel Code. These stress reports are available inhouse for review.
The Westinghouse analysis of the steam generator tube-tubesheet complex is included as part of the stress report requirement for ASME Code Class 1 Nuclear Pressure Vessels. The evaluation is based on the stress and fatigue limitations outlined in ASME Section III.

The stress analysis techniques utilized include all factors considered appropriate to conservative determination of the stress levels used in evaluation of the tube-tubesheet complex. The analysis of the tubesheet complex includes the effect of all appurtenances attached to the perforated region of the tubesheet that are considered appropriate for conservative analysis of the stresses for evaluation on the basis of the ASME Code Section III stress limitations. The evaluation involves the heat conduction and stress analysis of the tubesheet, channel head, secondary shell structure for particular steady design conditions for which code stress limitations are to be satisfied, and for discrete points during transient operation for which the temperature/pressure conditions must be known to evaluate maximum and minimum stresses for fatigue life usage. In addition, limit analyses are performed to determine tubesheet capability to sustain faulted conditions for which elastic analysis does not suffice. The analytic techniques utilized are computerized and significant stress problems are verified experimentally to justify the techniques when possible.

The major concern in fatigue evaluation of the tube weld is the fatigue strength reduction factor to be assigned to the weld root notch. For this reason, Westinghouse has conducted low cycle fatigue tests of tube material samples to determine the fatigue strength reduction factor, and has applied them to the analytic interaction analysis results in accordance with the accepted techniques in the Boiler and Pressure Vessel Code for experimental stress analysis.

The steam generator tube-tubesheet complex integrity is verified by analysis for most adverse conditions resulting from a rupture of either primary or secondary piping.

It has been established that for such accident conditions, where a primary-to-secondary-side differential pressure exists, the primary membrane stresses in the tubesheet ligaments, averaged across the ligament and through the tubesheet thickness, satisfy the conditions given in table 5.2-3 for this faulted event. Also, for such accident conditions, the primary membrane stress plus primary bending stress in the tubesheet ligaments, averaged across the ligament width at the tubesheet surface location giving maximum stress, must not exceed the faulted condition criteria. In the case of a primary pressure loss accident, the secondary primary pressure differential is somewhat higher than the primary secondary design pressure differential. However, rigorous analysis shows that no stresses in excess of those covered by the ASME Boiler and Pressure Vessel Code for faulted conditions are experienced by the tubesheet for this accident.

The tubes have been designed to the requirements of the ASME Boiler and Pressure Vessel Code assuming 2485 psig as the design pressure differential. Hence, neither a primary nor a secondary pressure-loss accident impose stresses beyond those normally expected and considered as normal operation by the Code. ASME Section VIII design curves for iron-chromium-nickel steel cylinders under external pressure indicate a collapse pressure of 2310 psi for tubes having the minimum properties required by ASTM specifications. This indicates a minimum factor of safety of 2.4 against collapse. Collapse tests of 7/8-in. diameter, 0.050-mil-wall straight tubes at room temperature indicate actual tube strengths are significantly higher than specification and a collapse pressure of 6000 psi was recorded for the straight tube. The code charts indicate a collapse pressure of 2740 psi for this tube. The difference is attributed to the fact that the yield strength of the tube tested was 44,000 psi and the code charts are based on a yield strength of approximately 29,000 psi at room temperature.

Consideration has been given to the superimposed effects of secondary-side pressure loss and the safe shutdown earthquake loading. For the case of the tubesheet, the safe shutdown earthquake loading will contribute an equivalent static pressure loading over the tubesheet of < 10 psi (for vertical shock). Such an increase is small when compared to the pressure differentials (up to 2485 psig) for which the tubesheet is designed and does not result in stresses exceeding the allowable stresses. The fluid dynamic forces on the internals under secondary steam-break accident conditions indicate, in the most severe case, that the tubes are adequate to constrain the motion of the baffle plates with some plastic deformation, while boundary integrity is maintained.

A complete tube-tubesheet complex analysis is also performed to verify structural integrity for a primary pressure loss accident plus the safe shutdown earthquake.

Although the ASME Boiler and Pressure Vessel Code provides for rules and techniques in analysis of perforated plates, it should be noted that the stress intensity levels for perforated plate are given for triangular perforation arrays. Westinghouse tubesheets contain square hole arrays. Hence, Westinghouse utilizes its own data and that obtained from Pressure Vessel Research Committee research in square array perforation patterns for development of similar charts for stress intensity factors and elastic constants. The resulting stress intensity levels and fatigue stress ranges are evaluated according to the stress limitation of the code.

The vessels, piping, valves, pumps, and associated supports of the RCPB are designated ANS Safety Class 1.

Portions of small diameter piping, tubing, valves, fittings, and support elements connected to the RCS pressurizer are designated ASME Boiler and Pressure Vessel Code, Section III, Code Class 2. Westinghouse Nuclear Safety Advisory Letter (NSAL) 07-09 Revision 1, “Safety Classification of Small Lines Connected to the Pressurizer Steam Space,” states that small lines and their associated component items connected to the pressurizer steam space were originally misclassified as ASME Boiler and Pressure Vessel Code Class 2 and met the requirements for being classified as Code Class 1. In response, Farley Nuclear Plant concurred with the assessment on a plant-specific basis and submitted an Inservice Inspection (ISI) alternative requesting the NRC authorization to allow the affected component items to remain Code Class 2 as originally designed and constructed in lieu of upgrading the affected component items to Code Class 1 based on the increased burden not corresponding to the commensurate increase in level of safety. The alternative was authorized for the fourth ISI interval. Later correspondence with the NRC identified this issue as a construction code issue and no further alternatives for later intervals are required.

Loading combination and allowable stresses for ASME Section III, Class 1 components, piping and supports are given in tables 5.2-3 through 5.2-7.

Valves in sample lines are not considered to be part of the RCS boundary, i.e., not ANS Safety Class 1. This is because the nozzles where these lines connect to the RCS are orificed to a 3/8-in. hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging.

5.2.1.10.1.1 Analytical Methods for Supports and Loop Analysis. The load combinations that are considered in the design of structural steel members of component supports are given in paragraph 5.2.1.5. The design is described in paragraph 5.5.14.2.

A. Deadweight

The deadweight loading imposed by the piping on the supports is defined to consist of the dry weight of the coolant piping and the weight of the water contained in piping during normal operation. In addition, the total weight of the primary equipment components, including water, forms a deadweight loading on the individual component supports.

B. Thermal Expansion

The free vertical thermal growth of the reactor vessel nozzle centerlines is considered to be an external anchor movement transmitted to the RCL. The weight of the water in the steam generator and reactor coolant pump is applied as an external force in the thermal analysis to account for equipment nozzle displacement as an external movement.
The cold and hot moduli of elasticity, the coefficient of thermal expansion at the metal temperature, external movements transmitted to the piping as described above, and the temperature rise above the ambient temperature define the required input data to perform the flexibility analysis for thermal expansion.

C. Earthquake Loads

The intensity and character of the earthquake motion that produces forced vibration of the equipment mounted within the containment building are specified in terms of the ground acceleration time history. The ground acceleration time history for earthquake motions is given in reference 33.

D. Pressure

The steady-state hydraulic forces based on the system initial pressure are applied as external loads to the RCL model for determination of the RCL/support system deflections and support forces.

E. Pipe Rupture Loads

Blowdown loads are developed in the broken and unbroken RCLs as a result of the transient flow, pressure fluctuations following a postulated LOCA in one of the RCL accumulator or RHR branch nozzles. The postulated LOCA is assumed to have 1-ms opening time to simulate the instantaneous occurrence.

F. Analytical Methods

The static and dynamic structural analyses assume linear elastic behavior and employ the displacement (stiffness) matrix method and the normal mode theory for lumped parameter, multimass structural representation to formulate the solution. The complexity of the physical system to be analyzed requires the use of a computer for solution. Herein lies the need for accurate and adequate representation of the physical system by means of an idealized (mathematical) model.

The loadings on the component supports are obtained from the analysis of an integrated RCL support system dynamic structural model as shown on figure 5.2-5.

G. Reactor Coolant Loop Model

The RCL model is constructed for the WESTDYN, WECAN, and the PS + CAEPIPE computer programs. These are special purpose programs designed for the static and dynamic analysis of redundant piping systems with arbitrary leads and boundary conditions. The RCL lumped mass model represents an ordered set of data that numerically describes the physical system to the WESTDYN, WECAN, and the PS + CAEPIPE programs. The node point coordinates and the incremental lengths of the elements are calculated. The lumping of distributed mass of a segment or elbow is accomplished by locating the total mass at the mass center of gravity.

A valid representation of the effect of the equipment motion on the RCL piping and its support system is ensured by modeling the mass and stiffness characteristics of the equipment in the overall RCL model. Since the reactor pressure vessel is very massive and relatively rigid, for LOCA analysis, it is represented by a fixed boundary condition for the RCL model. The requirement in the time-history dynamic analysis, that the external hydraulic forcing functions be applied at only mass points, influences the construction of the steam generator and reactor coolant pump model described below. A simplified reactor pressure vessel model is incorporated into the time-history seismic analysis.

The steam generator is represented by a multimass, lumped model. The lower mass position is located at approximately the intersection of the inlet and outlet nozzles of the steam generator. The other masses are located at various locations on the steam generator.

The reactor coolant pump is represented by a two-mass, lumped model. The lower mass position is located at the intersection of the pump suction and discharge nozzles. The upper mass position is located at the center of gravity of the pump motor.

H. Support Structure Models

The equipment support structure models are dual purpose since they are required to quantitatively represent (in terms of 6-x-6 stiffness matrix) the elastic restraints which the supports impose upon the loop; and to evaluate the individual support member stresses caused by the forces imposed upon the supports by the loop.

Models for the STRUDL[2] computer program are constructed for the steam generator lower, steam generator upper lateral, and reactor coolant pump lower support structures. The structure geometry and member properties are obtained from the certified construction structural drawings.

I. <u>Hydraulic Models</u>

The hydraulic model is constructed to quantitatively represent the behavior of the coolant fluid within the RCLs in terms of the concentrated time-dependent loads it imposes upon the loops.

In evaluating the hydraulic forcing functions during a LOCA, the pressure and the momentum flux terms are dominant. Inertia and the gravitational terms are neglected; however, they are taken into account to evaluate the local fluid conditions.

Thrust forces resulting from a LOCA are calculated in two steps using two digital computer codes. The first code, MULTIFLEX, calculates transient pressure, flowrates, and other coolant properties as a function of time. The second code, THRUST, uses the results obtained from the first code and calculates time history of forces at locations where there is a change in either direction or area of flow within the RCL. These locations for the broken loop are shown in figure 5.2-6.

In MULTIFLEX blowdown analysis, both the broken and the unbroken loops are represented. The NRC approved MULTIFLEX 1.0 computer code (reference 3) is used to generate the transient coolant properties throughout the RCS. The MULTIFLEX code calculates the thermal-hydraulic transient within the RCS and considers subcooled, transition, and two-phase (saturated) blowdown regimes. The code employs the method of characteristics to solve the conservation laws, assuming one-dimensional flow and a homogeneous liquid and vapor mixture. The RCS is divided into subregions in which each subregion is regarded as an equivalent pipe. A complex network of these equivalent pipes is used to represent the entire primary RCS.

A coupled fluid-structure interaction is incorporated into the MULTIFLEX code by accounting for the deflection of the constraining boundaries, which are represented by separate spring-mass oscillator systems. For steam generator and other RCS component analyses, MULTIFLEX provides pressure and other coolant property transients at select locations. For loop piping analyses, the time-history RCS properties as computed by MULTIFLEX are used as input to the THRUST code to calculate the LOCA hydraulic forces at various locations along the RCS piping for the broken and unbroken loops. In the THRUST calculation of blowdown forces, the RCS is represented by the same model employed in the MULTIFLEX code. Twenty-six node points are selected along the geometric model of the RCL where the vector forces and their coordinate components are calculated.

The force components at each aperture are vectorially summed to obtain the total force components in global coordinate system at the nodes. These forces are stored on electronic media and, after proper coordinate transformation, applied as external loadings on the RCL dynamic model.

J. Static Load Solutions

The static solutions for deadweight, thermal expansion, and pressure load conditions are obtained by using the WESTDYN computer program. The computer input consists of the RCL mode, stiffness matrices representing various supports for static behavior, and the appropriate load condition. Coordinate transformations for rotation from the local or support coordinate system to the global system are applied to the stiffness matrices prior to their input.

K. Time-History Dynamic Solution for Seismic Loading

The reduced modal analysis method and modal superposition method are used in the time-history seismic analyses. The reduced modal analysis is used to determine the natural frequencies and mode shapes for a linear, undamped structure. This analysis requires the specification of dynamic or active degrees of freedom (DOF) for the model, which are a subset of the total number of DOF. The selection of dynamic DOF must be such that the low frequency spectrum can accurately be represented while a reduced eigenvalue problem is solved. In other words, the selected dynamic (or active) DOF should be able to describe the frequency modes of interest.

The modal superposition method gives a time-history solution for the response of an arbitrary structure subjected to known nodal forces or ground acceleration time histories. The structure may include linear and nonlinear elements. The uncoupled modal equations are integrated analytically.

The input to the time-history seismic analysis is in the form of time-history seismic motions applied individually for all three components at the base of the soil springs in the north-south, east-west, and vertical directions. These time-history seismic motions were provided in reference 33. The total response is obtained by determining the maximum response from absolutely combining each of the two horizontal responses with the vertical seismic response.

L. Time-History Dynamic Solution for LOCA Loading

The initial displacement configuration of the mass points is defined by applying the initial steady-state hydraulic forces to the unbroken RCL model. For this calculation, the support stiffness matrices for the static behavior are incorporated into the RCL model. For dynamic solution, the unbroken RCL model is modified to simulate the physical severance of the pipe caused by the postulated LOCA under consideration. This model includes definition of the support stiffness matrices for dynamic behavior. The natural frequencies and normal modes for the modified RCL dynamic model are determined. After proper coordinate transformation to the RCL global coordinate system, the hydraulic forcing functions to be applied at each lumped mass point are stored on magnetic tape for later use as input to the FIXFM program. FIXFM is a part of program WESTDYN.

The initial displacement conditions, natural frequencies, normal modes, and the time-history hydraulic forcing functions from the input to the FIXFM program which calculates the dynamic time-history displacement response for the dynamic degrees of freedom in the RCL model. The displacement response is plotted at all mass points. The displacement response at support points is reviewed to validate the use of the chosen support stiffness matrices for dynamic behavior. The time-history displacement response from the valid solution is saved on electronic media for later use to compute the support loads and to analyze the RCL piping stresses.

M. Evaluation of Support Structures

The support loads are computed by multiplying the support stiffness matrix, and the displacement vector, at the support point. The support loads are saved on magnetic tape for use in support member evaluation.

The STRUDL computer program is used to obtain support stiffness matrices and member influence coefficients for the equipment supports. Unit forces along and unit moments about each coordinate axis are applied to the models at the equipment vertical centerline joints. Stiffness analysis is performed for each unit load for each model. Printed output includes all six components of displacement at the joint at which loads are applied and six force components at each end of each member in the support system.

Joint displacements for applied unit loads are formulated into flexibility matrices. These are inverted to obtain support stiffness matrices which are included in the RCL model.

Loads acting on the supports obtained from the RCL analysis (including time-history LOCA forces), support structure member properties, and influence coefficients at each end of each member, are input into the THESSE program.

This program accomplishes the following for each support case used:

1. Combines the various types of support plane loads to obtain operating condition loads (normal, upset, or faulted).

2. Multiplies member influence coefficients by operating condition loads to obtain all member internal forces and moments. The 6-x-6 force arrays are printed for each end of each member. Diagonal terms in the array are the maximum (or minimum) values of each internal member force component and the other terms are the corresponding values of all other components. In addition, all member force components are printed along with the time of occurrence for the LOCA time-history loading producing the highest stresses in each member. This output gives a complete tabulation of all worst force and stress conditions in each member in the supporting system. It also provides maximum loads on the supporting concrete.

3. Solves appropriate stress or interaction equations for the specified operating condition. Maximum normal stress, shear stress, and combined load interaction equation values are printed as a ratio of maximum actual values divided by limiting values. The time of occurrence of the maximum value of each equation is also printed for the faulted condition, which includes time-history LOCA forces. Stress and interaction equations are used with limits specified for the operating conditions considered.

The stresses that were calculated are given in tables 5.2-28 and 5.2-31 as a percentage of the allowable. The member number refers to the identification used in the computer code. The members are identified by general classifications, such as "lower bumpers" for the steam generator supports. Those members which have no stress entries such as steam generator upper support members for the normal condition see no load in that plant condition. The largest percentage of allowable for any member of the steam generator, reactor coolant pump, and pressurizer supports for the normal condition is 34 percent; for the upset conditions, 44 percent; and for the faulted conditions of the SEE, combined with LOCA, 92 percent.

The reactor vessel supports were analyzed using a detailed finite element model. The maximum horizontal and maximum vertical loads were simultaneously applied to the support model, and the corresponding stresses were determined. For the reactor vessel support box (figure 3M-2), the percentage of allowable stress for the normal condition is 41 percent; for the upset condition, 47 percent; and 38 percent for the faulted condition.

The reactor vessel support shoe (figure 3M-1) is stressed to 37 percent of the allowable stress for the upset condition and 48 percent of the allowable stress for the faulted condition.

N. LOCA Evaluation of the Control Rod Drive Mechanisms

The response of the control rod drive mechanisms (CRDM) to the postulated reactor vessel inlet nozzle and outlet nozzle limited displacement breaks has been evaluated. The time-history analysis of the mechanism has been performed for the vessel motion developed previously. A one-row model of the CRDMs was formulated with gaps at the upper CRDM support modeled as nonlinear elements. The CRDMs were represented by beam elements with lumped masses. The translation and rotation of the vessel head were applied to this model (see figure 5.2-22). The resulting loads and stresses were compared to allowables to verify the adequacy of the system. The highest loads occur at the head adapter, the location where the mechanisms penetrate the vessel head. The bending moments at this location are presented in table 5.2-32 for the longest and shortest CRDM. The combined effect, including seismic loads, is shown to be less than the allowable bending moment at this location.

The heat transfer capability of the steam generators is sufficient to transfer to the steam and power conversion system the heat generated during normal operation, and during the initial phase of plant cooldown under natural circulation conditions.

5.2.1.11 Analysis Method for Faulted Condition

When the components and systems for the Farley units were being designed, only general design requirements existed for faulted conditions. There were no specific stress limits or associated methods of analysis established for faulted conditions. To provide a conservative basis for the analysis of Class 1 components, the collapse curves given in the PSAR were developed. The criterion represented by the collapse curves has evolved into the criteria of table 5.2-6 of the FSAR. The methods and criteria in table 5.2-6 should thus be reviewed with respect to the criterion agreed to in the PSAR, rather than with the more recently derived methods and limits established in the nonmandatory Appendix F of the ASME Code, Section III. These methods of analysis, in conjunction with the faulted condition stress limits, ensure that the general design requirements of the NRC for faulted conditions will be met and the plant can thus be safely shut down under accident conditions.

For the RCL and components, the elastic system analysis option of table 5.2-6 was used. Elastic component analyses were used on all components except those discussed below.

Inelastic component analysis was used for the reactor coolant pump support feet. The pump casing with the pump support feet is shown on figure 5.2-20. The pump foot was analyzed for a set of umbrella loads which are greater than the loads expected in any plant. The umbrella loads are calculated for the faulted condition and each of the maxima of the six load components, F_x, F_y, ..., M_z, are assumed to occur simultaneously. For example, the maximum F is chosen by surveying many past plants, and this is applied simultaneously with the maximum F_x, F_y, ..., M_z, all determined similarly. The actual plant loads are calculated and compared to the umbrella loads. Conformance indicates adequacy of the component for the specific plant application. If conformance is not demonstrated, an individual plant analysis would be performed. Table 5.2-26 indicates the relationship between the Farley specific plant loads for three different faulted conditions (from three different break locations) and the umbrella loads for which the pump foot was designed. The actual plant loads are, in themselves, also conservative since the maximum for each of the six load components is determined and assumed to act concurrently with the others. For the LOCA condition, the dynamic time-history analyses show that the maximum values of the six load components do not act concurrently. The seismic event, although evaluated by response spectra analysis, is also dynamic and the load component maximums at the foot clearly will not coincide. Note from table 5.2-26 that the umbrella loads are greater than these actual plant loads by a factor ranging from 1.0 to 20.4. From the preceding discussion, the conservatisms in the actual plant loads and the adequacy of the umbrella loads are therefore demonstrated.

The entire casing foot was analyzed by means of a 3-dimensional stress analysis. The foot model utilized symmetry about the bolt hold radial centerline (figure 5.2-21). The completed model contains 1584 node points and 1518 3-dimensional solid elements with 4088 active degrees of freedom in the model. The 3-dimensional finite elements are a mixture of rectangular prisms, triangular prisms, and tetrahedrons. The vertical side and horizontal plate sections have a minimum of four elements through the thickness. The model therefore yields bending stresses as well as direct stresses through the thickness. The higher stress regions have a finer model mesh consisting of smaller tetrahedron and triangular prism elements.

The ANSYS computer code[(11)] plastic analysis options were employed. The plasticity program is based upon incremental strain equations with the Prandtl-Reuss flow rule[(12)]. The virgin stress-strain option was used to incur the true stress-true strain material curve. To yield the required accuracy, loading increments were computed to keep the size of the plastic strain increments near the size of the material yield strain. The smaller load steps keep the solution process from diverging from the input stress-strain curve.

The resulting faulted condition plastic analysis stress intensity was compared with the faulted condition criteria of 0.7 S_{ut} = 59, 950 psi for 304 SS at 600°F. This is the limit for the primary membrane plus tending stress intensities as given in table 5.2-6. Since the foot is similar to a beam-type structure, the average stress across the section is very low. The primary tending stresses therefore control. The true ultimate stress, S_{ut} is determined from the engineering ultimate stress (the engineering stress at the point of maximum load) by assuming constancy of volume. Using this assumption, the true ultimate stress (S_{ut}) is given by:

$$S_{ut} = S_u(1 + \varepsilon)$$

Where ε is the engineering strain corresponding to the point of maximum load.

The stresses in the pump foot-to-casing attachment zone and weld-filled region were not controlling. The maximum stress in the foot occurred in the horizontal plate member near the vertical to horizontal plate intersection and in line with the bolt. Since the faulted allowables are based upon primary stresses and not peak stresses, the stress components in the high stress region were linearized through the plate thickness. The resulting maximum stress intensity of the section was found from these linearized maximum principal stresses. The stress intensity was

$$(\sigma_I)_{max} = 59{,}614 \text{ psi}$$

which was less than the inelastic allowable.

The maximum localized outer-fiber strain corresponding to this stress was approximately 12-14 percent. The incremental strains, however, for each load step were kept to approximately 0.2 percent. The maximum deflection calculated by the statically-applied loads was approximately 1 in. at the radial symmetry line passing through the hole. If geometry modifications had been made for this deflection, the load induced in the high stress regions would have been lowered since the moment arm for the beamline structure would decrease. The present analysis is therefore considered conservative from the analysis as well as the loads standpoint.

The stress and deflection analysis is based on a static application of loads which are physically short duration, dynamically applied loads. For this reason, the actual deflections caused by the short duration peak loads could be expected to be much lower than those calculated by the static analysis. The actual plant loads are also, in general, considerably lower than the design loads so that this will further reduce the true magnitude of the deflections.

The reactor coolant pump outlet nozzle was analyzed for the faulted condition using the limit analysis option of table 5.2-6. These limits are identical to the limits of Appendix F of the ASME Boiler and Pressure Vessel Code, subparagraph F-1323.2(a)/NB-3213.22. A set of umbrella loads was used in the analysis. These umbrella loads were developed using methods similar to those described for the determination of the umbrella loads for the pump foot. These umbrella loads, along with the actual plant loads for the faulted seismic condition, combined with the four worst pipe-break cases, are given in table 5.2-27. (Note that the umbrella loads exceed the actual plant loads by ratios of 2.15 to 9.77.) A three-dimensional finite element model was developed and these worst-case umbrella loads were applied. The complete model contains 792 node points and 1512 elements with 4676 active degrees of freedom. The plastic options of the ANSYS computer code were used with an elastic perfectly-plastic stress-strain curve. An iterative loading technique utilizing 25 load steps took the model from the elastic condition to the maximum load. The maximum load was increased by 10/9 to reflect the criteria in table 5.2-6. This requires that the load be < 90 percent of the limit load. At the final load step, the load deflection curve was increasing, indicating that the nozzle could take additional loads. Therefore, the faulted limit analysis requirements had been satisfied.

The reactor vessel support pads are also qualified using the test option of table 5.2-6.

The reactor pressure vessel support pads and shoes are designed to restrain unidirectional horizontal motion in addition to supporting the vessel. The design of the shoes, which are in contact with pads attached to the nozzles of the vessel, allows radial growth of the vessel, but restrains the vessel from horizontal displacements since each shoe prevents tangential displacement of the vessel at the location of the support.

To duplicate the loads that act on the pads during faulted conditions, the tests, which utilized a 1/8 linear scale model of the support system (nozzle pad, shoes, shims, and hold down bolts), were performed by applying a unidirectional static load to the nozzle pad. The load on the nozzle pad was reacted by the support shoe which was mounted to the test fixture with the bolt-down bolts.

The above modeling and application of load thus duplicates the actual case and allows the maximum load capacity of the support system to be accurately established. The test load, L_t , was then determined by multiplying the maximum collapse load by 64 (ratio of prototype area to model area) and including temperature effects in accordance with the rules of the ASME Code Section III.

The loads on the shoes, as calculated in the analysis of the components for faulted conditions, are limited to the value of 0.80 L_t in table 5.2-6.

The tests performed and the limits established for the test load method ensure that the experimentally obtained value for L_t is accurate and that the support system for the reactor pressure vessel will perform its intended function.

5.2.1.12 Protection Against Environmental Factors

5.2.1.12.1 Missile Protection

A discussion of the protection provided for the principal components of the RCS against missiles is found in chapter 3.0.

5.2.1.12.2 Flooding Protection

External flooding protection for the containment and the RCS is provided as described in appendix 3A. Internally-generated flooding of the containment could be caused only by inadvertent generation of the safety injection system, including the containment spray system, or a LOCA condition. The maximum amount of water injected into the containment during a spurious spray system operation is the volume of water contained in the refueling water storage tank. All safety-related components inside the containment are designed to withstand the effect of a water spray solution containing boric acid and sodium hydroxide. The maximum level of water inside the containment that would result from the containment spray system would be below the level of the RCPB and any of the safety-related equipment. Therefore, the flooding of this equipment is effectively precluded.

5.2.1.12.3 Fire Protection

Fire protection for the RCS is provided by the following means: first, the minimum use of combustible materials within the containment reduces the possibility of fire; second, environmental design specifications for electrical components and cables in the RCSs and all safety-related equipment inside the containment are discussed in paragraph 3.11.2.1. These requirements in design minimize the possibility of electrical shorts because of environmental effects. If shorts do occur, the selective tripping feature described in subsection 8.3.1 instantly removes power to the faulty equipment, minimizing damage. Also discussed in this subsection is the single-failure criterion imposed on the safety-related equipment, which ensures adequate protection for the RCS.

5.2.1.13 <u>Compliance with Code Requirements</u>

A brief description of the analyses and methods used to assure compliance with the applicable codes is provided in paragraph 5.2.1.10.1.

5.2.1.14 <u>Stress Analysis for Emergency and Faulted Condition Loadings</u>

The stress analyses used for faulted condition loadings are discussed in paragraph 5.2.1.10.1. There are no emergency conditions specified.

5.2.1.15 <u>Stress Levels in Category I Systems</u>

The stress intensity evaluations for the normal, upset, and faulted conditions show that the stress intensities in the piping are below the code-allowable values established in the design specifications.

5.2.1.15.1 Normal and Upset Conditions

RCL piping minimum wall thickness, t_m, was calculated in accordance with equation 1, subparagraph NB-3641.1, of the code. The as-built pipe minimum wall thickness meets the code requirement.

The maximum combined primary stress intensity caused by DBE pressure, and weight in the RCL is 19,010 psi, which is less than the code allowable stress intensity value of (1.5 S_m) 26,700 psi, using equation 9 of NB-3652.

The primary-plus-secondary stress intensity range calculations outlined in the code were performed. They show compliance with the code stress and fatigue requirements.

The cumulative usage factors calculated in accordance with the rules described in the code are less than the allowable value of unity for all piping components. All normal, upset, and test conditions having contributions to the usage factors were included in this evaluation. The code limit on the fatigue damages, measured by cumulative usage factors, is satisfied at all locations on the RCL piping. The maximum cumulative factor obtained from the analysis is 0.980 at the reactor pressure vessel outlet nozzle.

The RCL piping stress intensity ranges and fatigue damages are in conformance with the requirements of the code for the fatigue damage evaluation performed under all normal, upset, and test conditions.

5.2.1.15.2 Faulted Condition

The primary stress intensity contribution during the faulted condition can be an increase in the operating pressure of the RCL. The maximum pressure variation above the normal operating pressure for all faulted condition transients is 780 psi, caused by a control rod ejection transient. This pressure increase indicates that the permissible pressure of 2.0 P, where P is the design pressure as defined in the design specification, is not exceeded for the faulted condition.

The calculated maximum values of stress intensity for high stress points in the unbroken legs of the broken loop and the unbroken loop piping meet the code allowable stress intensity value for equation 9 for all LOCA cases and main steam line rupture. The maximum primary stress intensity for the primary stress intensity for the faulted condition loading combinations listed in table 5.2-3 is 47,700 psi, which is less than the code allowable primary stress intensity value of (3 S_m) 53,400 psi.

Therefore, the reactor coolant piping as designed is adequate and will maintain its structural integrity and meet the safety-related design requirements under all specified operating conditions.

5.2.1.16 Analytical Methods for Stresses in Pumps and Valves

Pumps and valves within the RCS boundary are designed to meet the stress limits given in table 5.2-4. Analytical methods are in accordance with the applicable codes described in table 3.2-1.

5.2.1.17 Analytical Methods for Evaluation of Pump Speed and Bearing Integrity

Reactor coolant pump overspeed evaluations are covered in paragraph 5.5.1.3.

5.2.1.18 Operation of Active Valves Under Transient Loadings

Valves required to open or close during or following any specified plant design transient condition have been designed in accordance with various codes and procedures that have been widely used by the nuclear industry. These codes and procedures are based on engineering judgment, inservice performance, and fundamental principles of engineering mechanics, rather than the requirements of a detailed stress analysis. This basis has resulted in conservative designs which ensure that these components will function as required.

5.2.1.19 Field Run Piping

Normally, pipe 2-in. and under will be field run with the following exception:

- Piping classified under ASME Section III, Class 1.

These pipes require certain physical routing considerations for protection from such events as pipe break and missiles and provisions for other design considerations such as separation and redundancy. It is necessary, therefore, that all 2-in. and under piping in the above category not be permanently installed by the field until the field isometric sketch is reviewed and analyzed by the responsible design engineer on the project.

Piping classified under ASME Section III, Classes 2 and 3, and ANSI B31.1 that require seismic stress analysis, were routed on the piping design drawings and dimensioned in the field. Detail isometrics were prepared for those pipes that were dimensioned in the field and forwarded to the project for review and analyses by the responsible engineer for seismic stress, thermal stress, shielding, and thermal insulation requirements as needed. The approved isometrics were then released for permanent installation. Only piping in the ANSI B31.1 class that does not require seismic analysis is run and dimensioned in the field without design engineering approval being required for permanent installation.

5.2.2 OVERPRESSURIZATION PROTECTION

The RCS is protected against overpressurization by two independent relief systems whose operability is governed by the mode of plant operation. During startup and shutdown operations, when the RCS is in the solid condition, low temperature overpressurization protection is provided by an overpressurization mitigating system which utilizes the two RHR system relief valves. Detailed information concerning the design of the overpressurization mitigating system is discussed in paragraph 5.2.2.4.

The RCS is protected by pressure relief devices comprising the three pressurizer safety valves and the two power-operated relief valves (PORVs) in other modes of plant operation when the overpressurization mitigating system is not in use. The following provides a detailed description of the pressurizer safety valves and the PORVs.

5.2.2.1 Location of Pressure Relief Devices

Pressure relief devices for the RCS include the three pressurizer safety valves and two PORVs shown on drawings D-175037, sheet 2 and D-205037, sheet 2; these discharge to the pressurizer relief tank by common header. Other relief valves that discharge to the pressurizer relief tank are itemized in table 5.2-18.

5.2.2.2 Mounting of Pressure Relief Devices

The pressure relief devices, as specified in paragraph 5.2.2.1, are mounted and installed as follows:

A. The pressurizer safety valve inlet piping forms a loop to ensure a water seal on the valve seat. The water volume in the loop seal is minimized to keep the reaction forces on the downstream piping as low as possible.

B. The loop seal piping is insulated to maximize loop seal water temperature. This maximizes the water volume expected to flash to steam upon lifting of the safety valves and thus, reduces downstream forces on discharge piping.

C. A support is provided on the discharge piping as close as possible to each safety and relief valve discharge nozzle so that forces and moments (including pipe whip and reactions following an assumed discharge pipe rupture) will not jeopardize the integrity of the valves, the inlet lines to the valves, or the nozzles on the pressurizer.

D. The support on the valve discharge is connected to the pressurizer instead of adjacent structures in order to minimize differential thermal expansion and seismic interactions.

E. Each straight leg of discharge piping is supported to take the force along that leg.

5.2.2.2.1 Pressurizer Safety and Relief Analysis Loading Criteria and Methods of Analysis

During original plant licensing, static and dynamic analyses were performed to verify the adequacy of the pressurizer safety and relief valves for FNP.

Under NUREG 0737[18], Section II.D.1, “Performance Testing of BWR and PWR Relief and Safety Valves,” all operating plant licensees and applicants were required to conduct testing to qualify the RCS relief and safety valves under expected operating conditions for design-basis transients and accidents. In addition to the qualification of valves, the functionability and structural integrity of the as-built discharge piping and supports was also required to be demonstrated on a plant-specific basis.

In response to these requirements, a program for the performance testing of PWR safety and relief valves was formulated by EPRI[19]. The primary objective of the test program was to provide full scale test data confirming that functionability of the RCS PORVs and safety valves are capable of performing their design function for expected operating and accident conditions. The second objective of the program was to obtain sufficient piping thermal hydraulic load data to validate models utilized for plant-unique analysis of PSARV discharge piping systems. Based on the results of the aforementioned EPRI Safety and Relief Valve Test Program, additional thermal hydraulic analyses were required to adequately define the loads on the piping system due to valve actuation.
The results of the analysis for FNP were provided to the NRC in reference 20. NRC acceptance of the FNP analysis is documented in reference 21. A summary of the FNP evaluation follows.

5.2.2.2.1.1 <u>Thermal Hydraulic Modeling</u>. The safety valve discharge loads were calculated for the fluid transient condition that will produce the most severe loading on the piping system. This occurs during a high pressure transient where steam from the pressurizer forces the water in the water seal through the safety valve down the piping system to the relief tank. Forcing functions are normally generated for hot or cold loop seals depending on the temperature in the loop seal. The hot and cold loop seal conditions for Farley plants are consistent with the hot and cold loop seal conditions defined in 1982 EPRI tests. Thermal hydraulic analysis for the Farley pressurizer safety valve system was originally analyzed in 1982 for both the hot and cold loop seal conditions. The hydraulic forces generated when the safety valves open are much higher for the cold loop seal condition compared to those forces from the hot loop seal condition. To reduce the loads from cold loop seal condition, modification to piping insulation was necessary to ensure sufficient heat was conducted to the loop seal water. However, the resulting loop seal piping temperatures were not high enough for classification as a hot loop seal. The measured temperature profiles at the three loop seal systems fall between the bounds of hot and cold. The thermal hydraulic forces resulting from this intermediate temperature loop seal are significantly less than predicted for the cold loop seal condition.

5.2.2.2.1.2 <u>Thermal Hydraulic Analysis</u>. Based on the WCAP-10105[22] report “Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program,” (June 1982), the valve opening characteristics are not linear. The valve stem actually lifts partially, allowing the water seal to pass through the valve. Once the steam behind the water slug reaches the valve stem, the valve stem will lift up fully in about .04 s. These valve opening characteristics are consistent with Figure 4-12 of the WCAP-10105 report and the loop seal purge delay curve (Figure 8) for a Crosby 6M6 forged safety valve. The opening characteristics of the Crosby 6M16 safety valves in Farley plants behave similarly with the Crosby 6M6 safety valves. Furthermore, a review of EPRI data confirmed that the pressure increase ramp rate from 2 to 375 psi/s envelops the ramp rate for Farley.

Nonlinear opening area time-history valve characteristics are considered in the latest thermal hydraulic analysis. In addition, an average loop seal temperature of about 200°F, which is below the average Farley loop seal temperature, is used for the loop seal water slug properties. This method used along with programs ITCH and FORFUN was benchmarked against the previous EPRI test results and good correlations were documented. For the Farley plant specific application, the thermal hydraulic forces were generated using the nonlinear valve opening area time-history method. The application of this method results in a reduction in the hydraulic thrust forces due to the water slug being more slowly passed through the valve (with 5 to 10% opening area) before the valve is fully open. The water hammer effect is thus reduced.

The thermal hydraulic forces generated by considering time-history variable valve opening were determined for 5% and 10% initial valve opening areas. The forces with the 10% initial valve opening area are more conservative than those with the 5% initial valve opening area and are used to perform the time-history structural analysis of the pressurizer safety valve piping system. For the Farley plant-specific safety valves, the actual initial valve opening area is 5% as determined by documented valve characteristics calculations.

5.2.2.2.1.3 <u>Thermal Hydraulic Analysis Computer Programs</u>. The computer program used for the thermal hydraulic analysis was ITCH on Sun Workstation[23]. This program was upgraded several times from original program ITCHVALVE[24,25] since 1982 and was renamed to ITCHVENT once on the mainframe computer. The program ITCHVENT was converted to Sun workstation in 1992. Program ITCHVALVE was benchmarked against the EPRI test data. ITCHVALVE is a 1-D thermal hydraulic code that calculates the time-history fluid properties within the pressurizer safety and relief valve system for the condition when the safety or relief valves open. The thermal hydraulic forces are calculated by another program called FORFUN[26] considering the momentum changes for the fluid in each element of the piping segment.

5.2.2.2.2 Structural Modeling and Analysis Methods

The structural modeling and analysis of the pressurizer safety valve piping system were performed using the WECAN computer code[27]. The piping system was modeled by pipe, elbow, support stiffness elements with both elastic and elastic/plastic capabilities. Consistent mass effect was considered in the analysis. For the analysis of the piping system with combination of deadweight and safety valve thrust discharge loadings, WECAN dynamic transient time-history analysis option was chosen. The input time-history was determined by ITCH and FORFUN computer programs and was applied to the piping system structural model.

Figure 5.2-16 shows the structural model of the Unit 2 safety line system, which contains three 6-in. safety valves on three lines before meeting a 12-in. common header. The 12-in. common header leading to the pressurizer relief tank is also in the model. Part of the relief line piping was modeled in the structural system to account for the structural system interactions. Structural analyses were performed for both Units 1 and 2.

The time-history solution for the dynamic thrust analysis of safety valve discharge with loop seal water slug was obtained from WECAN computer programs using direct integration methods.

Since the purpose of this analysis is to determine the elastic behavior of the piping system under the extreme loading of valve thrust, the linear-elastic option of the WECAN program was used. The resulting stress at 8 equally spaced circumferential points of a given cross-section was calculated for a 1.0-s time history following the simultaneous discharge at the three safety valves.

5.2.2.2.3 Piping Component Systems Evaluation Criteria

The pressurizer safety and relief valve piping system was originally qualified to its design basis allowables prior to NUREG-0737 requirements. The design basis was the requirements of ASME B&PV Code Section III, 1971 edition, including summer 1971 addenda for Class I piping and the ANS B31.1-1967 Code with 1971 addenda for nonnuclear safety (NNS) piping. To take advantage of the knowledge gained in the industry on stress calculation methods, the ASME B&PV Code Section III, 1977 Edition through Summer Addenda 1979 was used for Class 1 piping as allowed by NCA-1140(f) of the 1971 Code and NCA-1140(b) of the 1977 code. In 1982, Westinghouse performed additional evaluations to address TMI-related issues by considering the cold loop seal loads for these piping systems[28]. Criteria used in that analysis was based on the recommendation from piping subcommittee of the PWR Pressurizer Safety and Relief Valve (PSARV) test program and was documented in a WCAP-10105[22]. Those criteria were reviewed and accepted by the NRC in a 1986 SER(29).

In the FNP evaluation, the loading combination and piping evaluation criteria of WCAP-10105 were applied with the exception of an allowable stress of 2.4 S_h for the emergency condition for the NNS portion of the piping system. This exception was approved by the NRC as documented in reference 21.

Using elastic analysis techniques, the Class I piping (which connects the pressurizer safety line nozzle to the 6-in. safety valve), was qualified to the allowables listed in table 5.2-40 with the effect of valve thrust under both emergency and faulted conditions. The NNS portions of the piping system area also qualified to meet the allowables listed in table 5.2-41. The most limiting stresses for the emergency conditions are shown in table 5.2-42.

5.2.2.2.4 Safety Valve Nozzles

One additional means to ensure that the safety valve remains operable after the loop seal water is discharged is to assess the valve nozzle loads with respect to the valve operability limit provided in the equipment specification. For emergency condition, the calculated valve nozzle loads from the combination of deadweight, pressure, and valve thrust effects are within the equipment specification allowable. This allowable requires the maximum total valve nozzle stress to be 75% of the yield stress of the nozzle material at temperature. In addition, it further requires that the maximum bending stress be 50% and the maximum torsion stress also be 50% of the yield stress of the nozzle at temperature.

5.2.2.2.5 Support Component Evaluation Loading and Load Combinations

5.2.2.2.5.1 Loading Conditions. The piping system loading conditions considered for the pipe support evaluation consisted of the valve thrust loadings discussed above in combination with the existing design basis deadweight, normal thermal expansion, transient thermal expansion, and the OBE & SSE seismic loadings.

Since the pipe supports had previously been qualified for the Normal, Upset, Emergency, and Faulted conditions, the supports were only evaluated for the worst case load combination including the valve thrust loads from the piping system analysis. The loading combination used for support evaluation is:

$$P = DW \pm Thm_{max/min} \pm \sqrt{SSE^2 + Thrst^2}$$

5.2.2.2.5.2 Support Component Evaluation Stress Acceptance Criteria. The purpose of the support evaluation was to demonstrate that the supports retained their integrity for the controlling combined loads. This was accomplished by generally limiting the actual support member stresses to the allowable stress limits established by the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF and Appendix F, 1974 Edition. The code of record, AISC 7th Ed., does not address the faulted loading combination. ASME Subsection NF was used for this evaluation since it is essentially the same as AISC for the normal and upset conditions, and it provides criteria for the extreme faulted loading combination. In addition, the Subsection NF criteria are consistent with the pipe support criteria utilized by most other nuclear plants.

In accordance with NRC IE Bulletin 79-02, concrete expansion anchors (CEA) on Class I pipe support base plates were limited to manufacturer's allowables, including a Factor of Safety of 4.0. However, for four CEAs on NNS Class Pipe Support Base Plates, the manufacturer's allowable including a factor of safety of 3.0, was applied. These bolts are identified in table 5.2-43. The use of this safety factor for the 4 bolts was approved by the NRC for this application as documented in reference 21.

5.2.2.2.5.3 Support Evaluation Results. Class I supports - the results of the pipe support evaluations based on the as-built support data provided to Westinghouse show that all the Unit 1 and Unit 2 pipe support standard Grinnell components, structural members, and base plate element stress levels are within the allowable stress limits of ASME Subsection NF and Appendix F and will maintain their structural integrity and stability for the faulted loading combination provided above. All concrete expansion anchor for class I supports have a minimum safety factor of 4.0.

NNS supports - all Unit 1 and Unit 2 NNS pipe supports satisfied the ASME Subsection NF and Appendix F faulted stress criteria. Therefore, all the NNS pipe supports will maintain their structural integrity for the specified loading combination. Most expansion anchors have safety factor > 4.0. Table 5.2-43 provides a summary of only those NNS class pipe supports which have concrete expansion anchors with safety factor < 4.0 but > 3.0 in their qualification.

5.2.2.3 Report on Overpressure Protection

The pressurizer is designed to accommodate pressure increases (as well as decreases) caused by load transients. The spray system condenses steam to prevent the pressurizer pressure from reaching the setpoint of the PORVs during a step reduction in power level of 10 percent of load.

The spray nozzles are located on the top of the pressurizer. Spray is initiated when the pressure controlled spray demand signal is above a given setpoint. The spray rate increases proportionally with increasing pressure rate and pressure error until it reaches a maximum value.

The pressurizer is equipped with PORVs which limit system pressure for a large power mismatch and thus prevent actuation of the fixed high pressure reactor trip. The relief valves are operated automatically or by remote manual control. The operation of these valves also limits the undesirable opening of the spring-loaded safety valves.

Remotely-operated block valves are provided to isolate the PORVs if excessive leakage occurs. The relief valves are designed to limit the pressurizer pressure to a value below the high pressure trip setpoint for all design transients up to and including the design percentage step load decrease with steam dump, but without reactor trip.

Output signals from the pressurizer pressure control channels are used for pressure control. These are used to control pressurizer spray and heaters and PORVs. Pressurizer pressure is sensed by fast response pressure transmitters with a time response of better than 0.2 s.

In the event of a complete loss of heat sink, i.e., no steam flow to the turbine, protection of the RCS against overpressure is afforded by pressurizer and steam generator safety valves along with any of the following reactor trip functions:

A. Reactor trip on turbine trip (if the turbine is tripped).

B. High pressurizer pressure reactor trip.

C. Overtemperature-ΔT reactor trip.

D. Low-low steam generator water level reactor trip.

Continued integrity of the RCS during the maximum transient pressure is assured by design within the applicable codes as discussed in reference 4. The code safety limit is 110 percent of the 2485 psig design limit.

A detailed functional description of the process equipment associated with the high pressure trip is provided in reference 5.

The upper limit of overpressure protection is based upon the peak surge into the pressurizer of the reactor coolant produced as a result of turbine trip under full load, assuming no reactor trip. The self-actuated safety valves are sized on the basis of steam flow from the pressurizer to accommodate this surge at a setpoint of 2500 psia and a total accumulation of 3 percent. Note that no credit is taken for the relief capability provided by the PORVs during this surge.

The RCS design and operating pressure, together with the safety, power relief and pressurizer spray valve setpoints, and the protection system setpoint pressures, are listed in table 5.2-19.

System components whose design pressure and temperature are less than the RCS design limits are provided with overpressure protection devices and redundant isolation means. System discharge from overpressure protection devices is collected in the pressurizer relief tank in the RCS. Isolation valves are provided at all connections to the RCS.

5.2.2.4 <u>RCS Pressure Control During Low Temperature Operation</u>

Administrative procedures have been developed to aid the operator in controlling RCS pressure during low temperature operation. However, to minimize the frequency of RCS overpressurization, an overpressure mitigating system is provided to mitigate pressure excursions initiated by inadvertent mass and/or heat additions when the RCS temperature is less than or equal to the low temperature overpressure protection (LTOP) System applicability temperature specified in the PTLR.

5.2.2.4.1 System Design

The overpressure mitigation system employs the RHR system relief valves (RHRSRV) to mitigate RCS overpressure transients. One relief valve is installed in each RHR suction line. The RHRSRVs are spring-loaded, bellows-type valves which have a setpoint of ≤ 450 psig. The current methodology requirements use a setpoint of 436 $\pm$13 psig. At 495 psig the valves deliver full design flow. There are two isolation valves between each of the RHRSRVs and the RCS. The autoclosure interlock of the RHR suction/isolation valves was deleted per WCAP-11746 analysis. An alarm will alert the operators if the RHR suction/isolation valve(s) is not fully closed and the RCS pressure exceeds the alarm setpoint. The RHR suction valves inside containment are open when the RCS temperature is less than or equal to the LTOP System applicability temperature specified in the PTLR, thereby aligning the RHR relief valves for RCS overpressurization protection. As additional protection against RCS overpressurization, power is removed from the RHR isolation valves in Modes 1, 2, and 3. Power is reinstated to the isolation valves prior to exceeding an RCS temperature of 180°F via strict administrative controls, which assure the operability of the RHR isolation valves and associated interlocks.

The RHR relief valves have no electrical components. The open-permissive circuits of the RHR motor-operated isolation valves meet the requirements of IEEE-279-1971. Power supplies for the RHR isolation valves, the pressurizer pressure sensors, and the RCS temperature sensors are designed so that no single failure of the electrical system or the loss of offsite power would isolate both of the RHR relief valves.

In addition, several control room alarms have been provided. A Seismic Category I alarm designed to the requirements of IEEE-279-1971 alerts the operator if the RHR isolation valves are not fully open when the RCS temperature is ≤ 300°F. Another alarm provides indication to the operator of any overpressure transient occurring when the RCS pressure > 450 psig.

5.2.2.4.2 Operating Basis Earthquake (OBE) Evaluation

The RHRSRVs and the associated discharge piping up to the pressurizer relief tank are designed in compliance with Regulatory Guide 1.29, Rev. 1. The RHRSRVs were manufactured by the Crosby Valve and Gage Company, which has certified that the performance of these valves will not be degraded by an OBE event with a horizontal acceleration of 1.725 g and 1.455 g and a vertical acceleration of 1.221 g. This certification is based on valves of similar construction and characteristics as the subject relief valves. The piping downstream of the RHR system isolation valves up to the RHRSRVs, including the RHRSRVs, meets the ANSI Nuclear Safety Criteria for the design of stationary pressurized-water reactor plants, August 1970 draft. The piping upstream of the RHR system isolation valves including these valves is Quality Group A per 10 CFR 50.55(a). The RHRSRV discharge piping up to the pressurizer relief tank and the pressurizer relief tank are Nonnuclear Class (B.31.1 piping); however, they are seismically supported.

Thus, the overpressure mitigating system is capable of functioning following a seismic event.

5.2.2.4.3 Pressure Transient Analyses

ASME, Section XI, Appendix G, establishes guidelines for RCS pressure during low temperature operation (≤ 350°F). The relief system discussed in paragraph 5.2.2.4.1 serves to mitigate overpressure excursions to within these allowable limits. The worst-case mass input event was assumed to be the inadvertent operation of three high-head safety injection pumps with a maximum total flowrate of 1000 gal/min at 0 psig backpressure at RCS temperatures ≥ 180°F. Due to Technical Specification restrictions that allow only one operable charging pump at RCS temperatures < 180°F, the worst-case mass injection is limited to the start of a single charging pump at RCS temperatures < 180°F. The worst heat input event was assumed to be the starting of a single reactor coolant pump with a temperature differential of 50°F existing between the RCS and the steam generator. The maximum calculated RCS pressures for these postulated worst mass and heat input events remained below the pressures allowed by the Appendix G curves for transients initiated below 325°F. For transients above 325°F, the pressurizer code safety valves would relieve pressure to prevent violation of Appendix G limits.

5.2.2.4.4 Administrative Procedures

Although the system described in paragraph 5.2.2.4.1 mitigates pressure excursions to address the allowable pressure limits, administrative procedures are employed to minimize the potential for the development of any transient that would challenge the system.

Of primary importance is the basic mode of operation of the plant. Normal operating procedures maximize the use of a pressurizer cushion (steam bubble) during periods of low temperature operation. A steam bubble is formed in the pressurizer at a cold leg temperature in the range of approximately 130 to 180°F when the plant is being started up. It is collapsed at a cold leg temperature of < 200°F when the plant is being cooled down.

This cushion dampens the plant response to potential transient generating inputs, thereby providing easier pressure control with slower response rates.

This cushion substantially reduces the severity of some potential transients such as RCP-induced heat input and slows the rate of pressure rise for others. This provides reasonable assurance that most potential transients can be terminated by operator action before an overpressure condition exists.

Administrative controls employed to minimize the potential for overpressure developing include the following:

A. Only one charging pump may be operational when the RCS temperature is < 180°F. Power is removed from the two nonoperating charging pumps when the RCS temperature is ≤ 180°F, except during pump sump operations, by removing the motor circuit breakers from their electrical power supply circuits.

B. The letdown heat exchanger control valve is placed in the manual control position prior to starting or stopping an RHR pump when the RCS is in a water solid condition.

C. The RHR suction isolation valves are open and the RHR relief valves are available to mitigate an overpressure event or the RCS is vented whenever the RCS temperature is 325°F or less.

D. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures is < 325°F unless 1) the pressurizer water volume is < 770 ft^3 (24% of wide range, cold, pressurizer level indication) or 2) the secondary water temperature of each steam generators is < 50°F above each of the RCS+ cold leg temperatures.

E. The accumulators are isolated and power is locked out from the accumulator isolation valve operators at RCS pressure below 1000 psig. These actions are completed prior to reducing RCS pressure to < 900 psig.

F. The low pressurizer pressure and low steam line pressure safety injection signals are blocked during heatup and cooldown to preclude an inadvertent ECCS actuation.

G. During cooldown all steam generators should be connected to the steam header to assure a uniform cooldown of the RCS loops.

H. NRC acceptance criteria for GL 90-06 is as follows: When an LTOP channel is inoperable and the RCS is not water-solid (water-solid is defined as a pressurizer level of 30% [cold calibrated], a trained, dedicated operator will be assigned to monitor and control RCS pressure. The operator will have two independent alarms available to identify the occurrence of an overpressure event, and will be specifically trained to respond to these alarms.

I. It is recommended that if all reactor coolant pumps have been stopped for more than 5 min during plant heatup, and the reactor coolant temperature is greater than the charging and seal injection water temperature, there should be no attempt to restart a pump unless a steam bubble is formed in the pressurizer. This precaution will minimize the pressure transient when the pump is started and the cold water previously injected by the charging pumps is circulated through the warmer reactor coolant components. The steam bubble will accommodate the resultant expansion as the cold water is rapidly warmed.

If all reactor coolant pumps are stopped and the RCS is being cooled down by the residual heat exchangers, a nonuniform temperature distribution may occur in the RCLs. No attempt should be made to restart a reactor coolant pump unless a steam bubble is formed in the pressurizer.

These special precautions back up the normal operational mode of maximizing periods of steam bubble operation so that cold overpressure transient prevention is continued during periods of transitional operations.

Recommended procedures for ECCS testing include the following to preclude the development of cold overpressurization transients:

A. The normal procedure for periodic ECCS pump performance testing is to test the pumps during normal operation or at hot shutdown conditions. Performance testing of the ECCS pumps with the RCS in a water-solid condition is prohibited.

B. The SI/LOSP test is performed during Mode 6 operation or with the reactor defueled.

C. The ECCS branch line flow verification and charging pump low discharge head flow tests are performed in Mode 6 with the reactor vessel head removed or with the reactor defueled.

The above procedural recommendations covering normal operations with a steam bubble, transitional operations where potentially water solid, followed by specific testing operations, provide in-depth cold overpressure prevention or mitigation, augmenting the installed overpressure relief system.

5.2.3 GENERAL MATERIAL CONSIDERATIONS

5.2.3.1 Material Specifications

The material specifications used for the principal pressure retaining applications in each component comprising the Reactor Coolant System boundary are listed in table 5.2-20 for Class 1 Primary Components and table 5.2-21 for Class I and II Auxiliary Components. These materials are procured in accordance with the specification requirements and include supplemental requirements of the applicable ASME Code rules.

The welding materials used for joining the ferritic base materials of the reactor coolant boundary conform to, or are equivalent to, ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18, and 5.20. They are tested and qualified to the requirements of ASME Section III rules.

The welding materials used for joining the austenitic stainless steel base materials of the reactor coolant boundary conform to ASME Material Specifications SFA 5.4 and 5.9. They are tested and qualified according to the requirements stipulated in subsection 5.2.5.

The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combinations of the reactor coolant boundary conform to ASME Material Specifications SFA 5.11 and 5.14. They are tested and qualified to the requirements of ASME Section III rules and are used only in procedures that have been qualified to these same rules.

5.2.3.2 Compatibility With Reactor Coolant

Materials used in components within the RCPB are listed in tables 5.2-20, 5.2-21, and 5.2-23. All of the ferritic low-alloy and carbon steels used in principal pressure-retaining applications are provided with a 0.125-in. minimum thickness of corrosion-resistant cladding on all surfaces that are exposed to reactor coolant. This cladding material has a chemical analysis which is at least equivalent to the corrosion resistance of types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy. The other base materials which are used in principal pressure-retaining applications that are exposed to the reactor coolant are austenitic stainless steel, nickel-chromium-iron alloy, and martensitic stainless steel. Ferritic low-alloy and carbon steel nozzles are safe-ended with stainless steel weld metal analysis A-7 or nickel-chromium-iron alloy weld metal F-Number 43 using weld buttering techniques followed by a post-weld heat treatment. The latter buttering material requires further safe-ending with austenitic stainless steel base material after completion of the post-weld heat treatment when the nozzle is larger than 4 in. nominal I.D. and/or the wall thickness is > 0.531 in.

The cladding on ferritic-type base materials receives a post-weld heat treatment.

All of the austenitic stainless steel and nickel-chromium-iron alloy base materials are used in the solution-anneal-heat-treat-condition. The heat treatments are as required by the material specifications. During subsequent fabrication, these pressure-retaining materials are not heated above 800°F other than instantaneously and locally by welding operations. The solution-annealed surge line material is subsequently formed by hot bending followed by a resolution-annealing heat treatment. Corrosion tests are performed in accordance with ASTM A 393.

5.2.3.3 Compatibility With External Insulation and Environmental Atmosphere

In general, all of the materials listed in tables 5.2-20 and 5.2-21, which are used in principal pressure retaining applications and are subject to elevated temperature during system operation, are in contact with thermal insulation that covers their outer surfaces.

The thermal insulation used on the RCS, including the pressure vessel, is of the stainless steel reflective-type.

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is anticipated from service experience, such as valve packing, pump seals, etc., materials that are compatible with the coolant are used. These are shown in tables 5.2-20 and 5.2-21. Ferritic materials exposed to coolant leakage can be observed as part of the inservice visual and/or nondestructive inspection program to ensure the integrity of the component for subsequent service.

5.2.3.4 Chemistry of Reactor Coolant[a]

The RCS chemistry specifications are given in table 5.2-22.

The RCS water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications.[a]

The chemical and volume control system (CVCS) provides a means for adding chemicals to the RCS to control the pH of the coolant during initial startup and subsequent operation, to scavenge oxygen from the coolant during startup, and to control the oxygen level of the coolant caused by radiolysis during all power operations subsequent to startup. The oxygen content and pH limits for power operations are shown in table 5.2-22.

The pH control chemical employed is lithium hydroxide. This chemical is chosen for its compatibility with the materials and water chemistry of borated water, stainless steel, zirconium, and Inconel systems. In addition, lithium is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The lithium hydroxide is introduced into the RCS via the charging flow. The solution is prepared in the laboratory and poured into the chemical mixing tank. Reactor makeup water is then used to flush the solution to the suction manifold of the charging pumps. The concentration of lithium hydroxide in the RCS is maintained as a function of boron concentration in the range specified for pH control. If the concentration exceeds this range, either the cation-bed demineralizer or the mixed-bed demineralizer is employed in the letdown line to reduce the lithium concentration to within range.

During reactor startup from the cold condition, hydrazine is employed as an oxygen scavenging agent. The hydrazine solution is introduced into the RCS in the same manner as described above for the pH control agent.

Dissolved hydrogen is employed during power operation to control and scavenge oxygen produced because of radiolysis of water in the core region. Sufficient partial pressure of hydrogen is maintained in the volume control tank so that the specified equilibrium concentration of hydrogen is maintained in the reactor coolant. A self-contained pressure

a. The Water Chemistry Control Program is credited as a license renewal aging management program (see chapter 18, subsection 18.2.2).

control valve maintains a minimum pressure in the vapor space of the volume control tank. This can be adjusted to provide the correct equilibrium hydrogen concentration.

Components with stainless steel sensitized in the manner expected during component fabrication and installation will operate satisfactorily under normal plant chemistry conditions in pressurized-water reactor systems because chlorides, fluorides, and, particularly, oxygen, are controlled to very low levels.

5.2.4 FRACTURE TOUGHNESS[a]

5.2.4.1 Compliance With Code Requirements

Assurance of adequate fracture toughness of ferritic materials in the reactor coolant system boundary is provided by compliance with Section III of the 1968 ASME Boiler and Pressure Vessel Code, plus applicable Addenda and Code Cases. Test results for reactor pressure vessel materials are given in tables 5.2-24 and 5.2-25.

5.2.4.2 Acceptable Fracture Energy Levels

The initial NDTT of plate materials in the reactor vessel beltline will not be greater than the criteria for fracture energy levels as given in paragraph 5.2.4.3.

Although two test specimens for weld metal used in weld seam 10-923 of Unit 2 exhibited impact energies of < 75 ft-lb at a test temperature of 10°F, it is expected that the upper shelf impact energy requirement of 75 ft-lb identified in paragraph IV.A.1.a of 10 CFR 50 Appendix G would easily be exceeded if tests had been performed at test temperatures representative of the upper shelf. A review of many weld test certificates provided by the vessel fabricator indicates that the upper shelf energy of welds of chemical composition and fabrication history similar to weld seam 10-923 and fabricated with the same type of wire and flux (type B-4 weld wire and Linde 0091 Flux) used in seam 10-923 exceeds 75 ft-lb by a considerable margin. Four examples of the vessel fabricator test results for weld material similar to that of seam 10-923 are shown in table 5.2-35. Like weld seam 10-923, two of these four examples did not exhibit 75 ft-lb for all test specimens at 10°F; however, at higher temperatures, 75 ft-lb was exceeded.

Individual data points obtained from Charpy V-notch impact tests for each of the base metal heats in the Farley Unit 2 reactor vessel beltline are presented in tables 5.2-36, 5.2-37, and 5.2-38.

The Farley Unit 2 pressurizer was designed and fabricated in accordance with the requirements of the 1971 Edition of the ASME Code Section III through the Winter 1970 Addendum. The current 10 CFR 50 Appendix G requirements, which became effective on August 16, 1973, are more stringent than the applicable code requirements for Farley Unit 2.

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a. Reactor vessel neutron embrittlement was evaluated as a TLAA for license renewal in accordance with 10 CFR 54.21 (see chapter 18, subsection 18.4.1).

The Farley Units 1 and 2 replacement steam generators were designed and fabricated in accordance with the requirements of the 1989 edition of the ASME Code Section III which includes provisions consistent with 10 CFR 50 Appendix G. References 34 and 35 provide design and fabrication details of the replacement steam generators.

The actual fracture toughness data for RCPB pressure-retaining applications in the pressurizer are tabulated in table 5.2-39. In all cases, the applicable ASME Code requirements, as well as the intent of 10 CFR 50 Appendix G, are satisfied.

SA 508 Class 2a material and SA 533 Class 2 material was used in the Farley Unit 2 pressurizer. Neither of these materials was used in primary-side (RCPB) pressure retaining applications of the Farley Unit 2 steam generators. The fracture toughness data for these materials are included in table 5.2-39. The adequacy of the fracture toughness properties of these materials has been documented in reference 10.

The following discussion demonstrates that the intent of the Appendix G, Paragraph III.B.3 requirements is satisfied.

<u>Reactor Vessel</u> - Combustion Engineering (CE) calibrated Charpy V notch test machines in accordance with Watertown Arsenal Standards every 6 months. Temperature instruments, calibrated in accordance with ASTM-E-23, were purchased every 3 months.

These calibrations were performed in accordance with the requirements of the ASME Code 1968 Edition through Summer 1970 Addenda (Appendix IX-221 and 260), which is the applicable Code for the Farley Unit 2 reactor vessel. The Charpy V notch test machine calibrations were recorded. The temperature instrument calibrations were not recorded; however, thermometers qualified to ASTM standards were purchased, used for the certified time period, and replaced with new qualified thermometers.

CE required that all of its vendors who furnished materials or parts (for Farley Unit 2) to be on an approved vendors list. Each vendor was required to have a quality control system in accordance with #N-335 of the 1968 ASME Code through Summer 1970 Addenda. Periodic audits of these vendors were performed by CE QA personnel.

It should be noted that the Farley Unit 2 reactor vessel was partially furnished by B & W. Material furnished by B & W was accepted on the basis of material certifications; therefore, no QA audits were performed for those by CE.

<u>Pressurizers</u> - Charpy V-notch test machine calibration at <u>W</u> Tampa plant was performed yearly using samples obtained from Watertown Arsenal. Temperature instrument calibration was performed with standards traceable to the National Institute of Standards and Technology.

All material suppliers have been either surveyed by ASME auditors or <u>W</u> Tampa Plant Product Assurance to obtain supplier certifications. A sampling of one of the major material suppliers indicated that Charpy V-notch test machine calibrations were recorded and that calibrated temperature instruments were purchased (as replacements) on a yearly basis.

The following discussion demonstrates that the intent of the Appendix G, Paragraph III.B.4 requirements is satisfied.

Reactor Vessel - The personnel performing the Charpy testing at Combustion Engineering were qualified by schooling, training, and many years of experience. Their qualifications to perform this work have been certified by qualified supervisory personnel. This meets the requirements of the applicable ASME Code 1968 Edition through Summer 1970 Addenda (Appendix IX 221d).

Pressurizer - Charpy impact tests were performed at W Tampa Plant by Level III and Level II personnel who had a minimum of 5 years directly-related testing experience.

Steam Generators - The replacement steam generators are constructed to an edition of the ASME code, Section III that has incorporated provisions consistent with 10 CFR 50 Appendix G. Compliance with the applicable portions of the ASME Code, Section III for the design and testing of pressure boundary materials, welding, and weld filler metal provides a vessel in compliance with the fracture toughness requirements of 10 CFR 50, Appendix G.

5.2.4.3 Operating Limitations During Startup and Shutdown

The heatup and cooldown curves for Units 1 and 2 are based on the fracture toughness properties of each vessel, as given in tables 5.2-24 and 5.2-25 and the calculation methods described in WCAP-14040-A, Revision 4[30] and WCAP-18124-NP-A[36]. Tables 5.2-24 and 5.2-25 indicate that the original maximum reference nil-ductility temperatures (RT_{NDT}) of the Unit 1 and 2 reactor vessels are not higher than +60°F. Allowable pressures as a function of the rate of temperature change and the actual temperature relative to the vessel RT_{NDT} are established according to the methods given in Appendix G, "Fracture Toughness for Protection Against Failure," of Section XI of the ASME Pressure Vessel and Boiler Code. As required by the Technical Specifications, curves showing RCS heatup and cooldown limitations are provided in the Pressure Temperature Limits Report (PTLR).

These curves are based on temperature scale relative to the limiting RT_{NDT} of the vessels, including appropriate estimates of ΔRT_{NDT} caused by radiation.[a] Predicted ΔRT_{NDT} values are derived by using the recommendations of Regulatory Guide 1.99, Revision 2 [16], and the maximum fluence at 1/4 and 3/4 of vessel wall thickness corresponding to the beltline material in question and the selected service period. Heatup and cooldown limits are then calculated using the most limiting RT_{NDT} for the selected service period in accordance with the methods described in WCAP-14040-A, Revision 4[30]. The selection of such a limiting RT_{NDT} ensures that all components in the RCS are operated conservatively in accordance with ASME code requirements. The heatup and cooldown curves are in compliance with the NRC acceptance criteria contained in Appendices G and H of 10 CFR Part 50 and Regulatory Guide 1.99, Revision 2.

The results of the radiation surveillance programs are used to verify that the predicted ΔRT_{NDT} is appropriate, or to make necessary changes if the ΔRT_{NDT} determined from the surveillance capsules is different from the predicted ΔRT_{NDT}.

The use of an RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material automatically provides additional conservatism for the nonirradiated regions. However, 10 CFR 50, Appendix G requires licensees to address the metal temperature of the closure head flange and vessel flange in the determination of heatup and cooldown rate limitations.

This rule states that the minimum metal temperature of the closure flange regions must be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20% of the preservice hydrostatic test pressure. The rule also states that a plant-specific fracture evaluation may be performed to justify less limiting requirements. As a result, a fracture analysis was performed for Unit 2.[(17)] The fracture analysis results are also applicable to Unit 1 since the pertinent parameters are identical for both units. The impact of the 10 CFR 50, Appendix G rule and the results of the fracture analysis are reflected in the heatup and cooldown curves shown in the PTLR.

5.2.4.4 Compliance With Reactor Vessel Materials Surveillance Program Requirements

Changes in fracture toughness of the core region plates, weldments, and associated heat-affected zones because of radiation damage will be monitored by a surveillance program which conforms with ASTM E-185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." The evaluation of the radiation damage in this surveillance program is based on pre-irradiation and post-irradiation testing of Charpy V-notch and tensile specimens carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals. For additional details of the irradiation surveillance program, refer to paragraph 5.4.3.6.

5.2.4.5 Reactor Vessel Annealing

See paragraph 5.4.3.7 for a discussion of reactor vessel annealing.

5.2.5 AUSTENITIC STAINLESS STEEL

The unstabilized austenitic stainless steel material specifications used for the RCS boundary, systems required for reactor shutdown, and systems required for emergency core cooling, are listed in tables 5.2-20 and 5.2-21.

The unstabilized austenitic stainless steel material specifications used for the reactor vessel internals that are required for emergency core cooling for any mode of normal operation, or under postulated accident conditions, and for core structural load bearing members, are listed in table 5.2-23.

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a. Reactor vessel neutron embrittlement was evaluated as a TLAA for license renewal (see chapter 18, subsection 18.4.1).

All of the above tabulated materials are procured in accordance with the specification requirements and include supplemental requirements of the applicable ASME Code rules.

5.2.5.1 Cleaning and Contamination Protection Procedures

It is required that all austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems be handled, protected, stored, and cleaned according to recognized and accepted methods and techniques.

The rules covering these controls are stipulated in the following Westinghouse Electric Corporation process specifications. These process specifications supplement the equipment specification and purchase order requirements of every individual austenitic stainless steel component or system which Westinghouse procures for a nuclear steam supply system, regardless of the ASME Code Classification. They are also given to the architect-erector and to the owner of the power plant for use within their scope of supply and activity.

To ensure that manufacturers and installers adhere to the rules in these specifications, surveillance of operations by Westinghouse personnel is conducted either in-residence, at the manufacturer's plant and the installer's construction site, or during periodic engineering and quality assurance visitations and audits at these locations.

The process specifications which establish these rules and which are in compliance with the more current American National Standards Institute N-45 Committee specifications are as follows:

Process Specification Number	
82560HM	Requirements for Pressure Sensitive Tapes for Use on Austenitic Stainless Steels.
83336K	Requirements for Thermal Insulation Used on Austenitic Stainless Steel Piping and Equipment.
83860LA	Requirements for Marking of Reactor Plant Components and Piping.
84350HA	Site Receiving Inspection and Storage Requirements for Systems, Material and Equipment.
84351NL	Determination of Surface Chloride and Fluoride on Austenitic Stainless Steel Materials.
85310QA	Packaging and Preparing Nuclear Components for Shipment and Storage.
292722	Cleaning and Packaging Requirements of Equipment for Use in the NSSS.

597756	Pressurized-Water Reactor Auxiliary Tanks Cleaning Procedures.
597760	Cleanliness Requirements During Storage, Construction, Erection and Startup Activities of Nuclear Power Systems.

The cleaning and contamination protection procedures for Bechtel-supplied equipment made of austenitic stainless steel materials are detailed in the individual equipment specifications. These procedures assure that austenitic stainless steel material is cleaned and protected against contaminants capable of causing stress corrosion cracking. The cleaning procedures consist of the removal of all mill scale, rust, grease, and other contaminants and cleaning with both solvent and demineralized water.

During storage of austenitic stainless steel components, special precautions are taken to ensure suitable environmental conditions. Strictly controlled working procedures are followed in order to maintain the necessary cleanliness of all austenitic stainless steel components.

5.2.5.2 Solution Heat Treatment Requirements

All of the austenitic stainless steels listed in tables 5.2-20, 5.2-21, and 5.2-23 are procured from raw material producers in the final heat-treated condition required by the respective ASME Code Section II material specification for the particular type or grade of alloy.

5.2.5.3 Material Inspection Program

All of the wrought austenitic stainless steel alloy raw materials which require corrosion testing after the final mill heat treatment are tested in accordance with ASTM A 393, using material test specimens obtained from specimens selected for mechanical testing. The materials are obtained in the solution-annealed condition.

5.2.5.4 Unstabilized Austenitic Stainless Steels

The unstabilized austenitic stainless steels used in the RCPB and components are listed in tables 5.2-20 and 5.2-21.

These materials are used in the as-welded condition, as discussed in paragraph 5.2.5.2. The control of the water chemistry is stipulated in paragraph 5.2.3.4. These chemistry controls, coupled with the satisfactory experience with components and internals using unstabilized austenitic stainless steel materials which have been post-weld heat treated above 800°F, show acceptability of these heat-treatments for stainless steel in the PWR chemistry environment[(7)]. Actual observations of post-weld, heat-treated, austenitic stainless steel after actual operation indicate no effects of such treatments. Internals heat-treated above 800°F from H. B. Robinson, Unit 2, Zorita, Connecticut Yankee, San Onofre, Beznau 1, R. E. Ginna, Yankee Rowe, Selni, and SENA have been examined after service and show acceptable material condition.

5.2.5.5 Avoidance of Sensitization

The unstabilized austenitic stainless steels used for core structural load bearing members and component parts of the RCPB are processed and fabricated using the most practicable and conservative methods and techniques to avoid partial or local severe sensitization.

After the material has been heat-treated as described in paragraph 5.2.5.2, the material is not heated above 800°F during subsequent fabrication, except as described in paragraph 5.2.3.2 and in the paragraphs below.

Methods and material techniques that are used to avoid partial or local severe sensitization are as follows:

A. Nozzle Safe Ends

   Weld deposit with Inconel (Ni-Cr-Fe weld metal F No. 43), then attach safe-end after final post-weld heat-treatment, which was used for the reactor vessel, pressurizer, accumulators, and replacement steam generators.

B. For internals, the austenitic stainless steels have been given a stress-relieving treatment above 800°F; i.e., a high temperature stabilizing procedure is used. This is performed in the temperature range of 1600-1900°F, with holding times sufficient to achieve chromium diffusion to the grain boundary regions to limit the effects of sensitization on Cr-carbide precipitation in the grain boundary. The stainless nozzles on the pressurizer were given a post-weld treatment associated with the fabrication of the head. No intergranular tests are planned because of satisfactory service experience, as noted in paragraph 5.2.5.4.

C. All welding is conducted using those procedures that have been approved by the ASME Code rules of Section III and IX.

D. All welding procedures have been qualified by nondestructive and destructive testing according to the ASME Code rules of Section III and IX.

   When these welding procedure tests are being performed on test welds that are made from base metal and weld metal materials that are from the same lot(s) of materials used in the fabrication of components, additional testing is frequently required to determine the metallurgical, chemical, physical, corrosive, etc., characteristics of the weldment. The additional tests that are conducted on a technical case basis are as follows: light and electron microscopy, elevated temperature mechanical properties, chemical check analysis, fatigue tests, intergranular corrosion tests, and static and dynamic corrosion tests within reactor water chemistry limitations.

E. The following welding methods have been tested individually and in multiprocess combinations as outlined in (D) above, using these prudent energy input ranges for the respective method, as calculated by the following formula:

$$H = \frac{E \times I \times 60}{S}$$

where E = volts

I = amperes

S = travel speed in in./min

H = joules/in.

WELDING PROCESS METHOD	ENERGY INPUT RANGE (Kilojoules/in.)
Manual shielded tungsten arc	20 to 50
Manual shielded metallic arc	15 to 120
Semi-automatic gas shielded metallic arc	40 to 60
Automatic gas shielded tungsten arc-hot wire	10 to 50
Automatic submerged arc	60 to 140
Automatic electron beam - soft vacuum	10 to 50

F. The interpass temperature of all welding methods is limited to 350°F maximum.

G. All full-penetration welds require inspections in accordance with Article 6 of the ASME Section III Code rules.

5.2.5.6 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitizing Temperatures

In general, it is not feasible to remove samples from fabricated production components to prepare specimens for retest to determine the susceptibility to intergranular attack. These tests are performed only on test welds when meaningful results would predicate production material performance and are as described in paragraph 5.2.5.5. No intergranular tests are planned because of satisfactory service experience (see paragraph 5.2.5.5).

5.2.5.7 Control of Delta Ferrite

The austenitic stainless steel welding material used for joining Class 1 pipe, pump, fittings, and applications is described in paragraph 5.2.3.1. The welding material conforms to ASME Weld Metal Analysis A-8 for all applications. Bare weld filler metal materials, including consumable inserts used in inert gas welding processes, conform to ASME SFA-5.9 and are procured to contain not less than 5-percent delta ferrite. All weld filler metal materials used in flux-shielded welding processes conform to ASME SFA-5.4 or SFA-5.9 and are procured in a wire flux combination to be capable of providing not less than 5-percent delta ferrite in the deposit.

All welding materials are tested by the fabricator using the specific process(es) and the maximum welding energy inputs to be employed in production welding. These tests are in accordance with the requirements of ASME Section II, Material Specification, and, in addition, include delta ferrite determinations. The delta ferrite determinations are made by calculation using the "Schaeffler or Modified Schaeffler Constitution Diagram for Stainless Steel Weld Metal."

When subsequent in-process delta ferrite determinations are required, and since the welding material conformance is proved by the initial material testing described above, any of the recognized methods for measurement of delta ferrite is acceptable by mutual agreement. In these instances, sound welds (as determined by visual, penetrant and volumetric examinations) that display more than 1-percent-average delta ferrite content are considered to be unquestionably acceptable. All other sound welds are considered acceptable also, providing there is no evidence of deviation from qualified procedure parameters or use of malpractices. If evidence of the latter prevails, sampling for chemical and metallurgical analysis is required to determine the integrity and acceptability of the weld(s). The sample size is required to be 10 percent of the welds, but not less than 1 weld, in the particular component or system. If any of these weld samples are defective, that is, fail to pass bend tests as prescribed by ASME Section IX, or if the chemical analysis deviates from the material specification, then all remaining welds are sampled and all defective welds are removed and replaced.

All other applications use type 308 or type 316 which normally contain 3 to 15% delta ferrite and 1 to 5% delta ferrite in the deposit analyses, respectively. The successful experience with austenitic stainless steel welds for these applications, supplemented by nondestructive examination, provides assurance for avoiding microfissuring in welds.

The qualification of welding procedures is discussed in paragraph 5.2.5.5.

5.2.6 PUMP FLYWHEEL

The integrity of the reactor coolant pump flywheel is assumed on the basis of the following design and quality assurance procedures.[a]

5.2.6.1 Compliance with NRC Regulatory Guide 1.14

The calculated stresses at operating speed are based on stresses caused by centrifugal forces. The stress resulting from the interference fit of the flywheel on the shaft is < 2000 psi at 0 speed, but this stress becomes 0 at approximately 600 rpm because of radial expansion of the hub. The primary coolant pumps run at approximately 1190 rpm and may operate briefly at

a. Reactor coolant pump flywheel fatigue is evaluated as a TLLA for license renewal (see chapter 18, paragraph 18.4.2.3).

overspeeds up to 109% (1295 rpm) during loss of outside load. For conservatism, however, 125% of operating speed was selected as the design speed for the primary coolant pumps. The flywheels are given a preoperational test of 125% of the maximum synchronous speed of the motor.

The flywheel consists of two plates, approximately 5-in. and 8-in. thick, bolted together. Each plate is fabricated from electro-slag refined A-533 Grade B Class I steel . Supplier certification reports are available for all plates and demonstrate the acceptability of the flywheel material on the basis of the following requirements of NRC Regulatory Guide 1.14.

A. The nil-ductility transition (NDT) temperature of the flywheel material should be no higher than +10°F.

B. The Charpy V notch (C_v) energy level in both the parallel and normal orientation with respect to rolling direction of the material should be at least 50 ft-lb at the normal operation temperature of the flywheel.

A lower bound K_{ID} reference curve (see figure 5.2-11) has been constructed from dynamic fracture-toughness data generated in A533 Grade B Class I steel [8]. All data points are plotted on the temperature scale relative to the NDT temperature. The construction of the lower bound below which no single test point falls, combined with the use of dynamic data when flywheel loading is essentially static, together represent a large degree of conservatism. Reference of this curve to the guaranteed NDT temperature of +10°F indicates that, at the predicted flywheel operating temperature of 110°F, the minimum fracture toughness is in excess of 100 KSI-in$^{1/2}$. This conforms to NRC Regulatory Guide 1.14 requirement (6.1) that the dynamic stress intensity factor must be at least 100 KSI-in$^{1/2}$.

Flywheel blanks are flame-cut from the plate, with allowance for exclusion of heat affected material. The finished flywheels are subjected to 100-percent volumetric ultrasonic inspection. The finished machined bores are also subjected to magnetic particle or liquid penetrant examinations.

Precautionary measures taken to preclude missile formation from primary coolant pump components ensure that the pumps will not produce missiles under any anticipated accident condition. Each component of the primary pump motors has 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.

Thus it is concluded that flywheel plate materials are suitable for use and can meet NRC Regulatory Guide 1.14 acceptance criteria on the basis of suppliers certification data.

5.2.7 REACTOR COOLANT PRESSURE BOUNDARY (RCPB) LEAKAGE DETECTION SYSTEMS

5.2.7.1 Leakage Detection Methods

The reactor coolant leakage detection system provides the capability of detecting the presence of significant radioactive or nonradioactive leakage from the RCLs to the containment atmosphere during normal operation. Variations in the particulate activity, gaseous activity, and specific humidity of the containment atmosphere above a preset level give positive indications in the control room to the reactor operators. A leakage estimate is then made from either the functional variation during the transients or the new steady state. These leakage detection provisions are sufficiently sensitive so that small increases in leakage rates can be detected while the total leakage rate is still below a value consistent with safe operation of the plant.

Instrumentation is also provided to monitor pressure and flow conditions in auxiliary system lines penetrating the RCPB. Protection is also provided against possible overpressurization resulting from excessive check valve leakage, either by relief valves or by circuits permitting periodic tests. Provisions are also made to isolate the primary grade water within the containment should excessive intersystem leakages occur.

The particulate and gaseous activity are monitored by the containment air particulate and radiogas monitors. The specific humidity is monitored by the condensate measuring system and the dewpoint temperature system.

5.2.7.1.1 Systems Descriptions

The reactor coolant leakage detection system consists of the air particulate monitor, the radiogas monitor, condensate measuring devices, and humidity detectors.

A. Containment Air Particulate Monitor

This monitor takes continuous-flowing air samples from the containment atmosphere and measures the air particulate beta radioactivity. The samples are drawn outside the containment in a closed, sealed system and are monitored by a beta scintillation detector assembly. The fixed filter paper collects 99 percent of the particulate matter > 1.0 μ in size, which is viewed by a hermetically sealed combination photomultiplier tube. This monitor is series connected to the containment radioactive gas monitor and uses the pumping system common to both. This monitor has a measuring range of 10^{-12} to 10^{-6} μCi/cc.

The detector assembly is in a completely closed housing. The signal will be processed by the skid mounted microprocessor and will be transmitted to the radiation monitoring system cabinet in the control room. Lead shielding is provided to reduce the background radiation to a level where it does not interfere with the detector's sensitivity.

The activity is indicated on meters and monitored by the plant process computer. High activity alarm indications are displayed on the radiation monitoring cabinets. Local alarms provide operational status of supporting equipment such as pumps, motors, and flow and pressure controllers. The activity is indicated by a control and display module instead of a meter.

The sensitivity of the air particulate monitor to an increase in reactor coolant leak-rate is dependent upon the magnitude of the normal baseline leakage into the containment.

For cases where the baseline reactor coolant leakage falls with the detectable limits of the air particulate monitor, the instrument can be adjusted to alarm on leakage increase from 2-to-5 times the baseline value.

B. Containment Radioactive Gas Monitor

This monitor measures the gaseous beta radioactivity in the containment by taking the continuous air sample from the containment atmosphere. The sample first passes through the air particulate monitor where particulate matter is removed, and then through a closed, sealed system to a gas monitor assembly. After passing through the gas monitor, the gas sample is returned to the containment atmosphere.

Each sample is constantly mixed in fixed, shielded volumes, where it is viewed by photomultiplier tubes. This monitor has a measuring range of 10^{-12} to 10^{-3} µCi/cc.

The detector is in a completely enclosed housing containing a beta-sensitive photomultiplier tube mounted in a constant gas volume container. Lead shielding is provided to reduce the background radiation level to a point where it does not interfere with the detector's sensitivity.

The detector outputs are transmitted to the radiation monitoring system cabinets in the control room. The activity is indicated by a control and display module and monitored by the Analog Data Management System computer or the plant process computer. High activity alarm indications are displayed on the control board annunciator in addition to the radiation monitoring system cabinets. Local alarms annunciate the supporting equipment's operational status.

The air particulate and radiogas monitors have a pump unit common to both monitors.

The pump unit consists of:

1. A pump to obtain the air sample.

2. A digital mass flow indicator/controller to adjust and indicate the flow rate.

3. A flow-control valve to provide steady flow.

4. A flow-alarm assembly to provide low- and high-flow alarm signals.

The air particulate and radiogas monitors will be qualified to function following a safe shutdown earthquake as described in paragraph 11.4.2.2.3.

C. Specific Humidity Monitoring Devices

The containment specific humidity monitoring devices offer another means of detection of leakage into the containment. The devices, namely the condensate measuring system and the dewpoint monitors, are not as sensitive as the air particulate and the radiogas monitors, but have the advantage of being sensitive to vapor originating from all sources: the reactor coolant system, the steam system, and the feedwater system. Thus, these devices are able to detect leakage from nonradioactive or radioactive sources during the initial period of plant operation when the coolant activity may be low.

D. Condensate Measuring System

The condensate measuring system permits measurements of liquid runoff from the drain pans under each containment fan cooler unit. It consists of a vertical standpipe, valves, and standpipe level instrumentation installed in the drain piping of the reactor containment fan cooler unit.

The condensation from the containment coolers flows to the vertical standpipes. A differential pressure transmitter provides standpipe level signals. The system provides measurement capability of condensate runoff by monitoring standpipe level increase versus time.

Depending on the number of reactor containment fan cooler units in operation, the sum of the drainage flowrate from each operating cooler unit represents the total normal condensation. With the initiation of an additional or abnormal leak, the containment atmosphere humidity and condensation runoff rate will begin to increase, the water level will rise in the vertical pipe, and the high-condensate level alarm will be actuated.

The containment specific humidity will increase proportionally to time and leakage until the dewpoint is reached at the fan cooler units cooling coils. With the increasing specific humidity, the heat removal capacity needed to cool the air steam mixture to its dewpoint temperature decreases. Increases in specific humidity and available heat removal capacity from the cooling coils will result in added condensate flow. The condensate flowrate then is a function of specific humidity. Through accurate measurements of condensate level and dewpoint variations or RCS inventory (i.e., water inventory balance calculations), a reliable indication of the reactor coolant leakage rate can be made.

Detection of hot water leakage can be obtained from the condensate flow and dewpoint increase during the transient. A better estimate of leakage can be determined from the steady-state condensate flow when equilibrium has been reached. The device will alarm on a level equivalent to or below a condensate flowrate corresponding to a postulated 1.0 gal/min RCS leakage considering a flashing factor of approximately 40%.

E. Dewpoint Temperature Monitoring

The dewpoint measuring system consists of ten dew-cell elements. One element is located at the inlet and outlet of each of the containment fan cooler units and one each in the upper and lower compartments of the containment. The basis of operation of these elements is the behavior of a hygroscopic salt in the presence of water vapor.

When dry lithium chloride is exposed to the atmosphere under average room conditions, it will absorb moisture and dissolve, forming a salt solution. If this solution is heated, the water tends to escape back to the atmosphere. A state of equilibrium is reached at a temperature where the tendency of water to escape is equal to the tendency of the salt to absorb moisture. At this equilibrium point, the temperature of the salt and the saturated solution (temperature of the dew-cell element) is a measure of the partial pressure of the water vapor surrounding atmosphere, i.e., dewpoint temperature. The range of the dewpoint temperature measuring system is 50° to 130°F. Its accuracy is ±1°F.

Because of the slow response of containment atmosphere specific humidity for an abnormal increase device, dewpoint temperature recordings may prove useful in establishing the location and the history of the leak.

5.2.7.2 Indication in Control Room

Positive indications in the control room of leakage of coolant from the RCS to the containment are provided by equipment which permits continuous monitoring of containment air activity and humidity.

5.2.7.3 Limits for Reactor Coolant Leakage

The limits for reactor coolant leakage are delineated in the FNP Technical Specifications.

5.2.7.4 Unidentified Leakage

The total, normally expected leakage from the RCS is expected to be about 40 lb/day. The sensitivities and response times of subsystems are as follows:

5.2.7.4.1 Sensitivities of Leakage Detection Systems

The following system sensitivities are based upon discrete values of input parameters and assumptions as documented in Westinghouse WCAP-8009[31]. Actual performance may vary based on plant conditions.

A. Containment Air Particulate Monitor

The containment air particulate monitor is the most sensitive instrument available for detection of reactor coolant leakage into the containment. This instrument is capable of detecting particulate activity in concentration as low as 10^{-9} to 10^{-6} μc/cc of containment air sampled.

Leakage rates of about 0.01 gal/min to leaks > 10 gal/min can be observed, assuming specific values of corrosion-product activity and no fuel cladding damage. Assuming a corrosion-product activity (Fe, Mn, Co, Cr) of 0.4 μc/cc, a low, but detectable, background of containment air particulate activity, and complete dispersion of leaking radioactive solids into the air, leak rates of about 0.01 gal/min are detectable within 50 min after they occur. A 1.0 gal/min leak would be detectable within 0.5 min.

B. Containment Radioactive Gas Monitor

The containment radioactive gas monitor is inherently less sensitive (threshold at 10^{-6} μc/cc) than the containment air particulate monitor and would function in the event that significant reactor coolant gaseous activity exists because of fuel cladding defects. Assuming a reactor coolant gaseous activity of 22 μc/cc (corresponding to about 0.1 percent fuel defects), the occurrence of a leak of 1.0 gal/min would double a zero leakage background in approximately 40 min.

C. Condensate Measuring and Dewpoint Monitoring System

These systems provide indications that allow determination of leakage losses from water and steam systems within the containment. The condensate measuring system collects and measures the moisture condensed from the containment atmosphere onto the cooling coils of the containment cooling units. The dewpoint and condensate measuring system provide a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves. Condensate flows ≥ 0.1 gal/min can be identified by the condensate measuring system. Dewpoints can be observed to within 1°F. Leaks smaller than 1 gal/min can be measured by periodic observation of the level changes in the condensate collection system. If leakage is to another closed system, it will be detected by the plant radiation monitors and/or inventory control. For a condensate high-level alarm setpoint corresponding to 0.1 gal/min per cooler flowrate, the occurrence of a 1.0-gal/min leak would be detected within 1 h, assuming approximately 40 percent of the leakage enters the containment atmosphere as vapor.

5.2.7.4.2 Unidentified Leakage From Through-Wall Cracks

The 1-gal/min maximum permissible leakage rate from unidentified sources within the RCPB is well below the leakage rates calculated for critical through-wall cracks in pipes of 3-in. diameter and larger. The lengths of through-wall cracks that are calculated to leak 0.5- gal/min in 2-in. lines, 1 gal/min in 3-in. lines, and 2 gal/min in lines of 4-in. diameter and larger are given in reference 1. Included in this report are the ratios of critical through-wall cracks to computed lengths for these leakage values, as a function of pipe diameter and wall thickness based on the application of the principles of fracture mechanics, as well as the mathematical model and data used in the analyses.

Although the 1-gal/min maximum permissible unidentified leakage rate is larger than the 0.5-gal/min leakage rate analyzed for cracks in 2-in. lines, core cooling analyses have shown that for "small breaks," that is, for breaks up to the equivalent of the cross-sectional area of a 4-in.-diameter line, acceptable peak clad temperature results are obtained.

5.2.7.5 Maximum Allowable Total Leakage

The maximum allowable total leakage from the RCPB from other than controlled sources is 10 gpm. This leakage rate is approximately 10 percent of the makeup control system while in the automatic mode of operation. Normal background leakage (40 lb/day) does not influence this value significantly. Gross leakage or condensate overflow accumulates in the containment sump, which has a removal rate of 50 gal/min, a more than adequate capacity.

5.2.7.6 Differentiation Between Identified and Unidentified Leaks

The methods described in paragraph 5.2.7.1.1 will allow detection of RCPB leakages occurring within the containment. The location of specific leaks will in general have to be determined visually, although the systems that are indicating that a leak exists should aid in determining its source.

5.2.7.7 Sensitivity and Operability Tests

The air particulate and radiogas monitors are provided with their own test circuitry which tests electronics and the photomultiplier tube. The electronics test provides a precalibrated pulse signal that can be recorded. A remotely-operated long half-life radiation check source is provided with energy emission ranges similar to the radiation energy spectra being monitored. The source-strength is sufficient to cause approximately 30 percent of full-scale indication. These units can be tested at any time at the discretion of the operator. For the condensate measuring system, the level indicators will be calibrated prior to plant operation. ***[HISTORICAL][****A measured flow of water was provided to the standpipe to check the operation of the level indicators during preoperational plant testing.*

5.2.8 INSERVICE INSPECTION PROGRAM

During the design phase of the nuclear plant, consideration was given to the provision of access for performance of the examinations required by IS-261 of Section XI of the ASME Boiler and Pressure Vessel Code (Winter 1971 edition).]

5.2.8.1 Provisions for Access to Reactor Coolant System Boundary

The provision of adequate access was verified by a review of all the drawings applicable to the layout and arrangement of the Reactor Coolant and Associated Auxiliary Systems within the boundaries established in accordance with the requirements of IS-120.

The general design features of the nuclear plant reactor vessel, system layout, and other major primary coolant components to ensure compliance with the requirements of IS-141 and IS-142 are as follows (Specific provision to be made for inspection access in the design of the reactor vessel, system layout and other major primary coolant components also is listed):

A. All reactor internals are completely removable. The tools and storage space required to permit reactor internals removal for these inspections have been provided.

B. The reactor vessel shell in the core area is designed with a clean, uncluttered cylindrical inside surface to permit future positioning of test equipment without obstruction.

C. The reactor vessel cladding was improved in finish by grinding to the extent necessary to permit meaningful examination of the vessel welds and adjacent base metal in accordance with the code.

D. The cladding-to-base-metal interface was ultrasonically examined to ensure satisfactory bonding to allow the volumetric inspection of the vessel welds and base metal from the vessel inside surface.

E. The reactor closure head is stored in a dry condition on the operating deck during refueling, allowing direct access for inspection.

F. The insulation on the vessel closure and lower heads is removable, allowing access for the visual examination of head penetrations.

G. All reactor vessel studs, nuts, and washers are removed to dry storage during refueling, allowing inspection in parallel with refueling operations.

H. Access holes are provided in the core barrel flange, allowing access for the remote visual examination of the clad surface of the vessel without removal of the lower internals assembly.

I. Removable plugs are provided in the primary shield, providing access for the surface and visual examination of the primary nozzle safe-end welds.

J. Manways are provided in the steam generator channel head to provide access for internal inspection.

K. A manway is provided in the pressurizer top head to allow access for internal inspection.

L. The insulation covering all component and piping welds and adjacent base metal is designed for ease of removal and replacement in areas where external inspection will be planned.

M. Removable plugs are provided in the primary shield concrete above the main coolant pumps to permit removal of the pump motor to provide internal inspection access to the pumps.

N. The primary loop compartments are designed to allow personnel entry during refueling operations, to permit direct inspection access to the external portion of piping and components.

The use of conventional, nondestructive, volumetric test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and the remote underwater accessibility to this component.

As indicated above, the only sophisticated remote inspection equipment currently required is for inspection of the reactor vessel. The baseline inspection was performed by Westinghouse, utilizing a remote reactor vessel ultrasonic inspection tool to perform the code-required inspection of the circumferential and longitudinal shell welds, the flange-to-vessel weld, the ligaments between the flange holes, the nozzle-to-vessel welds, and the nozzle-to-safe-end-to-pipe welds. Because of access restrictions imposed by the location of the lower radial core support blocks, only 50 percent of the total length of the lower head-to-shell weld was examined from inside the vessel. The remainder of the weld was examined manually from the outside of the vessel.

5.2.8.2 **Equipment for Inservice Inspections**

The vessel inspection tool has two major components, the superstructure which holds the examination assembly and the examination assembly which delivers the various ultrasonic transducers to the desired work point in the vessel. Design of the tool permits precise positioning for accurate scanning of the examination volume. Reconfiguration of the examination assembly and transducers permits examination of the desired welds and components in the reactor vessel, e.g., circumferential shell welds, flange-to-shell welds, lower head welds, and nozzle examinations. Appropriate ultrasonic transducers are installed in the examination assembly to detect and size indications.

5.2.8.3 Recording and Comparing Data

For reactor vessel automated ultrasonic examinations, the data recording and positioning system is fully integrated with the vessel inspection tool to provide precise locations for state-of-the art sizing and characterizing of indications.

For manual ultrasonic examinations, such as the examination of a circumferential pipe weld, procedures are used to ensure that inspection results are recorded in such a manner which will avoid any ambiguity in interpretation. Procedures specify the location of weld reference points and the way in which indications must be recorded with respect to these reference points.

The data from various examinations is collected into a comprehensive report tabulating all of the results in sufficient detail to ensure repeatability for each examination.

[HISTORICAL] [5.2.8.4 Reactor Vessel Acceptance Standards

For Unit 1, the reactor vessel acceptance standards used during preoperational mapping of the vessel by ultrasonic examination met the requirements of IS-232 of the 1971 Edition through Winter 1971 Addenda of Section XI of the code. For Unit 2, the reactor vessel acceptance standards used during preoperational mapping of the vessel by ultrasonic examination met the requirements of IWB-2100 of the 1974 Edition through Summer 1975 Addenda of Section XI of the code.]

5.2.8.5 Coordination of Inspection Equipment With Access Provisions

The only areas where it is expected that high radiation levels will prohibit the access of personnel for direct examination of component areas or systems is the reactor vessel. The special design provisions and tooling required to perform the code-required examinations in these areas have been discussed above.

5.2.8.6 Preservice and Inservice Inspection and Inservice Testing Programs

[HISTORICAL] [5.2.8.6.1 Preservice Inspection Programs

The Unit 1 Preservice and Inservice Inspection Program was based on the requirements of the 1971 Edition through the Winter 1971 Addenda of the ASME Code, Section XI for ASME Code Class 1 Components and the 1971 Edition through the Winter 1972 Addenda of the ASME Code, Section XI for ASME Code Class 2 Components. Prior to Unit 1 initial plant startup, a program of scheduled preservice inspection in accordance with the above codes was conducted on ASME Code Class 1 & 2 components to the extent practical with the exception of those requiring a visual examination for evidence of leakage during the system hydrostatic test. This requirement was fulfilled by normal plant construction and startup activities.

In consideration of the requirements of 10 CFR 50.55a(g), the Unit 2 Preservice Inspection Program for ASME Code Class 1, 2, and 3 components was performed with the intent to meet, to the extent practical, the requirements of the 1974 Edition through the Summer 1975 Addenda of the ASME Code, Section XI. The Section XI requirement for visual examination for evidence of leakage during system pressure tests *for ASME Code Class 3 components was satisfied by the hydrostatic test requirements of Section III of the ASME Code. The Unit 2 Preservice Inspection Program is described below.*

A. Preservice Inspection Program for Class 1 Components

Table 5.2-33 provides a tabulation of the Class 1 pressure-retaining components (and their supports) subject to the inspection requirements of Subsection IWB of Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code, 1974 Edition through the Summer 1975 Addenda. These components were inspected in accordance with the requirements of Subsection IWB to the extent practical, with the exception of those requiring a visual examination for evidence of leakage during the system pressure test. This requirement was fulfilled by ASME Section III hydrostatic test requirements as allowed by ASME Section XI, paragraph IWA-5210(a).

This tabulation identifies the components which were inspected, the Section XI item and category, and the method of examination. Where relief from the inspection requirements of Subsection IWB was requested, information was provided which identified the applicable code requirements and justification for the relief requested. Table IWB-2600 items not applicable to Farley Nuclear Plant Unit 2 have also been listed and identified in the interest of completeness. Repairs were made in accordance with the requirements of Section XI.

B. Preservice Inspection Program for Class 2 Components

Table 5.2-34 provides a tabulation of the Class 2 pressure-retaining components (and their supports) subject to the inspection requirements of Subsection IWC of Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addenda. These components were inspected in accordance with the requirements of Subsection IWC to the extent practical, with the exception of those requiring a visual examination for evidence of leakage during the system pressure test. This requirement was fulfilled by ASME Section III hydrostatic test requirements as allowed by ASME Section XI, paragraph IWA-5210(a).

This tabulation identifies the components which were inspected, the Section XI item and category, area examined, and the method of examination. When relief from the inspection requirements of Subsection IWC was requested, information was provided which identified the applicable code requirements, justification for the relief requested, and the inspection method used as an alternative. Table IWC-2600 items not applicable to Farley Nuclear Plant Unit 2 have also been listed in the interest of completeness. Repairs were made in accordance with the requirements of Section XI. Article IWC-3000, entitled "Evaluation of Examination Results," was in the course of preparation by the code committee and was not available for use. Therefore, the rules of IWA-3000 were used]

5.2.8.6.2 Inservice Inspection Programs[a]

The Units 1 and 2 Inservice Inspection Programs have been established in accordance with 10 CFR 50.55a(g). The inservice inspections are performed in accordance with Section XI of the ASME Code with certain exceptions whenever specific written relief or alternative to ASME Code requirements are granted by the NRC. The First Ten-Year Inservice Inspection (ISI) Program for each unit was established to meet, to the extent practical, the requirements of the 1974 Edition through the Summer 1975 Addenda of the ASME Code Section XI. Following completion of the first 10-year interval for Unit 1, the Second Ten-Year ISI Program was established. Following completion of the second 10-year interval, the Third Ten-Year ISI Program is effective from December 1, 1997 through November 30, 2007. The Code of record for the third 10-year interval is the ASME Code, Section XI, 1989 Edition.

For Unit 2, by letter dated August 31, 1988, the NRC granted approval of an exemption from certain requirements of 10 CFR 50.55a, regarding the update of the ISI Program. Rather than requiring update of the Unit 2 ISI Program to the Code of record in effect on July 30, 1991, the NRC approved updating the program 3 years early, as provided by 10 CFR 50.55a(g)(4)(iv), in conjunction with the Unit 1 ISI Program which was previously updated to the ASME Code, Section XI, 1983 Edition through Summer 1983 Addenda. Exemption to the requirements of 10 CFR 50.55a(g)(4)(ii) extended the date of record by which the Unit 2 program is required to be updated from July 30, 1991, which marks the completion of the first 10 years of commercial operation of Unit 2, through November 30, 1997, the date which marks completion of the second 10 years of commercial operation for Unit 1. In this way, the Code of record in effect through November 30, 1997, for Unit 1—the ASME Code, Section XI, 1983 Edition through Summer 1983 Addenda—is also applicable to Unit 2.

The updated Unit 2 program went into effect in March 1989 during the Unit 2 sixth refueling outage and will continue through the third 40-month period of the first 10-year interval and remain in effect through the first and second 40-month periods of the second 10-year interval until December 1, 1997, the completion date for the second 10-year interval for Unit 1. At this time, a new updated Unit 2 program is in effect from December 1, 1997 through November 30, 2007. In this way, the Code of record in effect for the third 10-year interval for Unit 1—the ASME Code 1989 Edition—is also applicable to Unit 2.

[HISTORICAL] *[Inservice inspection of the metallic liner and the pressure retaining concrete structure of the containments of both units met the requirements of Subsection IWE and IWL of the 1992 edition with 1992 addenda of ASME Section XI as discussed in paragraph 3.8.1.7, Testing and Inservice Surveillance Requirements for the third interval.* ***]***

Beginning at the fourth ISI interval, inservice inspection of the metallic liner and the pressure retaining concrete structure of the containments of both units meet the requirements of Subsections IWE and IWL of the appropriate edition of ASME Section XI as described in the Containment Inspection Plan.

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a. The ISI Program is credited as a license renewal aging management program (see chapter 18, subsection 18.2.1).

Reactor vessel examinations in accordance with the First Ten-Year ISI Program for each unit included the Mandatory Appendix I requirements entitled "Ultrasonic Examination." Reactor vessel examinations in accordance with the Unit 1 Second Ten-Year ISI Program and the Unit 2 updated ISI Program are bound to Article 4 of Section V entitled "Ultrasonic Examination When Dimensioning of Indications is Required." Reactor vessel examinations performed under the Unit 1 Third Ten-Year ISI program and the new Unit 2 Updated Program (effective from December 1, 1997 through November 30, 2007) will be accomplished per the requirements of Appendix I, Article I-2100 of the 1989 Edition of ASME Section XI.

While maintaining these requirements as the technical basis of the examination programs, Units 1 and 2 comply with the Augmented Reactor Vessel Examination Program developed in response to NRC Generic Letter 83-15 and Regulatory Guide 1.150, Revision 1. This program for Units 1 and 2 was submitted by letter from F. L. Clayton, Jr. (APC) to S. A. Varga (NRC) of October 26, 1983.

5.2.8.6.3 Inservice Testing Programs

The Units 1 and 2 Inservice Testing (IST) Programs have been established in accordance with 10 CFR 50.55a(f). The First Ten-Year IST Program for each unit was established to meet, to the extent practical, the requirements of the 1974 Edition through the Summer 1975 Addenda of the ASME Code Section XI. Following completion of the first 10-year interval for Unit 1, the Second Ten-Year IST Program was established. Following completion of the second 10-year interval, the Third Ten-Year IST Program is effective from December 1, 1997, through November 30, 2007. FNP received approval to use the ASME OM Code - 1990 Edition as the Code of record for the third 10-year interval.

For Unit 2, by letter dated August 31, 1988, the NRC granted approval of and exemption from certain requirements of 10 CFR 50.55a regarding the requirements for updating the IST Program. Rather than requiring update of the Unit 2 IST Program to the Code of record in effect on July 30, 1991, the NRC approved updating the program 3 years early, as provided by 10 CFR 50.55a(f)(4)(iv), in conjunction with the Unit 1 IST Program which was previously updated to the ASME Code, Section XI, 1983 Edition through Summer 1983 Addenda. Exemption to the requirement of 10 CFR 50.55a(f)(4)(ii) extended the date of record by which the Unit 2 program is required to be updated from July 30, 1991, which marks the completion of the first 10 years of commercial operation of Unit 2 through November 30, 1997, the date which marks completion of the second 10 years of commercial operation for Unit 1. In this way, the Code of record in effect through November 30, 1997, for Unit 1-the ASME Code, Section XI, 1983 Edition through the Summer 1983 Addenda-is also applicable to Unit 2.

The updated Unit 2 program went into effect in March 1989 during the Unit 2 sixth refueling outage and will continue through the third 40-month period of the first 10-year interval and remain in effect through the first and second 40-month periods of the second 10-year interval until December 1, 1997, the completion date for the second 10-year interval for Unit 1. At this time, a new updated Unit 2 program is in effect from December 1, 1997 through November 30, 2007. In this way, the Code of record in effect for the third 10-year interval for Unit 1-ASME OM Code-1990 Edition is also applicable to Unit 2.

5.2.8.7 Ultrasonic Calibration Blocks

The ASME Boiler and Pressure Vessel Code, Sections V and XI, were used for the design ultrasonic calibration blocks as described in the Units 1 and 2 ISI Programs.

5.2.9 LOOSE PARTS MONITORING PROGRAM (METAL IMPACT MONITOR SYSTEM)

The metal impact monitor system in the Farley Nuclear Plant is designed to detect loose parts in the RCS. The developmental prototype of the Westinghouse metal impact monitor is installed in the R. E. Ginna Plant to evaluate the long-term performance of the system in an operating plant. The system consists of a detector, preamplifier, signal processor (with audio and record outputs), and display alarm. The system is a general maintenance aid and is not necessary for safe operation of the Farley Nuclear Plant.

Detector

The detectors are high temperature accelerometers mounted on each steam generator and on the reactor vessel.

Preamplifier

Preamplification of the detector signal is performed with a signal conditioning amplifier. This consists of a remote charge preamp and a signal conditioner. The remote charge preamp is located in close proximity to the accelerometer, on the outside of the primary system component. The signal conditioner is located in the MIMs cabinet outside of containment. These amplifiers are used to convert the low level accelerometer charge signal to a voltage signal for transmission to the signal processing equipment outside the containment.

Signal Processor and Display

The metal impact monitor was designed so that rate, as well as energy, of metal debris impact can be monitored continuously. Rate and amplitude latching-type alarms are displayed on the front panel of the monitor. Common alarm outputs are provided for connection to the main control room annunciator panel. An audio system produces the sound equivalent in parallel to the impact signal.

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TABLE 5.2-1

HARDSHIP EXCEPTIONS TO 10 CFR 50.55a

Component	As-constructed Code	NRC-required Code	Differences
Reactor coolant pumps (Unit 1)[a]	1968 Pump and Valve Code, March 1970 Addenda	ASME B & PV Code, Section III 1971 Edition	(1) Major defect mapping-1968 P & V; 1/5 of the casting thickness. 1971 B & PV; lesser of 10% of casting thickness or 3/8 in. (2) Hydrostatic test pressure. Pumps will be tested to 4100 psi instead of 4900 psi.[c]
Class I Valves	1968 Pump & Valve Codes plus Addenda	ASME B & PV Code, Section III 1971 Edition plus Summer 1971 Addenda	Major differences in formal documentation required.
Thermocouple Lead Appurtenances	(b)	ASME B & PV Code, Section III 1968 Edition plus all Addenda thru Summer 1970	Formal documentation requirements

Notes

a. The reactor coolant pumps for FNP Unit Number 2 will conform with ASME B & PV Code, Section III, 1971 Edition plus Summer 1972 Addenda.

b. Prior to the Summer 1970 Addenda of the 1968 Edition of the ASME B & PV Code Section III, no specific code requirements existed for the internals vessel appurtenances. In lieu of any formal code requirements, the internals vessel appurtenances were designed to meet the intent of the 1968 Edition of the ASME B & PV Code Section III.

c. Summer 1972 Addenda hydrostatic test pressure requirement is 3750 psi.

TABLE 5.2-2 (SHEET 1 OF 2)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Normal Conditions	Occurrences
Heatup and cooldown at 100°F/h (pressurizer cooldown 200°F/h)	200 (each)
Unit loading and unloading at 5 percent of full power/min	18,300 (each)
Step load increase and decrease of 10 percent full power	2,000 (each)
Large step load decrease, with steam dump	200
Steady-state fluctuations	Infinite
Upset Conditions	
Loss of load, without immediate turbine or reactor trip	80
Loss of power (blackout with natural circulation in the reactor coolant system)	40
Loss of flow (partial loss of flow one pump only)	80
Reactor trip from full power	400
Inadvertent auxiliary spray	10
One-half safe shutdown earthquake	5
Faulted Conditions[a]	
Main reactor coolant pipe break	1
Steam pipe break	1

TABLE 5.2-2 (SHEET 2 OF 2)

Test Conditions	Occurrences
Steam generator tube rupture	(included above in reactor trip from full power)
Safe shutdown earthquake	1
Turbine roll test	10
Hydrostatic test conditions	
Primary Side	5
Secondary side	10
Primary side leak test	50

a. In accordance with the ASME Nuclear Power Plant Components Code, faulted conditions are not included in fatigue evaluations.

TABLE 5.2-2a

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/hr and 200 cooldown cycles at < 100°F/hr	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F
	200 pressurizer cooldown cycles at ≤ 200°F/hr	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F
	80 loss of load cycles, without immediate turbine or reactor trip	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER
	40 cycles of loss of offsite AC electrical power	Loss of offsite AC electrical ESF electrical system
	80 cycles of loss of flow in one reactor coolant loop	Loss of only one reactor coolant pump
	400 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	10 inadvertent auxiliary spray actuation cycles	Spray water temperature differential > 320°F
	50 leak tests	Pressurized to ≥ 2485 psig
	5 hydrostatic pressure tests	Pressurized to ≥ 3100 psig
Secondary System	1 steam line break	Break in a > 6-inch steam line
	10 hydrostatic pressure tests	Pressurized to ≥ 1356 psig

TABLE 5.2-3

LOAD COMBINATIONS AND OPERATING CONDITIONS

Load Combination	Operating Condition
Normal condition transients, deadweight	Normal condition
Upset condition transients, deadweight, 1/2 SSE	Upset condition
Faulted condition transients, deadweight, SSE, or SSE and pipe rupture loads	Faulted condition

TABLE 5.2-4 (SHEET 1 OF 2)

LOADING CONDITIONS AND STRESS LIMITS: CLASS 1 COMPONENTS

Loading Conditions[a]	Stress Intensity Limits	Note
Normal	(a) $P_m \leq S_m$	
	(b) $P_L \leq 1.5\ s_m$	
	(c) P_m (or P_L) + $P_B \leq 1.5\ S_m$	1
	(d) P_m (or P_L) + P_B + $Q \leq 3.0\ S_m$	2
Upset condition	(a) $P_m \leq S_m$	
	(b) $P_L \leq 1.5\ S_m$	
	(c) P_m (or P_L) + $P_B \leq 1.5\ S_m$	1
	(d) P_m (or P_L) + P_B + $Q \leq 3.0\ S_m$	2
Faulted condition	Faulted condition limits in table 5.2-6	

P_m = primary general membrane stress intensity.

P_L = primary local membrane stress intensity.

P_B = primary bending stress intensity.

Q = secondary stress intensity.

S_m = stress intensity value from ASME B&PV Code, Section III, Nuclear Vessels.

S_y = minimum specified material yield (ASME B&PV Code, Section III, Table N-421 or equivalent).

a. Emergency condition is not included since none have been specified.

TABLE 5.2-4 (SHEET 2 OF 2)

NOTES FOR TABLE 5.2-4

Note 1: The limits on local membrane stress intensity ($P \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) + $P_\beta \leq 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings do not exceed 2/3 of the lower bound collapse load as per paragraph N-417.6(b) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 2: In lieu of satisfying the specific requirements for the local membrane ($P_L \leq 1.5S_m$) or the primary plus secondary stress intensity (P_m(or P_L) + P_β + $Q \leq 3S_m$) at a specific location, the structural action may be calculated on a plastic basis and the design will be considered to be acceptable if shakedown occurs, as opposed to continuing deformation, and if the deformations which occur prior to shakedown do not exceed specified limits, as per paragraph N-417.6(a) (2) of the ASME B&PV Code, Section III, Nuclear Vessels.

TABLE 5.2-5

LOADING CONDITIONS AND STRESS LIMITS: NUCLEAR POWER PIPING

Loading Conditions[b]	Stress Intensity Limits[a]
Normal condition	(a) $P_m \leq S_m$ (b) $P_L \leq 1.5\ S_m$ (c) P_m (or P_L) + $P_\beta \leq 1.5\ S_m$ (d) P_m (or P_L) + P_β + P_e + $Q \leq 3.0\ S_m$ (e) $P_e \leq 3.0\ S_m$
Upset condition	(a) $P_m \leq S_m$ (b) $P_L \leq 1.5\ S_m$ (c) P_m (or P_L) + $P_\beta \leq 1.5\ S_m$ (d) P_m (or P_L) + P_β + P_e + $Q \leq 3.0\ S_m$ (e) $P_e \leq 3.0\ S_m$
Faulted condition	Faulted condition limits are shown in table 5.2-6.

P_m = primary general membrane stress intensity.

P_L = primary local membrane stress intensity.

P_B = primary bending stress intensity.

P_e = secondary expansion stress intensity.

Q = secondary membrane plus bending stress intensity.

S_m = allowable stress intensity from ASME Boiler & Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1971 (or 1977 Edition through Summer 1979 Addenda).

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a. Alternatively, the rules and simplified analysis of sub sub articles NB-3640 and NB-3650 of ASME B&PV Code, Section III, Nuclear Power Plant Components, 1971 (or 1977 Edition through Summer 1979 Addenda), may be used in lieu of the stated equations.

b. Emergency condition is not included since none have been specified.

TABLE 5.2-6

FAULTED CONDITION STRESS LIMITS FOR CLASS 1 COMPONENTS

System (or Subsystem) Analysis	Components Analysis	Stress Limits for Components		Test
		P_m	$P_m + P_b$	
ELASTIC	Elastic	Smaller of 2.4 S_m and 0.70 S_u	Smaller of 3.6 S_m and 1.05 S_u Note (b)	
	Plastic	Larger of 0.70 S_u or S_y 1/3(S_u - S_y) Note (c)	Larger of 0.70 S_{ut} or S_y + 1/3 (S_{ut} - S_y) Note (c)	
	Limit Analysis	0.9 L_1 Notes (a and c)		0.8 L_T
PLASTIC	Plastic Elastic	Larger of 0.70 S_U or S + 1/3 (S_u - S_y)	Larger of 0.70 S_{ut} or S + 1/3 (S_{ut} - S_y)	Notes (c and d)

Notes:

a. L_1 = Lower bound limit load with an assumed yield point equal to 2.3 S_m .

b. These limits are based on a bending shape factor of 1.5 for simple bending cases with different shape factors, the limits will be changed proportionally.

c. When elastic system analysis is performed, the effect of component deformation on the dynamic system response should be checked.

d. L_T = The limits established for the analysis need not be satisfied if it can be shown from the test of a prototype or model that the specified loads (dynamic or static equivalent) do not exceed 80 percent of L_T, where L_T is the ultimate load or load combination used in the test. In using this method, account should be taken of the size effect and dimensional tolerances similitude relationships) which may exist between the actual component and the tested models to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading for faulted conditions.

S_y = Yield stress at temperature.
S_u = Ultimate stress from engineering stress-strain curve at temperature.
S_u = Ultimate stress from true stress-strain curve at temperature.
S_m = Stress intensity from ASME Section III at temperature.

TABLE 5.2-7

ALLOWABLE STRESSES FOR PRIMARY EQUIPMENT SUPPORTS

Loading Conditions	Stress Limits
Normal	AISC, Seventh Edition[a], Part 1, Allowable Stresses
Upset	AISC, Seventh Edition Part 1, Allowable Stresses
Faulted	Stresses ≤ yield strength of material. Local yielding is permitted but limited so that the structural integrity of the system is maintained. As an alternative to the above, 80 percent of L_T (see table 5.2-6) may be used.

a. Specifications for the design, fabrication and erection of structural steel for buildings.

TABLE 5.2-8 (SHEET 1 OF 2)

ACTIVE AND INACTIVE[c] VALVES IN THE REACTOR COOLANT SYSTEM PRESSURE BOUNDARY

System	Location	Line	Type	Size	Actuation Type	Classification A-Active I-Inactive	Environmental Design Criteria[b]
RCS	8010 A,B, C	Pressurizer safety (to PRT)	Safety	6-in.	System pressure (over set point)	A	(Internal fluid characteristics specified)
RCS	0460	Letdown	Globe	3-in.	Air-operated	A	1,2
RCS	0459	Letdown	Globe	3-in.	Air-operated	A	1,2
CVCS	8378	Charging	Check	3-in.	Δp	A	2,3
CVCS	8347	Charging	Check	3-in.	Δp	A	2,3
CVCS	8153, 8154	Excess letdown	Globe	1-in.	Air-operated	A [a]	1,2
CVCS	8377	Aux. spray	Check	2-in.	Δp	A [a]	1,2
CVCS	8145	Aux. Spray	Globe	2-in.	Air-operated	A [a]	1,2
SIS	8998 A,B, C	SIS injection	Check	6-in.	Δp	A	2,3
SIS	8973 A,B, C	RHR supply	Check	6-in.	Δp	A	2,3
SIS	8948 A, B, C	Accumulator disch. to C.L.	Check	12-in.	Δp	A	2, 3
SIS	8956 A, B, C	Accumulator disch. to C.L.	Check	12-in.	Δp	A	2, 3

TABLE 5.2-8 (SHEET 2 OF 2)

System	Location Line		Type	Size	Actuation Type	Classification A-Active I-Inactive	Environmental Design Criteria[b]
SIS	8998 A, B, C	Cold leg LHSI	Check	6-in.	Δp	A	2, 3
SIS	8997 A, B, C	Cold leg HHSI	Check	2-in.	Δp	A	2, 3
SIS	8993 A, B, C	Hot leg conn.	Check	6-in.	Δp	A	2, 3
SIS	8988 A, B	Hot leg conn.	Check	6-in.	Δp	A	2, 3
CVCS	8346	Alternate charging	Check	3-in.	Δp	A	2, 3
CVCS	8348 A, B, C 8367 A, B, C	RCP Seal injection	Check	2-in.	Δp	A	2, 3
CVCS	8379	Alternate charging	Check	3-in.	Δp	A	2, 3
SIS	8990 A, B, C 8992 A, B, C 8995 A, B, C	HHSI Hot leg HHSI Hot leg HHSI Cold leg.	Check	2-in.	Δp	A	2, 3
WDS	8057 A, B, C 8058 A, B, C	RCDT Drain	Isolation	2-in.	Manual	I	2, 3

a. There is a possibility that these valves may be open when an accident occurs.

b. Environmental Design Criteria
1. Ambient Temperature: 50°-150°F
2. Ambient Atmosphere: 8-15 psia, 100 percent Relative Humidity, 50 R/hr - Gamma Radiation
3. Ambient Temperature: 120°-150°F

c. All other valves in this Reactor Coolant Pressure Boundary are considered inactive and are shown on FSAR project drawings D-175037 Sh. 1, D-175037 Sh. 2, D-175037 Sh. 3, D-205037 Sh. 1, D-205037 Sh. 2, and D-205037 Sh. 3.

TABLE 5.2-9

STRESSES CAUSED BY MAXIMUM STEAM GENERATOR TUBESHEET PRESSURE DIFFERENTIAL (2485 PSIG)

THIS TABLE HAS BEEN DELETED.

TABLE 5.2-10

STEAM GENERATOR PRIMARY-SECONDARY BOUNDARY COMPONENTS[b]

THIS TABLE HAS BEEN DELETED.

TABLE 5.2-11

STEAM GENERATOR
PRIMARY-SECONDARY BOUNDARY COMPONENTS
Condition: Primary Hydrotest - 3107/0 psig

THIS TABLE HAS BEEN DELETED.

TABLE 5.2-12

STEAM GENERATOR PRIMARY-SECONDARY BOUNDARY COMPONENTS

THIS TABLE HAS BEEN DELETED.

TABLE 5.2-13

STEAM GENERATOR
PRIMARY-SECONDARY BOUNDARY COMPONENTS[a] [b]
Condition: Loss of Secondary Pressure (Steam Line Break)
Faulted Condition 2485/0 psig, 660°F

THIS TABLE HAS BEEN DELETED.

TABLE 5.2-14

**51,500 SQ FT STEAM GENERATOR USAGE FACTORS (INDIVIDUAL TRANSIENTS)
PRIMARY AND SECONDARY BOUNDARY COMPONENTS[(b)]**

THIS TABLE HAS BEEN DELETED.

TABLE 5.2-15

51,500 SQ FT STEAM GENERATOR USAGE FACTORS (INDIVIDUAL TRANSIENTS) CENTER OF TUBESHEET[(a)]

THIS TABLE HAS BEEN DELETED.

TABLE 5.2-16

TUBESHEET STRESS ANALYSIS RESULTS FOR 51,500 SQ FT STEAM GENERATORS[a]

THIS TABLE HAS BEEN DELETED.

TABLE 5.2-17

**LIMIT ANALYSIS CALCULATION RESULTS
TABLES OF STRAINS, LIMIT PRESSURES,
AND FATIGUE EVALUATIONS
FOR 51,500 SQ FT STEAM GENERATORS**

THIS TABLE HAS BEEN DELETED.

TABLE 5.2-18

RELIEF VALVE DISCHARGE TO THE PRESSURIZER RELIEF TANK

Reactor Coolant System		
3	Pressurizer safety valves	D-175037 Sh.2 (Unit 1)
2	Pressurizer power-operated relief valves	D-205037 Sh.2 (Unit 2)
Safety Injection System		
1	SIS discharge to hot leg	D-175038 Sh.2 (Unit 1)
2	SIS discharge to cold legs	D-205038 Sh.2 (Unit 2)
Residual Heat Removal System		
2	RHR pump suction line from RCS hot legs	D-175041 Sh.1 (Unit 1) D-205041 Sh.1 (Unit 2)
Chemical and Volume Control System		
2	Charging pump suction	D-175039 Sh.6 (Unit 1) D-205039 Sh.2 (Unit 2)
1	Seal-water return line	D-175039 Sh.1 (Unit 1) D-205039 Sh.1 (Unit 2)
1	Letdown line	D-175039 Sh.1 (Unit 1) D-205039 Sh.1 (Unit 2)

TABLE 5.2-19

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS (PSIG)

Hydrostatic test pressure (cold)	3107
Design pressure	2485
Safety valves open	2485
High pressure reactor trip	2385
Power relief valves open	2335
High controller output alarm	100 psig + controller setpoint (nominal 2335)
High pressure alarm	2310
Proportional spray full on	2310
Pressurizer spray valve begin to open	2260
Proportional spray off	2260
Proportional heaters off	2250
Design nominal operating	2235
Proportional heaters full on	2220
Backup heaters on	2210
Low pressure alarm	2185
P11 interlock	2000
Low pressure reactor trip	1865
Pressurizer level and pressure coincidence	1850

TABLE 5.2-20 (SHEET 1 OF 3)

REACTOR COOLANT SYSTEM BOUNDARY MATERIALS
CLASS 1 PRIMARY COMPONENTS

Reactor Vessel Component	
Shell (other than core region)	SA-533 B, Class 1 (vacuum treated)
Shell plates (core region)	SA-533 B, Class 1 (vacuum treated)
Head forging	SA-508 Grade 3, Class 1
Shell, flange, and nozzle forgings nozzle safe ends	SA-508 Class 2 SA-182 Type F316
CRDM, Instrumentation port and RVLIS head adapters and vent pipe (lower part)	SB-167 UNS No. 6690
RVLIS and instrumentation port housings	SA-182, F316
Vent pipe (upper part)	SA-312, Type 316
Instrumentation tube appurtenances - lower head	SB-166 or -167 and SA-182 Type F304, F304L, or F316
Closure studs	SA-540 Class 3 Gr B23 or B24
Closure nuts	SA-540 Class 3 Gr B23 or B24
Closure washers	SA-540 Class 3 Gr B23 or B24
Core support pads	SB-166 with carbon less than 0.10%
Vessel supports, seal ledge	SA-516 Gr 70 quenched and tempered or SA-533 Gr A, B, or C. (Vessel supports may be of weld metal buildup of equivalent strength.)
Head lifting lugs	SA-533, Type B, Class 1
Steam Generator Components	
Pressure forgings	SA-508 Class 3 or 3a
Nozzle safe ends	SA-336 Class F Type 316LN

TABLE 5.2-20 (SHEET 2 OF 3)

Tubes	SB163 Ni-Cr-Fe, annealed
Closure bolting and studs	SA193 Gr B-7
Closure nuts	SA194 Gr 7
<u>Pressurizer Components</u>	
Pressure plates	SA533 Gr A, Class 2
Pressure forgings	SA508 Class 2 or 3
Nozzle safe ends	SA182 or 376 Type 316 or 316L and Ni-Cr-Fe Weld Metal F-Number 43
Closure bolting	SA193 Gr B-7
Pressurizer safety valve forgings	SA182 Type F316
<u>Reactor Coolant Pump</u>	
Pressure forgings	SA182 Type F304, F316 or F348
Pressure castings	SA351 Gr CF8, CF8A or CF8M
Tube and Pipe	SA213, SA376 or SA312 - Seamless Type 304 or 316
Pressure plates	SA240 Type 304 or 316
Bar material	SA479 Type 304 or 316
Closure bolting	SA193 Gr B7 or B8 or, SA540 Gr B23 or B24 or SA453 Gr 660
<u>Reactor Coolant Piping</u>	
Reactor coolant pipe	Code Case 1423-1 Gr F304N or 316N, or SA351 Gr CF8A or CF8M centrifugal castings
Reactor coolant fittings	SA351 Gr CF8A or CF8M

TABLE 5.2-20 (SHEET 3 OF 3)

Branch nozzles	SA182 Gr F304 or 316 or Code Case 1423-1 Gr F304N or 316N
Surge line and loop bypass	SA-376 Type 304 or 316 or Code Case 1423-1 Gr F304N or 316N
Auxiliary piping 1/2 in. through12 in. and wall schedules 40S through 80S (ahead of second isolation valve)	ANSI B36.19
All other auxiliary piping(ahead of second isolation valve)	ANSI B36.10
Socket weld fittings	ANSI B16.11
Piping flanges	ANSI B16.5
Welding materials	SFA 5.4 and 5.9 Type 308 or 308L
<u>Control Rod Drive Mechanism</u>	
Pressure housing	SA-182 Gr F316
Pressure forgings	SA-182 Gr F316
Bar material	SA-479 Type 304
Welding materials	SFA 5.9 Type 316L

TABLE 5.2-21 (SHEET 1 OF 3)

TYPICAL REACTOR COOLANT SYSTEM BOUNDARY MATERIALS AUXILIARY COMPONENTS

<u>Motor- and Manual-Operated Gate and Check Valves</u>	
Bodies	SA182 GR F316
Bonnets	SA182 Gr F316
Discs	SA182 Gr F316, SA351 Gr CF8M
Stems	SA638 Gr 660 Type 1 or B637 UNS N07718 (Inconel 718)
Closure bolts and nuts	SA453 Gr 660 and SA194 Gr B6
<u>Air-Operated Valves</u>	
Bodies	SA182 Type F316 or SA351 Gr CF8 or CF8M
Bonnets	SA182 Type F316 or SA351 Gr CF8 or CF8M
Discs	SA182 Type F316 or SA564 Gr630 Cond 1100 F heat treatment
Stems	SA182 Type F316 or SA564 Gr630 Cond 1100 F heat treatment
Closure bolts and nuts	SA453 Gr 660 and SA194 Gr B6
<u>Auxiliary Relief Valves</u>	
Forging	SA182 Type F316
Disc	SA479 Type 316
Miscellaneous valves (2-in. and smaller)	
Bodies	SA479 Type 316, SA351 Gr CF8 or SA182 Gr F316
Bonnets	SA479 Type 316 or SA351 Gr CF8
Discs	SA479 Type 316

TABLE 5.2-21 (SHEET 2 OF 3)

Stems	SA479 Type 410 or Type 304, SA276 Type 4
Closure bolts and nuts	SA453 Gr 660 and SA193 Gr B6
Auxiliary Heat Exchangers	
Heads	SA182 Gr F304 or SA240 Type 304 or 316
Flanges	SA182 G4 F304 or F316
Flange necks	SA182 Gr F304 or SA240 Type 316 or SA312 Type 304 Seamless
Tubes	SA213 TP304
Tubesheets	SA240 Type 304 or 316 or SA182 Gr F304 or SA105 Gr 2, or SA515 Gr 70 with Stainless Steel Weld Metal Analysis A-7 Cladding
Shells	SA351 Gr CF8
Pipe	SA312 Type 304 seamless & SA312 Type 316
Auxiliary Pressure Vessels Tanks, Filters, etc.	
Shells and heads	SA240 Type 304 or SA264 Type 304 Clad to SA516 Gr 70 or SA516 Gr 70 with Stainless Steel Weld Metal Analysis A-7 Cladding
Flanges and nozzles	SA182 Gr F304 and SA105 or SA350 Gr LF2 with Stainless Steel Weld Metal Analysis A-7 Cladding, SA312 Type 304 seamless
Piping	SA312 TP304 or TP316 seamless, SA312 Type 304 Welded
Pipe fittings	SA403 WP304 seamless
Closure bolts and nuts	SA193 Gr B7 or B8 and SA194 Gr 2H

TABLE 5.2-21 (SHEET 3 OF 3)

Auxiliary Pumps

Pump casings and heads	SA351 Gr CF8 or CF8M, SA182 Gr F304 or F316
Flanges and nozzles	SA182 Gr F304 or F316, SA403 Gr WP316L seamless
Piping	SA312 TP304 or TP316 seamless
Stuffing or packing box cover	SA351 Gr CF8 or CF8M, SA240 TP304 or TP316
Pipe fittings	SA403 Gr WP316L seamless
Closure bolts and nuts	SA193 Gr B6, B7 or B8M and SA194 Gr 2H or Gr 8M

TABLE 5.2-22

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical conductivity	Determined by the concentration of boric acid and alkali present.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to10.5 (low boric acid concentration) at 25°C.
Oxygen, ppm maximum	Oxygen concentration of the reactor coolant is maintained below 0.1 ppm for plant operation above 250°F. Hydrazine may be used to chemically scavenge oxygen during heatup.
Chloride, ppm, maximum	0.15
Fluoride, ppm, maximum	0.15
Hydrogen, cc(STP)/kg H_2O	25-50 (power operation)[a]
Total suspended solids, ppm, maximum	1.0
pH control agent ($Li^7 0H$) (ppm Li)	0.20 - 4.36 (power operation)
Boric acid, ppm B	Variable from 0 to approximately 2500

a. Hydrogen concentration during transients (including preparation for shutdown, plant restart, etc.) is controlled per plant procedures based on OEM (Westinghouse) recommendations.

TABLE 5.2-23

MATERIALS FOR REACTOR VESSEL INTERNALS FOR EMERGENCY CORE COOLING

Forgings	SA182 Type F304
Plates	SA240 Type 304
Pipes	SA312 type 304 seamless or SA376 Type 304
Tubes	SA213 Type 304
Bars	SA479 type 304 & 410
Castings	SA351 Gr CF8 or CF8A
Bolting	SA(Pending)Westinghouse PE Spec. 70041EA
Nuts	SA193 Gr B-8
Locking devices	SA479 type 304
Weld buttering	Stainless steel weld metal analysis A-7

TABLE 5.2-24

UNIT 1 REACTOR VESSEL TOUGHNESS PROPERTIES

Component	Code No.	Material Type	Cu (%)	P (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Upper Shelf Energy MWD[b]	NMWD[c]
Replacement closure head	02W79-1-1[f]	SA-508, Gr. 3, CL.1	0.05	0.005	0.76	-50	-50	-	212
Vessel flange	B6913-1	A508, CL.2	0.17	0.011	0.69	60[a]	60[a]	106[a]	-
Inlet nozzle	B6917-1	A508, CL.2	-	0.010	0.83	-18[e]	-18[e]	-	110
Inlet nozzle	B6917-2	A508, CL.2	-	0.008	0.80	29[e]	29[e]	-	80
Inlet nozzle	B6917-3	A508, CL.2	-	0.008	0.87	-48[e]	-48[e]	-	98
Outlet nozzle	B6916-1	A508, CL.2	-	0.007	0.77	-17[e]	-17[e]	-	96.5
Outlet nozzle	B6916-2	A508, CL.2	-	0.011	0.78	-29[e]	-29[e]	-	97.5
Outlet nozzle	B6916-3	A508, CL.2	-	0.009	0.78	-23[e]	-23[e]	-	100
Nozzle shell	B6914-1	A508, CL.2	0.16	0.010	0.684	30	30[a]	148	95.3
Inter. shell	B6903-2	A533,B,CL.1	0.13	0.011	0.60	0	0	151.5	99
Inter. shell	B6903-3	A533,B,CL.1	0.12	0.014	0.56	10	10	134.5	87
Lower shell	B6919-1	A533,B,CL.1	0.14	0.015	0.55	-20	15	133	86
Lower shell	B6919-2	A533,B,CL.1	0.14	0.015	0.56	-10	5	134	86
Bottom head ring	B6912-1	A508, CL.2	-	0.010	0.72	10	10[a]	163.5	-
Bottom head segment	B6906-1	A533,B,CL.1	0.15	0.011	0.52	-30	-30[a]	147	-
Bottom head dome	B6907-1	A533,B,CL.1	0.17	0.014	0.60	-30	-30[a]	143.5	-
Inlet/Outlet Nozzle to Nozzle shell seams (1-897 A → F)	Multiple[g]	Shielded Metal Arc Weld	0.04	-	1.08	-	10	-	73
Nozzle to inter. weld seam (10-894)	90099[f]	Sub Arc Weld	0.197	-	0.06	-	-56[d]	-	82.5
Inter. shell long. weld seam (19-894A&B)	M1.33	Sub Arc Weld	0.258	0.017	0.165	0[a]	-56[d]	-	149
Inter. to lower weld seams (11-894)	G1.18	Sub Arc Weld	0.205	0.011	0.105	0[a]	-56[d]	-	104
Lower shell long. weld seams (20-894A&B)	G1.08	Sub Arc Weld	0.197	0.022	0.060	0[a]	-56[d]	-	82.5

(a) Estimate per NUREG-0800 "USNRC Standard Review Plan" Branch Technical Position (BTP) MTEB 5-2. Note, the methodology in MTEB 5-2 is equivalent to BTP 5-3.
(b) Major working direction.
(c) Normal to major working direction.
(d) Estimate per 10 CFR 50.61.
(e) Estimate per BWRVIP-173-A, "BWR Vessel and Internals Project: Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials."
(f) Material identified by heat number.
(g) The inlet/outlet nozzle to nozzle shell welds were made with a multitude of weld metal heats. The most limiting Cu and Ni wt. %, RT_{NDT}, and USE values across all heats were chosen for conservatism.

TABLE 5.2-25 (SHEET 1 OF 2)

UNIT 2 REACTOR VESSEL TOUGHNESS DATA

Component	Code No.	Grade	Cu (%)	P (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
Replacement CL. HD	03W108-1-1[d]	SA-508, Gr. 3, CL.1	0.06	0.004	0.84	-60	-60	197	-
CL. HD. Dome	B7215-1	A533,B,CL.1	0.17	0.010	0.49	-30	16[a]	83[a]	128
CL. HD. Flange	B7207-1	A508,CL.2	0.14	0.011	0.65	60[a]	60[a]	>56[a]	>86[c]
VES. Flange	B7206-1	A508,CL.2	0.10	0.012	0.67	60[a]	60[a]	>71[a]	>109
Inlet Noz.	B7218-2	A508,CL.2	-	0.010	0.68	-55[b]	-55[b]	103[a]	158
Inlet Noz.	B7218-1	A508,CL.2	-	0.010	0.71	-55[b]	-55[b]	112[a]	172
Inlet Noz.	B7218-3	A508,CL.2	-	0.010	0.72	-60[b]	-60[b]	98[a]	150
Outlet Noz.	B7217-1	A508,CL.2	-	0.010	0.73	-47[b]	-47[b]	100[a]	154
Outlet Noz.	B7217-2	A508,CL.2	-	0.010	0.72	-71[b]	-71[b]	108[a]	167
Outlet Noz.	B7217-3	A508,CL.2	-	0.010	0.72	-43[b]	-43[b]	103[a]	158
Upper Shell	B7216-1	A508,CL.2	0.16	0.010	0.724	30	30[a]	96.2[a]	149
Inter Shell	B7203-1	A533,B,CL.1	0.14	0.010	0.60	-40	15	100	140
Inter Shell	B7212-1	A533,B,CL.1	0.20	0.018	0.60	-30	-10	100	134
Lower Shell	B7210-1	A533,B,CL.1	0.13	0.010	0.56	-40	18	103	128
Lower Shell	B7210-2	A533,B,CL.1	0.14	0.015	0.57	-30	10[d]	99	145
Trans. Ring	B7208-1	A508,CL.2	-	0.010	0.73	40	40[a]	89[a]	137
Bot. HD. Dome	B7214-1	A533,B,CL.1	0.11	0.007	0.48	-30	-2[a]	87[a]	134
Inlet/Outlet Nozzle to Upper Shell (1-926 A → F)	Multiple [c]	SMAW	0.07	-	1.04	-	10	97	-

TABLE 5.2-25 (SHEET 2 OF 2)

Component	Code No.	Grade	Cu (%)	P (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
Upper Shell to Inter Shell (10-923)	5P5622[d]	SAW	0.153	0.016	0.077	-40	-40	102	-
Inter. Shell Long Seam (19-923A)	A1.46	SMAW	0.027	0.009	0.947	0(a)	-56[d]	>131	-
Inter Shell Long Seam (19-923A&B)	A1.40	SMAW	0.027	0.010	0.913	-60	-60	>106	-
Inter Shell to Lower Shell(11-923)	G1.50	SAW	0.153	0.016	0.077	-40	-40	>102	-
Lower Shell Long Seams(20-923A&B)	G1.39	SAW	0.051	0.006	0.096	-70	-70	>126	-

(a) Estimate per NUREG 0800 "USNRC Standard Review Plan" Branch Technical Position MTEB 5-2.
(b) Estimated.
(c) Upper Shelf not available, value represents minimum energy at the highest test temperature.
(d) Estimate per 10 CFR 50.61.

TABLE 5.2-26

FAULTED CONDITION LOADS FOR THE REACTOR COOLANT PUMP FOOT

	F (kips)	F (kips)	F (kips)	M (kips)	M (kips)	M (kips)
Umbrella Loads	±2605	±3305	±3340	±7050	±7050	±4010
Faulted 1[a]	834	162	1334	2001	6023	337
Faulted 2[a]	601	711	752	3682	2657	560
Faulted 3[a]	876	170	1804	2859	7021	442
Ratio between umbrella loads and actual loads for the faulted condition						
Case 1	3.12	20.40	2.50	3.52	1.17	11.90
Case 2	4.33	4.65	4.44	1.91	2.65	7.16
Case 3	2.97	19.44	1.85	2.47	1.00	8.97

a. These faulted loads on the pump support feet are derived from both the pump tie rod and the support column loads. At a particular foot, the maximums from the tie rods and the columns are combined absolutely, although the time history LOCA analysis demonstrates clearly that the maxima from the columns and tie rods do not occur at the same time-point. A time history combination of the column and tie rod loads on a particular foot would significantly reduce these loads.

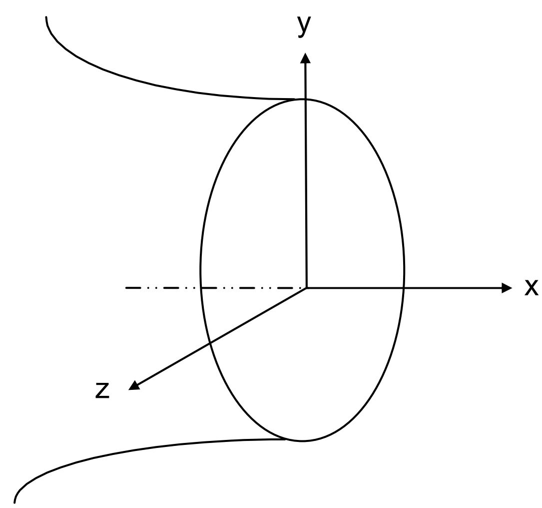
TABLE 5.2-27

REACTOR COOLANT PUMP OUTLET NOZZLE FAULTED CONDITION LOADS

	F_x	F_y	F_z	M_x	M_y	M_z
Umbrella	3005	915	930	28,070	72,770	97,850
Case 1	575	213	239	9,667	17,519	10,001
Case 2	428	116	274	13,532	24,535	11,672
Case 3	467	148	113	4,648	8,735	11,585
Case 4	926	184	154	3,568	12,592	13,273

Ratio Between Umbrella And Actual Loads For
The Faulted Condition

Case 1	6.97	4.30	3.89	3.01	4.15	9.77
Case 2	9.36	7.89	3.39	2.15	2.97	8.38
Case 3	8.58	6.18	8.23	6.25	8.33	8.45
Case 4	4.33	4.97	6.04	8.15	5.78	7.37



Coordinate System

TABLE 5.2-28

STEAM GENERATOR LOWER SUPPORT MEMBER STRESSES

Member		Member Stresses, Percent of Allowable, for Loading Condition:		
		Normal	Upset	Faulted
7 to 12	Bumpers	--	39	23
13, 14, 15	Beam	--	31	23
20 to 23	Columns	34	44	92

TABLE 5.2-29

STEAM GENERATOR UPPER SUPPORT MEMBER STRESSES

Member	Member Stresses, Percent of Allowable, for Loading Condition:		
	Normal	Upset	Faulted
25 to 29 Snubbers	--	--	--
34 to 69 Bumpers & Girder	--	18	18

TABLE 5.2-30

REACTOR COOLANT PUMP SUPPORT MEMBER STRESSES

Member	Member Stresses, Percent of Allowable, for Loading Condition:		
	Normal	Upset	Faulted
4 to 6 Tie Rod	--	26	36
7 to 9 Columns	30	31	42

TABLE 5.2-31

PRESSURIZER UPPER SUPPORT MEMBER STRESSES

Member		Member Stresses, Percent of Allowable, for Loading Condition:		
		Normal	Upset	Faulted
9	Upper Struts	--	13.	25.
10	Upper Struts	--	10.	30.
11	Upper Struts	--	11.	36.
12	Upper Struts	--	16.	29.

TABLE 5.2-32

CRDM HEAD ADAPTOR BENDING MOMENTS

	LOCA[a] (in-kip)	Combination of SSE and LOCA (in-kip)	% of Allowable
Longest CRDM	48.0	68.2	28.
Shortest CRDM	30.5	50.0	20.

a. Maximum moments are from reactor vessel inlet break.

[HISTORICAL][TABLE 5.2-33 (SHEET 1 OF 8)

FARLEY NUCLEAR PLANT UNIT 2 PRESERVICE INSPECTION PROGRAM ASME CODE CLASS 1 COMPONENTS

Table IWB-2600 Item No.	Table IWB-2500 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
B1.1	B-A	Reactor Vessel	Upper-to-middle-shell course circumferential weld	Volumetric	No
B1.1	B-A		Middle-to-lower-shell course circumferential weld	Volumetric	No
B1.1	B-A		Middle shell course longitudinal welds (2)	Volumetric	No
B1.1	B-A		Lower shell course longitudinal welds (2)	Volumetric	No
B1.2	B-B		Lower head-to-shell circumferential weld	Volumetric	No
B1.2	B-B		Lower head ring-to-disc circumferential weld	Volumetric	No
B1.3	B-C		Flange-to-vessel weld	Volumetric	No]
B1.4	B-D		Outlet nozzle-to-shell welds(3) and Nozzle inside-radiused sections (3)	Volumetric	No
B1.4	B-D		Inlet nozzle-to-shell welds (3) and nozzle inside-radiused sections (3)	Volumetric	No
B1.5	B-B		CRDM, Vent and incore instrumentation penetrations and CRDM seal welds	Visual	No
B1.6	B-F		Primary nozzle-to-safe-end welds	Volumetric & surface	No
B1.7	B-G-1		Closure studs (in place)	Not applicable	No-note b

TABLE 5.2-33 (SHEET 2 OF 8)

Table IWB-2600 Item No.	Table IWB-2500 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
B1.8	B-G-1	Reactor Vessel (Cont'd)	Closure studs and nuts	Volumetric & Surface	No
B1.9	B-G-1		Vessel flange ligaments	Volumetric	No
B1.10	B-G-1		Closure washers	Visual	No
B1.12	B-H		Integrally-welded supports	Not applicable	No - note c
B1.13	B-I-1		Closure head cladding	Visual &	No
B1.14	B-I-1		Vessel cladding	Visual	No
B1.15	B-N-1		Vessel interior surfaces and internals	Visual	No
B1.16	B-N-2		Interior attachments and core support structures	Not applicable	No - note d
B1.17	B-N-3		Core support structures	Visual	No
B1.18	B-0		Control rod drive housings	Volumetric	No
B1.19	B-P		Exempted components	Visual	No
B2.1	B-B	Pressurizer	Circumferential shell welds (5)	Volumetric	Yes - note a note m
B2.1	B-B		Longitudinal shell welds (3)	Volumetric	Yes - note a note m
B-2.2	B-D		Nozzle-to-vessel welds (6) and nozzle-to-vessel radiused sections (6)	Volumetric	Yes - note e note a note m
B2.3	B-E		Heater penetrations	Visual	No
B2.4	B-F		Nozzle-to-safe-end welds (6)	Surface & volumetric	No

TABLE 5.2-33 (SHEET 3 OF 8)

Table IWB-2600 Item No.	Table IWB-2500 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
B2.5	B-G-1	Pressurizer (Cont'd)	Pressure-retaining bolting (in place)	Not applicable	No - note g
B2.6	B-G-1		Pressure-retaining bolting	Not applicable	No - note g
B2.7	B-G-1		Pressure-retaining bolting	Not applicable	No - note g
B2.8	B-H		Integrally welded support	Volumetric	No
B2.9	B-I-2		Vessel cladding	Visual	No
B2.10	B-P		Exempted components	Visual	No
B2.11	B-G-2		Manway Bolting	Visual	No
B3.1	B-B	Steam Generators (3) (primary side)	Channel head-to-tubesheet weld (3)	Volumetric	No
B3.2	B-D		Nozzle-to-vessel welds and nozzle inside-radiused sections	Not applicable	No - note h
B3.3	B-F		Nozzle-to-safe-end welds (6)	Volumetric & surface	Yes - note f
B3.4	B-G-1		Pressure-retaining bolting (in place)	Not applicable	No - note g
B3.5	B-G-1		Pressure-retaining bolting	Not applicable	No - note g
B3.6	B-G-1		Pressure-retaining bolting	Not applicable	No - note g
B3.7	B-H		Integrally welded supports	Not applicable	No - note g
B3.8	B-I-2		Vessel cladding	Visual	No
B3.9	B-P		Exempted components	Visual	No

TABLE 5.2-33 (SHEET 4 OF 8)

Table IWB-2600 Item No.	Table IWB-2500 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
B3.10	B-G-2	Steam Generators (Cont'd)	Manway bolting	Visual	No
B4.1	B-F	Piping Pressure Boundary	Safe-end-to-pipe welds	Volumetric & surface	Yes -note i
B4.2	B-G-1		Pressure-retaining bolting (in place)	Not applicable	No - note g
B4.3	B-G-1		Pressure-retaining bolting	Not applicable	No - note g
B4.4	B-G-1		Pressure-retaining bolting	Not applicable	No - note g
B4.5	B-J		Circumferential and longitudinal pipe welds	Volumetric	Yes - notes i & j
B4.6	B-J		Branch pipe connection welds exceeding 6-inch diameter	Volumetric	Yes- note l
B4.7	B-J		Branch pipe connection welds 6-inch diameter and smaller	Surface	No
B4.8	B-J		Socket welds	Surface	No
B4.9	B-K-1		Integrally-welded supports	Volumetric	Yes- note k
B4.10	B-K-2		Support components	Visual	No
B4.11	B-P		Exempted components	Visual	No
B4.12	B-G-2		Pressure-retaining bolting	Visual	No
B5.1	B-G-1	Reactor Coolant Pump	Pressure-retaining bolts (in place)	Volumetric	No
B5.2	B-G-1		Pressure-retaining bolting	Volumetric & surface	No
B5.3	B-G-1		Pressure-retaining bolting	Visual	No

TABLE 5.2-33 (SHEET 5 OF 8)

Table IWB-2600 Item No.	Table IWB-2500 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
B5.4	B-K-1	Reactor Coolant Pump (Cont'd)	Integrally-welded supports	Not applicable	No - note g
B5.5	B-K-2		Support components	Visual	No
B5.6	B-L-1		Pump casing welds	Not applicable	No - note g
B5.7	B-L-2		Pump casing	Visual	No
B5.8	B-P		Exempted components	Visual	No
B5.9	B-G-2		Pressure-retaining bolting	Not applicable	No - note g
B6.1	B-G-1	Valve Pressure Boundary	Pressure-retaining bolting (in place)	Not applicable	No - note g
B6.2	B-G-1		Pressure-retaining bolting	Not applicable	No - note g
B6.3	B-G-1		Pressure-retaining bolting	Not applicable	No - note
B6.4	B-K-1		Integrally-welded supports	Not applicable	No - note g
B6.5	B-K-2		Support Components	Visual	No
B6.6	B-M-1		Valve-body welds	Not applicable	No-note g
B6.7	B-M-2		Valve bodies	Visual	No
B6.8	B-P		Exempted components	Visual	No
B6.9	B-G-2		Pressure-retaining bolting	Visual	No

TABLE 5.2-33 (SHEET 6 OF 8)

Notes

a. *For the pressurizer, the requirements of I-3121 of Section XI are impossible to meet. At the time the components were built, no excess material was saved for fabrication of calibration blocks. As an alternative, calibration blocks required for the ultrasonic examination of welds in these vessels will be fabricated from material of the same specification, product form, and heat treatment as one of the materials being joined as allowed by Article T-434.1.1 in Section V of the ASME Boiler and Pressure Vessel Code.*

b. *The reactor vessel closure studs are removed during the preservice inspection.*

c. *The reactor vessel supports are integral with the primary nozzles and the examination requirements of IWB-2600 is covered by item B1.4.*

d. *The requirements of IWB-2600 are applicable only to boiling water-type reactors and are thus not applicable to Farley Nuclear Plant.*

e. *The geometric configuration of the weld surface prevents ultrasonic examinations being performed to the extent required by IWB-2600. Angle beam examinations will be performed from the vessel head and on top of the weld. All of the weld, the heat affected zone, and the required amount of base metal on the shell side of the weld will be examined. Base metal on the nozzle side of the weld will be examined to the extent practical, which is approximately 25 percent. In addition, the welds will receive surface examination on those areas not scanned by UT.*

f. *Examination of the steam generator primary nozzle safe-end-to-pipe welds is limited by the nozzle geometry and surface condition, and by the limited surface preparation on the pipe side of the weld. The surface on the pipe side of the weld, which is a cast elbow, is machined for a distance of approximately 5-1/4 inches from the edge of the weld. Ultrasonic examination is limited to this distance from the edge of the weld. Examinations can be performed on the surface of the weld but are severely limited from the nozzle side by the configuration of weld build up and weld overlay.*

Ultrasonic examinations will be performed from both the pipe and weld surfaces as allowed by T-532 of Section V. All of the weld metal, including the weld root, will be inspected. Since no UT can be performed on the nozzle side of the weld, the extent of examination is limited to approximately 90 percent of the code-required area. Surface examinations will be performed on essentially 100 percent of the required area.

g. *There are no items in this category that require examination under the requirements of IWB-2600.*

h. *The steam generator nozzles are integrally forged with the channel head and thus do not contain any welds.*

TABLE 5.2-33 (SHEET 7 OF 8)

i. *The arrangements and details of the piping systems and components are such that some examinations as required by IWB-2600 are limited because of geometric configuration or accessibility. The welds will be ultrasonically examined by angle beams to the extent allowed by geometric configuration. In all cases, 100 percent of the weld material will be examined. Also, surface examinations will be performed to supplement limited volumetric examinations.*

Welds requiring supplemental surface examination, along with the estimated extent of volumetric examination, are as follows:

Loop 1 RTD return, weld	*#16 - 40%*
Loop 1 Cold Leg SIS, weld	*#8 - 60%*
Pressurizer Spray, Welds	*#42 - 70% and #43 - 70%*
Loop 3 RTD Return, weld	*#8 - 60%*
Pressurizer Relief, weld	*#14 - 50%*
Pressurizer Safety, welds	*#2 - 70%*
	# 5 - 80%
	#12 - 70%
	#16 - 80%
	#20 - 80%
	#24 - 70%
	#27 - 80%

Pressurizer safety welds 29, 31, 32, and loop 3, 2-in. safety injection (hot leg) weld 9 are inaccessible. However, field data in the form of radiography and dye penetrant will be utilized for preservice inspection as allowed by IWC-2100(b).

j. *In instances where the locations of pipe supports or hangers restrict the access available for the examination of pipe welds as required by IWB-2600, examinations will be performed to the extent practical unless removal of the support is permissible without unduly stressing the system.*

k. *The piping system integrally welded supports are attached to the pipe by fillet welds. The configurations of such welds are such that examinations cannot be performed to the extent required by IWB-2600 and only the base material of the pipe wall can be examined by ultrasonic techniques. Surface examination will be performed on the integrally welded attachments to supplement the limited volumetric examination.*

l. *The geometric configuration of the weld surface prevents ultrasonic examinations from being performed to the extent required by IWB-2600. Examinations will be performed to the extent practical from the pipe and nozzle surfaces adjacent to the weld. Surface examination of the weld will be performed to supplement the volumetric examinations.*

Welds requiring supplemental surface examination along with the estimated extent of volumetric examination, are as follows:

TABLE 5.2-33 (SHEET 8 OF 8)

Reactor Coolant Loop #1, weld #16BC - 80%
Reactor Coolant Loop #1, weld #21BC - 80%
Reactor Coolant Loop #2, weld #16BC - 80%
Reactor Coolant Loop #2, weld #21BC - 80%
Reactor Coolant Loop #3, weld #16BC - 80%
Reactor Coolant Loop #3, weld #21BC - 80%

m. *For the pressurizer, the requirements of I-3122 of Section XI cannot be met because of lack of cladding on the calibration blocks. However, only the top (O.D.) portions of the blocks are used for calibration. Specifically, the blocks contain side-drilled holes at depths of 1/4 T, 1/2 T, and 3/4 T. The blocks also contain a 2% T I.D. notch, but it is used only as a reference. Since the lack of cladding does not affect the ultrasonic calibration, the existing unclad calibration blocks will be utilized].*

[HISTORICAL][TABLE 5.2-34 (SHEET 1 OF 8)

FARLEY NUCLEAR PLANT UNIT 2 PRESERVICE INSPECTION PROGRAM
ASME CODE CLASS 2 COMPONENTS

Table IWC-2600 Item No.	Table IWC-2520 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
C1.1	C-A	Letdown Heat Exchanger (tube side)	Head-to-shell weld	Volumetric	No
C1.1	C-A		Shell-to-flange weld	Volumetric	No
C1.2	C-B		Nozzle-to-vessel weld	Not applicable	No - note a
C1.3	C-C		Integrally-welded supports	Not applicable	No - note b
C1.4	C-D		Tubesheet flange bolting	Not applicable	No - note b
C1.1	C-A	Excess Letdown Heat Exchanger (tube side)	Head-to-flange weld	Volumetric	Yes - note k
C1.2	C-B		Nozzle-to-vessel weld	Not applicable	No - note a
C1.3	C-C		Integrally-welded supports	Not applicable	No - note b
C1.4	C-D		Tubesheet flange bolting	Visual & volumetric	No
C1.1	C-A	Regenerative Heat Exchanger	Head-to-shell welds (6)	Volumetric	Yes - note g note k
C1.1	C-A		Shell-to-tubesheet welds (6)	Volumetric	Yes - note g note k
C1.2	C-B		Nozzle-to-vessel welds (12)	Not applicable	No - note a
C1.3	C-C		Integrally-welded supports	Surface	No
C1.4	C-D		Pressure-retaining bolting	Not applicable	No - note b
C1.1	C-A	Residual Heat Exchangers (2) (tube side)	Head-to-shell welds	Volumetric	No

TABLE 5.2-34 (SHEET 2 OF 8)

Table IWC-2600 Item No.	Table IWC-2520 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
C1.1	C-A		Shell-to-tubesheet welds	Volumetric	No
C1.2	C-B		Nozzle-to-vessel welds	Not accessible	Yes - note c
C1.3	C-C		Integrally-welded supports	Not applicable	No - note b
C1.4	C-D		Tubesheet flange bolting	Visual and volumetric	No
C1.1	C-A	Seal-Water Return Filter	Cover weldment-to-shell weld	Visual and surface	Yes - note d
C1.1	C-A		Head-to-shell weld	Visual and surface	Yes - note d
C1.2	C-B		Nozzle-to-vessel weld	Not applicable	No - note a
C1.3	C-C		Integrally-welded supports	Surface	No
C1.4	C-D		Pressure-retaining bolting	Not applicable	No - note b
C1.1	C-A	Volume Control Tank	Upper head-to-shell weld	Volumetric	No
C1.1	C-A		Lower head-to-shell weld	Volumetric	No
C1.2	C-B		Nozzle-to-vessel weld	Not applicable	No - note a
C1.3	C-C		Integrally-welded supports	Surface	No
C1.4	C-D		Manway bolting	Visual and volumetric	No
C1.1	C-A	Letdown Reheat Heat Exchanger (tube side)	Head-to-shell weld	Visual and surface	Yes - note d note k
C1.1	C-A		Shell-to-flange weld	Visual and surface	Yes - note d note k
C1.2	C-B		Nozzle-to-vessel weld	Not applicable	No - note a

TABLE 5.2-34 (SHEET 3 OF 8)

Table IWC-2600 Item No.	Table IWC-2520 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
C1.3	C-C		Integrally-welded supports	Surface	No
C1.4	C-D		Pressure-retaining bolting	Not applicable	No - note b
C1.1	C-A	Seal-Water-Heat Exchanger (tube side)	Head-to-shell weld	Visual and surface	Yes - note d
C1.1	C-A		Shell-to-flange weld	Visual and surface	Yes - note d
C1.2	C-B		Nozzle-to-vessel welds	Not applicable	No - note a
C1.3	C-C		Integrally-welded supports	Not applicable	No - note b
C1.4	C-D		Tubesheet flange bolting	Not applicable	No - note b
C1.1	C-A	Steam Generators (3) (shell side)	Upper head-to-shell weld	Volumetric	No
C1.1	C-A		Barrel-to-tubesheet weld	Volumetric	No
C1.2	C-B		Feedwater inlet nozzle-to-shell weld	Volumetric	No
C1.3	C-C		Integrally-welded supports	Not applicable	No - note b
C1.4	C-D		Pressure retaining bolting > 2 In.	Not Applicable	No - note b
C1.1	C-A	Reactor Coolant Filter	Cover weldment-to-shell weld	Visual and surface	Yes - note d

TABLE 5.2-34 (SHEET 4 OF 8)

Table IWC-2600 Item No.	Table IWC-2520 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
C1.1	C-A		Head-to-shell weld	Visual and surface	Yes - note d
C1.2	C-B		Nozzle-to-vessel weld	Not applicable	No - note a
C1.3	C-C		Integrally-welded supports	Surface	No
C1.4	C-D		Pressure-retaining bolting	Not applicable	No - note b
C1.1	C-A	Letdown Delay Tanks (2)	Head-to-shell welds	Volumetric No	
C1.2	C-B		Nozzle-to-vessel welds	Not applicable	No - note a
C1.3	C-C		Integrally-welded supports	Surface	No
C1.4	C-D		Pressure-retaining bolting	Not applicable	No - note b
C1.1	C-A	Excess Letdown Delay Tanks (2)	Head-to-shell welds	Volumetric	No
C1.2	C-B		Nozzle-to-vessel welds	Not applicable	No - note a
C1.3	C-C		Integrally-welded supports	Surface	No
C1.4	C-D		Pressure-retaining bolting	Not applicable	No - note b
C2.1	C-F; C-G	Piping Systems - note i	Circumferential butt welds	Volumetric	Yes - notes e & f
C2.2	C-F; C-G		Longitudinal weld joints in fittings	Volumetric	No
C2.3	C-F; C-G		Branch pipe-to-pipe welds	Volumetric	Yes - note e
C2.4	C-D		Pressure-retaining bolting	Visual and volumetric	No
C2.5	C-E-1		Integrally-welded supports	Surface	No
C2.6	C-E-2		Support components	Visual	No

TABLE 5.2-34 (SHEET 5 OF 8)

Table IWC-2600 Item No.	Table IWC-2520 Examination Category	System or Component	Area To Be Examined	Method of Examination	Section XI Code Relief Requested
C3.1	*C-F*	*Residual Heat Removal Pumps (2)*	*Pump casing welds*	*Not applicable*	*No - note b*
C3.2	*C-D*		*Pressure-retaining bolting*	*Visual and volumetric*	*No*
C3.3	*C-E-1*		*Integrally-welded supports*	*Not applicable*	*No - note b*
C3.4	*C-E-2*		*Support components*	*Visual*	*No*
C3.1	*C-F*	*Centrifugal Charging Pumps (3)*	*Pump casing welds*	*Volumetric*	*No*
C3.2	*C-D*		*Pressure-retaining bolting*	*Visual and volumetric*	*No*
C3.3	*C-E-1*		*Integrally-welded supports*	*Surface*	*Yes - note j*
C3.4	*C-E-2*		*Support components*	*Visual*	*No*
C4.1	*C-F; C-G*	*Valves*	*Valve-body welds*	*Not applicable*	*No - note b*
C4.2	*C-D*		*Pressure-retaining bolting*	*Visual and Volumetric*	*No*
C4.3	*C-E-1*		*Integrally-welded supports*	*Not applicable*	*No - note b*
C4.4	*C-E-2*		*Support components*	*Visual*	*No*

TABLE 5.2-34 (SHEET 6 OF 8)

Notes

a. This item is excluded from the examination requirements of IWC-2600 by application of the criteria given in IWC-1220.

b. There are no items in this category that require examination under the requirements of IWC-2600.

c. The nozzle to vessel welds of the residual heat exchangers are covered by a reinforcement ring and are not accessible for examination as required by IWC-2600. The geometric configuration is such that alternative NDE methods cannot be substituted. The nozzles will be subject to visual inspection for leakage.

d. The thickness of the materials utilized for the construction of this component (0.165 to 0.185 in.) is such that meaningful results could not be expected with ultrasonic examination as required by IWC-2600. Surface and visual examination of these welds will be performed as an alternative method.

e. The arrangement and details of the Class 2 piping system and components were designed and fabricated before the examination requirements of Section XI of the Code were formalized and some examinations as required by IWC-2600 are limited or not practical because of geometric configuration or accessibility. Generally these limitations exist at all fitting to fitting welds such as elbow to tee, elbow to valve, reducer to valve, etc. where geometry and sometimes surface conditions preclude ultrasonic coupling or access for the required scan length. The limitations exist to a lesser degree at pipe to fitting welds, where examination can only be fully performed from the pipe side, the fitting geometry limiting or even precluding examination from the opposite side. The welds will be ultrasonically examined by angle beam to the extent allowed by geometric configuration; however, 100 percent of the weld material will be examined. Also, surface examinations will be performed to supplement the limited volumetric examinations. Welds requiring supplemental surface examination, along with the estimated extent of examination, are as follows:

RHR, welds	*#31 - 50%*
	#32 - 50%
	#14 - 90%
	#11 - 30%
	#20 - 30%
	#18 - 50%

TABLE 5.2-34 (SHEET 7 OF 8)

In instances of branch pipe to pipe welds, ultrasonic examinations cannot be performed on the surface of the weld. Surface examination will be performed on 100 percent of the weld and adjacent base material. Welds requiring supplemental surface examination, along with the estimated extent of volumetric examination, are as follows:

Main Steam, welds	*#4-14 - 80%*	*#2-13 - 80%*
	#4-15 - 80%	*#2-14 - 80%*
	#1-5 - 80%	*#2-15 - 80%*
	#1-11 - 80%	*#2-16 - 80%*
	#1-12 - 80%	*#2-17 - 80%*
	#1-13 - 80%	*#3-5 - 80%*
	#1-14 - 80%	*#3-11 - 80%*
	#1-15 - 80%	*#3-12 - 80%*
	#1-16 - 80%	*#3-13 - 80%*
	#1-17 - 80%	*#3-14 - 80%*
	#2-5 - 80%	*#3-15 - 80%*
	#2-11 - 80%	*#3-16 - 80%*
	#2-12 - 80%	*#3-17 - 80%*

f. *In instances where the locations of pipe supports or hangers restrict the access available for the examination of pipe welds as required by IWC-2600, examinations will be performed to the extent practical unless removal of the support is permissible without unduly stressing the system.*

g. *The regenerative heat exchanger shell is fabricated from centrifugally cast austenitic steel material which limits ultrasonic examination as required by IWC-2600 to the half node technique. The geometric configuration of the weld surface and the location of adjacent nozzles and supports provide limitations to the extent of examination coverage. Surface examinations will be performed to supplement the volumetric examination.*

h. *The following components are exempt from the examination requirements of IWC-2520 by application of the criteria given in IWC-1220. These components will be examined in accordance with the requirements of IWC-2510.*

1. *CVCS seal water injection filters (2)*

2. *Safety injection accumulators (3)*

3. *Boron injection tank*

4. *Containment spray pumps (2)*

5. *Refueling water storage tank (RWST) and*

    a. *Suction piping from the RWST to the High Head Safety Injection Pumps.*

TABLE 5.2-34 (SHEET 8 OF 8)

b. *Suction piping from the RWST to the Low Head Safety Injection/Residual Heat Removal Pumps.*

c. *Suction piping from the RWST to the Containment Spray Pumps.*

i. *All Class 2 piping with a nominal diameter of 4 in. or less is excluded from the examination requirements of IWC-2520 by the application of the criteria given in IWC-1220.*

j. *Because of component and support designs, approximately 20 percent of each integrally-welded support is inaccessible for examination. The accessible portion of each support will receive visual and surface examinations.*

k. *Table IWC-2520, Category C-A and IWC-2600, Item C1.1 require volumetric examinations "uniformly distributed among three areas around the vessel circumference." The location of adjacent nozzles provides limitations to the extent of examination coverage. Consequently, the requirement for three uniformly distributed areas cannot be met. One or two areas will be inspected, as accessibility permits, instead of the required three areas. The required 20 percent of each circumferential weld will be volumetrically inspected except where material thickness precludes ultrasonic testing as stated in note 4.]*

TABLE 5.2-35 (SHEET 1 OF 2)

TYPE B-4 WELD WIRE AND LINDE 0091 FLUX TESTS

Example 1 - Type B-4 Weld Wire and Linde 0091 Flux (Test #1302)

Type CVN	Temp. °F	Impact and or Fracture Tests Values Ft/Lbs	%Shear	Mils Lat Exp	Temp. °F		Values Drop Weights		NDT
	-80	3	0	1	-50		1 F		-40°F
	-80	3	0	2	-40		1 F		
	-80	9	0	4	-30		2 NF		
	-40	26	10	19					
	-40	37	15	25					
	-40	38	15	24					
	+10	69	35	46	+100	117	90	83	
	+10	50	25	38	+100	114	90	82	
	+10	66	30	44	+100	120	90	83	
	+20	66	35	46	+160	124	100	83	
	+20	81	50	57	+160	136	100	89	
	+20	90	60	63	+160	135	100	88	

Example 2 - Type B-4 Weld Wire and Linde 0091 Flux (Test #1388)

Type CVN	Temp. °F	Impact and or Fracture Tests Values Ft/Lbs	%Shear	Mils Lat Exp	Temp. °F		Values Drop Weights		NDT
	-80	11	0	3	-60		1 F		-60°F
	-80	11	0	3	-50		2 NF		
	-80	13	0	4	-40		1 NF		
	-40	30	15	17					
	-40	27	15	15					
	-40	25	10	11					
	0	77	50	45	+100	143	100	84	
	0	72	50	40	+100	133	100	82	
	0	70	50	41	+100	145	100	86	
	+10	76	50	41	+180	143	100	82	
	+10	74	50	46	+180	149	100	86	
	+10	82	60	45	+180	148	100	85	
	+60	116	70	76					
	+60	118	70	74					
	+60	121	70	71					

TABLE 5.2-35 (SHEET 2 OF 2)

Example 3 - Type B-4 Weld Wire and Linde 0091 Flux (Test #1389)

Impact and or Fracture Tests

Type CVN	Temp. °F	Ft/Lbs	Values %Shear	Mils Lat Exp	Temp. °F		Values Drop Weights		NDT
	-60	16	0	9	-60		1 F		-60°F
	-60	15	0	7	-50		2 NF		
	-60	19	0	11	-40		1 NF		
	-40	20	5	11					
	-40	28	10	16					
	-40	32	15	22					
	-20	85	50	53	+60	132	80	77	
	-20	88	50	56	+60	149	100	84	
	-20	76	40	47	+60	123	80	74	
	0	77	40	47	+100	142	100	82	
	0	75	40	45	+100	148	100	84	
	0	99	60	52	+100	140	100	82	
	+20	117	70	74					
	+20	105	60	65					
	+20	114	70	74					

Example 4 - Type B-4 Weld Wire and Linde 0091 Flux (Test #1386)

Impact and or Fracture Tests

Type CVN	Temp. °F	Ft/Lbs	Values %Shear	Mils Lat Exp	Temp. °F		Values Drop Weights		NDT
	-80	16	0	7	-60		1 F		-60°F
	-80	18	0	8	-50		2 NF		
	-80	18	0	7	-40		1 NF		
	-40	38	20	26					
	-40	32	15	17					
	-40	34	15	19					
	0	79	40	52	+100	137	100	82	
	0	61	70	39	+100	132	100	82	
	0	95	70	60	+100	141	100	83	
	+10	96	70	62	+180	142	100	82	
	+10	101	70	60	+180	145	100	85	
	+10	84	60	58	+180	143	100	83	
	+60	118	80	78					
	+60	130	90	80					
	+60	117	80	75					

TABLE 5.2-36

FARLEY NUCLEAR PLANT UNIT 2 LOWER SHELL COURSE CHARPY V NOTCH DATA[a]

Plate Code No. B7210-1				Plate Code No. B7210-2			
Test Temp. (°F)	Energy (Ft-Lb)	Lat. Exp (Mils)	Shear (%)	Temp. (°F)	Energy (Ft-Lb)	Lat. Exp (Mils)	Shear (%)
-50	10	6	9	-50	15	11	9
-50	14.5	8	15	-50	12.5	8	9
-50	11	7	9	-50	11	8	9
20	33	25	29	0	26	24	30
20	47	35	34	0	27.5	27	34
20	46	33	34	0	45	35	32
75	48.5	38	59	30	51	39	30
75	50	40	59	30	40	34	34
75	62	47	64	30	47	39	30
110	86	67	80	100	67	52	79
110	75	57	75	100	80.5	58	75
110	69.5	54	67	100	85	60	75
150	100	69	100	150	100	76	100
150	95	71	100	150	101	74	100
150	93	67	100	150	97	75	100
210	96	70	100	210	98	69	100
210	105.5	74	100	210	102	76	100
210	107	75	100	210	95.5	72	100

a. Normal to major rolling direction of the plate.

TABLE 5.2-37

FARLEY NUCLEAR PLANT UNIT 2
INTERMEDIATE SHELL COURSE CHARPY V NOTCH DATA[a]

Plate Code No. B7203-1				Plate Code No. B7212-1			
Test Temp. (°F)	Energy (Ft-Lb)	Lat. Exp (Mils)	Shear (%)	Test Temp. (°F)	Energy (Ft-Lb)	Lat. Exp (Mils)	Shear (%)
-50	13.5	9	15	-50	18.5	11	12
-50	19	11	15	-50	15.5	11	12
-50	14	8	15	-50	19	11	12
0	28	25	30	0	35	27	27
0	34	26	28	0	34.5	27	25
0	44	36	34	0	30	27	25
20	55	41	40	30	43	35	32
20	51	38	45	30	48	36	35
20	43	32	34	30	52	39	43
75	50.5	50	56	100	76.5	55	73
75	61.5	40	52	100	74	56	73
75	65	46	61	100	70	54	69
150	91	68	100	150	95	67	100
150	97	76	100	150	98	68	100
150	92	70	100	150	106	76	100
210	105.5	69	100	210	89	68	100
210	97.5	74	100	210	94	70	100
210	95.5	72	100	210	88	69	100

a. Normal to major rolling direction of the plate.

TABLE 5.2-38
(SHEET 1 OF 2)

FARLEY NUCLEAR PLANT UNIT 2
NOZZLE SHELL COURSE CHARPY V NOTCH DATA[a]

FORGING CODE No. B7216-1

Test Temp. (°F)	Energy (Ft-Lb)	Lat. Exp (Mils)	Shear (%)
-80	2	0	0
-80	4	0	0
-80	8	4	0
-20	68	53	29
-20	37	25	16
-20	66	52	29
10	99	76	64
10	103	76	64
10	110	81	70
10	95	77	52
10	55	41	29
10	78	63	40
30	72	57	23
30	93	70	55
30	87	65	46
100	147	91	100
100	123	77	75

TABLE 5.2-38
(SHEET 2 OF 2)

Test Temp. (°F)	Energy (Ft-Lb)	Lat. Exp (Mils)	Shear (%)
100	110	80	70
180	146	90	100
180	151	88	100
180	149	88	100

a. Major working direction of forging.

TABLE 5.2-39

PRESSURIZER FRACTURE TOUGHNESS PROPERTIES

Component	Component Part	Test Number	Material Specification	Charpy V Notch (ft-lb)	Lateral Expansion (in)	Test Temperature (°F)	T_{ndt} (°F)	RT_{ndt} (°F)
Pressurizer	Lower head	T03626	SA 533 Gr. A C1.2	75, 83, 76	.060, .064, .062	70	10	10
(1561)	Surge nozzle forging	T03386	SA 508 C1.2	88, 100, 96	.067, .082, .078	70	10	10
	Upper head	T03748	SA 533 Gr. A C1.2	63, 75, 72	.060, .056, .062	70	10	10
	Manway nozzle forging	T03336	SA 508 C1.2	113, 129, 120	.085, .086, .077	70	10	10
	Safety nozzle forging	T03381-3	SA 508 C1.2	82, 82, 78	.065, .066, .063	70	10	10
	Safety nozzle forging	T03284-1	SA 508 C1.2	64, 64, 55	.042, .042, .036	10	(a)	
	Safety nozzle forging	T04281-10	SA 508 C1.2a	139, 136, 141	.089, .079, .087	120	60	60
	Relief nozzle forging	T03380-3	SA 508 C1.2	84, 83, 92	.067, .070, .076	70	10	10
	Spray nozzle forging	T03722	SA 508 C1.2	74, 86, 71	.064, .072, .059	70	10	10
	Manway cover	T04405	SA 533 Gr. A C1.1	75, 79, 81	.069, .068, .064	120	60	60
	Shell barrel	T03630	SA 533 Gr. A C1.2	58, 58, 62	.054, .054, .054	70	10	10
	Shell barrel	T03741	SA 533 Gr. A C1.2	51, 54, 54	.049, .044, .050	80	10	20
	Shell barrel	T03355	SA 533 Gr. A C1.2	60, 64, 66	.046, .058, .058	70	10	10

a. Drop weight test results not available.

TABLE 5.2-40

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER AND RELIEF VALVE PIPING - UPSTREAM OF VALVES CLASS I PIPING

Plant/System Operating Condition	Load Combination	Piping Allowable Stress Intensity
Normal	N	1.5 S_m
Upset	N + OBE	1.5 S_m
Upset	N + SOT_U	1.5 S_m
Upset	N + OBE + SOT_U	1.8 S_m/1.5 S_y(2)
Emergency	N + SSE + SOT_E	2.25 S_m/1.8 S_y(2)
Faulted	N + SSE + SOT_F	3.0 S_m

NOTES: 1. Use SRSS for combining dynamic load responses.

2. The smaller of the given allowable is to be used.

N	=	Sustained loads during normal plant operation
SOT	=	System operating transient
SOT_U	=	Relief valve discharge transient
SOT_E	=	Safety valve discharge transient
SOT_F	=	Max (SOT_U; SOT_E); or transition flow
OBE	=	Operating basis earthquake
SSE	=	Safe shutdown earthquake
S_h	=	Basic material allowable stress at maximum (hot) temperature
S_m	=	Allowable design stress intensity
S_y	=	Yield strength value

TABLE 5.2-41

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER AND RELIEF VALVE PIPING - DOWNSTREAM OF VALVES NNS PIPING

Plant/System Operating Condition	Load Combination	Piping Allowable Stress
Normal	N	1.0 S_h
Upset	N + OBE	1.2 S_h
Upset	N + SOT_U	1.2 S_h
Upset	N + OBE + SOT_U	1.8 S_h
Emergency	N + SOT_E	2.4 S_h*
Faulted	N + SSE + SOT_F	2.4 S_h

NOTE: Use SRSS for combining dynamic load responses.

*See reference (21)

N	=	Sustained loads during normal plant operation
SOT	=	System operating transient
SOT_U	=	Relief valve discharge transient
SOT_E	=	Safety valve discharge transient
SOT_F	=	Max (SOT_U; SOT_E); or transition flow
OBE	=	Operating basis earthquake
SSE	=	Safe shutdown earthquake
S_h	=	Basic material allowable stress at maximum (hot) temperature
S_m	=	Allowable design stress intensity
S_y	=	Yield strength value

TABLE 5.2-42

FARLEY UNITS 1 AND 2
SAFETY LINE PIPE STRESS AND STRAIN SUMMARY
FOR EMERGENCY CONDITION

Node Point	Piping Components	Code Maximum Stress (ksi)	Allowable Stress (ksi)
1290*	Butt weld at valve end nozzle	15.1	18.8
1460*	Long radius elbow	34.2	36.45
100**	Branch connection	32.9	44.67
690**	reducer	25.1+	44.67
1490**	Welded attachment at support R120***	54.97***	55.42

* ASME Class 1 piping, upstream of safety valves
** ASME NNS piping, downstream of safety valves
*** Based on ASME Code Case N-318 allowable
\+ Stress Index based on ANSI B31.1-1967, including 1971 Addenda

TABLE 5.2-43

FARLEY NUCLEAR PLANT - TMI ACTION NUREG-0737.II.D.1
UNITS 1 AND 2 PSARV LINE PIPE SUPPORTS
ANCHOR BOLT DATA FOR SUPPORTS WITH FACTOR OF SAFETY F.S. <4

Unit	Serial No.	Support Mark No.	Total No. of Bolts	No. of Bolts w/ F.S. ≥4	No. of Bolts w/ F.S. <4	Actual F.S.		Types of Bolts with F.S. <4
						Bolt #	F.S.	
1	1	RC-R61	4	2	2	#3 #4	3.57 3.57	#3 and #4 3/4" ϕ HILTI KWIK
2	1	2RC-131X	5	3	2	#2 #5	3.77 3.20	#2 AND #5 1/2" ϕ HILTI KWIK

THIS FIGURE HAS BEEN DELETED.

REV 28 10/18

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRIMARY-SECONDARY BOUNDARY COMPONENTS SHELL LOCATIONS OF STRESS INVESTIGATIONS FIGURE 5.2-1

THIS FIGURE HAS BEEN DELETED.

REV 28 10/18

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRIMARY AND SECONDARY HYDROSTATIC TEST STRESS HISTORY FOR THE CENTER HOLE LOCATION FIGURE 5.2-2

THIS FIGURE HAS BEEN DELETED.

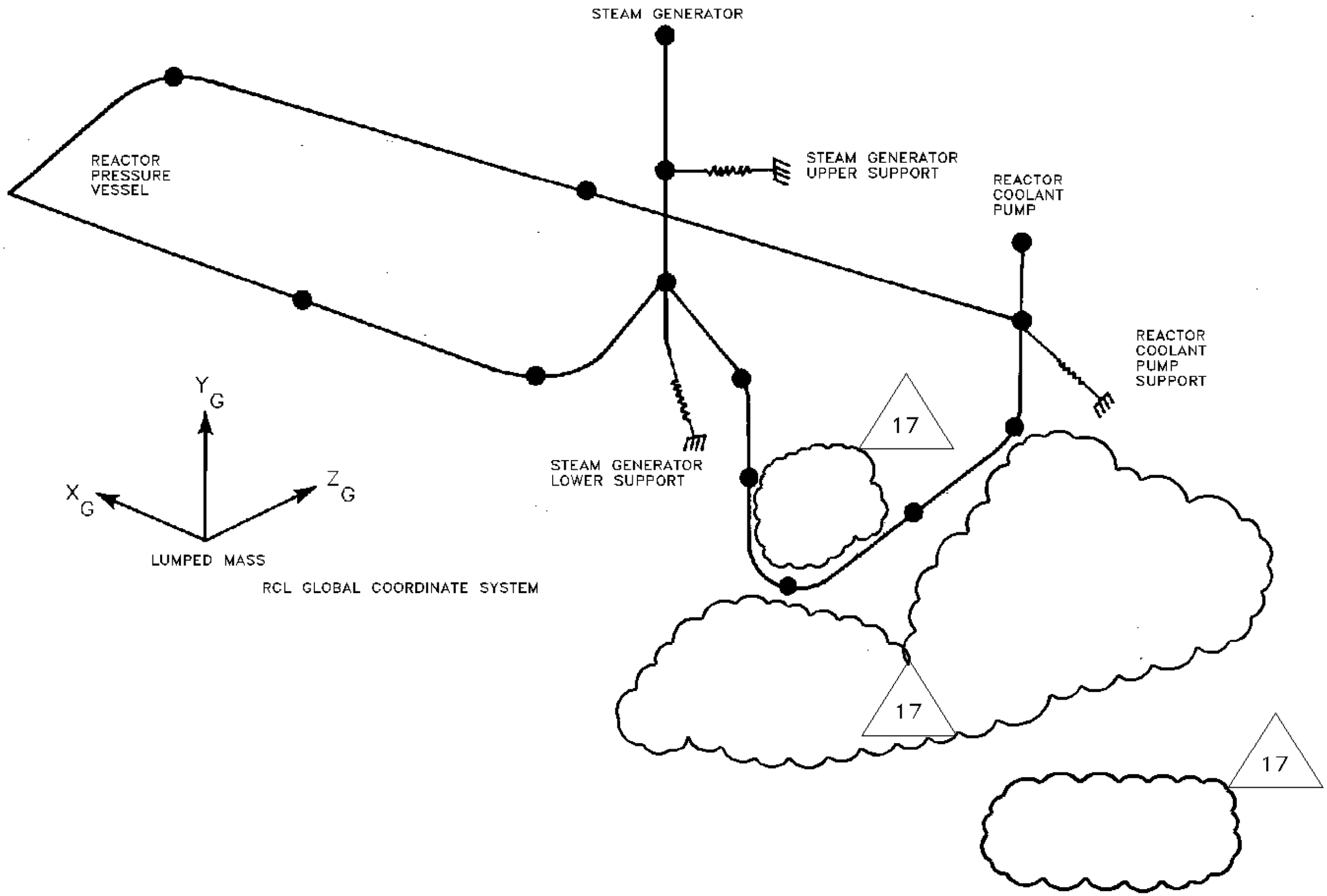
REV 28 10/18

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PLANT HEATUP AND LOADING OPERATIONAL TRANSIENTS (WITH STEADY-STATE PLATEAU) STRESS HISTORY FOR THE HOT SIDE CENTER HOLE LOCATION FIGURE 5.2-3

THIS FIGURE HAS BEEN DELETED.

REV 28 10/18

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LARGE STEP LOAD DECREASE AND LOSS OF FLOW STRESS HISTORY FOR THE HOT SIDE CENTER HOLE LOCATION FIGURE 5.2-4



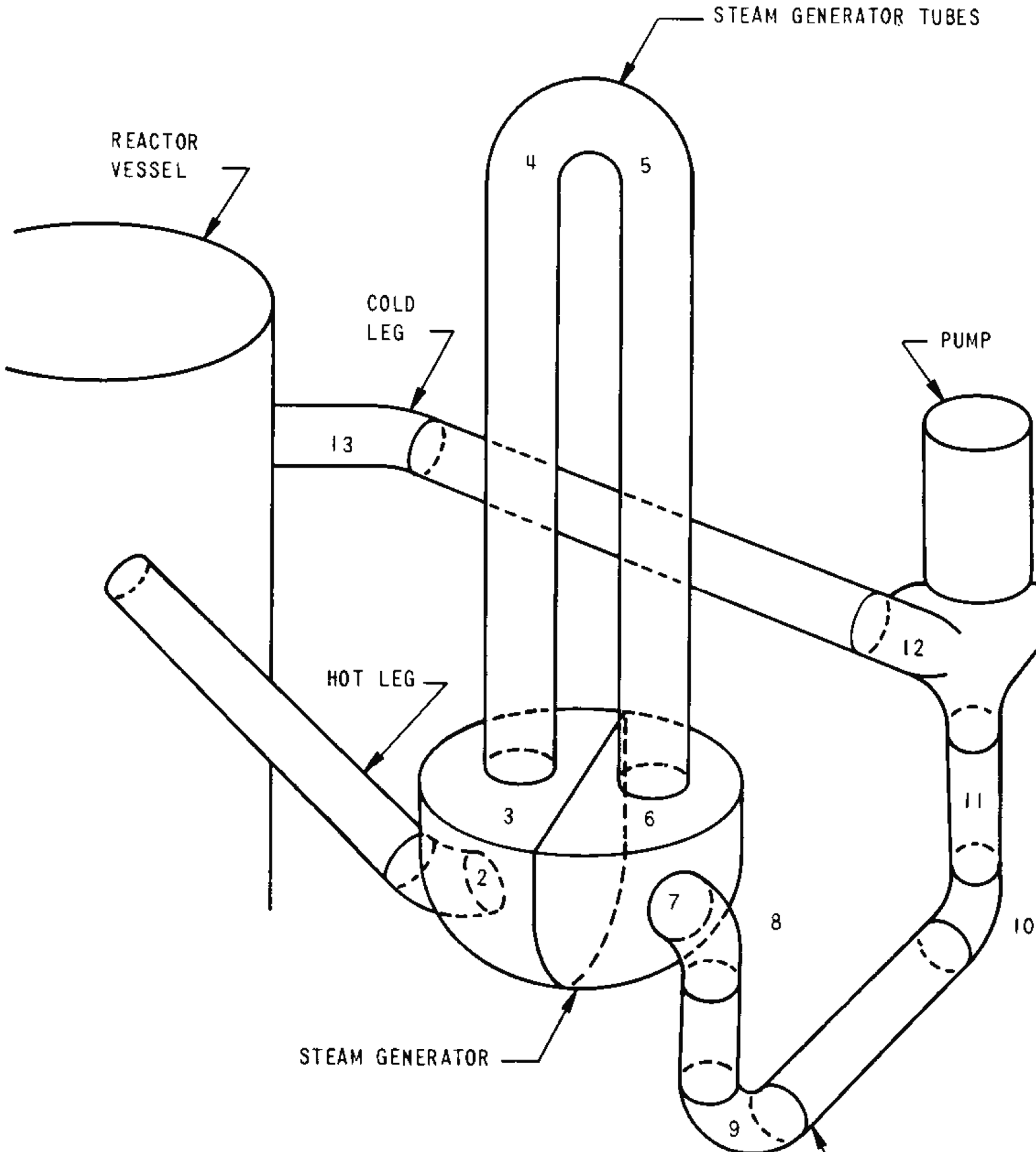
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR COOLANT LOOP/SUPPORTS SYSTEM DYNAMIC – STRUCTURAL MODEL

FIGURE 5.2-5



REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STHRUST RCL MODEL SHOWING HYDRAULIC FORCE LOCATIONS
		FIGURE 5.2-6

THIS FIGURE HAS BEEN DELETED

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REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR FIRST TIME 16 EFPY OF OPERATION FIGURE 5.2-7 (SHEET 1 OF 2)

THIS FIGURE HAS BEEN DELETED

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REV 21 5/08

FARLEY UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 14 EFPY

FIGURE 5.2-7 (SHEET 2 OF 2)

THIS FIGURE HAS BEEN DELETED

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REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNIT 1 ALA REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 16 EFFECTIVE FULL POWER YEARS OF OPERATION

FIGURE 5.2-8 (SHEET 1 OF 2)

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FARLEY UNIT 2 REACTOR COOLING SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 14 EFPY

FIGURE 5.2-8 (SHEET 2 OF 2)

THIS FIGURE HAS BEEN DELETED.

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSEL

FIGURE 5.2-9

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNIT 1 FAST NEUTRON FLUENCE (E > 1 MEV) AS A FUNCTION OF FULL POWER SERVICE LIFE

FIGURE 5.2-10 (SHEET 1 OF 2)

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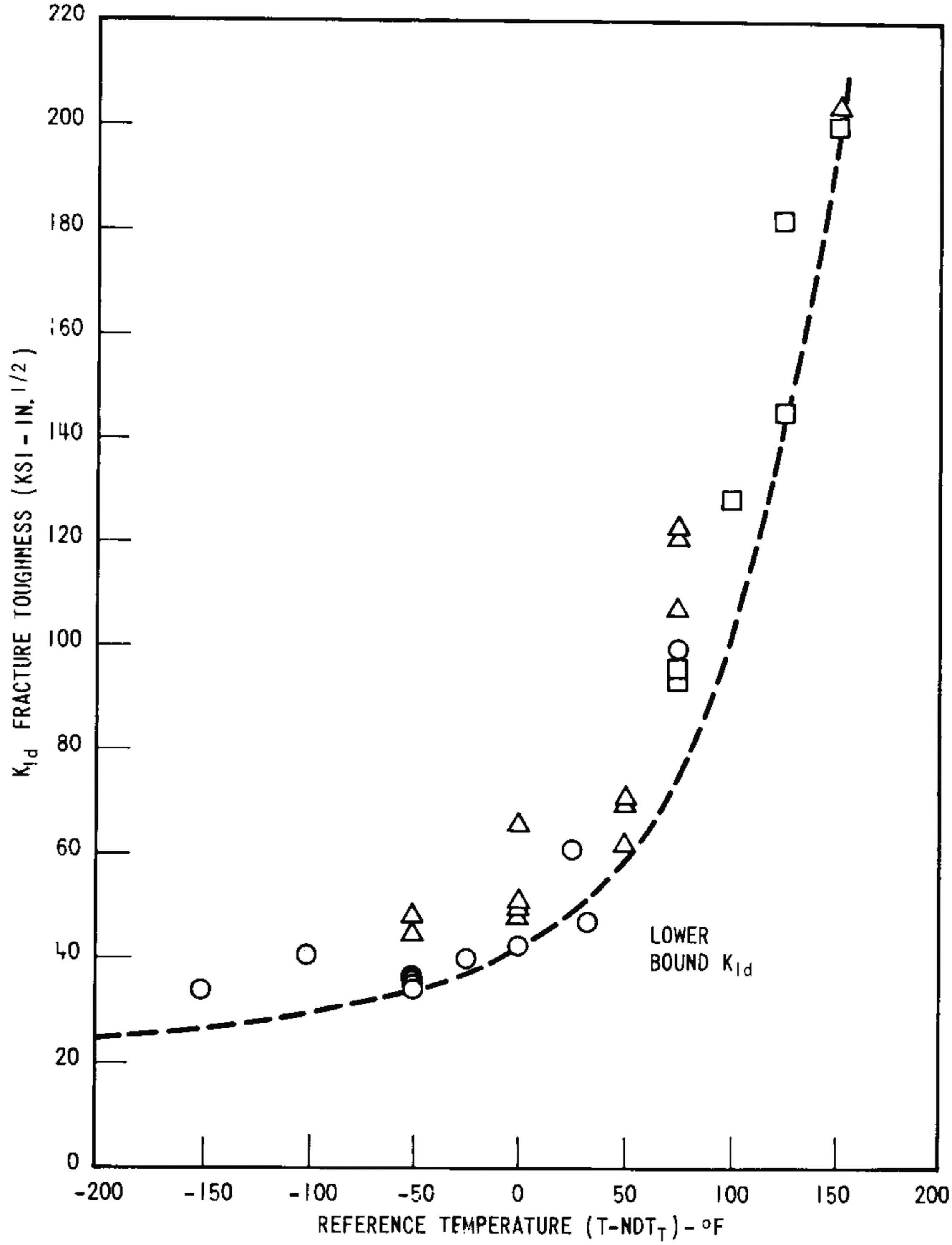
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNIT 1 FAST NEUTRON FLUENCE (E > 1 MEV) AS A FUNCTION OF FULL POWER SERVICE LIFE (45° LOCATION)

FIGURE 5.2-10 (SHEET 2 OF 2)



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Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

K_{1d} LOWER BOUND FRACTURE TOUGHNESS
A533V (REFERENCE WCAP 7623) GRADE B CLASS 1

FIGURE 5.2-11

THIS FIGURE HAS BEEN DELETED.

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Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONDENSATE MEASURING SYSTEM

FIGURE 5.2-12

THIS FIGURE HAS BEEN DELETED

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TOOL DETAILS (VESSEL SCANNER)

FIGURE 5.2-13

THIS FIGURE HAS BEEN DELETED

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TOOL DETAILS
(NOZZLE AND FLANGE SCANNER)

FIGURE 5.2-14

THIS FIGURE HAS BEEN DELETED

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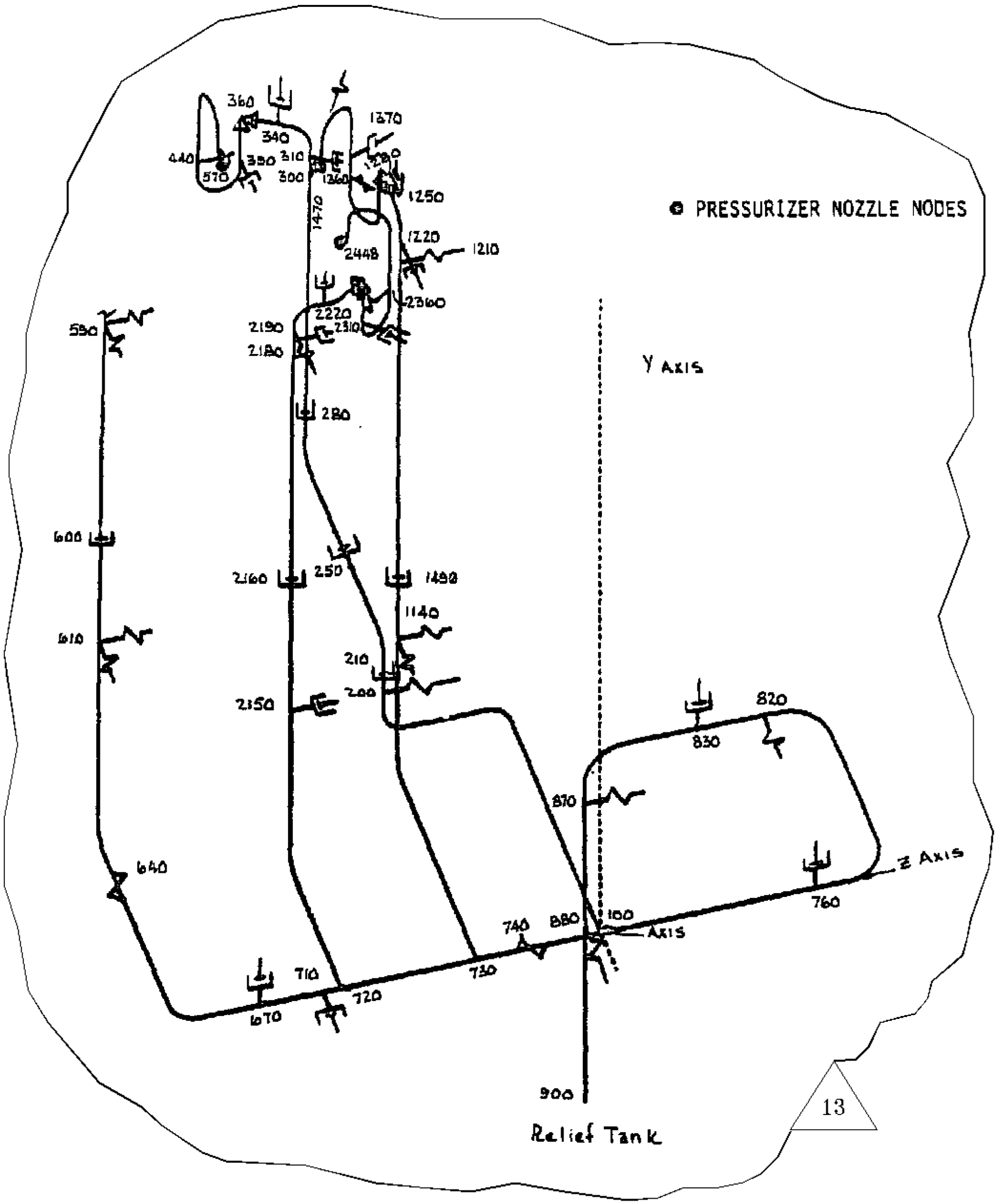
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SAMPLE WELD DATA SHEET

FIGURE 5.2-15



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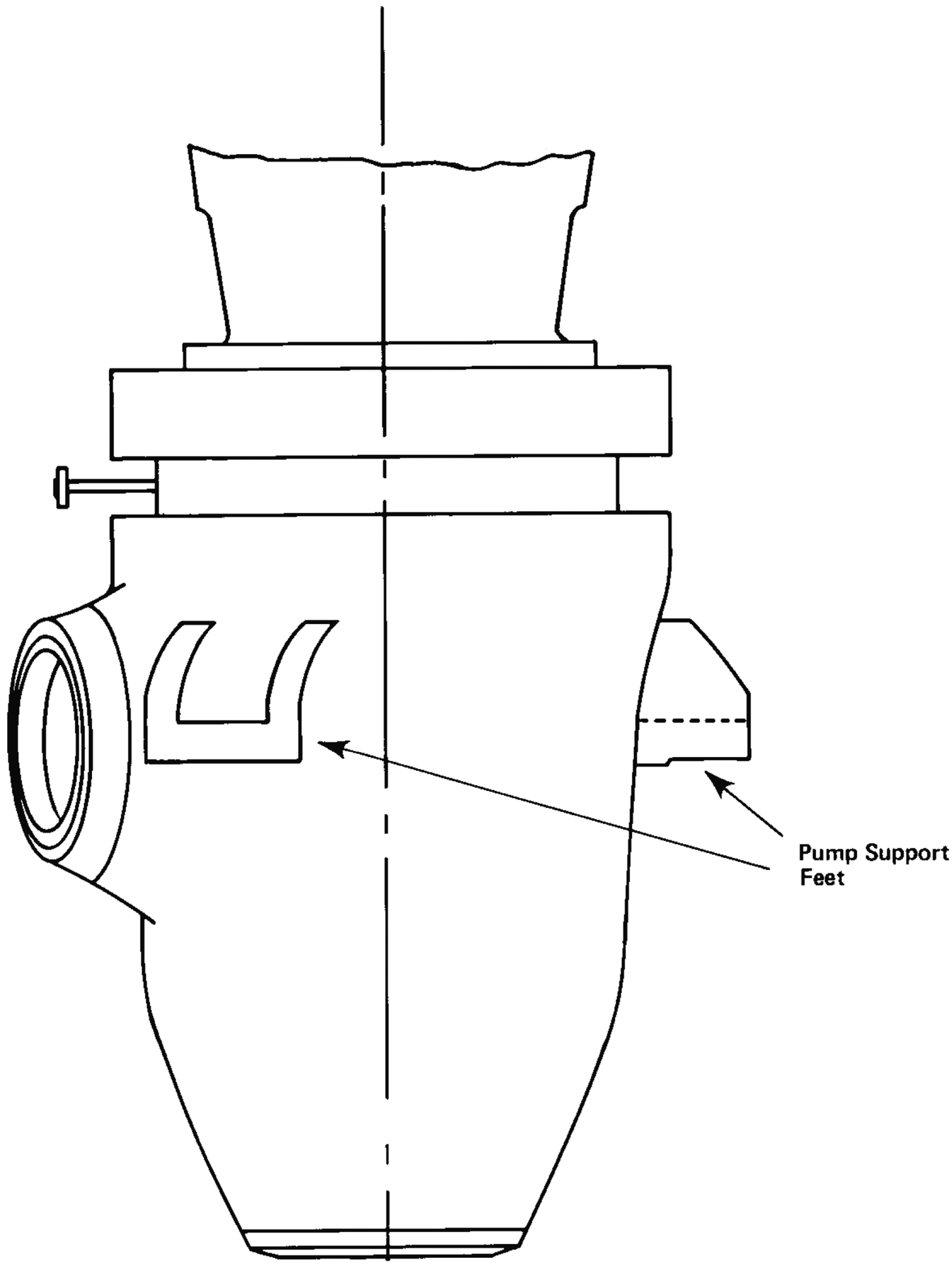
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURIZER SAFETY LINE STRUCTURAL MODEL

FIGURE 5.2-16



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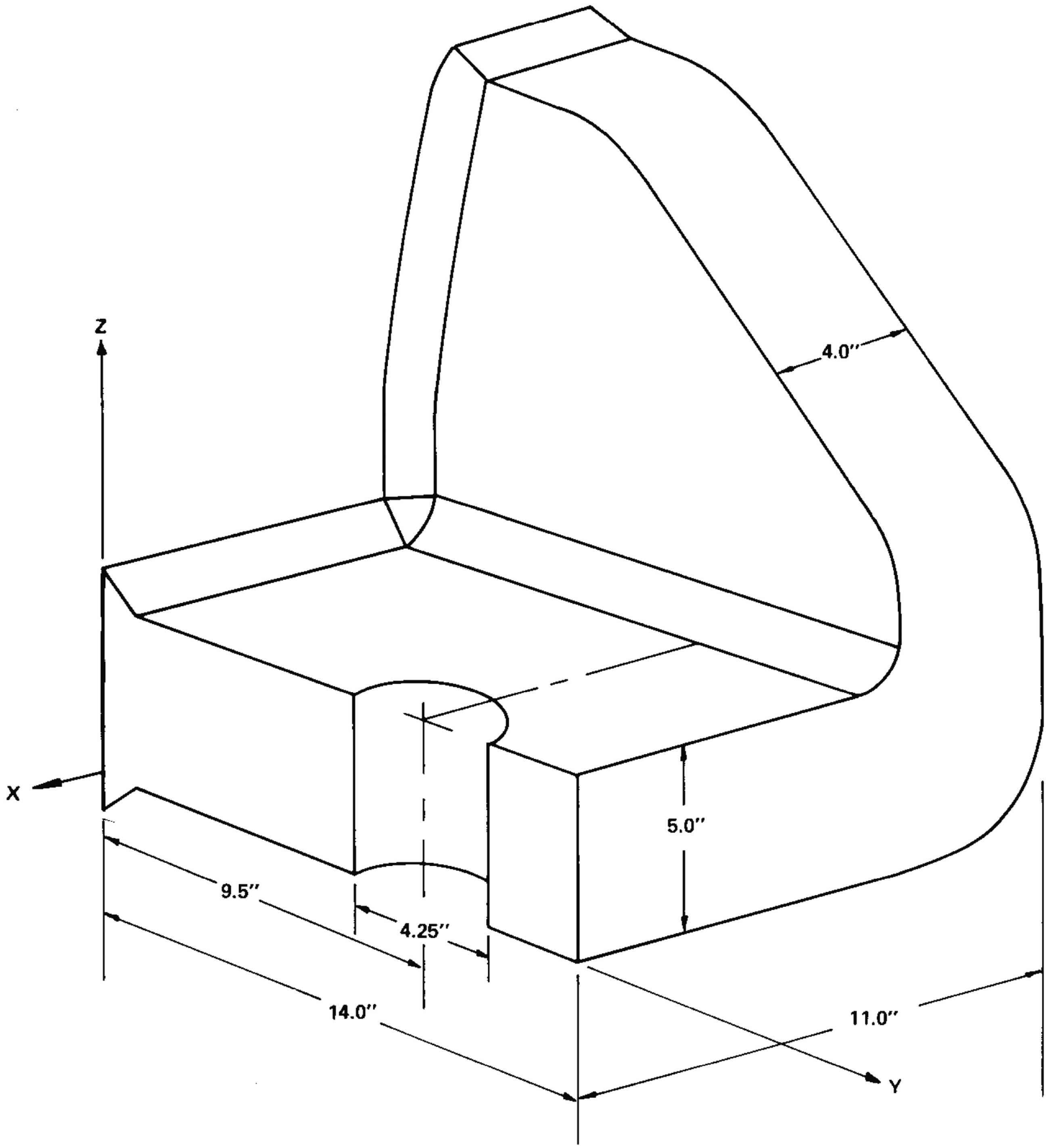
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR COOLANT PUMP CASING WITH SUPPORT FEET

FIGURE 5.2-20



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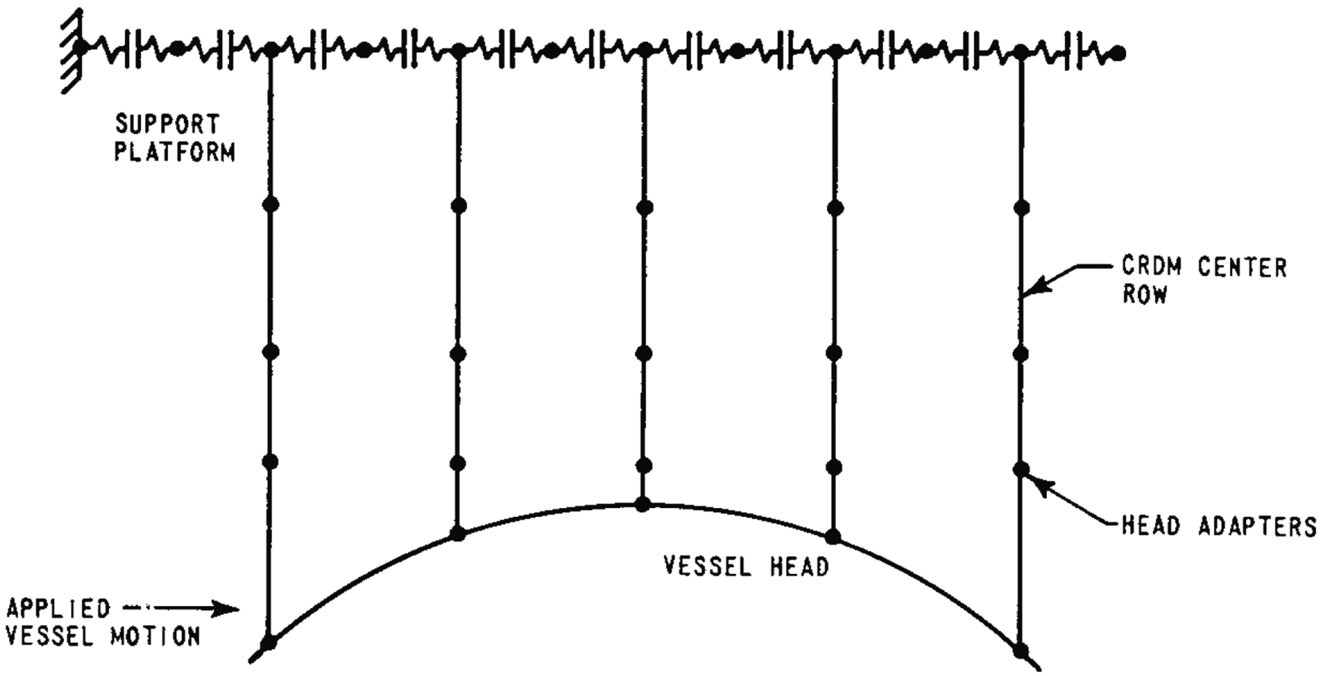
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

BOLT HOLD RADIAL CENTERLINE

FIGURE 5.2-21



REV 21 5/08

NONLINEAR CRDM CENTER ROW MODEL

FIGURE 5.2-22

5.3 THERMAL HYDRAULIC SYSTEM DESIGN

5.3.1 ANALYTICAL METHODS AND DATA

The thermal and hydraulic design bases of the reactor coolant system (RCS) are described in sections 4.3 and 4.4 in terms of core heat generation rates, departure from nucleate boiling ratio (DNBR), analytical models, peaking factors, and other relevant aspects of the reactor.

5.3.2 OPERATING RESTRICTIONS ON PUMPS

Plant operating experience and instrument inaccuracy are used to establish a pressure range which ensures that all RCP support conditions are met and that the LTOP relief valves are not challenged during RCP start, the ensuing transient, and any subsequent operation.

5.3.3 BOILING WATER REACTOR (BWR)

5.3.4 TEMPERATURE-POWER OPERATING MAP

The effects of reduced core flow because of inoperative pumps is discussed in subsections 5.5.1, 15.2.5, and 15.3.4.

Natural circulation capability of the system is shown in table 5.3-1.

The issue of steam formation in the RCS was made part of TMI Action Plan Requirement II.K.2.17. The potential for voids being generated in the RCS during anticipated transients is accounted for in present analysis models. The transient analyses performed using these models demonstrate that steam voids will not result in unacceptable consequences during anticipated transients.

5.3.5 LOAD FOLLOWING CHARACTERISTICS

The RCS is designed on the basis of steady-state operation at full-power heat load. The reactor coolant pumps utilize constant speed drives as described in section 5.5, and the reactor power is controlled to maintain average coolant temperature at a value which is a linear function of load, as described in section 7.7.

5.3.6 TRANSIENT EFFECTS

Transient effects are evaluated as follows: complete loss of forced reactor coolant flow (15.3.4); partial loss of forced reactor coolant flow (15.2.5); startup of an inactive loop (15.2.6); loss of load (15.2.7); loss of normal feedwater (15.2.8); loss of offsite power (15.2.9); and accidental depressurization of the reactor coolant system (15.2.12).

5.3.7 THERMAL AND HYDRAULIC CHARACTERISTICS SUMMARY TABLE

The thermal and hydraulic characteristics are given in tables 4.3-1, 4.4-1, and 4.4-2.

TABLE 5.3-1

NATURAL CIRCULATION REACTOR COOLANT FLOW VERSUS REACTOR POWER

Reactor Power (% Full Power)	Reactor Coolant Flow (% Nominal Flow)
3.5	4.8
3.0	4.6
2.5	4.4
2.0	4.1
1.5	3.7

5.4 REACTOR VESSEL AND APPURTENANCES

Section 5.4 has been divided into four principal subsections: viz., design bases, description, evaluation, and testing and inspections for the reactor vessel and its appurtenances, consistent with the requirements of the introductory paragraph 5.4 of the Standard Format and Content Guide, Revision 1. The following specific information required by the guide is cross-referenced below.

Guide Reference		FSAR Section
5.4.1	Protection of Closure Studs	5.4.2.2
5.4.2	Special Processes for Fabrication and Inspection	5.4.2.1, 5.4.4
5.4.3	Features for Improved Reliability	5.4.1, 5.4.2.1
5.4.4	Quality Assurance Surveillance	5.4.2, 5.4.4
5.4.5	Materials and Inspections	5.2.3, 5.4.4
5.4.6	Reactor Vessel Design Data	Table 5.4-1
5.4.7	Reactor Vessel Schematic (BWR)	Not applicable

5.4.1 DESIGN BASES

5.4.1.1 Codes and Specifications

The vessel is Safety Class 1. Design and fabrication of the reactor vessel is carried out in strict accordance with ASME Section III, Class 1. Material specifications are in accordance with the ASME Code requirements and are given in subsection 5.2.3.

5.4.1.2 Design Transients

Cyclic loads are introduced by normal power changes, reactor trip, and startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the projected plant life. [a] Vessel analyses result in a usage factor that is < 1.

a. Metal fatigue is evaluated as a time-limited aging analysis (TLAA) for license renewal (see chapter 18, subsection 18.4.2).

With regard to the thermal and pressure transients involved in the loss-of-coolant accident (LOCA), the reactor vessel is analyzed to confirm that the delivery of cold emergency core cooling water to the vessel following a LOCA does not cause a loss of integrity of the vessel.

The design specifications require analysis to prove that the vessel is in compliance with the fatigue limits of Section III of the ASME Pressurized Vessel and Boiler Code. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The heatup and cooldown rates are 100°F per hour. These rates are reflected in the vessel design specifications.

A control rod housing failure does not cause propagation of failure to adjacent housings or to any other part of the reactor coolant system (RCS) boundary.

Design transients are discussed in detail in paragraph 5.2.1.5.

5.4.1.3 Protection Against Nonductile Failure

Protection against nonductile failure is discussed in subsection 5.2.4.

5.4.1.4 Inspection

The internal surface of the reactor vessel is capable of inspection periodically, using visual and/or nondestructive techniques over the accessible areas. During refueling, the vessel cladding is capable of being inspected in certain areas between the closure flange and the primary coolant inlet nozzles and, if deemed necessary, the core barrel is capable of being removed, making the entire inside vessel surface accessible.

The closure head is examined to meet 10 CFR 50.55a requirements. Optical devices permit a selective inspection of the cladding, control rod drive mechanism (CRDM) nozzles, and the gasket seating surface. The knuckle transition piece, which is the area of highest stress of the closure head, is accessible on the outer surface for visual inspection, dye penetrant or magnetic particle testing, and ultrasonic testing. The closure studs can be inspected periodically using visual, magnetic particle, and/or ultrasonic techniques.

5.4.2 DESCRIPTION

The reactor vessel purchase order was originally placed with Babcock and Wilcox on May 18, 1967. Another purchase order was placed with Combustion Engineering on November 15, 1969, for final completion of the vessel. The reactor vessel is Safety Class 1. Design and fabrication of the vessel is in strict accordance with ASME Section III, Class 1. Material specifications, given in subsection 5.2.3, are in accordance with the ASME Code requirements.

The reactor vessel is cylindrical, with a welded hemispherical bottom head and a removable, bolted, flanged and gasketed hemispherical upper head (drawings U-419289 and U-611139). The reactor vessel flange and head are sealed by two hollow, metallic O-rings. Seal leakage is detected by means of two leakoff connections; one between the inner and outer ring and one outside the outer O-ring. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains head adapters. These head adapters are tubular members, attached by partial-penetration welds to the underside of the closure head. Inlet and outlet nozzles are spaced evenly around the vessel. Outlet nozzles are arranged on the vessel to facilitate optimum layout of RCS equipment. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear incore instrumentation. Each nozzle consists of a tubular member made of an Inconel stainless steel composite tube. Each tube is attached to the inside of the bottom head by a partial-penetration weld.

Internal surfaces of the vessel that are in contact with primary coolant are weld overlaid with 0.156-in. minimum of stainless steel or Inconel. The exterior of the reactor vessel is insulated with canned stainless steel reflective sheets. The insulation is 3 in. thick and contoured to enclose the top, sides, and bottom of the vessel. All the insulation modules are removable, but access to vessel-side insulation is limited by the surrounding concrete.

The chemical composition of the reactor vessel's beltline region (as defined by Paragraph II. H, Appendix G, 10 CFR 50) base and weld materials for Unit No. 1 is identified in tables 5.4-3 and 5.4-5; for Unit No. 2, consult tables 5.4-7 and 5.4-9. The materials' locations are shown in figure 5.4-3 for Unit No. 1 and figure 5.4-4 for Unit No. 2. Also included in table 5.4-4 and 5.4-6 is projected End of License (54 EFPY) Upper Shelf Energy Values for the reactor vessel beltline region plates and welds, respectively, for Unit No. 1. Tables 5.4-8 and 5.4-10 provide the same information for Unit No. 2. Additional information may be found in each unit’s Pressure Temperature Limits Report (PTLR).

5.4.2.1 **<u>Fabrication Processes</u>**

A. The use of severely sensitized stainless steel as a pressure boundary material has been prohibited and has been eliminated by either a select choice of material or by programming the method of assembly. This restriction on the use of sensitized stainless steel has been established to provide the primary system with preferential materials suitable for:

   1. Improved resistance to contaminants during shop fabrication, shipment, construction, and operation.

   2. Application in critical areas.

B. Minimum preheat requirements have been established for pressure boundary welds using low-alloy weld material. Special preheat requirements have been added for stainless steel cladding of low-stressed areas. Preheat must be maintained until post-weld heat treatment, except for overlay cladding where it may be lowered to ambient temperature under restrictive conditions. The purpose of placing limitations on preheat requirements is the addition of precautionary measures to decrease the probabilities of weld cracking by decreasing temperature gradients, lowering susceptibility to brittle transformation, preventing hydrogen embrittlement, and reducing peak hardness.

C. The CRDM head adaptor threads and surfaces of the guide studs are chrome plated to prevent possible galling of the mated parts.

D. At all locations in the reactor vessel where stainless steel and Inconel are joined, the final joining beads are Inconel weld metal in order to prevent cracking.

E. Core region shells fabricated of plate material have longitudinal welds and are angularly located away from the peak neutron exposure experienced in the vessel.

F. During fabrication, Combustion Engineering (CE) utilized several types of tests and reports to ensure the acceptability of welding materials and their conformance to applicable specifications as described below:

1. Vendor Mill Test Reports - CE has reported that each shipment of bare wire and shielded metal arc electrodes is accompanied by a vendor mill test report from the manufacturer of the material. The report identifies the degree to which the material conforms to the applicable purchase specifications.

2. Wire Alloy Verification Tests - CE has a quality program which requires that each end of each coil of welding wire be tested for selected chemical elements to assure that the material is homogeneous and conforms to the purchase specification. Additionally, CE specifies that only one heat of wire and one weld splice may be used in any one coil of wire.

3. Weld Deposit Test Plates - After receiving welding wire and flux material from the vendor, CE also prepares a weld test plate which may then be utilized during shop fabrication. This test plate is analyzed chemically and mechanically to assure compliance to all required codes and specifications in the as-deposited condition. This test is considered the most significant indication of material acceptability. Shielded metal arc electrodes are also tested in this manner. The tables in Section V of the CE generic report identify the number and dates of tests performed on weld deposit test plates.

   In regard to other tests, such as procedure qualifications, welder performance tests, or in process checks, CE has reported that it does not maintain a record of the specific lots of shielded metal electrode or combinations of heat or submerged arc wire and lot of flux which are used for procedure qualification of welder performance test. This is in accordance with the ASME Code. The Code requires only that materials with similar properties be used for this type of testing. Therefore, only accepted materials are used for procedure qualification test results and welder performance test results. These tests are not to justify the acceptability of specific heats or lots of material. However, all procedures and all welders/welding operators used in reactor vessel fabrication have been qualified to the applicable requirements of the codes and customer specifications.

G. Concerning selecting and testing of weld materials, CE responded to IE Bulletin 78-12, 12A, and 12B "Atypical Weld Material in Reactor Pressure Vessel Welds," with the following information:

1. The type, form, identifying heat and lot numbers, and manufacturer of weld material is found in Section V of the generic CE response to the NRC. Both wire/flux combination and 8018 electrode information are presented.

2. The acceptance criteria established for weld material and completed weldments are found in Section VI of the CE response.

3. Section V of the Combustion Engineering generic report provides summary tables of the results of the test performed on weld deposit test plates for combinations of submerged arc wire and lot of flux, and for lots of shielded metal electrodes.

   The acceptance criteria for these tests are described in Section VI. Unless otherwise noted, the results of the test met the applicable acceptance criteria.

   In the CE generic test report, representative test results are presented in Section VIII (submerged arc wire/flux) and Section IX (8018 shielded metal arc electrodes). At least one report for each heat of submerged arc wire is provided. For shielded metal arc electrodes, a test report is provided for every tenth lot.

4. The tables in Section V include a column entitled "Refer to Attached Nonconformance Report." An entry in this column indicates that a nonconformance was identified and dispositioned. Details of the nature of the nonconformance and its disposition are included in Section VII. All nonconformances were found to be properly dispositioned.

H. Since CE does not maintain an inventory of archive materials for the welds represented in the generic report, Westinghouse has inventoried archive surveillance weldment material which could be used for verification purposes. This material consists of full-thickness weldments of weld wire heat number BOLA (MMA welding process). This information is contained in WCAP-8956, dated August 1977.

Principal design parameters of the reactor vessel are given in table 5.4-1.

5.4.2.2 Protection of Closure Studs

Refueling procedures require the studs, nuts, and washers to be removed from the reactor closure and to be placed in storage racks during preparations for refueling. The storage racks are then removed from the refueling cavity and stored at convenient locations on the containment operating deck prior to reactor closure removal and refueling cavity flooding. Therefore, the reactor closure studs are never exposed to the borated refueling cavity water.

The stud holes in the reactor flange are sealed with special plugs before removing the reactor closure, thus preventing leakage of the borated refueling water into the stud holes.

5.4.3 EVALUATION

5.4.3.1 Steady-State Stresses

Evaluation of steady-state stresses is discussed in paragraphs 5.2.1.5 and 5.2.1.10.

5.4.3.2 Fatigue Analysis Based on Transient Stresses

Fatigue analysis based on transient stresses is discussed in paragraph 5.2.1.10.

5.4.3.3 Thermal Stresses Caused By Gamma Heating

The stresses caused by gamma heating in the vessel wall are also calculated by the vessel vendor and combined with the other design stresses. They are compared with the code allowable limit for mechanical plus thermal stress intensities to verify that they are acceptable. The gamma stresses are low and thus have a negligible effect on the stress intensity in the vessel.

5.4.3.4 Thermal Stresses Caused By Loss-of-Coolant Accident

In the event of a large LOCA, the RCS rapidly depressurizes, and the loss of coolant may empty the reactor vessel. If the reactor is at normal operating conditions before the accident, the reactor vessel temperature is approximately 550°F; and if the plant has been in operation for some time, part of the reactor vessel is irradiated. At an early stage in the depressurization transient, the emergency core cooling system (ECCS) rapidly injects cold coolant into the reactor vessel. This results in thermal stress in the vessel wall. To evaluate the effect of the stress, three possible modes of failure are considered: ductile yielding, brittle fracture, and fatigue.

A. Ductile Mode

The failure criterion used for this evaluation is that there will be no gross yielding across the vessel wall, using the material yield stress specified in Section III of the ASME Boiler and Pressure Vessel Code. The combined pressure and thermal stresses during injection through the vessel thickness as a function of time have been calculated and compared to the material yield stress at the times during the safety injection transient.

The results of the analyses showed that local yielding may occur only in approximately the inner 18 percent of the base metal and in the vessel cladding, complying with the above criterion.

B. Brittle Mode

The possibility of a brittle fracture of the irradiated core region has been considered, utilizing fracture mechanics concepts. This analysis is performed assuming the effects of water temperature, heat transfer coefficients, and fracture toughness as a function of time, temperature, and irradiation.

Both a local crack effect and a continuous crack effect have been considered, with the latter requiring the use of a rigorous finite-element axisymmetric code. It is concluded from the analysis that if the nuclear steam supply system (NSSS) sustains a large LOCA, the integrity of the reactor pressure vessel would be maintained and the plant could be shut down in an orderly manner.

C. Fatigue Mode

The failure criterion used for the failure analysis is as presented in Section III of the ASME Boiler and Pressure Vessel Code. In this method, the piece is assumed to fail once the combined usage factor at the most critical location for all transients applied to the vessel exceeds the code allowable usage factor of one.

The location in the vessel below the nozzle level which will see the emergency core cooling water and have the highest usage factor will be the incore instrumentation tube attachment welds to the vessel bottom head. As a worst case assumption, the incore instrumentation tubes and attachment penetration welds are considered to be quenched to the cooling water temperature while the vessel wall maintains its initial temperature before the start of the transient. The maximum possible pressure stress during the transient is also taken into account. This method of analysis is quite conservative and yields calculated stresses greater than would actually be experienced. The resulting usage factor for the instrument tube welds, considering all the operating transients, including the safety injection transient occurring at the end of the plant life, is below 0.2, which compares favorably with the code-allowable usage factor of 1.0.

It is concluded from the results of these analyses that the delivery of cold emergency core cooling water to the reactor vessel following a LOCA does not cause any loss of integrity of the vessel.

5.4.3.5 Heatup and Cooldown

Heatup and cooldown requirements for the reactor vessel material are discussed in section 5.2.

5.4.3.6 Irradiation Surveillance Program[a]

In the surveillance program, the evaluation of the radiation damage is based on preirradiation testing of Charpy V-notch and tensile specimens and postirradiation testing of Charpy V-notch and tensile fracture mechanics test specimens.

The surveillance material for beltline region base material for the Unit 1 reactor vessel is from lower shell plate B6919-1. The surveillance weldment is representative of the intermediate shell longitudinal seams (19-894A and B). The surveillance weld is fabricated with the same heat of weld wire as seams 19-894A and B but with a different flux. The program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature and fracture mechanics approaches.

The reactor vessel surveillance program uses six specimen capsules. Removal of these specimen capsules was completed for Unit 1 at refueling outage 1R21 and for Unit 2 at refueling outage 2R18. Refer to paragraph 5.4.3.6.3 (Unit 2) and 5.4.3.6.4 (Unit 1) for a description of the ex-vessel neutron dosimetry system used following removal of the last specimen capsule. The capsules are located in guide baskets welded to the outside of the neutron shield pads and are positioned directly opposite the center portion of the core. Sketches of an elevation and plan view showing the location and dimensional spacings of the capsules with relation to the core, neutron shield pads, and vessel and weld seams are shown in figures 5.4-1 and 5.4-2, respectively. The capsules can be removed when the vessel head is removed and can be replaced when the internals are removed. The six capsules contain reactor vessel steel specimens, oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting shell plate located in the core region of the reactor vessel and associated weld metal and weld heat-affected zone metal. The six capsules contain 54 tensile specimens, 360 Charpy V-notch specimens (which include weld metal and weld heat-affected zone material), and 72 CT specimens. Archive material sufficient for two additional capsules will be retained.

Dosimeters, including Ni, Cu, Fe, Co-Al, Cd-shielded Co-Al, Cd-shielded Np-237, and Cd-shielded U-238, are placed in filler blocks drilled to contain them. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low melting point alloys are included to monitor the temperature of the specimens. The specimens are enclosed in a tight-fitting stainless steel sheath to prevent corrosion and ensure good thermal conductivity. The complete capsule is helium leak-tested.

Vessel material sufficient for at least two capsules will be kept in storage should the need arise for additional replacement test capsules in the program. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent will be made for surveillance material and as-deposited weld metal.

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a. The Reactor Vessel Surveillance Program is credited as a license renewal aging management program (see chapter 18, subsection 18.2.5).

Details of the surveillance specimens are given in tables 5.4-11 and 5.4-12. The only respects in which the Farley Nuclear Plant (FNP) program deviates from the surveillance requirements of Appendix G and H, 10 CFR 50, is that the surveillance weldment is not selected per ASTM E-185-73 as required by Appendix H.

Weld metal representative of the intermediate shell longitudinal weld seam is included in the FNP Unit 2 surveillance program. The surveillance weldment has a copper content of 0.028 percent (see table 5.4-12). It should be noted that the vessel girth weld has a copper content of 0.153 percent, and therefore is more limiting than the surveillance weldment. However, the base metal in the surveillance program (from intermediate shell plate B7212-1) has a copper content of 0.20 percent. The NRC approved an exemption to the requirements of 10 CFR 50, Appendix H, paragraph II.B, explicitly approving the selection of the surveillance weldment for Farley Unit 2 on the basis that (1) the limiting beltline material was contained in the surveillance program and (2) conservative methods of analysis, contained in Regulatory Guide 1.99, to determine the radiation characteristics of the limiting beltline weld were available. As a compensatory measure, the NRC required that the Unit 2 operating limits be based on the (1) actual shift in reference temperature for plate B7212-1 as determined by impact testing; or (2) the predicted shift in reference temperature for weld seam 11-923 as determined in Regulatory Guide 1.99. The NRC determined that the Unit 2 surveillance program, with the exemptions noted above, satisfies the NRC acceptance criteria contained in GDC 31 and GDC 32.

The following dosimeters and thermal monitors are included in each of the six capsules:

Dosimeters

Iron

Copper

Nickel

Cobalt-aluminum (0.15 percent Co), cobalt-aluminum (cadmium- shielded)

U-238 (cadmium-shielded)

Np-237 (cadmium-shielded)

Thermal Monitors

97.5-percent Pb, 2.5-percent Ag (579°F melting point)

97.5-percent Pb, 1.75-percent Ag, 0.75-percent Sn (590°F melting point)

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the vessel wall, with the specimens being located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition temperature shift measurements are representative of the vessel at a later time in life. CT specimens are not tested as part of the surveillance program. The CT specimens are stored at Westinghouse in the event that additional testing becomes necessary.

The calculated cumulative maximum fast neutron exposure at the vessel inner surface is < 6.05 x 10^{19} n/cm^2 and 6.00 x 10^{19} n/cm^2 (E > 1 MeV) for Units 1 and 2, respectively.[(17)] The reactor vessel surveillance capsules are located as shown in figure 5.4-2. The capsule lead factors are given in the Pressure Temperature Limits Report (PTLR) for each unit.

Correlations between the calculations and the measurements on the irradiated samples in the capsules are described in paragraph 5.4.3.6.1. They have indicated good agreement. The calculations of the integrated flux at the vessel wall are conservative. The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure and withdrawal schedule is made by use of data on all capsules withdrawn. The schedule for removal of the capsules for postirradiation testing is given in the PTLR for each unit. NRC approval is required prior to changing a surveillance capsule withdrawal schedule. [Reference: NRC Administrative Letter 97-04.]

5.4.3.6.1 Measurement of Integrated Fast Neutron (E > 1 MeV) Flux at the Irradiation Samples

The use of passive neutron sensors such as those included in the internal surveillance capsule dosimetry sets does not yield a direct measure of the energy dependent neutron flux level at the measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average flux level and, hence, time integrated exposure (fluence) experienced by the sensors may be developed from the measurements only if the sensor characteristics and the parameters of the irradiation are well known. In particular, the following variables are of interest:

1 - The measured specific activity of each sensor
2 - The physical characteristics of each sensor
3 - The operating history of the reactor
4 - The energy response of each sensor
5 - The neutron energy spectrum at the sensor location

In this section, the procedures used to determine sensor specific activities, to develop reaction rates for individual sensors from the measured specific activities and the operating history of the reactor, and to derive key fast neutron exposure parameters from the measured reaction rates, are described.

5.4.3.6.1.1 **<u>Determination of Sensor Reaction Rates</u>.** The specific activity of each of the radiometric sensors is determined using established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor is determined by means of a lithium drifted germanium, Ge(Li), gamma spectrometer. In the case of the surveillance capsule multiple foil sensor sets, these analyses are performed by direct counting of each of the individual wires; or, as in the case of U-238 and Np-237 fission monitors, by direct counting preceded by dissolution and chemical separation of cesium from the sensor.

The irradiation history of the reactor over its operating lifetime is obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" or from other plant records. In particular, operating data are extracted on a monthly basis from reactor startup to the end of the capsule irradiation period. For the sensor sets utilized in the surveillance capsule irradiations, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation are determined from the following equation:

$$R = \frac{A}{N_0 F Y \sum_j \frac{P_j}{P_{ref}} C_j \left[1 - e^{-\mu_j}\right] e^{-\mu_j}}$$

where:

A	=	measured specific activity (dps/gm)
R	=	reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus)
N_0	=	number of target element atoms per gram of sensor
F	=	weight fraction of the target isotope in the sensor material
Y	=	number of product atoms produced per reaction
P_j	=	average core power level during irradiation period j (MW)
P_{ref}	=	maximum or reference core power level of the reactor (MW)
C_j	=	calculated ratio of ϕ (E > 1.0 MeV) during irradiation period j to the time weighted average ϕ (E > 1.0 MeV) over the entire irradiation period
λ	=	decay constant of the product isotope (s^{-1})
t_j	=	length of irradiation period j (s)
t_d	=	decay time following irradiation period j (s)

and the summation is carried out over the total number of monthly intervals comprising the total irradiation period.

In the above equation, the ratio P_j/P_{ref} accounts for month-by-month variation of power level within a given fuel cycle. The ratio C_j is calculated for each fuel cycle and accounts for the change in sensor reaction rates caused by variations in flux level due to changes in core power spatial distributions from fuel cycle to fuel cycle. For a single cycle irradiation, C_j is usually taken to be 1.0. However, for multiple cycle irradiations, particularly those employing low leakage fuel management, the additional correction must be utilized.

5.4.3.6.1.2 Corrections to Reaction Rate Data. Prior to using the measured reaction rates in the least squares adjustment procedure discussed in paragraph 5.4.3.6.1.3, additional corrections are made to the U-238 measurements to account for the presence of U-235

impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation.
In addition to corrections made for the presence if U-235 in the U-238 fission sensors, corrections are also made to both the U-238 and Np-237 sensor reaction rates to account for gamma ray induced fission reactions occurring over the course of the irradiation.

5.4.3.6.1.3 **Least Squares Adjustment Procedure.** Values of key fast neutron exposure parameters are derived from the measured reaction rates using the FERRET least squares adjustment code.(2) The FERRET approach uses the measured reaction rate data, sensor reaction cross-sections, and a calculated trial spectrum as input and proceeds to adjust the group fluxes from the trial spectrum to produce a best fit (in a least squares sense) to the measured reaction rate data. The "measured" exposure parameters along with the associated uncertainties are then obtained from the adjusted spectrum.

In the FERRET evaluations, a log-normal least squares algorithm weighs both the trial values and the measured data in accordance with the assigned uncertainties and correlations. In general, the measured values f are linearly related to the flux ϕ by some response matrix A:

$$f_i^{(s,a)} = \sum_g A_{ig}^{(s)} \varphi_g^{(\alpha)}$$

where i indexes the measured values belonging to a single data set s, g designates the energy group, and α delineates spectra that may be simultaneously adjusted. For example,

$$R_i = \sum_g \sigma_{ig} \varphi_g$$

relates a set of measured reaction rates R_i to a single spectrum ϕ_g by the multigroup reaction cross-section σ_{ig}. The log-normal approach automatically accounts for the physical constraint of positive fluxes, even with large assigned uncertainties.

In the least squares adjustment, the continuous quantities (i.e., neutron spectra and cross-sections) are approximated in a multigroup format consisting of 53 energy groups. The trial input spectrum is converted to the FERRET 53 group structure using the SAND-II code.(3) This procedure is carried out by first expanding the 47 group calculated spectrum into the SAND-II 620 group structure using a SPLINE interpolation procedure in regions where group boundaries do not coincide. The 620 point spectrum is then recollapsed into the group structure used in FERRET.

The sensor set reaction cross-sections, obtained from the ENDF/B-VI dosimetry file(4), are also collapsed into the 53 energy group structure using the SAND-II code. In this instance, the trial spectrum, as expanded to 620 groups, is employed as a weighting function in the cross-section collapsing procedure. Reaction cross-section uncertainties in the form of a 53 x 53 covariance matrix for each sensor reaction are also constructed from the information contained on the ENDF/B-VI data files. These matrices include energy group to energy group uncertainty correlations for each of the individual reactions.

Due to the importance of providing a trial spectrum that exhibits a relative energy distribution close to the actual spectrum at the sensor set locations, the neutron spectrum input to the FERRET evaluation is obtained from plant specific calculations for each dosimetry location. While the 53 x 53 group covariance matrices applicable to the sensor reaction cross-sections are developed from the cross-section data files, the covariance matrix for the input trial spectrum is constructed from the following relation:

$$M_{gg'} = R_n^2 + R_g R_{g'} P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty (i.e., complete correlation) for the set of values. The fractional uncertainties R_g specify additional random uncertainties for group g that are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\delta_{gg'} + \theta e^{-H}$$

where:

$$H = \frac{(g-g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes short range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1 when g = g' and 0 otherwise.

5.4.3.6.2 Calculation of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

Fast neutron exposure calculations for the reactor geometry are carried out using forward discrete ordinates transport techniques. Forward calculations provide the absolute exposure rate values using fuel cycle-specific core power distributions. In addition, the calculations provide the relative energy distribution of neutrons for use as input to neutron dosimetry evaluations as well as for use in relating measurement results to the actual exposure at key locations in the pressure vessel wall. The results of these calculations provide a direct comparison with all dosimetry results obtained over the operating history of the reactor.

The absolute cycle-specific data together with relative neutron energy spectra distributions provided the means to:

1 - Evaluate neutron dosimetry from surveillance capsule locations
2 - Enable a direct comparison of analytical prediction with measurement
3 - Determine plant-specific bias factors to be used in the evaluation of the best estimate exposure of the reactor pressure vessel
4 - Establish a mechanism for projection of pressure vessel exposure as the design of each new fuel cycle evolves

5.4.3.6.2.1 **Reference Forward Calculation.** The forward transport calculation for the reactor is carried out in r,θ,z geometry using the RAPTOR-M3G three-dimensional discrete ordinates code[5] and the BUGLE-96[10] cross-section library. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set. The reference forward calculation is based on core powers associated with completed fuel cycles and a core power of 2831 MWt for anticipated MUR-PU fuel cycles.

5.4.3.6.3 Ex-Vessel Neutron Dosimetry System (Unit 2 Only)

The ex-vessel neutron dosimetry system provides for continuing neutron fluence measurement after sufficient specimen material exposure has been achieved and the last of the six internal surveillance capsules has been removed from the reactor vessel. It enables verification of fast neutron exposure distributions within the reactor vessel wall beltline region and establishes a mechanism to enable long-term monitoring of this portion of the reactor vessel as required per 10 CFR 50 Appendix H. These fluence data can also support potential license renewal activities.

The neutron dosimetry is located external to the reactor vessel, allowing for ease of dosimetry removal and replacement. It is installed in the annular air gap between reactor vessel insulation and the primary concrete shield wall. The ex-vessel neutron dosimetry system is a passive system consisting of six aluminum dosimeter capsules containing radiometric monitors and four stainless steel bead chains, which are supported by tubular brackets attached to a titanium support bar. These four bead chain loops are mechanically secured at the bottom of the cavity slab by eye nuts which are welded to the dead weight floor anchor. The dosimetry titanium support bar is located approximately 6 in. above the top of the active fuel and is supported from below by two ¼-in. x 2-in. vertical posts (approximately 38 ft -6 in.) with cross brace and tension cables and is mechanically secured at the bottom of cavity slab by dead weight floor anchor. The dead weight floor anchor is, in turn, attached to Hilti KHB expansion anchors installed in the floor under the reactor vessel at plant elevation 80 ft + 3 in. The system is shown on drawings U-611432 and U-611433.

The ex-vessel neutron dosimetry measures fluence for approximately 1/8 of the vessel wall circumference, positioned relative to well known reactor features. Neutron transport calculations then determine the fluence for the entire vessel beltline wall. The system assists in the evaluation of radiation damage to the reactor vessel beltline region by measuring the fluence to this region, which can be used to predict the shift in the reference nil ductility transition temperature (RT_{NTD}). When used in conjunction with previously removed dosimetry from the internal surveillance capsules and with the results of neutron transport calculations, the ex-vessel neutron measurements allow the projection of embrittlement gradients through the reactor vessel wall with minimum uncertainty. Minimizing the uncertainty in the neutron exposure projections will help to assure that the reactor can be operated in the least restrictive mode possible with respect to:

- 10 CFR 50 Appendix G pressure/temperature limit curves for normal heatup and cooldown of the RCS;

- Emergency Response Guideline (ERG) pressure/temperature limit curves; and

- Pressurized thermal shock (PTS) RT_{NDT} screening criteria.

Comprehensive sensor sets are employed at discrete locations within the reactor cavity to characterize the neutron energy spectrum variations axially and azimuthally over the beltline region of the reactor vessel. In addition, the stainless steel gradient chains are used in conjunction with the encapsulated dosimeters to complete the mapping of the neutron environment between the discrete locations chosen for spectrum determinations.

The first replacement of irradiation dosimetry with transport stainless steel bead chain is at 2RF19 (September 2008). An irradiation interval of five fuel cycles between replacements is typical.

5.4.3.6.4 Ex-Vessel Neutron Dosimetry System (Unit 1 Only)

The ex-vessel neutron dosimetry system provides for continuing neutron fluence measurement after sufficient specimen material exposure has been achieved and the last of the six internal surveillance capsules has been removed from the reactor vessel. It enables verification of fast neutron exposure distributions within the reactor vessel wall beltline region and establishes a mechanism to enable long-term monitoring of this portion of the reactor vessel as required per 10 CFR 50 Appendix H. These fluence data can also support potential license renewal activities.

The neutron dosimetry is located external to the reactor vessel, allowing for ease of dosimetry removal and replacement. It is installed in the annular air gap between reactor vessel insulation and the primary concrete shield wall. The ex-vessel neutron dosimetry system is a passive system consisting of six aluminum dosimeter capsules containing radiometric monitors and four stainless steel gradient chains, which are bead chains connecting and supporting the dosimeter capsules. The bead chains are in turn supported by an arrangement of stainless steel hardware – tubular brackets on a support bar suspended by chains from plates welded to the reactor cavity liner plate. The bead chains are mechanically secured below the reactor vessel. The system is shown on drawings U-419920 and U-419916 to U-419918.

The ex-vessel neutron dosimetry measures fluence for approximately 1/8 of the vessel wall circumference, positioned relative to well known reactor features. Neutron transport calculations then determine the fluence for the entire vessel beltline wall. The system assists in the evaluation of radiation damage to the reactor vessel beltline region by measuring the fluence to this region, which can be used to predict the shift in the reference nil ductility transition temperature (RT_{NTD}). When used in conjunction with previously removed dosimetry from the internal surveillance capsules and with the results of neutron transport calculations, the ex-vessel neutron measurements allow the projection of embrittlement gradients through the reactor vessel wall with minimum uncertainty. Minimizing the uncertainty in the neutron exposure projections will help to assure that the reactor can be operated in the least restrictive mode possible with respect to:

- 10 CFR 50 Appendix G pressure/temperature limit curves for normal heatup and cooldown of the RCS;

- Emergency Response Guideline (ERG) pressure/temperature limit curves; and
- Pressurized thermal shock (PTS) RT_{NDT} screening criteria.

Comprehensive sensor sets are employed at discrete locations within the reactor cavity to characterize the neutron energy spectrum variations axially and azimuthally over the beltline region of the reactor vessel. In addition, the stainless steel gradient chains are used in conjunction with the encapsulated dosimeters to complete the mapping of the neutron environment between the discrete locations chosen for spectrum determinations.

5.4.3.7 Capability for Annealing the Reactor Vessel

There are no special design features that would prohibit the in situ annealing of the vessel. If the unlikely need for an annealing operation was required to restore the properties of the vessel material opposite the reactor core because of neutron irradiation damage, a metal temperature of approximately 750°F would be applied. A typical annealing operation would involve the use of a special heater assembly designed to raise the affected vessel area to the required temperature for the necessary holding period.

5.4.3.8 PWR Supplemental Surveillance Program

As part of a coordinated industry research initiative, Farley agreed to provide previously tested RPV surveillance specimens for reconstitution and reinsertion into a host reactor for further irradiation. The intent of this research is to obtain light water reactor (LWR) irradiation test data for PWR vessel materials at irradiation (fluence) levels expected to occur near or beyond the extended period of operation for the PWR fleet's 60-year renewed licenses. Results from this research will support future embrittlement trend correlations with actual LWR data versus reliance on test reactor data which tends to over predict LWR embrittlement behavior. Farley has authorized the use of the following broken Charpy V-notch (CVN) specimen halves from tested surveillance capsules:

- 15 broken CVN halves of plate B6919-1 from Farley Unit 1 tested capsule Z.
- 16 broken CVN halves of weld heat 33A277 from Farley Unit 1 tested capsule Z.
- 16 broken CVN halves of plate B7212-1 from Farley Unit 2 tested capsule V.
- 16 broken CVN halves of weld heat BOLA from Farley Unit 2 tested capsule V.

These materials will be reconstituted along with materials donated form CVN halves selected from other US PWRs for use in constructing two surveillance capsules with the intent to insert into two host reactors for approximately 10 years of additional irradiation, at which time the capsules will be withdrawn and tested. This research program does not represent a change to any element of the Farley surveillance program intended for compliance to 10 CFR 50 Appendix H. The number of CVN specimens donated represents less than 10% of the broken specimens in storage and, thus, those remaining in storage are substantially more than adequate to meet the requirement to retain archive material sufficient for the reconstitution of two additional capsules described in paragraph 5.4.3.6.

In addition to the materials donated for reconstitution, Farley agreed to allow Farley Unit 1 to serve as a host reactor for one of the two supplemental surveillance capsules fabricated under this program. The capsule was designed in accordance with Westinghouse standard surveillance capsules in order to fit and install with no design change to the reactor internals needed. Design drawings are documented in PWR Owner's Group report [8]. Installation was completed during the 27th refueling outage of Farley Unit 1 with the intent for the capsule to remain in the vessel for approximately 10 years of irradiation when it will then be removed for material testing. A description of the overall program is contained in the PSSP Capsule Fabrication Report [9] which detains the contents of the reconstituted CVN specimens and intent of the research program.

5.4.4 TESTS AND INSPECTION

[HISTORICAL][*The reactor vessel quality assurance program is given in table 5.4-2.*

5.4.4.1 Ultrasonic Examinations

A. *During fabrication, angle beam inspection of 100 percent of plate material is performed to detect discontinuities that may be undetected by longitudinal wave examination, in addition to the design code straight beam ultrasonic test.*

B. *The reactor vessel is examined after hydrotesting to provide a baseline map for use as a reference document in relation to later inservice inspections.*

5.4.4.2 Penetrant Examinations

The partial penetration welds for the control rod drive mechanism head adapter are inspected by dye penetrant after the first layer of weld metal, after each one-third of the thickness of weld metal, and at the final surface. Bottom instrumentation tube partial penetration welds are inspected by dye penetrant after the first layer and at each one-fourth inch of weld metal. Core support block attachment welds are inspected by dye penetrant after the first layer of weld metal.

This is required to detect cracks or other defects to lower the weld surface temperatures for cleanliness, and to prevent microfissures.

5.4.4.3 Magnetic Particle Examination

A. *All surfaces of quenched and tempered materials are inspected on the inside diameter prior to cladding and on the outside diameter after hydrotesting. This serves to detect possible defects resulting from the forming and heat-treatment operations.*

B. The attachment welds for the vessel supports are inspected at the final surface. Lifting lug welds are inspected after the first layer of weld metal, after each one-half inch of weld thickness, and at the final weld surface. The refueling seal ledge weld is inspected after back chipping and at the final surface of the weld.]

5.4.4.4 Inservice Inspection

The full penetration welds in the following areas of the installed irradiated reactor vessel are available for visual and/or nondestructive inspection.

A. Vessel shell - the inside surface.

B. Primary coolant nozzles - the inside surface.

C. Closure head - the inside and outside surface.

D. Bottom head - the outside surface.

E. Closure studs, nuts, and washers.

F. Field welds between the reactor vessel nozzles and the main coolant piping.

G. Vessel flange seal surface.

The design considerations that have been incorporated into the system design to permit the above inspections are as follows:

A. All reactor internals are completely removable. The tools and storage space required to permit these inspections are provided.

B. The closure head is stored dry on the reactor operating deck during refueling to facilitate direct visual inspection.

C. All reactor vessel studs, nuts, and washers are removed to dry storage during refueling.

D. Removable plugs are provided in the primary shield. The insulation covering the nozzle welds may be removed.

E. Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.

The reactor vessel presents access problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacture in preparation for the periodic nondestructive tests. These are as follows:

A. Shop ultrasonic examinations are performed on all internally-clad surfaces to acceptance and repair standards to ensure an adequate cladding bond to allow later ultrasonic testing of the base metal from inside the surface. The size of cladding bonding defect allowed is 3/4-in. diameter.

B. The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit future positioning of the test equipment without interference.

C. After the shop hydrostatic testing, selected areas of the reactor vessel are ultrasonically tested and mapped to facilitate the inservice inspection program.

5.4.4.5 Inspection of Rod Cluster Control Assemblies (RCCAs)

Rod Cluster Control Assembly visual and eddy current inspections are performed by removing the RCCA from its fuel assembly and lowering it through an inspection guide fixture temporarily mounted on the top of the spent-fuel racks. This guide fixture contains eddy current transducers and an optional TV camera. Eddy current testing is an NDE method relying on the interaction of induced alternating currents and fields with RCCA defects to produce noticeable changes in the search coil (eddy current probe) impedance.

The guide fixture is mounted on top of a vacant fuel rack cell and protrudes over adjacent cells, which shall also be vacant since adequate flow for fuel cooling purposes may not otherwise exist.

The inadvertent drop of the test equipment into the spent-fuel pit is bounded by the load drop analysis in section 9.1. Since the weight of the test equipment (approximately 150 lb) is less than the combined weight of a fuel assembly with control rods and handling tool, it does not result in a load on the fuel racks that must be included in the stress analysis and no additional seismic analysis is required.

REFERENCES

1. Bencini, R. L., "Joseph M. Farley Units 1 and 2 Radiation Analyses and Neutron Dosimetry Evaluation," WCAP-14687, June 1996.

2. F. A. Schmittroth, FERRET Data Analysis Code, HEDL-TME 79-40, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

3. McElroy, W. N., et al., “A Computer-Automated Iterative Method of Neutron Flux Spectra Determined by Foil Activation,” AFWL-TR-67-41, Volumes I-IV, Air Force Weapons Laboratory, Kirkland AFB, NM, July 1967.

4. RSIC Data Library Collection DLC-178, “SNLRML Recommended Dosimetry Cross-Section Compendium,” July 1994.

5. Westinghouse Report WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-MEG and FERRET,” July 2018

6. RSIC Data Library Collection DLC-175, “BUGLE-93, Production and Testing of the VITAMIN-B6 Fine Group and the BUGLE-93 Broad Group Neutron/Photon Cross-Section Libraries Derived from ENDF/B-VI Nuclear Data,” April 1994.

7. WCAP-18402-P, Revision 0, “Measurement Uncertainty Recapture Power Uprate program Licensing Report,” (Proprietary)

8. “PWR Supplemental Surveillance Program Capsule Drawings for Farley Unit 1 and Shearon Harris,” PWROG-16038-P Revision 0, July 2016.

9. “Materials Reliability Program: PWR Supplemental Surveillance Program (PSSP) Capsule Fabrication,” Report MRP-412, EPRI Product 3002007964, September 2016.

10. RSIC Data Library Collection DLC-185, “BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications,” March 1996.

TABLE 5.4-1

REACTOR VESSEL DESIGN PARAMETERS

Design/operating pressure (psig)	2485/2235
Design temperature (°F)	650
Overall height of vessel and closure head (ft-in.) (bottom head OD to top of control rod mechanism adapter)	42 - 7-3/16
Thickness of insulation (min, in.)	3
Number of reactor closure head studs	58
Diameter of reactor closure head studs (in.)	6
ID of flange (in.)	149-9/16
OD of flange (in.)	184
ID at shell (in.)	157
Inlet nozzle ID (in.)	27-1/2
Outlet nozzle ID (in.)	29
Clad thickness (min, in.)	5/32
Lower head thickness (min, in.)	5
Vessel beltline thickness (min, in.)	7-7/8
Closure head thickness (in.)	6-3/16

[HISTORICAL] [TABLE 5.4-2

REACTOR VESSEL QUALITY ASSURANCE PROGRAM

		RT[a]	UT[a]	PT[a]	MT[a]
Forgings					
1.	Flanges		yes		yes
2.	Studs		yes		yes
3.	Head adapters		yes	yes	
4.	Head adapter tube		yes	yes	
5.	Instrumentation tube		yes	yes	
6.	Main nozzles		yes		yes
7.	Nozzle safe ends		yes	yes	
Plates			yes		yes
Weldments					
1.	Main seam	yes	yes		yes
2.	CRD head adapter connection			yes	
3.	Instrumentation tube connection			yes	
4.	Main nozzles	yes	yes		yes
5.	Cladding		yes	yes	
6	Nozzle safe ends (forging)	yes	yes	yes	
7	Head adapter forging to head adapter tube	yes		yes	
8.	All ferritic welds accessible after hydrotest		yes		yes
9.	All nonferritic welds accessible after hydrotest		yes	yes	
10.	Seal ledge				yes
11.	Head lift lugs				yes
12.	Core pad welds		yes	yes	yes

a. RT - Radiographic
UT - Ultrasonic
PT - Dye penetrant
MT - Magnetic particle]

TABLE 5.4-3

IDENTIFICATION OF UNIT NO. 1 REACTOR VESSEL BELTLINE REGION BASE MATERIAL

Component	Code No.	Heat No.	Material Spec. No.	Composition (Wt. %) C	Mn	P	S	Si	Ni	Mo	Cu	Cr	AL
Inter. shell	B6903-2	C6294	A533B, CL.1	0.20	1.32	0.011	0.013	0.21	0.60	0.55	0.13	-	0.017
Inter. shell	B6903-3	C6308	A533B, CL.1	0.21	1.29	0.014	0.015	0.16	0.56	0.56	0.12	-	0.019
Lower shell	B6919-1	C6940	A533B, CL.1	0.20	1.39	0.015	0.015	0.18	0.55	0.56	0.14	-	0.025
Lower shell	B6919-2	C6897	A533B, CL.1	0.20	1.39	0.015	0.018	0.19	0.56	0.53	0.14	-	0.018

TABLE 5.4-4

PREDICTED END OF LICENSE (54 EFPY) UPPER SHELF ENERGY VALUES
FARLEY UNIT NO. 1 REACTOR VESSEL BELTLINE PLATES (Ref. 7)

Beltline Material	Wt. % Cu	1/4T Fluence (10^{19} n/cm^2)	Unirradiated USE (ft-lb)	Decrease in USE (%)	Projected EOL USE (ft-lb)
Intermediate Shell Plate B6903-2	0.13	3.77	99	30	69.3
Intermediate Shell Plate B6903-3	0.12	3.77	87	29	61.8
Lower Shell Plate B6919-1	0.14	3.76	86	32	58.5
Lower Shell Plate B6919-2	0.14	3.76	86	32	58.5

TABLE 5.4-5

IDENTIFICATION OF UNIT NO. 1 REACTOR VESSEL BELTLINE REGION WELD METAL

Weld Location	Weld Process	Weld Wire		Flux		Composition (Wt. %)							
		Type	Heat No.	Type	Lot No.	C	Mn	P	S	Si	Mo	Cu	Ni
Inter. shell long seams 19-894 A&B	Sub-arc	B4	33A277	Linde 1092	3889	0.11	1.27	0.015	0.010	0.14	0.49	0.258	0.165
Inter. shell to lower shell Circle Seam 11-894	Sub-arc	B4	6329637	Linde 0091	3999	0.14	1.15	0.011	0.014	0.19	0.53	0.205	0.105
Lower shell long seams 20-894 A&B	Sub-arc	B4	90099	Linde 0091	3977	0.15	1.12	0.022	0.012	0.23	0.49	0.197	0.060

TABLE 5.4-6

PREDICTED END OF LICENSE (54 EFPY) UPPER SHELF ENERGY VALUES
FARLEY UNIT NO. 1 REACTOR VESSEL BELTLINE WELDS (Ref. 7)

Beltline Material	Wt. % Cu	1/4T Fluence (10^{19} n/cm^2)	Unirradiated USE (ft-lb)	Decrease in USE (%)	Projected EOL USE (ft-lb)
Intermediate Shell Longitudinal Welds 19-894 A & B using Surveillance Capsule Data	0.258	1.14	149	20	119.2
Circumferential Weld 11-894	0.205	3.75	104	47	55.1
Lower Shell Longitudinal Welds 20-894 A & B	0.197	1.15	82.5	36	52.8

TABLE 5.4-7

IDENTIFICATION OF UNIT NO. 2 REACTOR VESSEL BELTLINE REGION BASE MATERIAL (wt%)

Component	Code No.	Heat No.	Material Spec. No.	C	Mn	P	S	Si	Ni	Mo	Cu	Cr	Al
Inter. shell	B7203-1	C6309-2	A533B, CL.1	0.20	1.30	0.010	0.013	0.19	0.60	0.55	0.14	-	0.020
Inter. shell	B7212-1	C7466-1	A533B, CL.1	0.21	1.30	0.018	0.016	0.24	0.60	0.49	0.20	0.15	0.040
Lower shell	B7210-1	C6888-2	A533B, CL.1	0.24	1.28	0.010	0.014	0.20	0.56	0.56	0.13	-	0.020
Lower shell	B7210-2	C6293-1	A533B, CL.1	0.19	1.30	0.015	0.015	0.18	0.57	0.59	0.14	-	0.026

TABLE 5.4-8

PREDICTED END OF LICENSE (54 EFPY) UPPER SHELF ENERGY VALUES FARLEY UNIT NO. 2 REACTOR VESSEL BELTLINE PLATES (Ref. 7)

Beltline Material	Wt. % Cu	1/4T Fluence (10^{19} n/cm^2)	Unirradiated USE (ft-lb)	Decrease in USE (%)	Projected EOL USE (ft-lb)
Intermediate Shell Plate B7203-1	0.14	3.74	100	32	68
Intermediate Shell Plate B7212-1 using Surveillance Capsule Data	0.20	3.74	100	42	58
Lower Shell Plate B7210-1	0.13	3.74	103	30	72.1
Lower Shell Plate B7210-2	0.14	3.74	99	32	67.3

TABLE 5.4-9

IDENTIFICATION OF UNIT NO. 2 REACTOR VESSEL BELTLINE REGION WELD METAL

Weld Location	Welding Process	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	C	Mn	P	S	Si	Mo	Cu	V	Ni
Inter. shell long. seam 19-923A	SMAW	E8018C3	HODA	-	-	0.09	1.00	0.009	0.010	0.38	0.25	0.027	0.010	0.947
	SMAW	E8018C3	BOLA	-	-	0.09	0.95	0.004	0.014	0.34	0.23	0.027	0.006	0.913
Inter. shell long. seam 19-923B	SMAW	E8018C3	BOLA	-	-	0.09	0.95	0.004	0.014	0.34	0.23	0.027	0.006	0.913
Inter. shell to lower shell circle seam 11-923	Sub-arc	B4	5P5622	Linde 0091	1122	0.17	1.29	0.016	0.008	0.19	0.57	0.153	0.009	0.077
Lower shell long. seams 20-923 A&B	Sub-arc	B4	83640	Linde 0091	3490	0.16	1.22	0.006	0.011	0.19	0.57	0.051	0.006	0.096

TABLE 5.4-10

PREDICTED END OF LICENSE (54 EFPY) UPPER SHELF ENERGY VALUES
FARLEY UNIT NO. 2 REACTOR VESSEL BELTLINE WELDS (Ref. 7)

Beltline Material	Wt. % Cu	1/4T Fluence (10^{19} n/cm^2)	Unirradiated USE (ft-lb)	Decrease in USE (%)	Projected EOL USE (ft-lb)
Intermediate Shell Longitudinal Welds 19-923A, Heat # HODA	0.27	1.15	131	20	104.8
Intermediate Shell Longitudinal Welds 19-923A &B Heat # BOLA using Surveillance Capsule data	0.027	1.15	148	10	133.2
Circumferential Weld 11-923, Heat # 5P5622	0.153	3.74	102	40	61.2
Lower Shell Longitudinal Welds 20-923 A & B, Heat # 83640	0.051	1.17	126	20	100.8

TABLE 5.4-11 (SHEET 1 OF 2)

SURVEILLANCE MATERIAL
BELTLINE LOCATION AND FABRICATION HISTORY - FARLEY UNIT NO. 1

Surveillance Material	Beltline Location of Surveillance Material	Heat-Treatment	
Base metal	Inter. shell plate B6919-1	1550 - 1650°F 1200 - 1250°F 1125 - 1175°F	4 hr-WQ 4 hr-AC 40 hr-FC to 600°F
Weld metal	Inter. shell longitudinal Weld seams 19-894 A & B	1125 - 1175°F	16 hr-FC

SURVEILLANCE TEST SPECIMENS - TYPE, ORIENTATION, AND QUANTITY PER TEST CAPSULE - FARLEY UNIT NO. 1

Surveillance Material	Specimen Orientation	Charpy-V	Tensile	1/2T-CT	Bend Bar
Base metal (plate B6919-1)	Transverse	15	3	4	1
Base metal (plate B6919-1)	Longitudinal	15	3	4	-
Weld metal	Transverse	15	3	4	-
HAZ metal (plate B6919-1)	Longitudinal	15	-	-	-

TABLE 5.4-11 (SHEET 2 OF 2)

Surveillance Material	Beltline Location of Surveillance Material	Heat-Treatment
Base metal	Inter. shell plate B7212-1	1550 - 1650°F - 4 h-WQ, 1200 - 1250°F - 4 h-AC, 1125 - 1175°F - 18 h-FC
Weld metal[(a)]	Inter. shell long. weld seam	1125 - 1175°F - 13 h-FC
HAZ metal	Inter. shell plate B7212-1	1125 - 1175°F - 13 h-FC

SURVEILLANCE TEST SPECIMENS - TYPE, ORIENTATION, AND QUANTITY PER TEST CAPSULE - FARLEY UNIT NO. 2

Surveillance Material	Specimen Orientation	Charpy-V	Tensile	1/2T-CT
Base metal (plate B7212-1)	Transverse	15	3	4
Base metal (plate B7212-1)	Longitudinal	15	3	4
Weld metal	Transverse	15	3	4
HAZ metal (plate B7212-1)	Longitudinal	15	-	-

a. Surveillance weldment fabricated using plate B7212-1 and B7203-1. Surveillance weldment was fabricated using the same type of wire (E8018C3) and the same heat of wire (heat No. BOLA) as was used to fabricate the intermediate shell longitudinal weld seam (19-923B) in the vessel. The same welding procedures (MA-511-D and A-244-110-8) were used by the vessel supplier to fabricate the surveillance weldment and the intermediate shell longitudinal weld seam (19-923B).

TABLE 5.4-12 (SHEET 1 OF 2)

SURVEILLANCE MATERIAL CHEMICAL COMPOSITION (wt%) – FARLEY UNIT NO. 1

Element	Plate B6919-1		Weld Metal
	Combustion Engineering Analysis	Westinghouse Analysis	Westinghouse Analysis
C	0.20	--	0.13
S	0.015	0.013	0.009
N_2	--	0.003	0.005
Co	0.008	0.16	0.018
Cu	0.14	0.10	0.014
Si	0.18	0.28	0.27
Mo	0.56	0.51	0.50
Ni	0.55	0.56	0.19
Mn	1.39	1.40	1.06
Cr	--	0.13	0.063
V	--	<0.001	0.003
P	0.015	0.015	0.016
Sn	--	0.008	0.005
A1	0.025	--	0.009

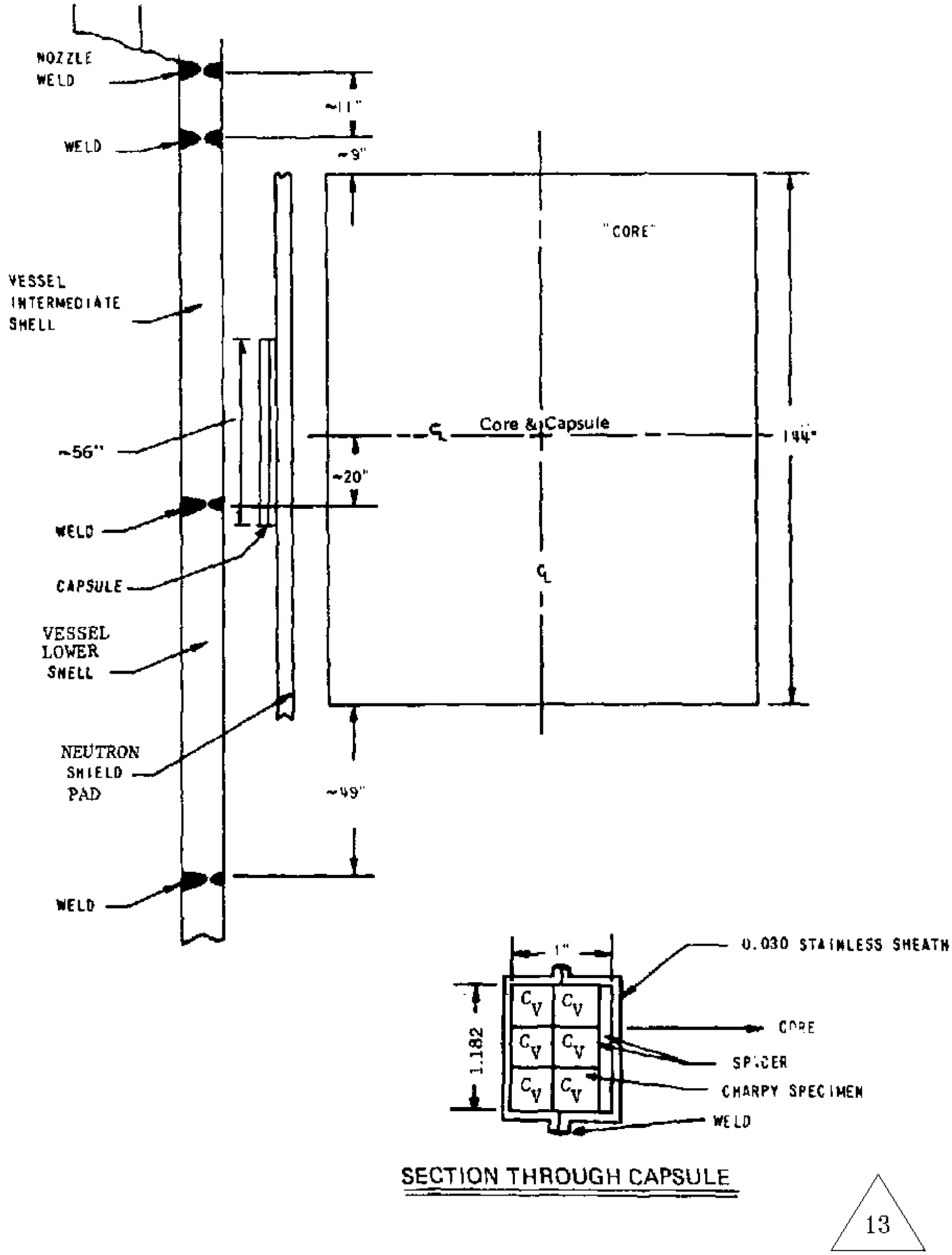
The surveillance weld was fabricated from sections of plate B6919-1 and adjoining intermediate shell plate B6903-2, using weld wire representative of that used in the original fabrication.

TABLE 5.4-12 (SHEET 2 OF 2)

SURVEILLANCE MATERIAL CHEMICAL COMPOSITION (wt%) – FARLEY UNIT NO. 2

Element	Plate B7212-1	Weld Metal
C	0.21	<0.086
Mn	1.30	0.95
P	0.018	0.004
S	0.016	0.014
Si	0.24	0.34
Ni	0.60	0.89
Cr	0.15	<0.01
Mo	0.49	0.23
Cu	0.20	0.028
V	0.003	0.006
Co	0.027	0.010
Sn	0.011	0.002
A1	0.040	0.003
N_2	0.006	0.007

The surveillance weldment was fabricated with the same type of wire and the same heat of wire (wire type E8018C3 and wire heat No. BOLA) as was used to fabricate the longitudinal weld seam (19-923 B) in the intermediate shell course of the vessel. The same welding procedures were used to fabricate the surveillance weldment and the vessel weld seam (19-923 B).



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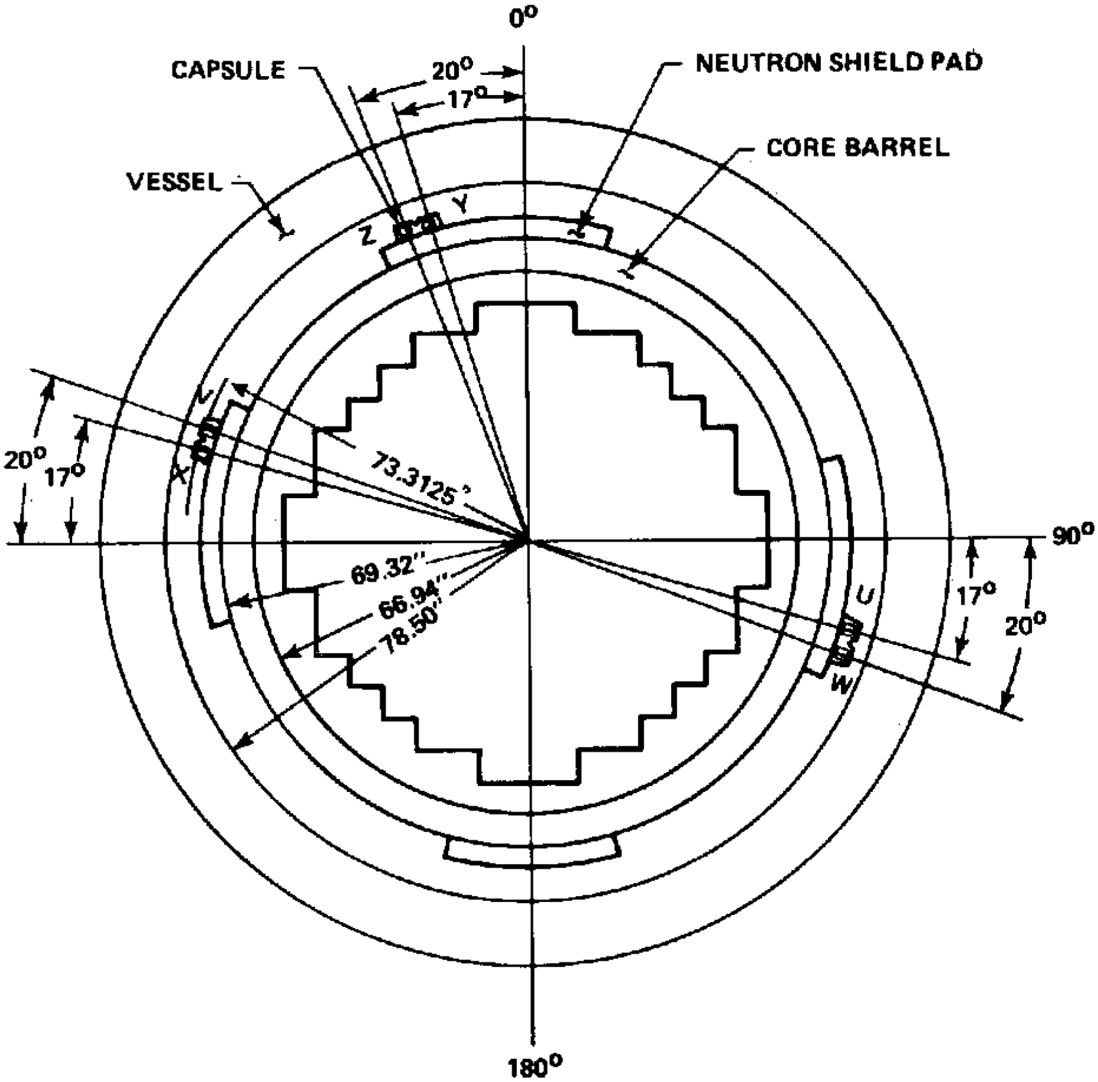
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SURVEILLANCE CAPSULE ELEVATION VIEW

FIGURE 5.4-1



VERTICAL WELD SEAM LOCATIONS

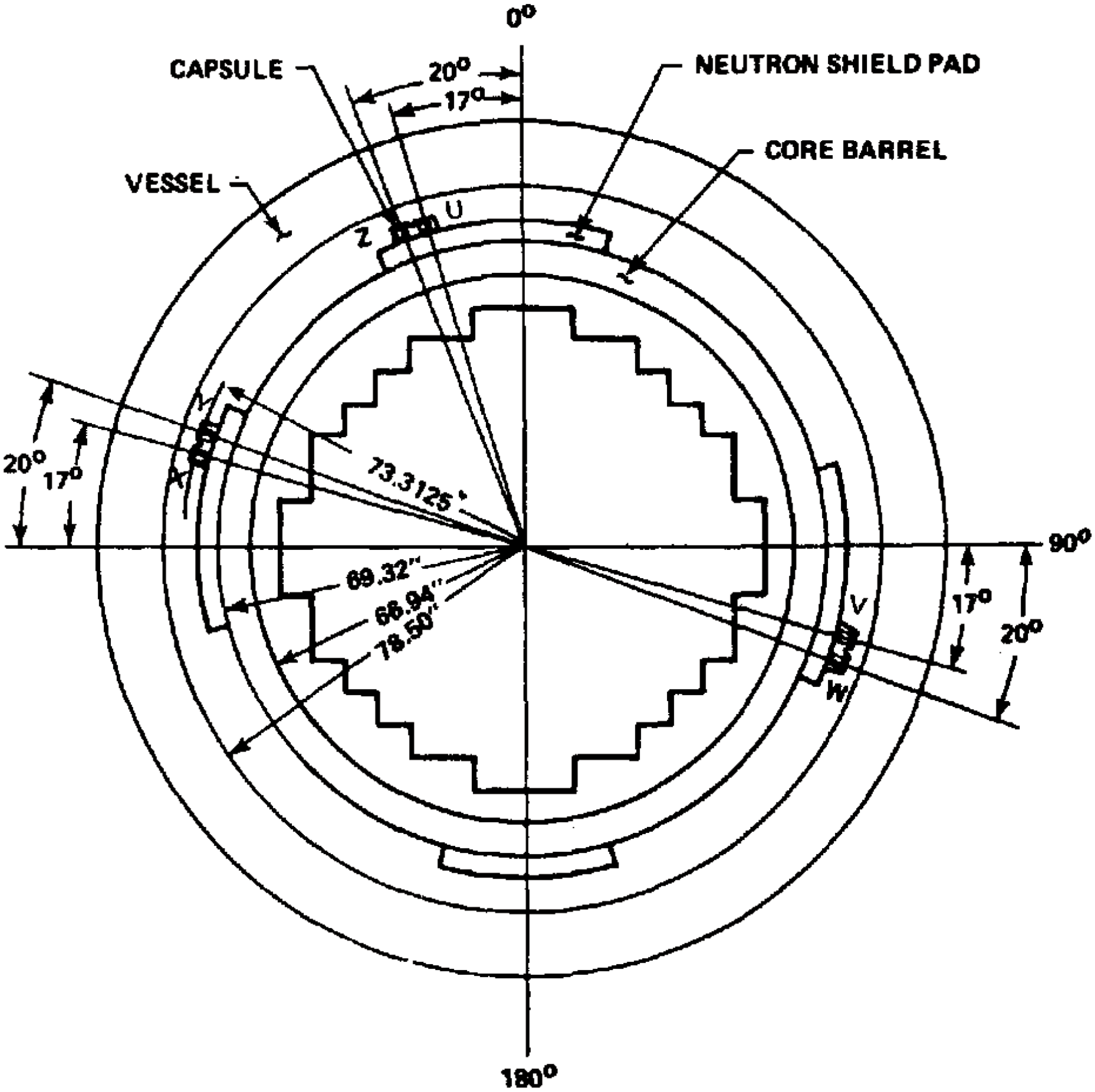
INT. SHELL 45° & 225°

LOWER SHELL 135° & 315°

13

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	UNIT 1 SURVEILLANCE CAPSULE PLAN VIEW
		FIGURE 5.4-2 (SHEET 1 OF 2)



VERTICAL WELD SEAM LOCATIONS

INT. SHELL 45° & 225°

LOWER SHELL 135° & 315°

13

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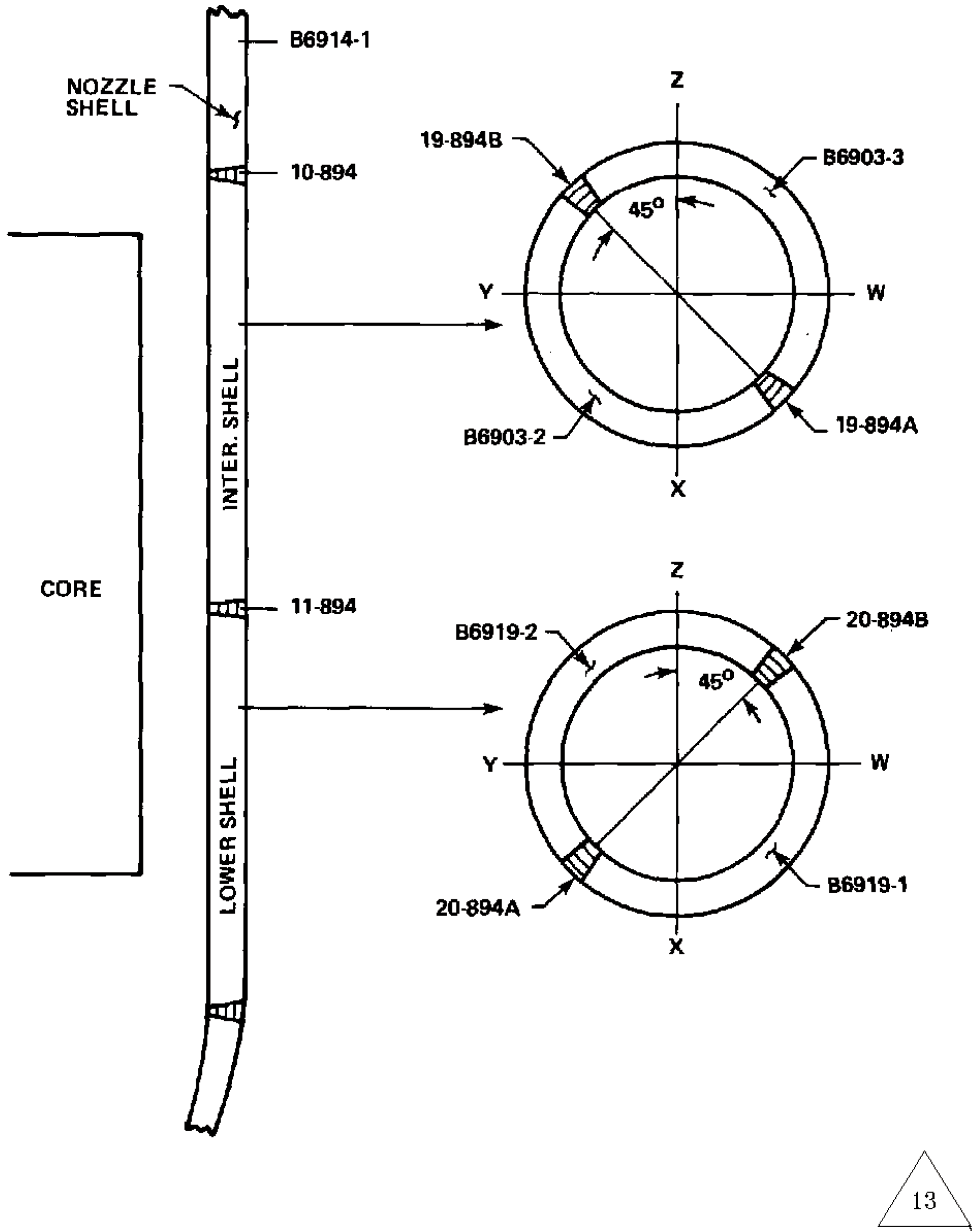
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNIT 2 SURVEILLANCE CAPSULE PLAN VIEW

FIGURE 5.4-2 (SHEET 2 OF 2)



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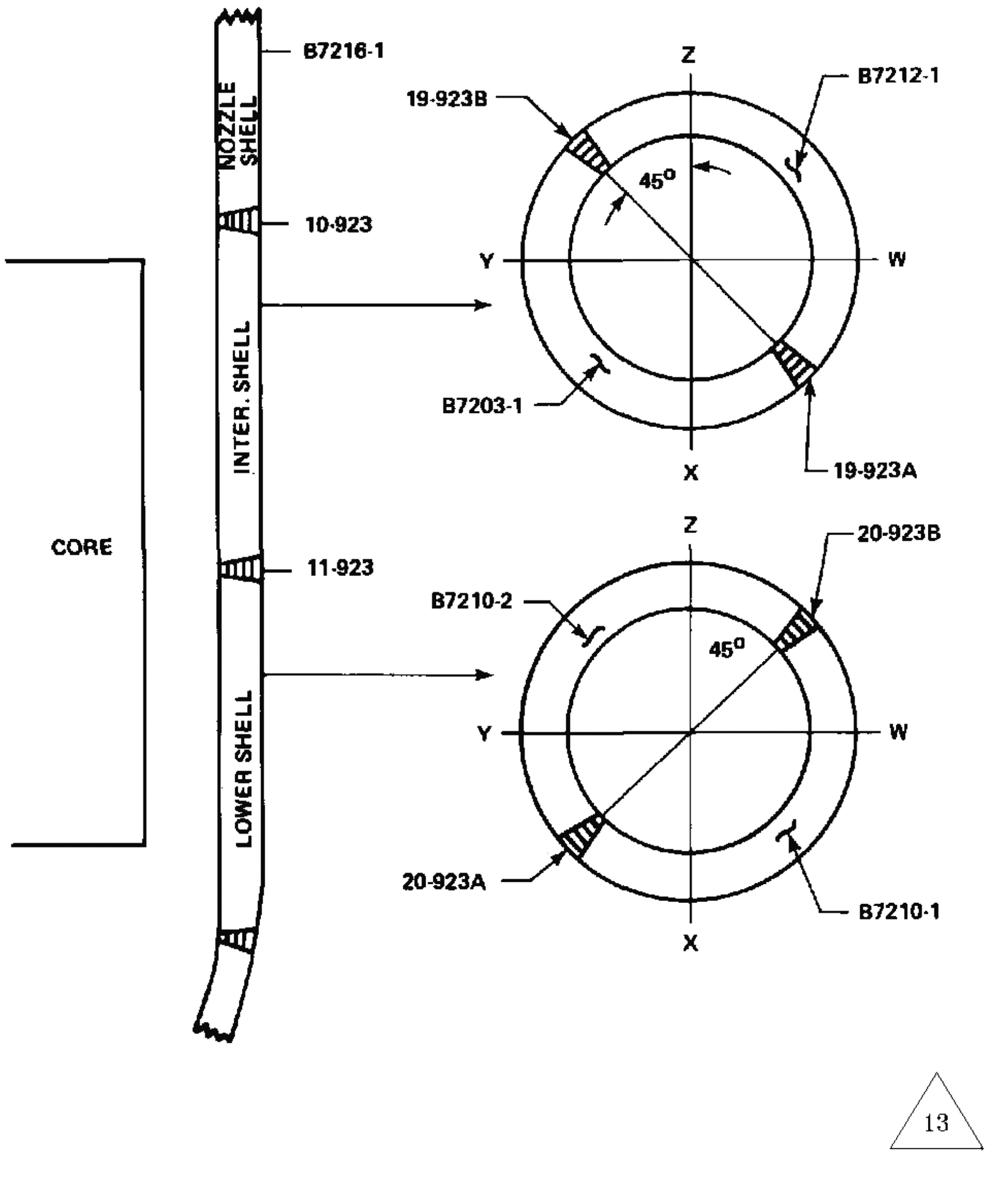
IDENTIFICATION AND LOCATION OF FARLEY UNIT NO. 1
REACTOR VESSEL BELTLINE REGION MATERIAL

FIGURE 5.4-3



REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	IDENTIFICATION AND LOCATION OF FARLEY UNIT NO. 2 REACTOR VESSEL BELTLINE REGION MATERIAL FIGURE 5.4-4

5.5 COMPONENT AND SUBSYSTEM DESIGN

5.5.1 REACTOR COOLANT PUMPS

5.5.1.1 Design Bases

The reactor coolant pump (RCP) ensures an adequate core cooling flowrate, and hence sufficient heat transfer, to maintain departure from nucleate boiling ratio (DNBR) greater than the safety analysis limit within the parameters of operation. The required net positive suction head (NPSH) is, by conservative pump design, always less than that available by system design and operation.

Sufficient pump rotation inertia is provided by a flywheel, in conjunction with the impeller and motor assembly, to provide adequate flow during coastdown. This flow following an assumed loss of pump power provides the core with adequate cooling.

The pump is capable of operation without mechanical damage at overspeeds up to and including 125 percent of normal speed. The RCP is shown in figure 5.5-1. The RCP design parameters are given in table 5.5-1.

Code requirements are provided in section 3.2. Material requirements are discussed in subsection 5.2.3.

5.5.1.2 Design Description

The RCP is a vertical, single-stage, centrifugal, shaft seal pump designed to pump large volumes of main coolant at high temperatures and pressures.

The pump consists of three areas from bottom to top. They are the hydraulics, the shaft seals, and the motor.

A. The hydraulic section consists of an impeller, diffuser, casing, thermal barrier, heat exchanger, lower radial bearing, main flange, motor stand, and pump shaft.

B. The shaft seal section consists of four devices. They are the No. 1 controlled leakage film-riding face seal, the No. 2 and No. 3 rubbing face seals, and a shutdown seal assembly. The shutdown seal is housed within the No. 1 seal area and is a passive device activating only on high temperature if seal cooling is lost. These seals are contained within the main flange and seal housing.

C. The motor section consists of a vertical, solid shaft; a squirrel-cage, induction-type motor; an oil-lubricated, double Kingsbury-type thrust bearing; two oil-lubricated radial bearings; and a flywheel.

Attached to the bottom of the pump shaft is the impeller. The reactor coolant is drawn up through the impeller, discharged through passages in the diffuser, and out through the discharge nozzle in the side of the casing. Above the impeller is a thermal barrier heat exchanger which limits heat transfer between hot system water and seal injection water.

High-pressure seal injection water is introduced through the thermal barrier wall. A portion of this water flows through the seals; the remainder flows down the shaft and through and around the bearing and thermal barrier, where it acts as a buffer to prevent system water from entering the radial bearing and seal section of the unit. The heat exchanger provides a means of cooling system water to an acceptable level in the event that seal injection flow is lost. The water-lubricated, journal-type bearing mounted above the thermal barrier heat exchanger has a self-aligning spherical seat.

The RCP 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.

The motor is an air-cooled, Class B thermalastic epoxy-insulated, squirrel-cage induction motor. The rotor and stator are of standard construction and are cooled by air. Six resistance temperature detectors are located throughout the stator to sense the winding temperature. The top of the motor consists of a flywheel and an antireverse-rotation device.

Each of the RCP assemblies is equipped for continuous monitoring of RCP shaft and frame vibration levels. Shaft vibration is measured by two relative shaft probes mounted on top of the pump seal housing; the probes are located 90° apart in the same horizontal plane and mounted near the pump shaft. Frame vibration is measured by two velocity seismoprobes located 90° apart and mounted at the top of the motor support stand. Proximeters and converters provide output of the probe signals, which are displayed on meters in the electrical penetration room and annunciated in the control room. These meters automatically indicate the highest output from the relative shaft probes and the frame seismoprobes. Manual selection allows monitoring of individual probes. Indicator lights display caution and danger limits of vibration, and are adjustable over the full range of the meter scale.

All parts of the pump in contact with the reactor coolant are austenitic stainless steel except for seals, bearings, and special parts. Component cooling water is supplied to the two oil coolers on the pump motor and to the pump thermal barrier heat exchanger.

The pump shaft, seal housing, thermal barrier, main flange, and motor stand can be removed from the casing as a unit without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

The performance characteristic shown in figure 5.5-2 is common to all of the fixed-speed, mixed-flow pumps, and the "knee" at about 45-percent design flow introduces no operational restrictions, since the pumps operate at full speed.

5.5.1.3 Design Evaluation

5.5.1.3.1 Pump Performance

The RCPs are sized to deliver flow at rates which equal or exceed the required flowrates. Initial reactor coolant system (RCS) tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial plant operation.

The reactor trip system ensures that pump operation is within the assumptions used for loss-of-coolant flow analyses, which also assures that adequate core cooling is provided to permit an orderly reduction in power if flow from a RCP is lost during operation.

An extensive test program has been conducted for several years to develop the controlled-leakage, shaft-seal for pressurized-water reactor (PWR) applications. Long-term tests were conducted on less than full-scale prototype seals as well as on full-size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled-leakage, shaft seal pump design.

The support of the stationary member of the No. 1 seal ("seal ring") is such as to allow large deflections, both axial and tilting, while still maintaining its controlled gap relative to the seal runner. Even if all the graphite were removed from the pump bearing, the shaft could not deflect far enough to cause opening of the controlled-leakage gap. The "spring rate" of the hydraulic forces associated with the maintenance of the gap is high enough to ensure that the ring follows the runner under very rapid shaft deflections.

Testing of pumps with the No. 1 seal entirely removed (full reactor pressure on the No. 2 seal) shows that relatively small leakage rates would be maintained for long periods of time (approximately 100 h) even if the No. 1 seal fails entirely. The plant operator is warned of this condition by the increase in the No. 1 seal leakoff and has time to safely shut down the reactor without significant leakage of reactor coolant to the containment. Thus, it may be concluded that gross leakage from the pump would not occur even if seals were to suffer physical damage.

The effect of loss of offsite power on the pump itself is to cause a temporary stoppage in the supply of injection flow to the pump seals and also of the cooling water for seal and bearing cooling. The emergency diesel generators are started automatically upon loss of offsite power so that component cooling flow is automatically restored. Seal water injection flow is automatically restored when the charging pump is started by the diesel sequencer.

In the event of a station blackout (SBO), the shutdown seal will deploy on high seal cooling water temperature to limit leakage from the RCP seal package. Leakage is limited when a thermal actuator retracts causing a polymer seal to clamp down around the No. 1 seal sleeve (reference 8).

5.5.1.3.2 Coastdown Capability

It is important to reactor operation that the reactor coolant continues to flow for a short time after reactor trip. In order to provide this flow in a station blackout condition, each RCP is provided with a flywheel. Thus, the rotating inertia of the pump, motor, and flywheel is employed during the coastdown period to continue the reactor coolant flow. The coastdown flow transients are provided in the figures in section 15.3.

The pump is designed for the safe shutdown earthquake (SSE) at the site, and the integrity of the bearings is described in paragraph 5.5.1.3.4. Hence, it is concluded that the coastdown capability of the pumps is maintained even under the most adverse case of a blackout coincident with SSE. Core flow transients and figures are provided in subsection 15.2.5 and 15.3.4.

5.5.1.3.3 Flywheel Integrity

Demonstration of integrity of the RCP flywheel is discussed in subsection 5.2.6.

5.5.1.3.4 Bearing Integrity

The design requirements for the RCP bearings are primarily aimed at ensuring a long life with negligible wear, in order 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, do not begin to approach loads that cannot be adequately carried for short periods of time.

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

Low oil levels in the motor bearings signal an alarm in the control room and require the operators to initiate immediate and supplementary actions. Each motor bearing contains embedded temperature detectors, so initiation of failure separate from loss of oil is indicated and alarmed in the control room as a high bearing temperature. This requires pump shutdown. Even if these indications were ignored and the bearing proceeded to failure, the low melting point of Babbitt metal on the pad surfaces would ensure that no sudden seizure of the bearing could occur. In this event, the motor continues to drive, as it has sufficient reserve capacity to operate even under such conditions. However, it demands excessive currents and, at some stage, is shutdown because of high current demand.

The RCP shaft is designed so that its critical speed is well above the operating speed.

5.5.1.3.5 Locked Rotor

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 fails in torsion just below the coupling to the motor, disengaging the flywheel and motor from the shaft. This constitutes a loss of coolant flow in the loop. Following such a postulated seizure, the motor continues to run without any overspeed, and the flywheel maintains its integrity because it is still supported on a shaft with two bearings. Flow transients and figures are provided in subsection 15.4.4.

There are no credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing is precluded by graphite in the bearing. Any seizure in the seals results in a shearing of the antirotation pin in the seal ring. Although actuation of the shutdown seal on a rotating assembly could minimally and temporarily affect RCP coastdown, the capability to provide sufficient cooling flow to the reactor core will be unaffected. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions are initially given by high temperature signals from the bearing water temperature detector and excessive No. 1 seal leakoff indications, respectively. Following these signals, pump vibration levels are checked. Excessive vibration indicates mechanical trouble and the pump is shut down for investigation.

5.5.1.3.6 Critical Speed

It is considered desirable to operate below first critical speed, and the RCPs 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.

5.5.1.3.7 Missile Generation

Each component of the pump is 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.

5.5.1.3.8 Pump Cavitation

The minimum NPSH required by the RCP at running speed is approximately 192-ft head (approximately 85 psi). In order for the controlled leakage seal to operate correctly, it is necessary to have a differential pressure of approximately 200 psi across the seal. This results in a requirement for a minimum of 325 psi pressure in the primary loop before the RCP may be operated. This is taken into consideration in the operating instructions. At this pressure, the NPSH requirement is exceeded and no limitation on pump operation occurs from this source.

5.5.1.3.9 Pump Overspeed Considerations

For turbine trips actuated by either the reactor trip system or the turbine protection system (except for turbine trips resulting from thrust bearing failure), the generator and RCPs are maintained connected to the external network for 30 s to prevent any pump overspeed condition.

An electrical fault requiring immediate trip of the generator (with resulting turbine trip) could result in an overspeed condition. The turbine control system and the turbine intercept valves limit the overspeed to less than 120 percent. As additional backup, the turbine protection system has a mechanical overspeed protection trip usually set at about 110 percent.

5.5.1.3.10 Antireverse-Rotation Device

Each of the RCPs is provided with an antireverse-rotation device in the motor. This antireverse mechanism consists of eleven pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and two shock absorbers.

After the motor has come to a stop, one pawl engages the ratchet plate and, as the motor tends to rotate in the opposite direction, the ratchet plate also rotates until stopped by the shock absorbers. The rotor remains in this position until the motor is energized again. After the motor has come up to speed, the ratchet plate is returned to its original position by the spring return.

When the motor is started, the pawls drag over the ratchet plate until the motor reaches approximately 80 rpm. At this time, centrifugal forces acting on the pawls produce enough friction to prevent the pawls from rotating, and thus hold the pawls in the elevated position until the speed falls below the above value. Considerable shop testing and plant experience with the design of these pawls have shown high reliability of operation.

5.5.1.3.11 Shaft Seal Leakage

During normal operation, leakage along the RCP shaft is controlled by three shaft seals arranged in series so that reactor coolant leakage to the containment is essentially zero. Charging flow is directed to each RCP via a seal water injection filter. Information on the design of the seal water injection filter is provided in table 9.3-6. It enters the pumps through the thermal barrier wall and is directed to a point between the pump shaft bearing and the thermal barrier cooling coils. Here the flow enters the shaft annulus; a portion flows down past the thermal barrier cooling cavity and labyrinth seals and into the reactor cooling system; the remainder flows up the pump shaft annulus cooling the lower shaft bearing. This flow provides a back pressure on the No. 1 seal and a controlled flow through the seal. Above the seal, most of the flow leaves the pump via the No. 1 seal discharge line. Minor flow passes through the No. 2 seal and discharge line. A backflush injection of 400 cc/h from a head tank flows into the No. 3 seal between its "double dam" seal area. At this point the flow divides, with half flushing through one side of the seal and out the No. 2 seal leakoff, while the remaining half flushes through the other side and out the No. 3 seal leakoff. This arrangement ensures essentially zero leakage of reactor coolant or trapped gases from the pump.

During an SBO or loss of all seal cooling event, 550 ºF RCS water begins to travel up the RCP annulus and displace the much cooler (~140 ºF) seal injection water. Once the temperature of the seal package reaches the shutdown seal (SDS) actuation temperature range, the SDS deploys. This action alone limits leakage of reactor coolant via the RCP.

5.5.1.3.12 Seal Discharge Piping

Discharge pressure from the No. 1 seal is reduced to that of the volume control tank. Water from each pump No. 1 seal is piped to a common manifold, through the seal water return filter and through the seal water heat exchanger, where the temperature is reduced to that of the volume control tank. The No. 2 and No. 3 leakoff lines permit normal No. 2 and 3 seal leakage to flow to the reactor coolant drain tank.

5.5.1.3.13 Spool Piece

The application of a removable spool piece in the RCP shaft serves to facilitate the inspection and maintenance of the pump seal system without breaking any of the fluid, electrical, or instrumentation connections to the motor and without removal of the motor (see figure 5.5-3). Thus it serves to reduce plant downtime for pump maintenance, and also to reduce personnel radiation exposure because of the reduced time in the proximity of the primary coolant loop.

5.5.1.3.14 Motor Air Coolers

The internal parts of the motor are cooled by air. Integral vanes on each end of the rotor draw air in through cooling slots in the motor frame. This air passes through the motor with particular emphasis on the stator end turns. It is then ducted to the external air/water heat exchangers. Each motor has two such coolers mounted diametrically opposite each other. In passing through the coolers, the air is cooled to below 122°F so that minimum heat is rejected to the containment from the motors.

5.5.1.4 <u>Tests and Inspections</u>

Support feet are cast integral with the casing to eliminate a weld region.

The design enables disassembly and removal of the pump internals for usual access to the internal surface of the pump casing. Inservice inspection is discussed in subsection 5.2.8.

Additionally, as required by the Technical Specifications, each RCP flywheel is inspected in accordance with the Reactor Coolant Pump Flywheel Inspection Program.

Castings for the three RCPs of Unit 1 were mapped for major defects greater than one-fifth of the wall thickness prior to weld repair in accordance with paragraph 314.5.5 of the Pump and Valve Code, 1968 Edition. Since fabrication, paragraph NB-2539.6 of ASME Section III, 1971 Edition, became effective and requires mapping of major defects of 3/8 in. and/or 10 percent of the wall thickness. However, remapping according to the latter code was impossible because some lesser defects not previously defined as "major" were weld repaired from the cast condition. To confirm integrity of the casings, they were completely radiographed in excess of code requirements.

The RCP quality assurance program is given in table 5.5-2.

5.5.2 STEAM GENERATOR

5.5.2.1 Design Bases

Steam generator design data are given in table 5.5-3. The design stress limits, transient conditions, and combined loading conditions for the steam generator are given in subsection 5.2.1. Estimates of radioactivity levels anticipated in the secondary side of the steam generators during normal operation, and the bases for the estimates, are given in chapter 11. The accident analysis of a steam generator tube rupture is discussed in chapter 15.

The internal moisture separation equipment is designed to ensure that moisture carryover does not exceed 0.10 percent by weight under the following conditions:

1. Steady-state operation up to 100 percent of full-load steam flow, with water at the normal operating level.

2. Loading or unloading at a rate of 5 percent of full-power steam flow per minute in the range of 15 percent to 100 percent of full-load steam flow.

3. A step load change of 10 percent of full power in the range from 15 percent to 100 percent full-load steam flow.

The steam generator tube-tubesheet complex meets the stress limitations and fatigue criteria specified in the ASME Code Section III. Code and materials requirements of the steam generator are given in section 5.2.

The water chemistry in the reactor side is selected to provide the necessary boron content for reactivity control and to minimize corrosion of RCS surfaces. The water chemistry of the steam side is discussed in subsection 10.3.5.

5.5.2.2 Design Description

The steam generator shown on figure 5.5-4 is a vertical shell and U-tube evaporator 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 tubesheet. Manways are provided for access to both sides of the divided head.
Feedwater flows directly into the annulus formed by the shell and tube bundle wrapper before entering the boiler section of the steam generator. Water/steam mixture then flows upward

through the tube bundle and into the steam drum section. A set of centrifugal moisture separators, located above the tube bundle, removes most of the entrained water from the steam. Steam dryers are employed to increase the steam quality to a minimum of 99.90 percent (0.10-percent moisture). The moisture separators recirculate the separated water which mixes with the feedwater as it passes through the annulus formed by the shell and tube bundle wrapper.

The steam drum has two bolted and gasketed access openings for inspection and maintenance of the dryers, which can be disassembled and removed through the opening.

The unit is primarily carbon steel. The heat transfer tubes and the divider plate are Inconel, and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. The primary side of the tubesheet is weld clad with Inconel.

5.5.2.3 Design Evaluation

5.5.2.3.1 Forced Convection

The limiting case for heat transfer capability is the "nominal 100-percent design" case. The steam generator effective heat transfer coefficient is based on the coolant conditions of temperature and flow for this case, and includes a conservative allowance for tube fouling. Adequate tube area is provided to ensure that the full design heat removal rate is achieved.

5.5.2.3.2 Natural Circulation Flow

Upon loss of power to the RCPs, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops (RCLs). The natural circulation flow was calculated by a digital code for the conditions of equilibrium flow and maximum loop flow impedance. The model used has given results within 15 percent of the measured flow values obtained during natural circulation tests conducted at the Yankee-Rowe plant and has also been confirmed at San Onofre, Connecticut Yankee, and the Ginna Nuclear Power Plants. The natural circulation flow ratio as a function of reactor power is given in table 15.2-2.

Tube and tubesheet stress analyses of the steam generator are given in section 5.2.

Calculations confirm that the steam generator tubesheet will withstand the loading (which is quasistatic rather than a shock loading) caused by loss of reactor coolant.

5.5.2.3.3 Corrosion

The steam generators are fabricated with thermally treated Alloy 690, a nickel-chromium-iron alloy (ASME SB-163). Industry wide corrosion testing and specification development programs have shown that Alloy 690 has substantially increased resistance to corrosion compared to the mill annealed Alloy 600. These programs have justified the selection of thermally treated Alloy

690 for the steam generator tubes. Operating histories throughout the industry have not indicated the potential for Inconel-690 steam generator tubes to experience degradation.

[HISTORICAL] [*Farley Nuclear Plant corrosion experience with Inconel-600 primary side stress corrosion cracking at the expanded and unexpanded tube transition adjacent to the top of the tubesheet, in the tubesheet region, and in the small radius U-bends. Secondary side degradation includes tube denting, tube wastage, tube pitting, tube fretting, and secondary side stress corrosion cracking in the tube support plate intersections and in the sludge pile regions and in the free spans.****]***

The primary approach to controlling degradation has been through the control of water chemistry.[a] However, where water chemistry has been ineffective in controlling the degradation or where changes to the chemistry program occurred after the degradation started, other approaches have been used. Should similar forms of steam generator degradation occur with Inconel-690, it will be managed in a similar fashion.

Farley Nuclear Plant Technical Specifications provide details for performing thorough inspections of steam generator tubes. Preventive maintenance activities have been implemented, aimed towards maximizing the steam generator tubing corrosion resistance and reliability.[a] If a tube is determined to be degraded beyond specified limits at the time of inspection, it is plugged to remove it from service.

Plugging results in a reduction in RCS flow through the steam generators. At the completion of each outage, the level of plugging is evaluated to ensure the as-left operating conditions are within evaluated limits.

5.5.2.3.4 Flow-Induced Vibration

In the design of Westinghouse steam generators, consideration has been given to the possibility of degradation of tubes because of mechanical or flow-induced excitation. This consideration includes detailed analysis of the tube supporting system, as well as an extensive research program with tube vibration model tests.

The major cause of vibratory failure in heat exchanger components is that caused by hydrodynamic excitation of the fluid outside the tube. Consideration is given by Westinghouse to three regions where the possibility of flow-induced vibration may exist:

A. At the entrance of downcomer feed to the tube bundle (cross-flow).

B. Along the straight sections of the tube (parallel flow).

C. In the curved tube section of the U-bend (cross-flow).

Two types of flow exist, cross-flow and parallel flow. For the case of parallel flow, an analysis is made to determine the vibratory deflections. Analysis of the steam generator tubes indicates the flow velocities to be sufficiently below that required for damaging fatigue or impacting vibratory amplitudes. The support system, therefore, is deemed adequate to preclude parallel-flow excitation. For the case of cross-flow excitation, it is noted in the literature that several techniques for the analysis of the tube vibration exist. The design problem is to ascertain that the tube natural frequency is well above the vortex shedding frequency. In order to avoid resonant vibration, adequate tube supports are provided.

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a. The Water Chemistry Control and Steam Generator Programs are credited as license renewal aging management programs (see chapter 18, subsections 18.2.2 and 18.2.7).

Since the problem of cross-flow-induced vibration was of major concern in the design of shell and tube heat exchangers, Westinghouse has given consideration to the experimental evaluation of the behavior of tube arrays under cross-flow. While consideration was given to instrumentation of actual units in service, the hostile environment would limit the amount and quality of information obtained. As a result, it was deemed prudent to undertake a research program that would allow the study of fluid elastic vibrator behavior of tubes in arrays. A wind tunnel was built specifically for this purpose and Westinghouse has invested approximately three years of research in the study of this problem. The research facilities for the tube vibration study have expanded with the construction of a water tunnel facility.

The results of this research and work done by others[(1),(2),(3)] confirm the vortex shedding mechanism. More significant, however, is the evaluation of a fluid elastic mechanism[(4)] not associated with vortex shedding. This is not commonly understood from the literature and could be a source of vibration failure. Westinghouse steam generators are evaluated on this basis, in addition to the aforementioned techniques, and have been found to be adequately designed. Testing has also been conducted using specific parameters of the steam generator and the results show the support system to be adequate.

[HISTORICAL] **[***Historically, flow induced vibration has not been a significant contributor to steam generator tube plugging in the Series 51 steam generator. Instances of wear at the tube support plate intersections have been rare in the Series 51 generator. The only region of the tube which has seen noticeable numbers of indications is the U-bend region, at the anti-vibration bar (AVB) supports. Excitation of the tubes resulted in fretting wear against the AVBs. Fretting wear has typically been tracked over several operating cycles until the depth of the indication warranted the tube to be removed from service. To address the issue of U-bend wear, the original AVBs in the Farley Nuclear Plant's steam generators were replaced with an expandable design. In this design the contact area between the AVB and the tube was increased, thereby lowering contact loads. The replacement AVB design included two separate flat bars which were drawn against each other. The surfaces between the bars were opposing tapers, and as the bars were subsequently drawn towards each other, the width of the assembly increased. This design effectively eliminated the tube to AVB gap, thereby preventing any wear mechanism.***]** The steam generator design has an enhanced antivibration bar design to provide for a more stable tube bundle and limit the potential for wear and high cyclic fatigue of the tubes. As required by Technical Specifications, periodic eddy current inspections will provide assurance that steam generator tube wear will be appropriately managed.

Vibration effects are eliminated during normal operation by the supporting system. Under loss-of-coolant accident (LOCA) conditions, vibration is of short duration and there is no endurance problem. Further consideration is given to the possibility of mechanically-excited vibration, in which resonance of external forces with tube natural frequencies must be avoided. It is believed that the transmissibility of external forces, either through the structure or from fluid within the tubes, is negligible and should cause little concern.

5.5.2.4 Tests and Inspections

[HISTORICAL] [*The steam generator quality assurance program is given in table 5.5-4.*

Radiographic inspection and acceptance standards are in accordance with the requirements of Section III of the ASME Code.

Liquid penetrant inspection is performed on weld-deposited tubesheet cladding, channel head cladding, tube-to-tubesheet weldments, and weld-deposit cladding.

Liquid penetrant inspection and acceptance standards are in accordance with the requirements of Section III of the ASME Code.

Magnetic particle inspection is performed on the tube sheet forging, channel head forging, nozzle forgings, and the following weldments:

A. Nozzle-to-shell.

B. Support brackets.

C. Instrument connections (primary and secondary).

D. Temporary attachments after removal.

E. All accessible pressure containing welds after hydrostatic tests.**]**

The heat transfer tubing is subjected to eddy current testing according to Farley Technical Specifications. The purpose of this testing is to detect tube degradation at an early stage so that corrective action can be taken to minimize further degradation and reduce the potential for significant primary-to-secondary leakage.

If a tube imperfection of sufficient size is discovered, it may be plugged as detailed in the Farley Technical Specifications.

5.5.3 REACTOR COOLANT PIPING

5.5.3.1 Design Bases

The RCS piping is designed and fabricated to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions. Code and material requirements are provided in section 3.2 and subsection 5.2.3, respectively. Subsection 5.2.5 discusses sensitization and its prevention, cleaning procedures, storage, etc., that prevent stress-corrosion cracking.

Materials of construction are specified to minimize corrosion/erosion and to ensure compatibility with the operating environment.

The piping in the RCS pressure boundary is Safety Class 1 and is designed and fabricated in accordance with ASME III.

Stainless steel pipe conforms to ANSI B36.19 for sizes 1/2 in. through 12 in. and wall thickness schedules 40S through 80S. Stainless steel pipe outside of the scope of ANSI B36.19 conforms to ANSI B36.10.

The minimum wall thicknesses of the loop pipe and fittings are not less than that calculated using the ASME III Class 1 formula of Paragraph NB-3641.1 (3) with an allowable stress value of 17,550 psi.

The pipe wall thickness for the pressurizer surge line is to Schedule 160. A full structural weld overlay has been applied on the outside surface of the pressurizer surge nozzle and extends to a portion of the stainless steel piping on the downstream side of the nozzle safe-end.

The minimum pipe bend radius is 5 nominal pipe diameters; ovality does not exceed 6 percent.

All butt welds, nozzle welds, and boss welds are of a full-penetration design.

The mechanical properties of representative material heats in the final heat-treat condition are determined by test at 650°F design temperature per ASTM E-21 or equivalent. In particular, the hot yield strength (0.2 percent offset) at 650°F equals or exceeds 19,850 psi.

Processing and minimization of sensitization are discussed in subsection 5.2.5.

Flanges conform to ANSI B16.5.

Socket weld fittings and socket joints conform to ANSI B16.11.

Inservice inspection is discussed in subsection 5.2.8.

5.5.3.2 Design Description

Principal design data for the reactor coolant piping are given in table 5.5-5.

Pipe and fittings are cast seamless, without longitudinal welds or electroslag welds, and comply with the requirements of ASME Section II (Parts A and C), Section III, and Section IX.

The RCS piping is specified in the smallest sizes consistent with system requirements. In general, high fluid velocities are used to reduce piping sizes. This design philosophy results in the reactor inlet and outlet piping diameters given in table 5.5-5. The line between the steam generator and the pump suction is larger in order to reduce pressure drop and improve flow conditions to the pump suction.

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. There is no electroslag welding on these components. All smaller piping which comprises part of the RCS boundary, such as the pressurizer surge line, spray and relief line, loop drains, and connecting lines to other systems are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank (PRT) is carbon steel. All joints and connections are welded, except for the pressurizer relief and the pressurizer code-safety valves, where flanged joints are used. Thermal sleeves are installed at points in the system where high thermal stresses could develop because of rapid changes in fluid temperature during normal operational transients.

These points include:

A. Charging connections and auxiliary charging connections at the primary loop from the chemical and volume control system (CVCS).

B. Return line connections from the RHRS at the RCLs.

C. Both ends of the pressurizer surge line.

D. Pressurizer spray line connection at the pressurizer.

Thermal sleeves are not provided for the remaining injection connections of the emergency core cooling system (ECCS), since these connections are not in normal use. Additionally, a stress analysis has been performed to demonstrate that those points in the system where thermal sleeves are installed are qualified to withstand all applicable design transients without the thermal sleeves. The results of this analysis indicate that all critical locations in the piping meet the ASME Code Requirements and that the piping will maintain its structural integrity without thermal sleeves.

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

A. RHR pump suction, which is 45 degrees down from the horizontal centerline. This enables the water level in the RCS to be lowered in the reactor coolant pipe while operation of the residual heat removal system (RHRS) continues, should this be required for maintenance.

B. Loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.

C. The differential pressure taps for flow measurement are downstream of the steam generators on the first 90-degree elbow. The tap arrangement is discussed in the instrumentation section of this description.

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

A. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the RCL flow adds to the spray driving force.

B. The reactor coolant sample system taps protrude into the main stream to obtain a representative sample of the reactor coolant.

C. The narrow range temperature detectors are located in resistance temperature detector wells that extend into the reactor coolant pipes.

D. The wide-range temperature detectors are located in resistance temperature detector wells that extend into the reactor coolant pipes.

Each hot leg has three narrow range, thermowell mounted, dual element, fast response RTDs located in approximately the same plane 120 degrees apart. These RTDs extend into the reactor coolant fluid, sensing the temperature at three distinct locations within the hot leg pipe. One element provides the hot leg temperature measurement, and the other element is an installed spare. These three measurements are electronically averaged to provide a representative T_{hot} indication.

Where possible, the thermowells in each loop are located within the scoops previously used to supply temperature samples to the RTD bypass manifold. The scoops were modified by machining a flow hole in the end of the scoop to facilitate the flow of water through the existing holes in the leading edge of the scoop and past the temperature sensitive tip of the RTD. Due to physical limitations, several hot leg RTDs are located in independent thermowells near the original scoop location.

The cold leg is provided with a dual element, narrow range, thermowell mounted, fast response RTD. One element provides the cold leg temperature measurement, and the other element is an installed spare.

The original cold leg RTD bypass penetration nozzle was modified to accept the thermowell, and the original crossover leg was capped.

The RCS boundary piping includes those sections of piping interconnecting the reactor vessel, the steam generator, and the RCP. It also includes the following:

A. Charging line and alternate charging line from the cold leg branch connections on the RCLs to the second check valve.

B. Letdown line and excess letdown line from the branch connections on the RCLs between the steam generator and pump to the second downstream valve.

C. Pressurizer spray lines from the reactor coolant cold legs to spray nozzle on the pressurizer vessel.

D. RHR lines from three RCS cold legs out to the second check valve and from two RCS hot legs out to the second valve.

E. Safety injection lines from the RCS hot and cold legs out to the second check valve.

F. Accumulator lines from the RCL hot legs to the second check valve.

G. Loop fill, loop drain, sample, and instrument lines to or from the RCLs out to the second valve.

H. Pressurizer surge line from one RCL hot leg to the pressurizer vessel inlet nozzle.

I. Pressurizer spray scoop, sample connection with scoop, reactor coolant RTD thermowell installation boss, and the RTD thermowells.

J. All branch connection nozzles attached to RCLs.

K. Pressure relief lines from nozzles on top of the pressurizer vessel up to and through the power-operated pressurizer relief valves and pressurizer safety valves.

L. Seal injection water and labyrinth differential pressure lines to or from the RCP inside reactor containment out to the second valve.

M. Auxiliary spray line from the pressurizer spray line header out to the second valve.

N. Sample lines from pressurizer to the isolation valve.

Details of the materials of construction and codes used in the fabrication of the reactor coolant piping and fittings are discussed in section 5.2.

5.5.3.3 Design Evaluation

Piping load and stress evaluation for normal operating loads, seismic loads, blowdown loads, and combined normal, blowdown, and seismic loads are discussed in section 5.2.

5.5.3.3.1 Material Corrosion/Erosion Evaluation

The water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications.

An upper limit of about 60 ft/s is specified for internal coolant velocity to avoid the possibility of accelerated erosion. All pressure containing welds, out to the second valve that delineates the reactor coolant pressure boundary, are available for examination with removable insulation.

Components with stainless steel will operate satisfactorily under normal plant chemistry conditions in PWR systems because chlorides, fluorides, and particularly, oxygen, are controlled to very low levels.

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to desired reactor coolant water quality listed in table 5.2-22. Maintenance of the water quality to minimize corrosion is accomplished using the CVCS and sampling system which are described in chapter 9.0.

5.5.3.3.2 Sensitized Stainless Steel

Sensitized stainless steel is discussed in section 5.2.

5.5.3.3.3 Contaminant Control

Contamination of stainless steel and Inconel by copper, low melting temperature alloys, and lead is prohibited. Contamination due to exposure to mercury is possible if one or more temporary underwater lights used in the refueling cavity, transfer canal, and the spent-fuel pool were to fail catastrophically. The lights approved for use in these areas are manufactured by ROS, model HPS-1000, and contain up to 3 mg of mercury each in double encapsulated bulbs. The use of up to twelve of these lights at any one time has been evaluated as acceptable. Colloidal graphite is the only permissible thread lubricant.

Prior to application of thermal insulation, the austenitic stainless steel surfaces are cleaned and analyzed to a halogen limit of 0.0015 mg $C1/cm^2$ and 0.0015 mg F/cm^2.

5.5.3.4 <u>Tests and Inspections</u>

The RCS piping quality assurance program is given in table 5.5-6.

Radiographic examination is performed throughout 100 percent of the wall volume of each pipe and fitting in accordance with NB-2573 of Section III of the ASME Code for all pipe 27-1/2 in. and larger. All unacceptable defects are eliminated in accordance with the requirements of Paragraphs NB-2578 and NB-2579 of ASME III.

Magnetic particle inspection and acceptance standards are in accordance with requirements of Section III of the ASME Code.

An ultrasonic test is performed on the tubesheet forging, tubesheet cladding, secondary shell, and heat plate and nozzle forgings.

The heat transfer tubing is subjected to an eddy-current test.

Hydrostatic tests are performed in accordance with Section III of the ASME Code.

In addition, the heat transfer tubes are subjected to a hydrostatic test pressure prior to installation into the vessel, which is not less than 1.25 times the primary side design pressure multiplied by the ratio of the material allowable stress at the testing temperature.

Manways are provided to give access to both the primary and secondary sides.

Regulatory Guide 1.83, "Inservice Inspection of Pressurized-Water Reactor Steam Generator Tubes," provides recommendations concerning the inspection of tubes, which cover inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, and required actions based on findings. The minimum requirements for inservice inspection of steam generators are established as part of the Technical Specifications.

5.5.4 MAIN STEAM LINE FLOW RESTRICTIONS

5.5.4.1 Design Basis

The outlet nozzle of each steam generator is provided with a flow restrictor designed to limit steam flowrate consequent to a main steam line rupture. The sudden increase in steam flow resulting from the pipe rupture creates backpressure at the flow restrictor, which limits further increase in flow.

Design bases for the steam generator flow restrictors are:

A. To provide plant protection in event of a steam line rupture by limiting steam flow from the break, which limits the cooling rate of the primary system which, in turn, precludes DNB and minimizes fuel clad damage, as shown in chapter 15.

B. Minimize unrecovered pressure loss across the restrictor during normal operation.

5.5.4.2 Description

The flow restrictor (figure 5.5-5) is an assembled cluster of seven venturi nozzles installed within the steam outlet nozzle of the steam generator. Holes forged into the steam nozzle position an outer circle of six nozzles and one central nozzle. The venturi nozzles are forged Inconel and are welded to Inconel clad on the nozzle forging.

5.5.4.3 Evaluation

The equivalent area of the steam generator outlet is reduced by an approximate factor of 4.4 and the resultant pressure drop through the restrictors at 100-percent steam flow is approximately 5 psi. The steam-side weld to the outlet nozzle is in compliance with manufacturing and quality control requirements of Section III of the ASME Code. The restrictor is constructed of material specified in Section III of the ASME Code.

5.5.4.4 Tests and Inspections

The restrictors are not a part of the steam system boundary. No tests or inspections of the restrictors beyond those performed during fabrication are required.

5.5.5 MAIN STEAM LINE ISOLATION SYSTEM

The plant has no system designated as the main steam line isolation system. This function is served by the main steam line isolation valves and their bypass valves. These valves are installed in the steam lines as shown in drawings D-175033, sheet 1; D-175033, sheet 2; D-170114, sheet 1; D-170114, sheet 2; D-205033, sheet 1; D-205033, sheet 2; and D-200007, Main Steam Supply System, and isolate the steam generators and the main steam lines inside the containment on signal from the engineered safety features actuation system (ESFAS).

The following subdivisions provide information on design bases; system description; design evaluation; and test and inspections for these valves.

5.5.5.1 Design Bases

The main steam line isolation valves and their bypass valves are designed to stop forward flow and to isolate the steam generators and the main steam lines on signal initiated by ESFAS under any of the following conditions:

A. High steam line flow in coincidence with low-low-T_{avg} or low steam line pressure (break in main steam line).

B. High pressure in the containment.

The main steam line isolation valves and the bypass valves are designed and manufactured in accordance with the requirements of the ASME Code for Pumps and Valves for Nuclear Power, Draft, November 1968 including March 1970 addenda, Class II. The quality group classifications that apply to the valves are listed in subsection 3.2.2. The valves are designed for a minimum life of 40 years.[a] All valves, their air vent lines, and air supply lines up to and including the check valves as shown in drawings D-175033, sheet 1; D-175033, sheet 2; D-170114, sheet 1; D-170114, sheet 2; D-205033, sheet 1; D-205033, sheet 2; and D-200007 are designed to meet seismic Category I requirements. The valves are of fail-closed design.

The three-way solenoid valves, which control the air supply to the main steam isolation valves, are manufactured using similar high temperature components, including Class electrical coils, as the containment isolation valve solenoid valves. The main steam isolation solenoid valves are tested under simulated accident conditions outlined in the Environmental Qualification Program. Information concerning the Environmental Qualification Program is contained in FSAR subsection 3.11.3.

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a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 18).

5.5.5.2 System Description

The following components are provided for main steam line isolation:

A. Main steam line isolation valves.

B. Main steam bypass valves.

The main steam isolation valves are installed in the main steam lines from the steam generators. Two valves are installed in each main steam line downstream from the safety relief valves outside the containment.

The isolation valves are 32-in., 600-lb, full-flow, swing-check, nonreturn-type valves with pneumatic actuators. The data for these valves are listed in table 5.5-15. Each valve is provided with solenoid valves in the air supply and stroke test lines as shown in drawings D-175033, sheet 1; D-175033, sheet 2; D-170114, sheet 1; D-170114, sheet 2; D-205033, sheet 1; D-205033, sheet 2; and D-200007.

During normal plant operation the valves are kept open against a spring force by air pressure under the piston in the actuator cylinder. In case of high pressure in the containment, or high steam line flow in coincidence with low-low (T_{avg}) or low steam line pressure, the air pressure in the cylinder is relieved and the valve is closed by action of the spring to prevent the forward flow of steam through the valve.

Plant instrument air at 80-100 psig pressure is supplied to the actuator cylinder. Each of the redundant isolation valves has its own means of venting the air supply to relieve the cylinder pressure and close the valve. Each isolation valve is provided with a three-way solenoid valve, normally open operation, to supply air in its air supply line. A three-way solenoid valve, normally closed operation, to supply air is provided in its stroke test line. Each of the redundant isolation sets of supply and vent solenoid valves is supplied from a separate 125-V-dc power system and receives a separate signal from the ESFAS.

The two valves in the isolation valve bypass lines are also closed by a signal from the ESFAS to prevent steam from escaping through these lines. Closure signals for these valves are shown in table 6.2-20.

5.5.5.3 Design Evaluation

The main steam line isolation valves are capable of isolating the steam generators within 7 s of receiving the signal from the ESFAS. In the event of a steam line break, this action prevents continuous uncontrolled steam release from more than one steam generator. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a failure of one of the isolation valves.

Two redundant control signals are supplied by the ESFAS to close the redundant isolation valve and the bypass line valves. The two valves in the bypass line for each pair of isolation valves are closed simultaneously with the isolation valves to prevent steam from escaping through this line.

The loss of air supply at the valve operator will cause the closure of isolation valves and the bypass line valves.

5.5.5.4 Tests and Inspection

The isolation valves are subjected to radiographic, magnetic particle, liquid penetrant, and hydrostatic tests and wall thickness measurement in accordance with the ASME Code for Pumps and Valves. Valve seats were tested in accordance with Manufacturers Standardization Society of the Valve and Fittings Industry and the seat leakage across the valve did not exceed 2 cc/h/in. of diameter.

All the valves were tested during the plant startup program to verify that they function properly. Inservice tests and inspection of the solenoid valves will be performed throughout the life of the plant. Periodic tests will be performed as required by the technical specifications.

5.5.6 REACTOR CORE ISOLATION COOLING SYSTEM

This system is not applicable to PWR plants.

5.5.7 RESIDUAL HEAT REMOVAL SYSTEM

The RHRS transfers heat from the RCS to the component cooling system to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal plant cooldown, and maintains this temperature until the plant is started up again.

As a secondary function, the RHRS also serves as part of the ECCS during the injection and recirculation phases of a LOCA.

The RHRS also is used to transfer refueling water between the refueling water storage tank (RWST) and the refueling cavity before and after the refueling operations.

The relief valves of the RHRS are used as part of the overpressure mitigating system described in paragraph 5.2.2.4.

5.5.7.1 Design Bases

RHRS design parameters are listed in table 5.5-7.

The RHRS is designed to remove residual heat from the core and reduce the temperature of the RCS during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the RCS is reduced by transferring heat from the RCS to the steam and power conversion system through the use of the steam generator.

The RHRS is placed in operation approximately 4 h after reactor shutdown when the temperature and pressure of the RCS are approximately 350°F and 425 psig, respectively. Assuming that two heat exchangers and two pumps are in service and that each heat exchanger is supplied with component cooling water at a flowrate of 4160 gal/min, the RHRS is capable of reducing the temperature of the reactor coolant from 350°F to 140°F within 38h. The heat load handled by the RHRS during the cooldown transient includes residual heat from the core and RCP heat. The design residual heat load is based on the residual heat fraction that exists at 20h following reactor shutdown from an extended run at full power.

5.5.7.2 System Description

The RHRS as shown in drawings D-175041 and D-205041 consists of two residual heat exchangers, two RHR pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet lines to the RHRS are connected to the hot legs of two RCLs, while the return lines are connected to the cold legs of each of the RCLs. These return lines are also the ECCS low head injection lines (see drawings D-175038, sheet 1 and D-205038, sheet 1).

The RHRS suction lines are isolated from the RCS by two motor-operated valves in series and a relief valve, all located inside the containment. Each discharge line is isolated from the RCS by three check valves located inside the containment and by a normally open motor-operated valve located outside the containment. The check valves and the motor-operated valve on each discharge line are not part of the RHRS. These valves are shown on drawings D-175038, sheet 1; D-205038, sheet 1; D-175038, sheet 2; and D-205038, sheet 2 as part of the ECCS.

During system operation, reactor coolant flows from the RCS to the RHR pumps, through the tube side of the residual heat exchangers, and back to the RCS. The heat is transferred to the component cooling water circulating through the shell side of the residual heat exchangers.

Coincident with operation of the RHRS, a portion of the reactor coolant flow may be diverted from downstream of the residual heat exchangers to the CVCS low pressure letdown line for cleanup and/or pressure control. By regulating the diverted flowrate and the charging flow, the RCS pressure may be controlled. Pressure regulation is necessary to maintain the pressure range dictated by the fracture prevention criteria requirements of the reactor vessel and by the No. 1 seal differential pressure and NPSH requirements of the RCPs.

The RCS cooldown rate is manually controlled by regulating the reactor coolant flow through the tube side of the residual heat exchangers. A line containing a flow control valve bypasses each residual heat exchanger and is used to maintain a constant return flow to the RCS. Instrumentation is provided to monitor system pressure, temperature and total flow.

The RHRS is also used for filling the refueling cavity before refueling. After refueling operations, water is pumped back to the RWST until the water level is brought down to the flange of the reactor vessel. The remainder of the water is removed via a drain connection at the bottom of the refueling canal.

When the RHRS is in operation, the water chemistry is the same as that of the reactor coolant. Provision is made for the sampling system to extract samples from the flow of reactor coolant downstream of the residual heat exchangers. A local sampling point is also provided on each RHR train between the pump and heat exchanger.

The RHRS functions in conjunction with the high head portion of the ECCS to provide injection of borated water from the RWST into the RCS cold legs during the injection phase following a LOCA.

In its capacity as the low-head portion of the ECCS, the RHRS provides long-term recirculation capacity for core cooling following the injection phase of the LOCA. This function is accomplished by aligning the RHRS to take fluid from the containment sump, cool it by circulation through the residual heat exchangers, and directly supply it to the core, as well as via the centrifugal charging pumps in the CVCS.

The use of the RHRS as part of the ECCS is more completely described in section 6.3.

5.5.7.2.1 Component Description

The materials used to fabricate RHRS components are in accordance with the applicable code requirements. All parts of components in contact with borated water are fabricated or clad with austenitic stainless steel or equivalent corrosion resistant material.

Component codes and classifications are given in section 3.2 and component parameters are listed in table 5.5-8.

A. RHR Pumps

Two pumps are installed in the RHRS. The pumps are sized to deliver reactor coolant flow through the residual heat exchangers to meet the plant cooldown requirements. The use of two separate RHR trains ensures that cooling capacity is only partially lost should one pump become inoperative.

To ensure that the RHR pumps do not overheat or vibrate at low flows, a miniflow return line is provided from the downstream side of each RHR heat exchanger to the pump suction lines (see drawings D-175041 and D-205041). A control valve (FCV-602A, B) located in each miniflow line is actuated by a flow switch (FIS-602A, B). The miniflow valves open when RHR pump flow decreases to 750 gal/min (Unit 1), 1334 gal/min (Unit 2), and close when the flow increases to 1399 gal/min (Unit 1), 2199 gal/min (Unit 2).

A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high pressure alarm is also actuated by the pressure sensor.

The two pumps are vertical, centrifugal units with mechanical seals on the shafts. All pump surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion-resistant material.

B. Residual Heat Exchangers

Two residual heat exchangers are installed in the system. The heat exchanger design is based on heat load and temperature differences between reactor coolant and component cooling water existing 20 h after reactor shutdown when the temperature difference between the two systems is small.

The installation of two heat exchangers in separate and independent RHR trains ensures that the heat removal capacity of the system is only partially lost if one train becomes inoperative.

The residual heat exchangers are of the shell and U- tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tubesheet to prevent leakage of reactor coolant.

C. RHRS Valves

Valves that perform a modulating function are equipped with two sets of packings and an intermediate leakoff connection that discharges to the drain header.

Manual and motor-operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open. Leakage connections are provided where required by valve size and fluid conditions.

5.5.7.2.2 System Operation

A. Reactor Startup

Generally, while at cold shutdown condition, residual heat from the reactor core is being removed by the RHRS. The number of pumps and heat exchangers in service depends upon the heat load at the time.

At initiation of the plant startup, the RCS is completely filled and the pressurizer heaters are energized. The RHRS is connected to the CVCS via the low pressure letdown line to control reactor coolant pressure. As an alternative to the water-solid condition, a steam bubble may be drawn in the pressurizer to provide sufficient system pressure to start a RCP. During this time, the RHRS acts as an alternate letdown path. The manual valves downstream of the residual heat exchangers leading to the letdown line of the CVCSs are opened. The pressure control valve in the line from the RHRS to the letdown line of the CVCS is then manually adjusted in the control room to permit letdown flow. Failure of any of the valves in the line from the RHRS to the CVCS has no safety implications, either during startup or cooldown.

After the RCPs are started, pressure control via the RHRS and the low pressure letdown line is continued as the pressurizer steam bubble is formed. Indication of steam bubble formation is provided in the control room by the damping out of the RCS pressure fluctuations, and by pressurizer level indication. The RHRS is then isolated from the RCS and the system pressure is controlled by normal letdown, pressurizer spray, and pressurizer heaters.

B. Power Generation and Hot Standby Operation

During power generation and hot standby operation, the RHRS is not in service but is aligned for operation as part of the ECCS.

C. Reactor Shutdown

The initial phase of reactor cooldown is accomplished by transferring heat from the RCS to the steam and power conversion system through the use of the steam generators.

When the reactor coolant temperature and pressure are reduced to approximately 350°F and 425 psig, approximately 4 h after reactor shutdown, the second phase of cooldown starts with the RHRS being placed in operation.

Startup of the RHRS includes a warmup period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through residual heat exchangers. The mixed mean temperature of the return flows is controlled. Coincident with the manual adjustment, each heat exchanger bypass valve is regulated to give the required total flow.

The reactor cooldown rate is limited by RCS equipment cooling rates based on allowable stress limits, as well as the operating temperature limits of the component cooling system. As the reactor coolant temperature decreases, the reactor coolant flow through the residual heat exchangers is increased by adjusting the control valve in each heat exchanger's tube-side outlet line.

As cooldown continues, the pressurizer is filled with water and the RCS is operated in the water-solid condition.

At this stage, pressure control is accomplished by regulating the charging flowrate and the rate of letdown from the RHRS to the CVCS.

After the reactor coolant pressure is reduced and the temperature is 140°F or lower, the RCS may be opened for refueling or maintenance.

D. Refueling

One RHR pump is utilized during refueling to pump borated water from the RWST to the refueling cavity. The other is used in cooldown alignment for decay heat removal. During this operation, the isolation valves in the inlet lines of the RHRS are closed, and the isolation valves to the RWST are opened.

The reactor vessel head (RVH) is lifted and placed on the storage stand. The refueling water is then pumped into the reactor vessel through the normal RHRS return lines and into the refueling cavity through the open reactor vessel. After the water level reaches normal refueling level, the inlet isolation valves are opened, the RWST supply valves are closed, and RHR is resumed.

During refueling, the RHRS is maintained in service with the number of pumps and heat exchangers in operation as required by the Technical Specifications.

Following refueling, the RHR pumps are used to drain the refueling cavity to the top of the reactor vessel flange by pumping water from the RCS to the RWST.

5.5.7.3 Design Evaluation

5.5.7.3.1 System Availability and Reliability

The system is provided with two RHR pumps and two residual heat exchangers arranged in separate independent flow paths. If one of the two pumps or one of the two heat exchangers is not operable, safe cooldown of the plant is not compromised; however, the time required for cooldown is extended.

The two separate flow paths provide redundant capability of meeting the safeguard function of the RHRS. The loss of one RHRS flow path would not negate the capability of the ECCS since the two flow paths provide full redundancy for safeguard requirements.

To ensure reliability, the two RHR pumps are connected to two separate electrical buses so that each pump receives power from a different source. If a total loss of offsite power occurs while the system is in service, each bus is automatically transferred to a separate emergency diesel power supply. A prolonged loss of offsite power would not adversely affect the operation of the RHRS.

5.5.7.3.2 Leakage Provisions and Activity Release

In the event of a LOCA, fission products may be recirculated through part of the RHRS exterior to the containment. If the RHR pump seal should fail, the water would spill out on the floor in a shielded compartment. Each pump is located in a separate, shielded room. If one of the rooms is flooded, this would have no effect on the other since there are no interconnections. In addition, in each room provisions are made for draining spillage into a sump which is provided with dual pumps and suitable level instrumentation so that the spillage can be pumped to the waste processing system.

5.5.7.3.3 Overpressurization Protection

Each inlet line to the RHRS is equipped with a pressure relief valve sized to relieve the combined flow of all the charging pumps at the relief set pressure.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve the maximum possible backleakage through the valves separating the RHRS from the RCS. These relief valves have nominal setpoints of 600 psig and are located in the ECCS (see drawings D-175038, sheet 1 and D-205038, sheet 1).

The design of the RHRS includes two isolation valves in series on each inlet line from the high pressure RCS to the lower pressure RHRS. Each isolation valve on each inlet line is interlocked with one of the two independent RCS pressure signals to provide an open permissive to these valves. The valve in each inlet line closest to the RHRS is also interlocked with the pressurizer vapor space temperature sensor to provide an additional open permissive. The open permissive interlock prevents the valves from being opened when the RCS pressure is > ~ 383 psig and the pressurizer vapor space temperature is > 475°F (pressurizer vapor space temperature interlock is applicable only to the valves closest to the RHRS). The autoclosure interlock for these valves was deleted per WCAP-11746 analysis. This interlock is described in more detail in subsection 7.6.2.

5.5.7.3.4 Shared Function

The safety function performed by the RHRS is not compromised by its normal function, which is normal plant cooldown. The valves associated with the RHRS are normally aligned to allow immediate use of this system in its safeguard mode of operation. The system has been designed in such a manner that two redundant flow circuits are available, ensuring the availability of at least one train for safety purposes.

The normal plant cooldown function of the RHRS is accomplished through a suction line arrangement which is independent of any safeguard function. The normal cooldown return lines are arranged in parallel redundant circuits and are utilized also as the low head safeguards injection lines to the RCS. Utilization of the same return circuits for safeguards, as well as for normal cooldown, lends assurance to the proper functioning of these lines for safeguard purposes.

5.5.7.3.5 Radiological Considerations

The highest radiation levels experienced by the RHRS are those that would result from a LOCA.

Following a LOCA, the RHRS is used as part of the ECCS. During the recirculation phase of emergency core cooling, the RHRS is designed to operate for up to a year pumping water from the containment sump, cooling it, and returning it to the containment to cool the core.

Since, except for some valves and piping, the RHRS is located outside the containment, most of the system is not subjected to the high levels of radioactivity in the containment post-accident environment.

The operation of the RHRS does not involve a radiation hazard for the operators, since the system is controlled remotely from the control room. If maintenance of the system is necessary, that portion of the system requiring maintenance is isolated by remotely-operated valves and/or manual valves with stem extensions which allow operation of the valves from a shielded location. The isolated piping is drained before maintenance is performed.

5.5.7.4 Tests and Inspections

Periodic visual inspections and preventive maintenance are conducted during plant operation according to normal industrial practice.

The instrumentation channels for the RHR pump flow instrumentation devices are calibrated during each refueling operation if a check indicates that recalibration is necessary.

Because of the role that the RHRS has in sharing components with the ECCS, the RHR pumps are tested as a part of the ECCS testing program (see subsection 6.3.4).

5.5.8 REACTOR COOLANT CLEANUP SYSTEM

The CVCS provides reactor coolant cleanup and is discussed in chapter 9.0. The radiological considerations are discussed in chapter 11.0.

5.5.9 MAIN STEAM LINE AND FEEDWATER PIPING

Main steam line piping is covered in section 10.3, Main Steam Supply System. Feedwater piping is covered in subsection 10.4.7, Condensate and Feedwater System.

5.5.10 PRESSURIZER

5.5.10.1 Design Bases

The general configuration of the pressurizer is shown in figure 5.5-6. The design data of the pressurizer are given in table 5.5-9. Codes and material requirements are provided in section 5.2.

5.5.10.1.1 Pressurizer Surge Line

The surge line is sized to limit the pressure drop between the RCS and the safety valves with maximum allowable discharge flow from the safety valves. Overpressure of the RCS does not exceed 110 percent of the design pressure.

The surge line and the thermal sleeves at each end are designed to withstand the thermal stresses resulting from volume surges which occur during operation.

The pressurizer surge line nozzle diameter is given in table 5.5-9 and the pressurizer surge line dimensions are shown on drawings D-175037, sheet 2 and D-205037, sheet 2.

5.5.10.1.2 Pressurizer

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

A. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

B. The water volume is sufficient to prevent the heaters from being uncovered during a step load increase of 10 percent at full power.

C. The steam volume is large enough to accommodate the surge resulting from the design step load reduction of load with reactor control and steam dump without the water level reaching the high-level reactor trip point.

D. 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.

E. The pressurizer does not empty following reactor and turbine trip.

F. The emergency core cooling signal is not activated during reactor trip and turbine trip.

5.5.10.2 Design Description

5.5.10.2.1 Pressurizer Surge Line

The pressurizer surge line connects the pressurizer to one reactor hot leg. The line enables continuous coolant volume pressure adjustments between the RCS and the pressurizer.

5.5.10.2.2 Pressurizer

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.

The surge line nozzle and removable electric heaters are installed in the bottom head. The heaters are removable for maintenance or replacement. A thermal sleeve is provided to minimize stresses in the surge-line nozzle. A screen at the surge-line nozzle and baffles in the lower section of the pressurizer prevents an insurge of cold water from flowing directly to the steam/water interface and assists mixing.

Spray line nozzles and relief and safety valve connections are located in the top head of the vessel. Spray flow is modulated by automatically-controlled, air-operated valves. The spray valves also can be operated manually by a switch in the control room.

A small, continuous spray flow is provided by means of normal valve seat leakage through the power-operated spray valves and by manual throttling of the bypass spray valves. This flow serves to ensure that the pressurizer liquid is homogeneous with the coolant and to prevent excessive cooling of the spray piping. The presence of flow is ensured by monitoring the low temperature alarms on the pressurizer spray and surge lines.

During an outsurge from the pressurizer, flashing of water to steam and generating of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. During an insurge from the RCS, the spray system, which is fed from two cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the setpoint of the power-operated relief valves (PORVs) for normal design transients. Heaters are energized on high-water level during insurge to heat the subcooled surge water that enters the pressurizer from the RCL.

The quick opening PORVs are designed to limit the pressurizer pressure such that a reactor trip will not occur, and to limit the frequency of unnecessary releases from the safety relief valves. A more detailed discussion of the PORVs is given in subsection 5.5.13.

Material specifications are provided in table 5.2-20 for the pressurizer and the surge line. Design transients for the components of the RCS are discussed in paragraph 5.2.1.5. Additional details on the pressurizer design cycle analysis are given in paragraph 5.5.10.3.5.

A. Pressurizer Support

The skirt-type support is attached to the lower head and extends for a full 360 degrees around the vessel. The lower part of the skirt terminates in a bolting flange with bolt holes for securing the vessel to its foundation. The skirt-type support is provided with ventilation holes around its upper perimeter to ensure free convection of ambient air past the heater, plus connector ends for cooling.

B. Pressurizer Instrumentation

Refer to chapter 7 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

C. Spray Line Temperatures

Temperatures in the spray lines from two loops are measured and indicated. Alarms from these signals are actuated by low spray-water temperature. Alarm conditions indicate insufficient flow in the spray lines.

D. Safety and Relief Valve Discharge Temperatures

Temperatures in the pressurizer safety and relief valve discharge lines are measured and indicated. An increase in a discharge line temperature is an indication of leakage through the associated valve.

5.5.10.3 Design Evaluation

5.5.10.3.1 System Pressure

Whenever a steam bubble is present within the pressurizer, RCS pressure is maintained by the pressurizer. Analyses indicate that proper control of pressure is maintained for the operating conditions.

A safety limit has been set to ensure that the RCS pressure does not exceed the maximum transient value allowed under the ASME Code, Section III, and thereby ensures continued integrity of the RCS boundary.

Evaluation of plant conditions of operation which follows indicates that this safety limit is not reached.

During startup and shutdown, the rate of temperature change is controlled by the operator. Limits for the rate of pressurizer temperature change are contained in the Technical Requirements Manual. During reactor core shutdown, the maximum heating by pump energy is limited. The installed pressurizer electrical heating capacity provides additional controlled heatup energy.

When the pressurizer is filled with water, i.e., near the end of the second phase of plant cooldown and during initial system heatup, RCS pressure is maintained by the RHRS.

5.5.10.3.2 Pressurizer Performance

The pressurizer has a minimum free internal volume. The normal operating water volume at full-load conditions is approximately 54.9 percent. Under part-load conditions, the water volume in the vessel is reduced for proportional reductions in plant load to approximately 21.4-percent level at zero power level. The various plant operating transients are analyzed and the design pressure is not exceeded with the pressurizer design parameters as given in table 5.5-9.

5.5.10.3.3 Pressure Setpoints

The RCS design and operating pressure, together with the safety, power relief, and pressurizer spray valves setpoints and the protection system setpoint pressures are listed in table 5.2-19. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics.

5.5.10.3.4 Pressurizer Spray

Two separate, automatically-controlled spray valves with remote, manual overrides are used to initiate pressurizer spray. In parallel with each spray valve is a manual bypass valve which permits adjustment of the flow through each spray line. The required continuous spray flow through both spray lines is provided by means of normal seat leakage through the power-operated spray valves and manual throttling of the bypass spray valves. The small continuous spray flow is needed 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 flow. The layout of the common spray line piping to the pressurizer forms a water seal which prevents the steam buildup back to the control valves. The spray rate is selected to prevent the pressurizer pressure from reaching the operating setpoint of the power relief valves during a step reduction in power level of 10 percent of full load.

The pressurizer spray lines and valves are large enough to provide adequate spray, using as the driving force the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop, so that the velocity head of the RCL flow adds to the spray driving force. The spray valves and spray line connections are arranged so that the spray will operate when one RCP is not operating. The line may also be used to assist in equalizing the boron concentration between the RCLs and the pressurizer.

A flow path from the CVCS 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 RCPs are not operating. The thermal sleeves on the pressurizer spray connection and the spray piping are designed to withstand the thermal stresses resulting from the introduction of cold spray water.

5.5.10.3.5 Pressurizer Design Analysis

The occurrences for pressurizer design cycle analysis are defined as follows:

A. The temperature in the pressurizer vessel is always, for design purposes, assumed to equal saturation temperature for the existing RCS pressure, except in the pressurizer steam space subsequent to a pressure increase. In this case, the temperature of the steam space will exceed the saturation temperature since an isentropic compression of the steam is assumed.

   The only exception to the above occurs when the pressurizer is filled solid during plant startup and cooldown.

B. The temperature shock on the spray nozzle is assumed to equal the temperature of the nozzle minus the cold leg temperature, and the temperature shock on the surge nozzle is assumed to equal the pressurizer water space temperature minus the hot leg temperature.

C. Pressurizer spray is assumed to be initiated instantaneously to its design value as soon as the RCS pressure increases 40 psi above the nominal operating pressure. Spray is assumed to be terminated as soon as the RCS pressure falls 40 psi below the normal operating pressure.

D. Unless otherwise noted, pressurizer spray is assumed to be initiated once per occurrence of each transient condition. The pressurizer surge nozzle is also assumed to be subject to one temperature transient per transient condition, unless otherwise noted.

E. At the end of each transient, except the faulted conditions, the RCS is assumed to return to a load condition consistent with the plant heatup transient.

F. Temperature changes occurring as a result of pressurizer spray are assumed to be instantaneous. Temperature changes occurring on the surge nozzle are also assumed to be instantaneous.

G. Whenever spray is initiated in the pressurizer, the pressurizer water level is assumed to be at the no load level.

5.5.10.4 Tests and Inspections

The pressurizer is designed and constructed in accordance with ASME Section III.

To implement the requirements of ASME Section XI, the following welds are designed and constructed to present a smooth transition surface between the parent metal and the weld metal. The path is ground smooth for ultrasonic inspection.

A. Support skirt to the pressurizer lower head.

B. Surge nozzle to the lower head.

C. Nozzles to the safety, relief, and spray lines.

D. Nozzle-to-safe-end attachment welds.

E. All girth and longitudinal full penetration welds.

F. Manway attachment welds.

The liner within the safe-end nozzle region extends beyond the weld region to maintain a uniform geometry for ultrasonic inspection.

Peripheral support rings are furnished for the removable insulation modules.

The pressurizer quality assurance program is given in table 5.5-10.

5.5.11 PRESSURIZER RELIEF TANK

5.5.11.1 Design Bases

Design data for the PRT are given in table 5.5-11. Codes and materials of the tank are given in section 5.2.

The tank is designed to accept a steam discharge from the pressurizer equal to 110 percent of the volume above the full power pressurizer water level setpoint. 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, assuming an initial temperature of 120°F and increasing to a final temperature of 205°F. If the temperature in the tank rises above 120°F during plant operation, the tank is cooled by spraying cool water and draining out the warm mixture to the waste processing system.

The PRT will normally be cooled by circulating the contents through the reactor coolant drain tank heat exchanger (RCDTH). The heat transfer capacity of the RCDTH is sufficient to cool the contents of the PRT to 120°F within 8 h following a design steam discharge. A backup for this cooling mode is provided by spraying in cool water and draining out the tank to the recycle holdup tank via the RCDT pumps.

5.5.11.2 Design Description

The PRT condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from smaller relief valves located inside or outside the containment and the reactor vessel head vent system (RVHVS) is also piped to the relief tank. The tank normally contains water and a predominantly nitrogen atmosphere.

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. A flanged nozzle is provided on the tank for the pressurizer discharge line connection.

5.5.11.2.1 Pressurizer Relief Tank Pressure

The PRT pressure transmitter provides an alarm signal on the main control board, should there be a steam discharge into the PRT when the vent valve is open.

5.5.11.2.2 Pressurizer Relief Tank Level

The PRT level transmitter supplies a signal for an indicator and high- and low-level alarms.

5.5.11.2.3 Pressurizer Relief Tank Water Temperature

The temperature of the water in the PRT is checked and an alarm, actuated by high temperature, informs the operator that cooling of the tank contents is required.

5.5.11.3 Design Evaluation

The volume of water in the tank is capable of absorbing heat from the pressurizer discharge during a loss of load from full power without a turbine trip. Water temperature in the tank is maintained at the nominal containment temperature.

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 design safety valve discharge described above. The tank and rupture disc holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20 percent of the setpoint pressure at full flow.

5.5.12 VALVES

5.5.12.1 Design Bases

Design parameters for valves within the reactor coolant pressure boundary are given in table 5.5-12. As noted in subsection 5.2.1, all valves out-to-and-including the second valve are normally closed or capable of automatic or remote closure. Valve closure time is such that for any postulated component failure outside the system boundary, the loss of reactor coolant would not prevent orderly reactor shutdown and cooldown, assuming makeup is provided by normal makeup systems. Normal makeup systems are those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, operation, or cooldown. If the second of two normally open check valves is considered the boundary, means are provided to periodically assess backflow leakage of the first valve when closed. For a check valve to qualify as the system boundary, it must be located inside the containment system.

Materials of construction are specified to minimize corrosion/erosion and to ensure compatibility with the environment.

5.5.12.2 Design Description

All valves in the RCS that are in contact with the coolant are constructed primarily of stainless steel. Other materials in contact with the coolant, such as those for hard surfacing and packing, are special materials.

All manual and motor-operated valves of the RCS which are 3-in. and larger are provided with double-packed stuffing boxes and stem intermediate-lantern-gland-leakoff connections. All throttling control valves, regardless of size, are provided with double-packed stuffing boxes and with stem-leakoff connections. All leakoff connections are piped to a closed collection system if the valve normally contains radioactive fluid and operates above 212°F. Leakage to the atmosphere is essentially zero for these valves.

Gate valves at the engineered safety features interface are wedge design and are essentially straight through.

The wedges are flex-wedge or solid. All gate valves have backseat and outside screw and yoke. Globe valves, "T" and "Y" style, are full-ported with outside screw and yoke construction. Check valves are spring-loaded, lift-piston types for sizes 2-in. and smaller, and swing-type for sizes 2-1/2 in. and larger. All check valves which contain radioactive fluid are stainless steel and do not have body penetrations other than the inlet, outlet, and bonnet. The check hinge is serviced through the bonnet.

The accumulator check valve is designed with a low pressure drop configuration, with all operating parts contained within the body. The disc has limited rotation to provide a change of seating surface and alignment after each valve opening.

Valves at the RHRS interface are provided with interlocks that meet the intent of IEEE-279. This interlock is discussed in detail in subsection 5.5.7 above and section 7.6.

The isolation valves between the accumulators and the RCS boundary are provided with interlocks that meet the intent of IEEE-279 and ensure automatic valve opening when RCS pressure exceeds a specified pressure or on safety injection signal. These interlocks are discussed in detail in section 7.6.

The reactor coolant boundary valve quality assurance parameters are given in table 5.5-13.

5.5.12.3 Design Evaluation

Stress analysis of the RCL/support system, discussed in subsections 3.7.3 and 5.2.1, ensures acceptable stresses for all valves in the reactor coolant pressure boundary under every anticipated condition.

Reactor coolant chemistry parameters are specified to minimize corrosion. Periodic analyses of coolant chemical composition, specified in the Technical Requirements Manual, ensure that the reactor coolant meets these specifications. The upper-limit coolant velocity of about 60 ft/s precludes accelerated corrosion.

Valve leakage is minimized by design features as discussed above.

All RCS boundary valves required to perform a safety function during the short-term recovery from transients or events considered in the respective operating condition categories will operate in 10 s or less.

5.5.12.4 Tests and Inspections

Hydrostatic tests are performed in accordance with ASME B&PV Code Section XI, as modified by ASME Code Case N-498-1. Seat leakage and operation tests are performed on reactor coolant boundary valves in accordance with the Technical Specifications.

There are no full-penetration welds within valve-body walls. Valves are accessible for disassembly and internal visual inspection. The valve quality assurance program is given in table 5.5-13. In addition, minimum wall thicknesses are being dimensionally verified and documented for valves 1 in. or larger NPS.

Inservice inspection is discussed in subsection 5.2.8.

5.5.13 SAFETY AND RELIEF VALVES

5.5.13.1 Design Bases

The combined capacity of the pressurizer safety valves is designed to accommodate the maximum surge resulting from loss of load. This objective is met without reactor trip or any operator action, provided the steam safety valves open as designed when steam pressure reaches the steam-side safety setting.

The pressurizer PORVs are designed to limit pressurizer pressure to a value below the high pressure reactor trip setpoint for 50-percent step load changes with steam-dump valves. The PORVs may be used to vent the pressurizer in order to remove entrained and undissolved gas from the RCS during a fill and vent operation. The PORVs may also be used to establish low-head safety injection and allow the accumulators to discharge in the event of an inadequate core cooling transient by reducing RCS pressure. The PORVs are a backup means of RCS depressurization and, for Farley Unit 2, meet the NRC acceptance criterion contained in Table 1 of Branch Technical Position RSB 5-1, Revision 1.

5.5.13.2 Design Description

The pressurizer safety valves are of the totally enclosed pop-type. The valves are spring-loaded, self-activated, have backpressure compensation features, and are designed to functionally operate during post-SSE conditions. These valves are provided with position indicating switches which provide the status of the valve (open/closed) in the control room.

The 6-in. pipe connecting each pressurizer nozzle to its respective code safety valve is shaped in the form of a loop seal. Condensate, as a result of normal heat losses, accumulates in the loop. The water prevents any leakage of hydrogen gas or steam through the safety valve seats. If the pressurizer pressure exceeds the set pressure of the safety valves, they start lifting, and the water from the seal discharges during the accumulation period. This loop pipe is insulated to reduce relief valve discharge hydrodynamic loads on the piping.

The PORVs are quick opening, operated automatically or by remote control, and are designed to functionally operate during post-SSE conditions. Each PORV is air-operated and equipped with two solenoid valves in their respective air lines. The solenoids are powered from train A and train B 125-V dc buses, respectively. These buses are powered through associated battery chargers which can also be supplied from the train A or B diesel generators upon loss of offsite power, or from 125-V dc batteries. A backup, seismically designed air supply is also provided for the PORVs by two nitrogen accumulators. The PORVs are designed to fail closed on a loss of air supply.

Remotely operated block valves are provided to isolate the PORVs if excessive leakage develops. Positive position indication for the block valves is provided in the control room from Limitorque limit switches in the valve operators. Temperatures in the pressurizer safety and relief valves discharge lines are measured and indicated. An increase in a discharge line temperature is an indication of leakage through the associated valve.

The reactor coolant leakage detection system, as discussed in section 5.2, provides sufficient sensitivity to detect increases in leakage rates while the total leakage rate is below a value consistent with safe operation of the plant. To detect PORV and safety valve leakage, the following indication is provided on the main control board: temperature detectors on the PORV and safety valve discharge lines; and temperature, pressure, and level indication for the PRT, to which PORV and safety valve discharge piping is routed.

Design parameters for the pressurizer safety- and PORVs are given in table 5.5-14.

5.5.13.3 Design Evaluation

The pressurizer safety valves prevent RCS pressure from exceeding 110 percent of system design pressure, in compliance with the ASME Code Section III.

The PORVs, the safety relief valves, and the reactor high-pressure trip setpoints meet the NRC acceptance criteria of NUREG-0737, item II.K.3.2.

The pressurizer PORVs prevent actuation of the fixed reactor high-pressure trip for all design transients up to and including the design step load decrease with steam dump. The relief valves also limit unnecessary opening of the spring-loaded safety valves.

See paragraph 5.2.2.2 for a discussion of the installation details and adequacy of the support design.

5.5.13.4 Tests and Inspections

Preoperational tests and inspections performed on safety and relief valves included valve performance tests and inspections. Periodic operational tests include hydrostatic, seat leakage, and set pressure tests for safety valves and hydrostatic and stroke tests for PORVs.

There are no full-penetration welds within the valve-body walls. Valves are accessible for disassembly and internal visual inspection.

5.5.14 COMPONENT SUPPORTS

Component supports allow virtually unrestrained lateral thermal movement of the loop during plant operation and provide restraint to the loops and components during accident conditions. The loading combinations and design stress limits are discussed in paragraph 5.2.1.9. The fracture toughness material properties of the steam generator and RCP supports are discussed in a Westinghouse report entitled, "Fracture Toughness and Potential for Lamellar Tearing of Steam Generator and Reactor Coolant Pump Support Materials," which was submitted to the NRC by reference 5. The design maintains the integrity of the RCS boundary for normal and accident conditions and satisfies the requirements of the piping code. Results of piping and supports stress evaluation are presented in paragraph 5.2.1.10.

5.5.14.1 Description

The support structures are of welded steel construction and are either a linear type or plate-and-shell type. Vessel skirts and saddles are fabricated from plate-and-shell elements to accommodate a biaxial stress field. Linear supports are tension and compression struts, beams, and columns. Attachments are of integral and nonintegral types. Integral attachments are welded, cast, or forged to the pressure boundary component by lugs, shoes, rings, and skirts.

Nonintegral attachments are bolted or pinned, or bear on the pressure boundary component.

The supports permit unrestrained thermal growth of the supported systems but restrain vertical, lateral, and rotational movement resulting from seismic and pipe-break loadings. This is accomplished using pin-ended columns for vertical support and girders, bumper pedestals, and tie-rods for lateral support.
Shimming and grouting enable adjustment of all support elements during erection to achieve correct fit-up and alignment. Final setting of equipment is by shim and grouting at the concrete steel support interface rather than at the equipment support interface. For the Model 54F replacement steam generators, shims are included between the tops of the steam generator support columns and the steam generator feet to achieve correct fit-up and alignment.

A. Vessel

Supports for the reactor vessel (figure 5.5-7) are individual, air-cooled, rectangular-box structures beneath the vessel nozzles bolted to the primary shield wall concrete. Each box structure consists of a horizontal top plate that receives loads from the reactor vessel shoe, a horizontal bottom plate supported by and transferring loads to the primary shield wall concrete, and connecting vertical plates. The supports are air-cooled to maintain the supporting concrete temperature at or below 190°F at a flow rate of 2000 ft^3/min with an air temperature of 120°F to meet the acceptance criteria for the localized concrete temperature of 200°F. However, recognizing the potential degradation of the RPV supports subjected to sustained temperatures higher than 150°F, FNP has committed (NEL letter #00-279 to USNRC) to an augmented program to inspect the structural components including portions of the reactor vessel system (RVS) in the containment buildings as part of the maintenance rule structural monitoring program. This program will ensure that significant cracking of RVS that could affect the structural support of the reactor vessel or cause out of plumbness conditions will be detected and corrected [NRC commitment CTS #10533].

The lower supports for the steam generator (figure 5.5-8) consist of four vertical pin-ended columns bolted to the bottom of the steam generator support pads and lateral support girders and pedestals that bear against horizontal bumper blocks bolted to the side of the generator support pads. The upper lateral steam generator support consists of a ring girder around the generator shell supported by struts on three sides. Loads are transferred from the equipment to the ring girder by means of a number of bumper blocks between the girder and generator shell.

B. Pump

The RCP supports (figure 5.5-9) consist of three pin-ended structural steel columns and three lateral tie-bars. A large-diameter bolt connects each column and tie-rod to a pump support pad. The outer ends of all three tie-rods have slotted pinholes to permit unrestrained lateral movement of the pump during plant heatup and cooldown, but provide lateral restraint for accident loading.

C. Pressurizer

The pressurizer (figure 5.5-10) is supported at its base by bolting the flange ring to the supporting concrete slab. In addition, upper lateral support is provided near the vessel center of gravity by four "V frames" or struts extending horizontally from the compartment walls and bearing against the vessel lugs.

5.5.14.2 Evaluation

Detailed evaluation ensures the design adequacy and structural integrity of the RCL and the primary equipment supports system. This detailed evaluation is made by comparing the analytical results with established criteria for acceptability. Structural analyses are performed to demonstrate design adequacy for safety and reliability of the plant in case of a large or small seismic disturbance and/or LOCA conditions. Loads that the system is expected to encounter often during its lifetime, including thermal, weight, pressure, and 1/2 SSE, are applied and stresses are compared to allowable values, as described in paragraph 5.2.1.10.

The SSE and design basis LOCA, resulting in a rapid depressurization of the system, are required design conditions for public health and safety.

For SSE and LOCA loadings, the basic criteria ensure that the severity will not be increased, thus maintaining the system for a safe-shutdown condition. The rupture of a RCL pipe will not violate the integrity of the unbroken leg of the loop. To ensure the integrity and stability of the RCL support system and a safe shutdown of the system under LOCA and the worst combined (Normal + SSE + LOCA) loadings, the stresses in the unbroken piping of a broken loop and the unbroken loop piping and the supports system are analyzed. The results of design analysis are provided in paragraph 5.2.1.10.

5.5.14.3 Tests and Inspections

Weld inspection and standards are specified in accordance with Appendix IX, Section III, ASME Code. Welder qualifications and welding procedures are specified in accordance with Section IX, ASME Code.

5.5.15 REACTOR VESSEL HEAD VENT SYSTEM

5.5.15.1 Design Basis

The basic function of the RVHVS is to remove noncondensable gases or steam from the RVH and the RCS, which may inhibit core cooling during natural circulation for events beyond the present design basis. This system is designed to mitigate a possible condition of inadequate core cooling, or impaired natural circulation, resulting from the accumulation of noncondensable gases in the RCS. The existing PORV system functions as the RCS vent for the pressurizer. See subsection 5.5.13 for a discussion of the PORV system. Venting capability of the RCS hot legs is not required since the hot legs are not a high point in the system. The design of the RCS satisfies the requirements of NUREG-0737.

5.5.15.2 System Description

The RVHVS is designed to remove noncondensable gases or steam from the reactor vessel via remote manual operations from the control room. The system discharges to the PRT. The RVHVS is designed to vent a volume of hydrogen at system design pressure and temperature approximately equivalent to one-half of the RCS volume in 1 h.

The RVHVS will be connected to the existing 1-in. vent pipe, which is located near the center of the RVH. The system consists of two parallel-flow paths with redundant 1-in., open/close, solenoid-operated isolation valves and a 3/8-in. flow-limiting orifice in each flow path. The venting operation uses only one of these flow paths at any one time. The flow diagram of the RVHVS is shown in drawings D-175037, sheet 1; D-205037, sheet 1; D-175037, sheet 2; and D-205037, sheet 2; and the equipment design parameters are listed in table 5.5-16.

The system design with two valves in series in each flow path minimizes the possibility of reactor coolant pressure boundary leakage. The exception is with a manual vent (one normally closed valve and a blind flange).

The isolation valves in one flow path will be powered by one vital power supply and the valves in the second flow path will be powered by a second vital power supply. The isolation valves are fail/closed, normally closed-active valves. The isolation valves are also included in the Westinghouse valve operability program which is an acceptable alternative to Regulatory

Guide 1.48. These valves are qualified to IEEE-323-1974, IEEE-344-1975, and IEEE-382-1972.

All piping and equipment used in the reactor head vent system from the connection to the existing vent pipe to the orifices are designed and fabricated in accordance with ASME, Section III, Class 1 requirements. From the orifices up to and including the second isolation valves, all equipment is designed and fabricated in accordance with ASME, Section III, Class 2 requirements. The piping downstream of the second isolation valves is nonnuclear safety. The materials used in construction of the new portion of the RVHVS were fabricated and tested in accordance with SRP section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

The RVHVS is operated from the control room. The isolation valves have stem position switches. The position indication from each valve is monitored in the control room by status lights.

5.5.15.3 Design Evaluation

The series/parallel arrangement of the four valves in the RVHVS precludes failure of the system through the failure of one of the valves to operate properly. If one single-active failure prevents a venting operation through one flow path, the redundant path is available for venting. Similarly, the two isolation valves in each flow path provide a single-failure method of isolating each of the venting subsystems. With two valves in series, the failure of any one valve or power supply will not inadvertently open a vent path, and power lockout to any valve is not considered necessary. Thus, the combination of safety-grade train assignments and valve failure modes will not prevent vessel head venting or venting isolation with any single-active failure.

The inadvertent opening of both isolation valves in one of the RVHVS flow paths will result in discharge to the PRT. No damage to safe shutdown equipment will result.

The orifice forms the Safety Class 1 to Safety Class 2 transition. The system is orificed to limit the blowdown from a break downstream of either of the orifices to within the capacity of one of the centrifugal charging pumps, thus preventing a net loss of coolant and satisfying the requirements of NUREG-0737.

The only missile which has a potential for impacting the RVHVS is a control rod drive mechanism (CRDM) missile. However, a break of the RVH vent line upstream of the orifices would result in a small LOCA of not greater than 1-in. diameter. This break is similar to those analyzed in WCAP-9600 (reference 6). Since this break would behave in a manner similar to the hot leg break case presented in reference 6, the results presented therein are applicable. This postulated vent line break would therefore result in no calculated core uncovery.

The system provides for venting the RVH by using only safety-grade equipment. The RVHVS satisfies applicable requirements and industry standards, including ASME Code classification, safety classification, single-failure criteria, and environmental qualification. Additional information and analyses of the RVHVS are contained in reference 7.

5.5.15.4 Tests and Inspections

Periodic visual inspections and preventive maintenance are conducted during the refueling operation according to normal industrial practice. Surveillance requirements for the RVHVS are specified in the Technical Requirements Manual.

REFERENCES

1. Paidoussis, M., "The Amplitude of Fluid-Induced Vibration of Cylinders in Axial Flow, "AECL-2225.

2. Chen, Y., "Flow-Induced Vibration and Noise in Tube-Bank Heat Exchangers Due to von Karmen Streets," ASME Publication, 1967.

3. Nelms, M. and Segaser, C., "Survey of Nuclear Reactor System Primary Circuit Heat Exchangers," ORNL-4399.

4. Connors, H. J. Jr., "Fluidelastic Vibration of Tube Arrays Excited by Cross-flow," Proceedings of Symposium on Flow-Induced Vibration in Heat Exchangers, ASME Winter Annual Meeting, New York, December 1970.

5. Letter from F. L. Clayton, Jr. (Alabama Power) to J. F. Stolz (NRC), dated May 30, 1978.

6. WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System," (specifically case F, section 3.2), June 1979.

7. Letter from F. L. Clayton, Jr. (Alabama Power) to B. J. Youngblood (NRC), dated June 25, 1981.

8. PWROG-14001-P, Revision 0, “PRA Model for the Generation III Westinghouse Shutdown Seal,” June 2014.

TABLE 5.5-1

REACTOR COOLANT PUMP DESIGN PARAMETERS

Design pressure (psig)	2485
Design temperature (°F)	650
Capacity per pump (gpm)	88,500
Developed head (ft)	264
NPSH required (ft)	170
Suction temperature (°F)	543.3
RPM nameplate rating	1200
Discharge nozzle, ID (in.)	27-1/2
Suction nozzle, ID (in.)	31
Overall unit height (ft-in.)	26-10
Water volume (ft^3)	57
Moment of inertia (ft-lb)	82,000
Weight, dry (lb)	197,000
Motor	
Type	AC induction, single speed, air cooled
Power (H.P.)	6000
Voltage, volts	4000
Insulation class	
Hot loop operation	Class B
Cold loop operation	Class F
Phase	3
Frequency (Hz)	60
Starting current	5120 @ 4000V
Input, hot reactor coolant (kW)	4870
Input, cold reactor coolant (kW)	6165
Seal water injection (gpm)	8
Seal water return (gpm)	3

[HISTORICAL] [TABLE 5.5-2

REACTOR COOLANT PUMP QUALITY ASSURANCE PROGRAM

	$RT^{(a)}$	$UT^{(a)}$	$PT^{(a)}$	$MT^{(a)}$
Castings	*yes*		*yes*	
Forgings				
1. Main shaft		*yes*	*yes*	
2. Main flange bolts		*yes*		*yes*
3. Flywheel (rolled plate)		*yes*	*yes for bore*	
Weldments, Pressure Boundary				
1. Circumferential	*yes*		*yes*	
2. Instrument connections			*yes*	

---

a. RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle]

TABLE 5.5-3

STEAM GENERATOR DESIGN DATA[a]

Number of steam generators per Unit	(No.)	3
Design pressure, RCS / Steam	(psig)	2,485/1,085
RCS hydrostatic test pressure (tube side - cold)	(psig)	3,107
Design temperature, reactor coolant / steam	(°F)	650/600
Reactor coolant flow	(lb/h)	32.7×10^6
Total head transfer surface area	(ft^2)	54,500
Heat transferred	(Btu/h)	$3,168 \times 10^6$
Steam Conditions at full load, outlet nozzle:		
Steam flow	(lb/h)	4.08×10^6
Steam temperature	(°F)	515.5
Steam pressure	(psig)	781
Maximum moisture carryover	(wt %)	0.10
Feedwater	(°F)	443.4
Overall height	(ft-in.)	67-9
Shell OD, upper/lower	(in.)	177/136
Total number of U-tubes (plugged and unplugged)	(No.)	3,592
U-tube outer diameter	(in.)	0.875
Tube wall thickness, (minimum)	(in.)	0.050
Number of manways/ID	(No.)	4 (16 inch)
Number of inspection ports ID	(No.)	2 (4 inch)
Number of inspection handholes ID	(No.)	6 (6 inch)

		Rated Load	No Load
Reactor coolant water volume	(ft^3)	1,168	1,168
Primary-side fluid heat content	(Btu)	30.6×10^6	29.9×10^6
Secondary-side water volume	(ft^3)	2,167	3,618
Secondary-side steam volume	(ft^3)	3,645	2,193
Secondary-side fluid heat content	(Btu)	6.05×10^7	9.71×10^7

a. Quantities are for each steam generator.

[HISTORICAL] [TABLE 5.5-4 (SHEET 1 OF 2)

STEAM GENERATOR QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]
Tubesheet				
Forging		*yes*		*yes*
Cladding		*yes*[b]	*yes*[c]	
Channel Head				
Forging		*yes*		*yes*
Cladding			*yes*	
Secondary Shell and Head				
Plates		*yes*		
Tubes	*yes*			*yes*
Nozzles (forgings)		*yes*		*yes*
Weldments				
Shell, circumferential	*yes*			*yes*
Cladding (channel head-tube sheet joint cladding restoration)			*yes*	
Steam and feedwater nozzle-to-shell	*yes*			*yes*
Support brackets				*yes*
Tube-to-tubesheet			*yes*	
Instrument connections (primary and secondary)				*yes*
Temporary attachments after removal				*yes*
After hydrostatic test (all welds and complete channel head – where accessible)				*yes*

TABLE 5.5-4 (SHEET 2 OF 2)

	RT[a]	*UT[a]*	*PT[a]*	*MT[a]*
Nozzle safe ends (if forgings)	*yes*		*yes*	

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

b. Flat surfaces only.

c. Weld deposit areas only. ***]***

TABLE 5.5-5

REACTOR COOLANT PIPING DESIGN PARAMETERS

	Unit 1	Unit 2
Reactor inlet piping, ID (in.)	27.5	27.5
Reactor inlet piping, nominal wall thickness (in.)	2.2975	2.3225
Reactor outlet piping, ID (in.)	29	29
Reactor outlet piping, nominal wall thickness (in.)	2.420	2.445
Coolant pump suction piping, ID (in.)	31	31
Coolant pump suction piping, nominal wall thickness (in.)	2.575	2.600
Pressurizer surge line piping, ID (in.)	11.188	11.188
Pressurizer surge line piping, nominal wall thickness (in.)	1.406	1.406
Water volume, all loops and surge line (ft^3)	1030	1030
Design/operating pressure (psig)	2485/2235	2485/2235
Design temperature (°F)	650	650
Design temperature, pressurizer surge line (°F)	680	680
Design pressure, pressurizer relief line		
From pressurizer to safety valve (psig)	2485	2485
From safety valve to relief tank (psig)	600	600
Design temperature, pressurizer relief line		
From pressurizer to safety valve (°F)	650	650
From safety valve to relief tank (°F)	600	600

[HISTORICAL] [TABLE 5.5-6

REACTOR COOLANT PIPING QUALITY ASSURANCE PROGRAM

	RT[a]	*UT*[a]	*PT*[a]
Fittings and pipe (castings)	*yes*		*yes*
Fittings and pipe (forgings)		*yes*	*yes*
Weldments			
Circumferential	*yes*		*yes*
Nozzle to runpipe (except no RT for nozzles less than 4 in.)	*yes*		*yes*
Instrument connections			*yes*

---

a. RT - Radiographic
UT - Ultrasonic
PT - Dye penetrant]

TABLE 5.5-7

DESIGN BASES FOR RESIDUAL HEAT REMOVAL SYSTEM OPERATION

Residual Heat Removal System Startup	~4 hours after reactor shutdown
Reactor Coolant System initial pressure (psig)	~425
Reactor Coolant System initial temperature (°F)	~350
Component cooling water design temperature (°F)	105
Cooldown time, hours after initiation of RHRS operation	~34
Reactor Coolant System temperature, at end of cooldown (°F)	140
Decay heat generation at 20 hours after reactor shutdown (Btu/h)	60.8×10^6

TABLE 5.5-8

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pump

Number	2
Design pressure, (psig)	600
Design temperature (°F)	400
Design flow (gpm)	3750
Design head (ft)	280

Residual Heat Exchanger

Number	2	
Design heat removal capacity	29.5 x 10^6Btu/h	
	Tube-side	Shell-side
Design pressure (psig)	600	150
Design temperature (°F)	400	200
Design flow (lb/h)	1.87 x 10^6	2.8 x 10^6
Inlet temperature (°F)	140	105
Outlet temperature (°F)	124.3	115.6
Material	Austenitic stainless steel	Carbon steel
Fluid	Reactor coolant	Component cooling water

TABLE 5.5-9

PRESSURIZER DESIGN DATA

Item	Value
Design pressure (psig)	2485
Design temperature (°F)	680
Surge line nozzle diameter (in.)	14
Heatup rate of pressurizer using heaters only (°F/h)	55
Internal volume (ft^3)	1400

[HISTORICAL] [TABLE 5.5-10

PRESSURIZER QUALITY ASSURANCE PROGRAM

Item	*RT*[a]	*UT*[a]	*PT*[a]	*MT*[a]
Heads				
Plates	*yes*			*yes*
Cladding			*yes*	
Shell				
Plates		*yes*		*yes*
Cladding			*yes*	
Heaters				
Tubing[b]		*yes*	*yes*	
Centering of element	*yes*			
Nozzle		*yes*	*yes*	
Weldments				
Shell, longitudinal	*yes*			*yes*
Shell, circumferential	*yes*			*yes*
Cladding			*yes*	
Nozzle safe-end (forging)	*yes*		*yes*	
Instrument connections			*yes*	
Support skirt				*yes*
Temporary attachments after removal				*yes*
All welds, heads, and shell after hydrostatic test				*yes*
Final assembly				
All accessible exterior surfaces after hydrostatic test				*yes*

a. RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle

b. Or a UT and ET.]

TABLE 5.5-11

PRESSURIZER RELIEF TANK DESIGN DATA

Item		Value
Design pressure (psig):	Internal	100
	External	15
Rupture disc release pressure (psig)		100 ± 5%
Design temperature (°F)		340
Total rupture disc relief capacity (lb/h at 100 psig)		1.14×10^6

TABLE 5.5-12

REACTOR COOLANT SYSTEM BOUNDARY VALVE DESIGN PARAMETERS

Item	Value
Normal operating pressure (psig)	2235
Design pressure (psig)	2485
Preoperational plant hydrotest (psig)	3107
Design temperature (°F)	650

[HISTORICAL] [TABLE 5.5-13

REACTOR COOLANT SYSTEM VALVES QUALITY ASSURANCE PROGRAM

Boundary Valves, Pressurizer Relief and Safety Valves	*RT[(a)]*	*UT[(a)]*	*PT[(a)]*
Castings	*yes*		*yes*
Forgings (no UT for valves 2 in. and smaller)		*yes*	*yes*

a. RT - Radiographic
UT - Ultrasonic
PT - Dye penetrant]

TABLE 5.5-14

PRESSURIZER VALVES DESIGN PARAMETERS

Pressurizer Spray Control Valves	
Number	2
Design pressure (psig)	2485
Design temperature (°F)	650
Design flow for valves full open, each (gpm)	300
Pressurizer Safety Valves	
Number	3
Minimum relieving capacity, ASME rated flow (lb/h)(per valve)	345,000
Set pressure (psig)	2485
Fluid	Saturated steam
Backpressure:	
Normal (psig)	3 to 5
Expected during discharge (psig)	350
Pressurizer Power Relief Valves	
Number	2
Design pressure (psig)	2485
Design temperature (°F)	650
Relieving capacity at 2350 psig (lb/h) (per valve)	210,000
Fluid (2335 psig)	Saturated steam

TABLE 5.5-15

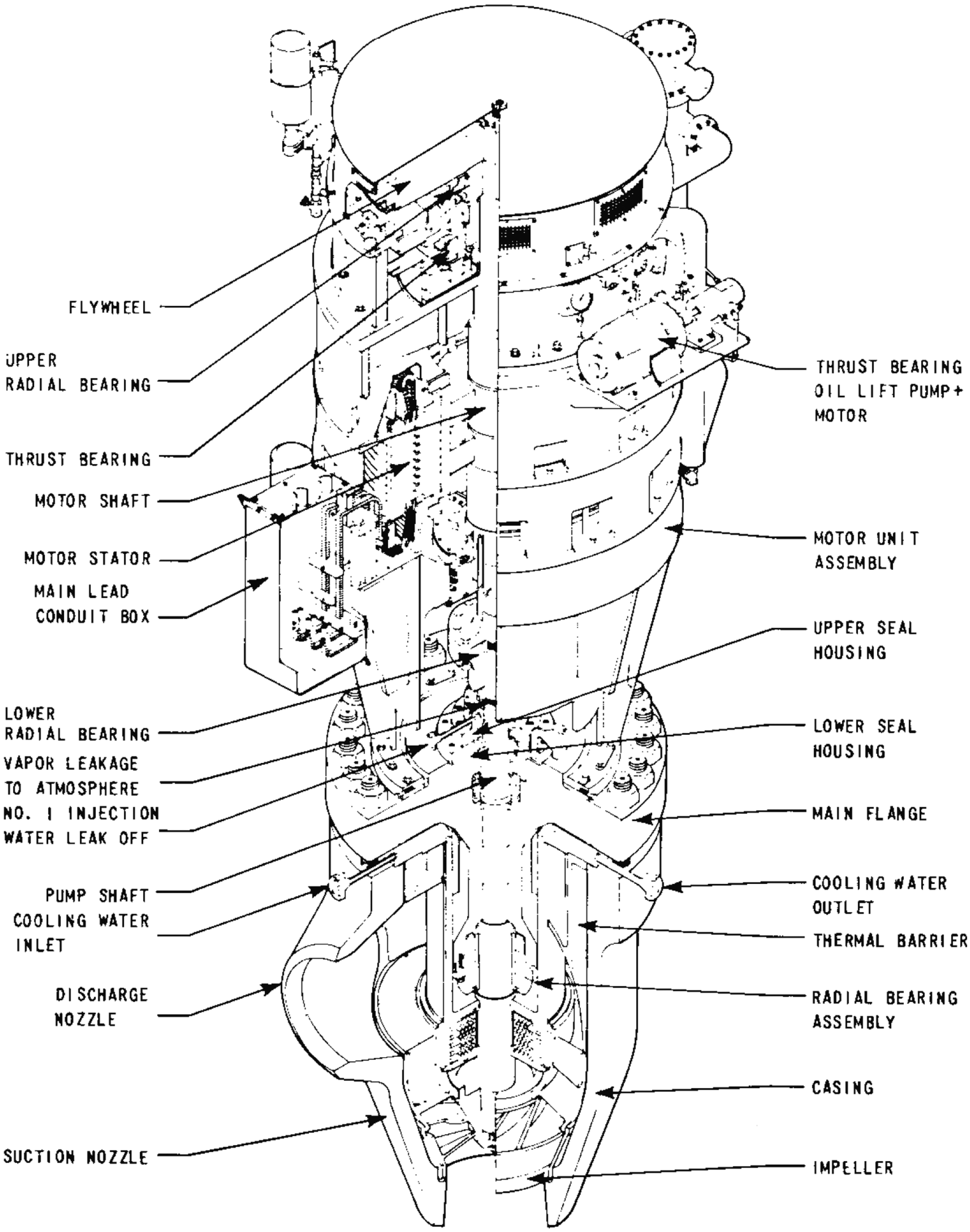
MAIN STEAM VALVE DESIGN PARAMETERS MAIN STEAM ISOLATION VALVES

Number	6
Design Pressure (psig)	1085
Design temperature (°F)	600
Normal Operating Flow (lb/h)	3.875×10^6
Main Steam Bypass Valves	
Number	6
Design pressure (psig)	1085
Design Temperature (°F)	600
Actuator Type	Piston

TABLE 5.5-16

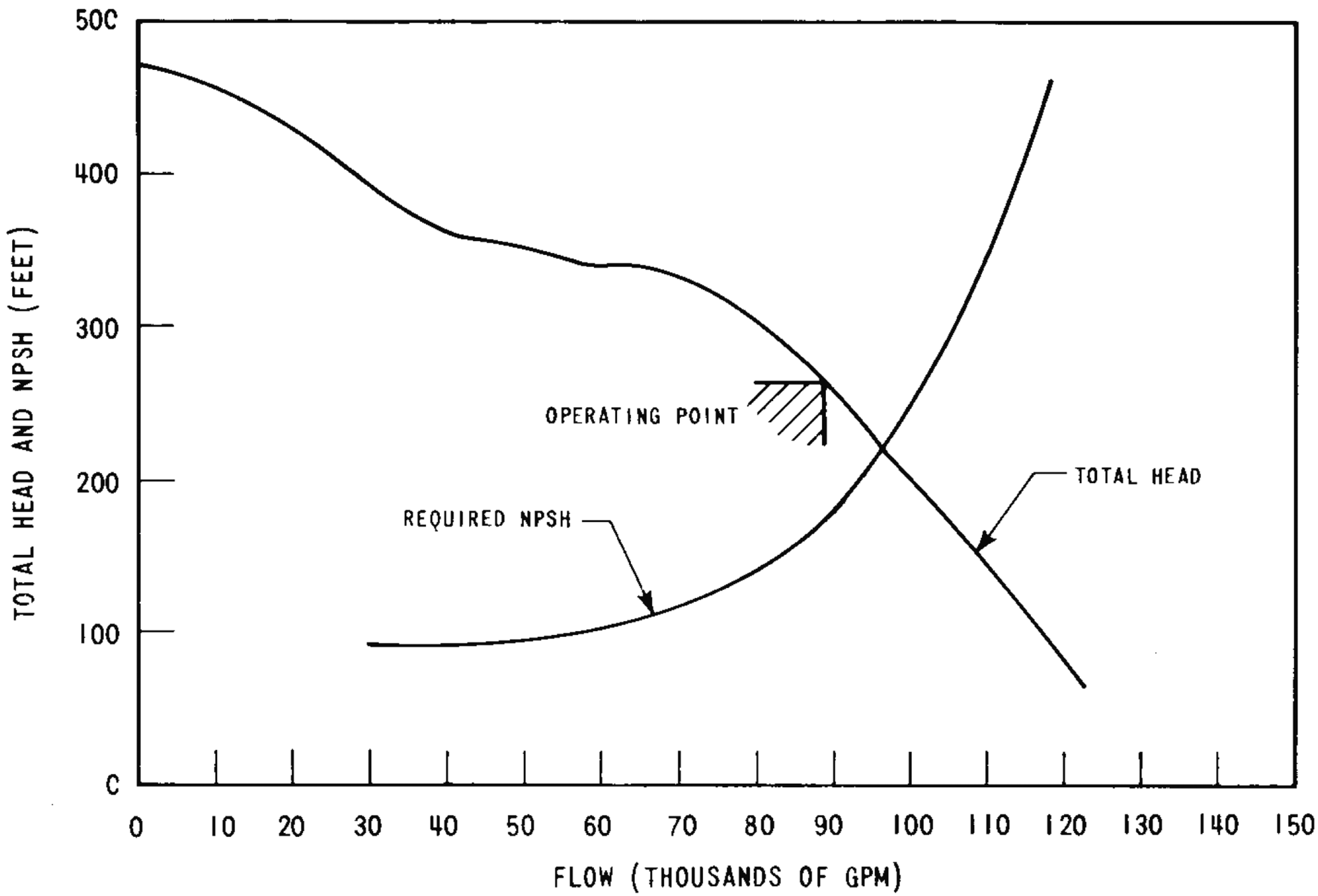
REACTOR VESSEL HEAD VENT SYSTEM EQUIPMENT DESIGN PARAMETERS

Reactor Vessel Head Vent Subsystem	
Valves	
Number (includes one manual valve)	5
Design pressure (psig)	2485
Design temperatures (°F)	650
Piping	
Vent line, nominal diameter (in.)	1
Design pressure (psig)	2485
Design temperature (°F)	650
Maximum normal operating temperature (°F)	620



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	REACTOR COOLANT CONTROLLED LEAKAGE PUMP FIGURE 5.5-1



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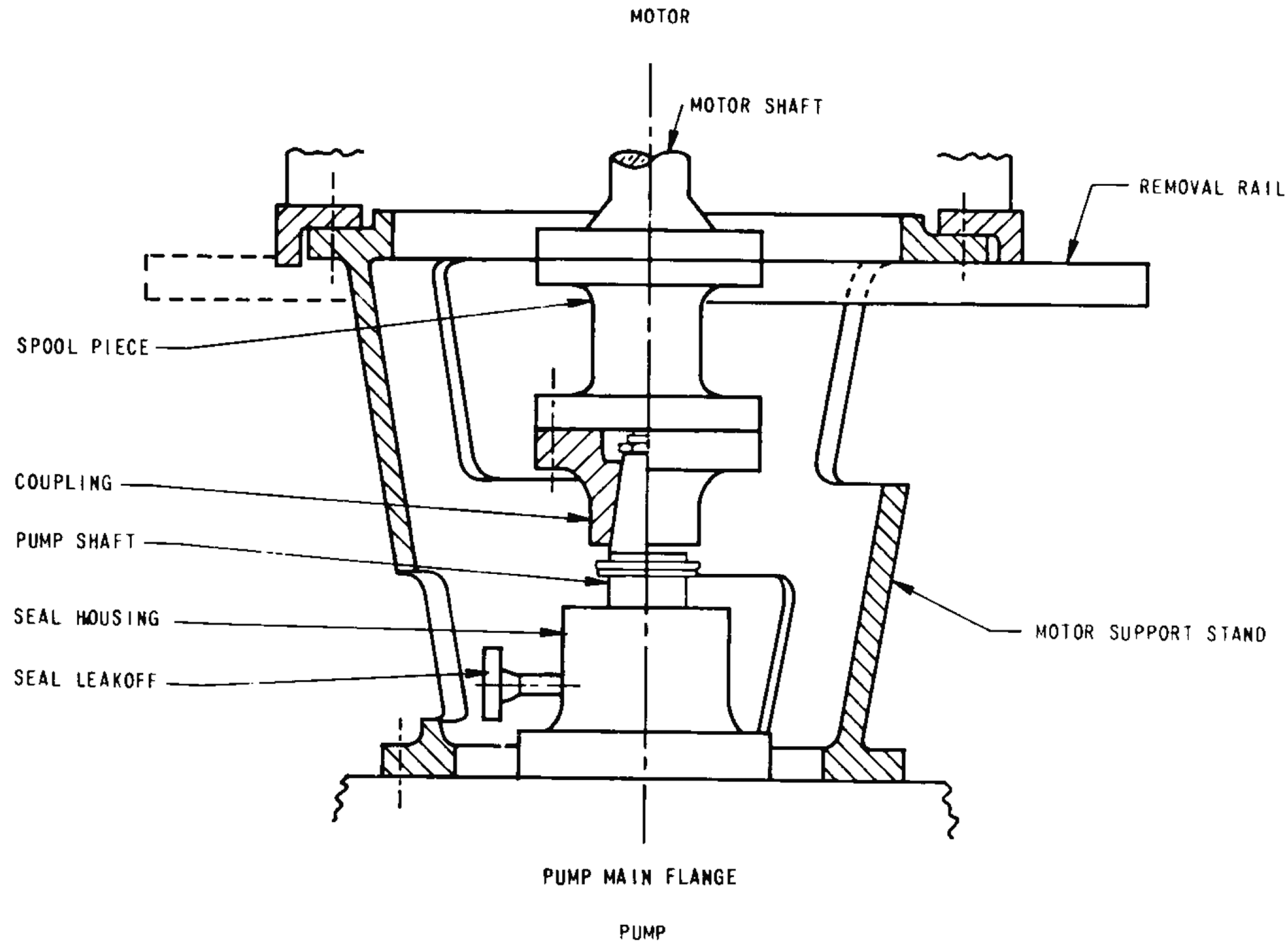
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR COOLANT PUMP PERFORMANCE CURVE

FIGURE 5.5-2



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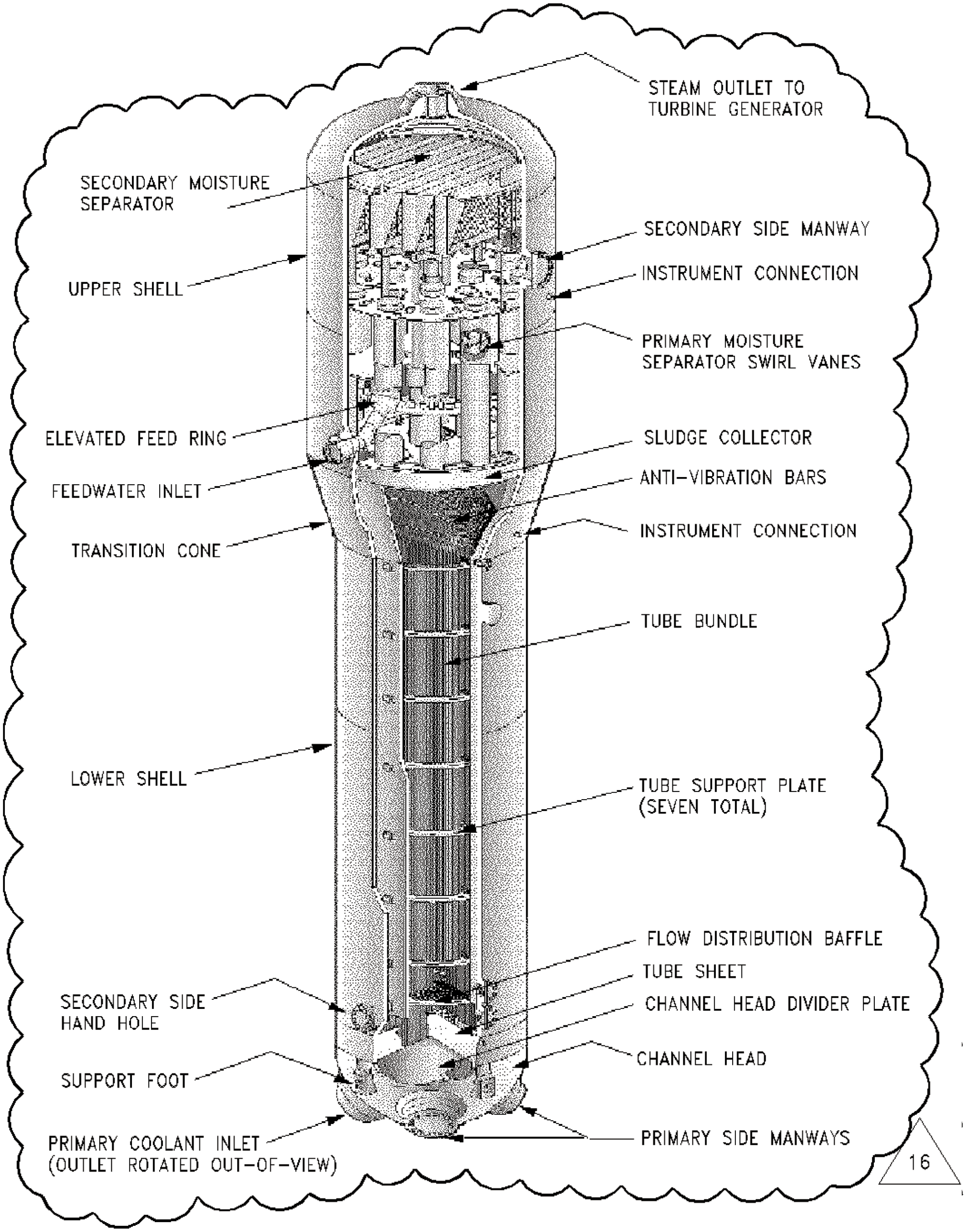
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR COOLANT PUMP SPOOL PIECE AND MOTOR SUPPORT STAND

FIGURE 5.5-3



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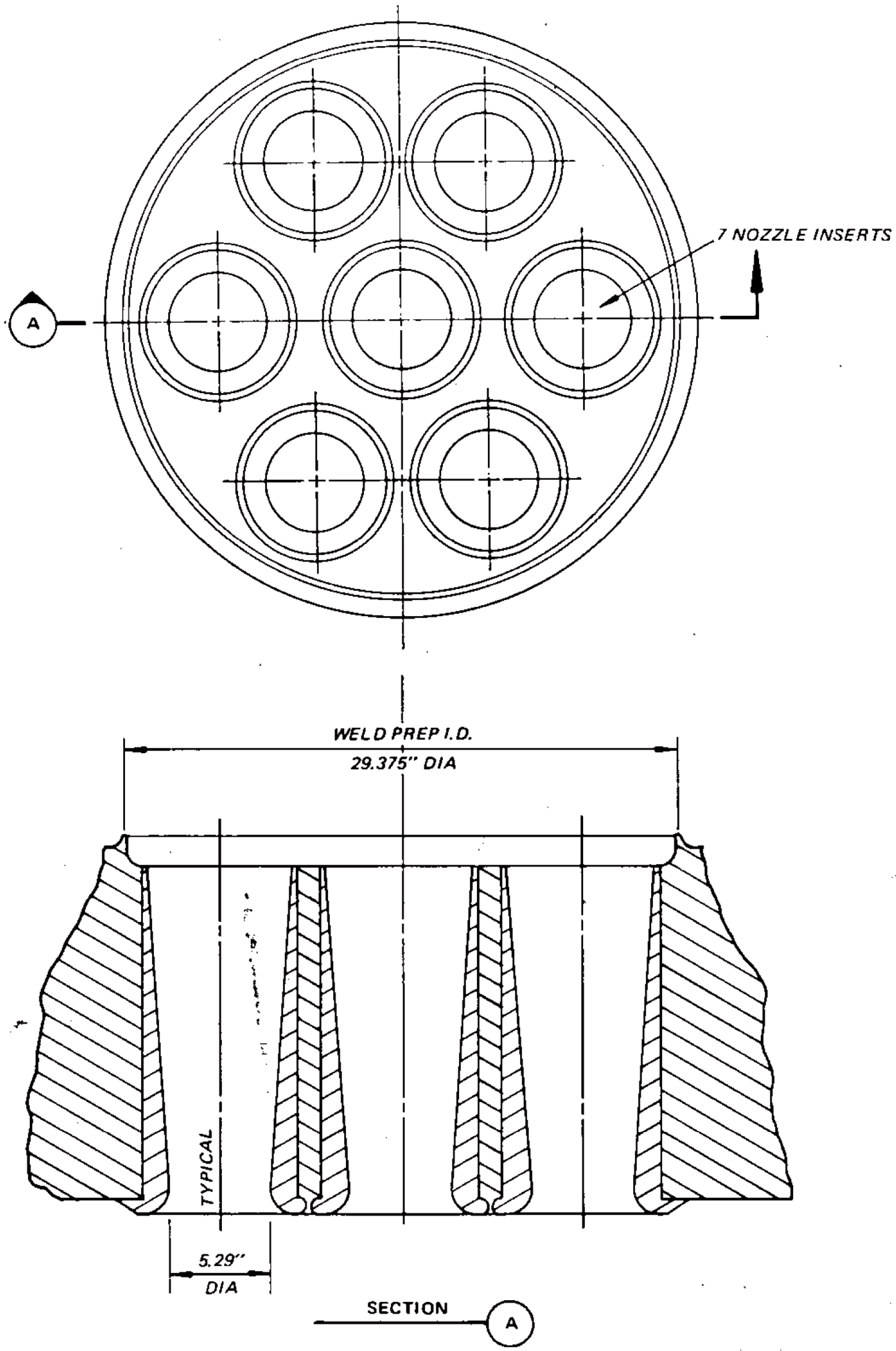
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STEAM GENERATOR
		FIGURE 5.5-4



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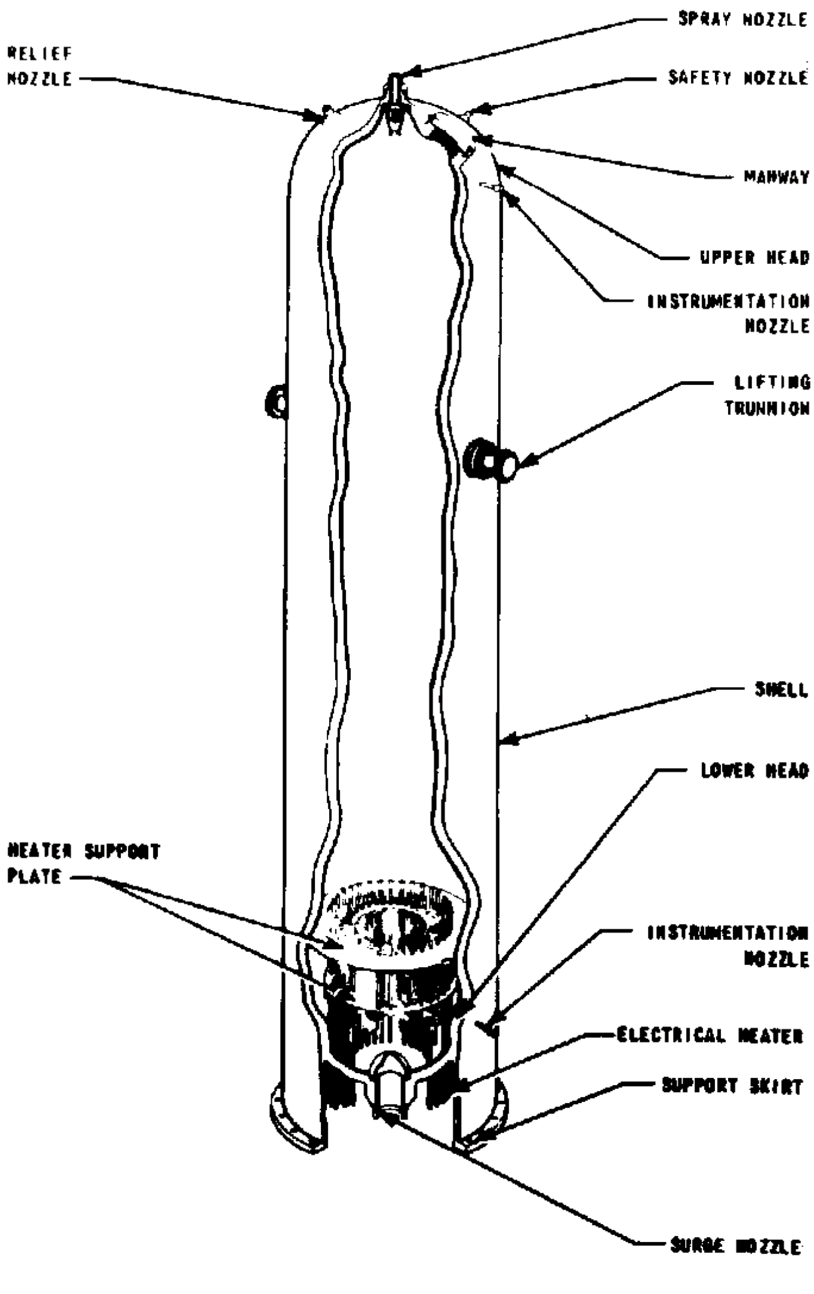
UNIT 1 AND UNIT 2

STEAM GENERATOR FLOW LIMITING DEVICE

FIGURE 5.5-5



13

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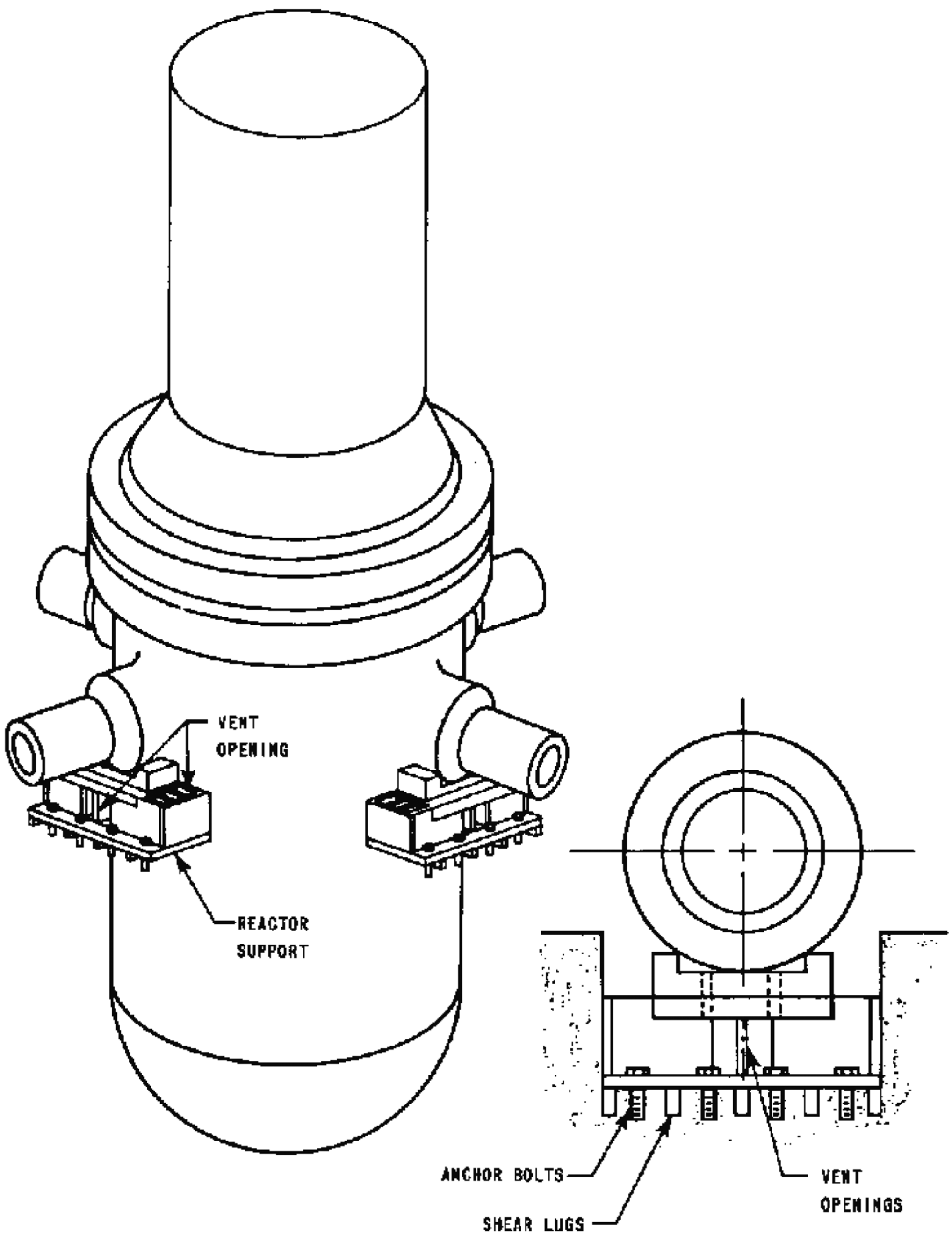
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURIZER

FIGURE 5.5-6



13

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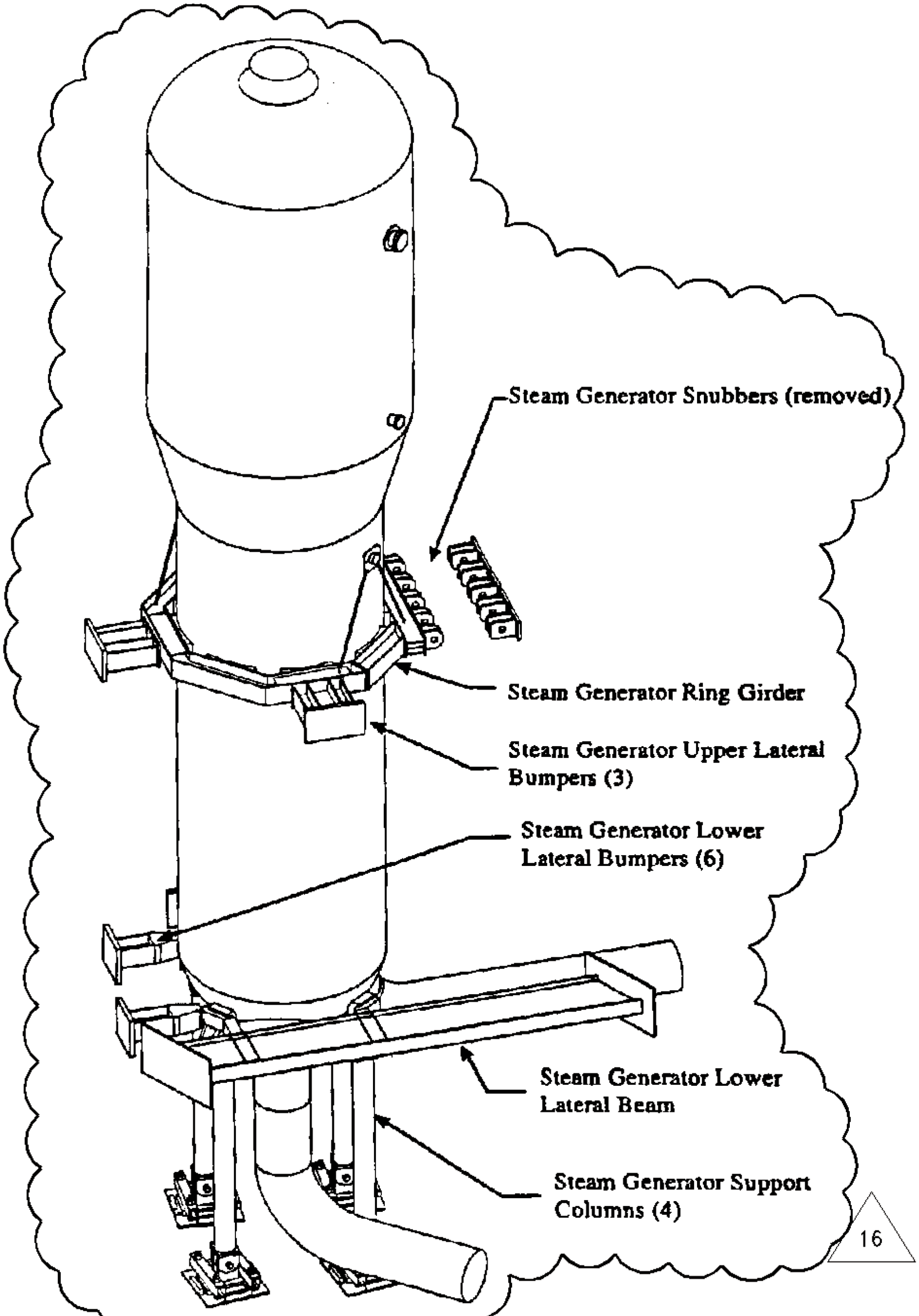
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR VESSEL SUPPORTS

FIGURE 5.5-7



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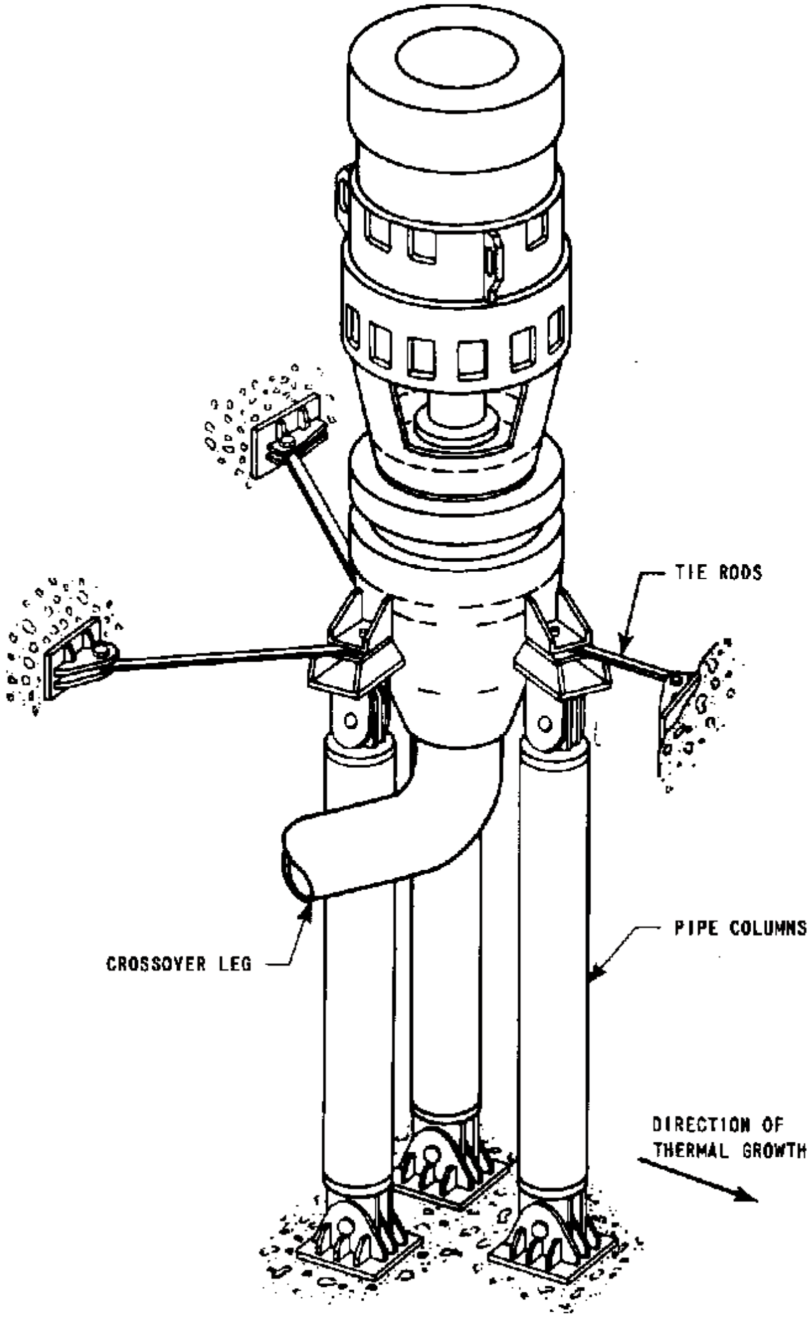
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	DRY CONTAINMENT STEAM GENERATOR SUPPORTS FIGURE 5.5-8



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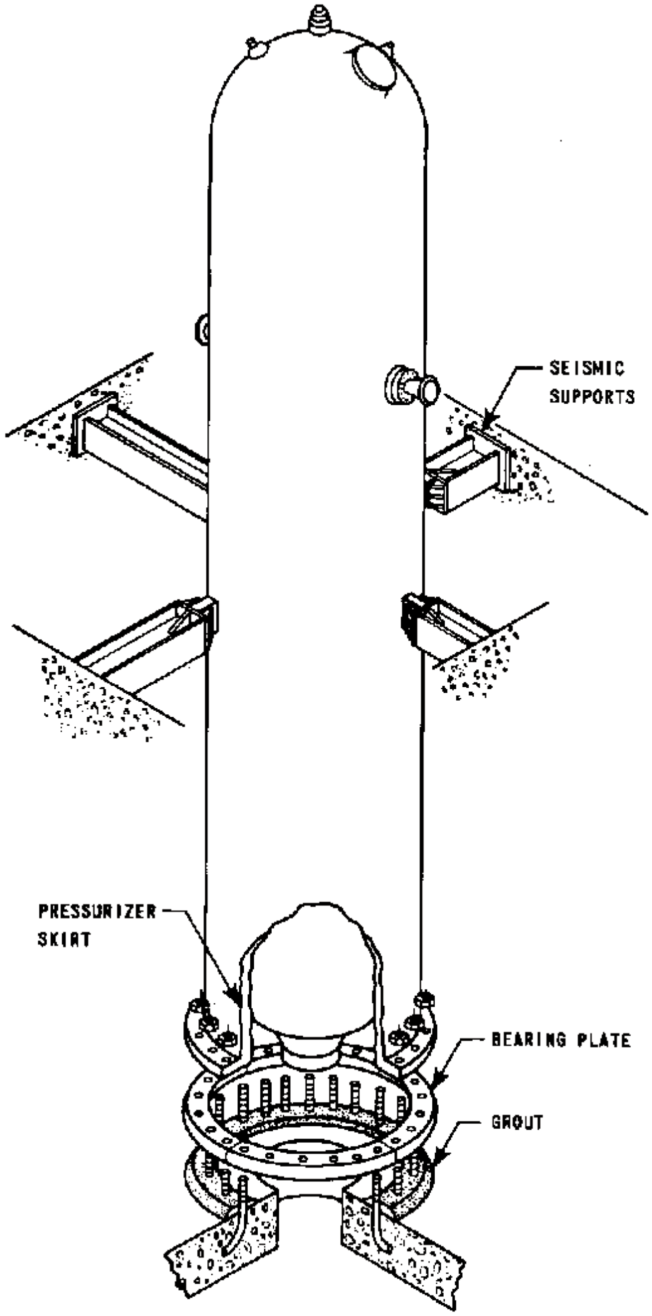
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR COOLANT PUMP SUPPORTS

FIGURE 5.5-9



13

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JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

PRESSURIZER SUPPORTS

FIGURE 5.5-10

THIS FIGURE HAS BEEN DELETED

15

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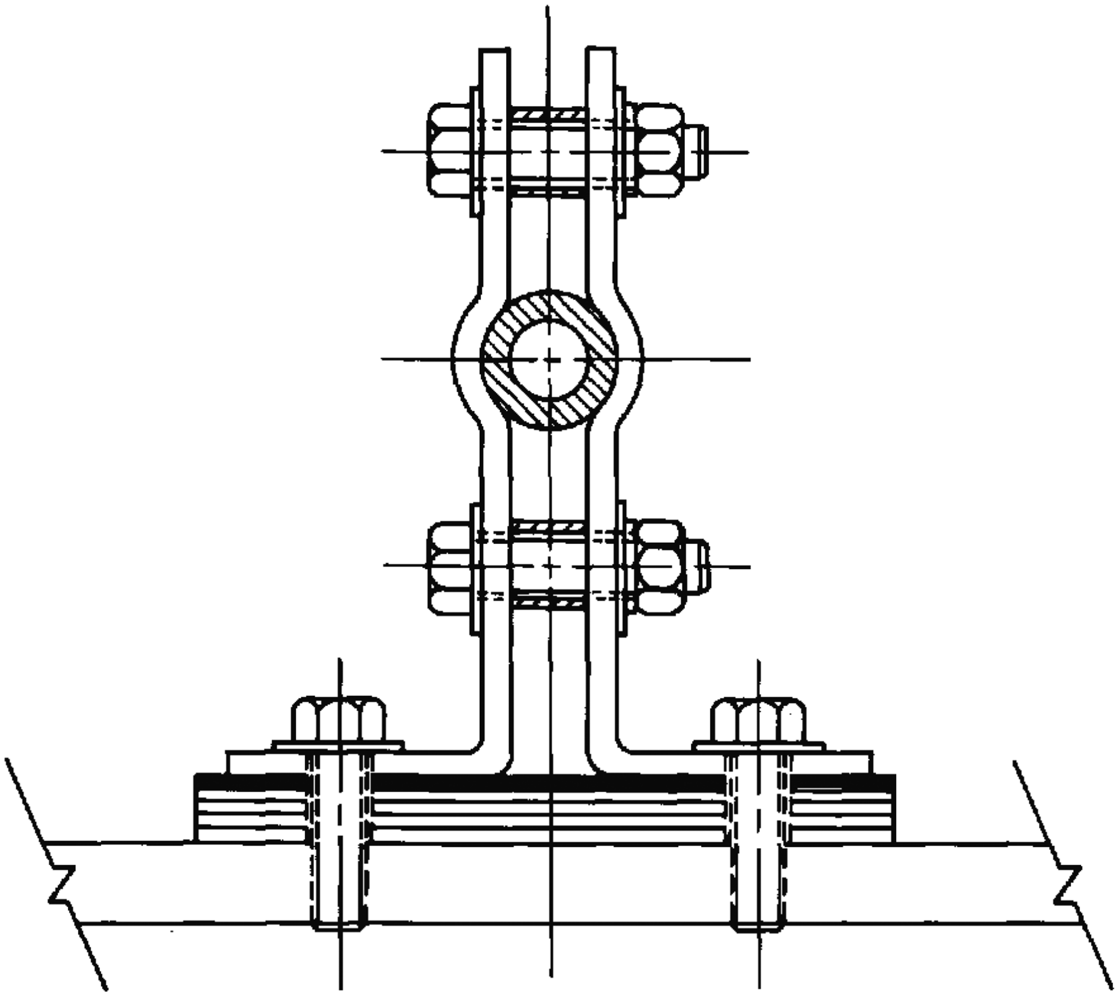
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CROSSOVER LEG RESTRAINTS

FIGURE 5.5-11



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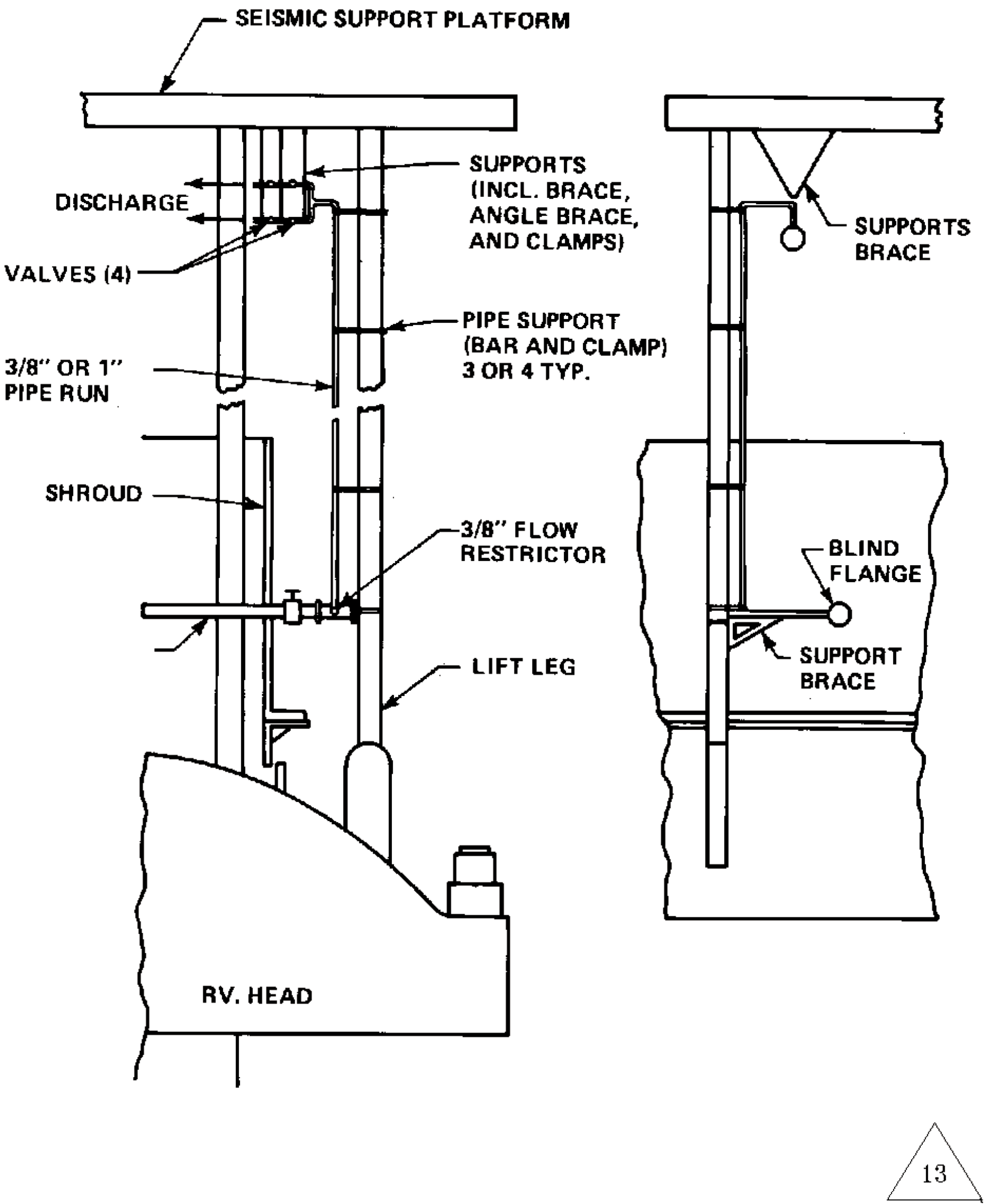
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CRDM SEISMIC SUPPORT PLATFORM
PIPE SUPPORT CLAMP

FIGURE 5.5-12



REV 21 5/08

SIDEVIEW RVHVS AND SUPPORTS

FIGURE 5.5-13

5.6 INSTRUMENTATION APPLICATION

Process control instrumentation is provided for the purpose of acquiring data on the pressurizer and, on a per-loop-basis, for the key process parameters of the reactor coolant system (RCS) (including the reactor coolant pump motors), as well as for the residual heat removal system. The pick-off points for the reactor coolant system are shown in drawings D-175037, sheet 1, D-205037, sheet 1, D-175037, sheet 2, D-205037, sheet 2, D-175037, sheet 3, and D-205037, sheet 3, and for the residual heat removal (RHR) system, in drawings D-175041 and D-205041. In addition to providing input signals for the protection system and the plant control systems, the instrumentation sensors furnish input signals for monitoring and/or alarming purposes for the following parameters:

A. Temperatures.

B. Flows.

C. Pressures.

D. Water levels.

E. Vibration.

In general, these input signals are used for the following purposes:

A. Provide input to the reactor trip system for reactor trips as follows:

1. Overtemperature-ΔT.

2. Overpower-ΔT.

3. Low-pressurizer pressure.

4. High-pressurizer pressure.

5. High-pressurizer water level.

6. Low primary coolant flow.

The following fluid parameter generates an input to the reactor trip system. While not part of the reactor coolant system, it is included here for information. (This is not a complete listing of reactor trip system inputs.)

7. Low-low steam generator level.

B. Provide input to the engineered safety features (ESF) actuation system as follows:

1. High differential pressure between any steam line and the other steam lines.

2. Low steam line pressure.

   Although it is not part of the RCS, the following parameter, which also is sensed to generate an input to the reactor trip system, is included here for purposes of completeness.

3. High steam flow coincident with low-low T_{avg}.

C. Furnish input signals to the nonsafety-related systems, such as the plant control systems and surveillance circuits so that:

1. Reactor coolant average temperature (T_{avg}) will be maintained within prescribed limits. The resistance temperature detector instrumentation is identified on drawings D-175037, sheet 3, and D-205037, sheet 3.

2. Pressurizer level control, using T_{avg} to program the setpoint, will maintain the coolant level within prescribed limits.

3. Pressurizer pressure will be controlled within specified limits.

4. Steam dump control, using T_{avg} control, will accommodate sudden loss of generator load.

5. Information is furnished to the control room operator and at local stations for monitoring.

The following is a functional description of the system instrumentation. Unless otherwise stated, all indicators, recorders, and alarm annunciators are located in the plant control room.

A. Temperature Measuring Instrumentation

1. Mechanical

   The individual loop temperature signals required for input to the reactor control and protection system are obtained using resistance temperature detectors (RTDs) installed in each reactor coolant loop.

   a. Hot Leg

      The hot leg temperature measurement on each loop is accomplished with three fast response, narrow range, dual element RTDs mounted in thermowells. One element of the RTD is considered active, and the other element is held in reserve as a spare. To accomplish the sampling function of the RTD bypass manifold system and to minimize the need for additional hot leg piping penetrations, the thermowells are located within the three existing RTD bypass manifold scoops wherever possible. A hole is machined through the end of each scoop so that water flows in through the existing holes in the leading edge of the scoop, past the RTD, and out through the new hole. Due to physical limitations, several hot leg RTDs are located in independent thermowells near the original scoop locations. These three RTDs measure the hot leg temperature which is used to calculate the reactor coolant loop differential temperature (ΔT) and average temperature (T_{avg}). One wide range RTD element is utilized in each hot leg. These elements, installed in dry thermowells, penetrate the reactor coolant piping and extend into the flow stream. The wide range RTDs provide temperature indication on temperature recorders.

b. Cold Leg

One fast response, narrow range, dual element RTD is located in each cold leg at the discharge of the reactor coolant pump (RCP) (as replacements for the cold leg RTDs located in the bypass manifold). Temperature streaming in the cold leg is minimized by the mixing action of the RCP. The cold leg RTD measures the cold leg temperature which is used to calculate reactor coolant loop ΔT and T_{avg}. The existing cold leg RTD bypass penetration nozzle was modified to accept the RTD thermowell. One element of the RTD is considered active, and the other element is held in reserve as a spare. One wide range RTD element is utilized in each cold leg. These elements, installed in dry thermowells, penetrate the reactor coolant piping and extend into the flow stream. The wide range RTDs provide temperature indication on temperature recorders.

c. Crossover Leg

The RTD bypass manifold return line has been capped at the nozzle on the crossover leg.

2. Electrical

a. Control and Protection System

The hot leg RTD measurements (three per loop) are electronically averaged in the process protection system. The averaged T_{hot} signal is then used with the T_{cold} signal to calculate reactor coolant loop ΔT and T_{avg} which are used in the reactor control and protection systems. This is accomplished by additions to the existing process protection system equipment. The T_{hot} and T_{cold} spare RTD elements are wired to the control rooms and terminated at the 7300 rack input terminals. This arrangement allows online accessibility to the spare elements for RTD cross calibrations and facilitates connection of the spare RTD element in the event of an RTD element failure.

The previous RCS loop temperature measurement system used dedicated direct immersion RTDs for the control systems. This was done largely to satisfy the IEEE Standard 279-1971 which applied single failure criteria to control and protection system interaction. The new thermowell mounted RTDs are used for both control and protection. In order to continue to satisfy the requirements of IEEE Standard 279-1971, the T_{avg} and ΔT signals generated in the protection system are electrically isolated and transmitted to the control system into median signal selectors for T_{avg} and ΔT, which select the signal which is in between the highest and lowest values of the three loop inputs. This precludes an unwarranted control system response that could be caused by a single signal failure.

3. Pressurizer Temperature

There are two temperature detectors in the pressurizer, one located in the vapor or steam space and one located in the water or liquid space. Both detectors supply signals to temperature indicators and high-temperature alarms. The steam space detector, located near the top of the pressurizer, may be used during startup to determine water temperature when the pressurizer is completely filled with water. The steam space temperature is also used as part of an open permissive interlock to prevent the residual heat removal system isolation valves from being opened when the pressurizer steam space temperature is greater than 475°F. The liquid space temperature is used to determine the pressurizer spray differential temperature during heat up and cool down.

4. Surge Line Temperature

This detector supplies a signal for a temperature indicator and a low-temperature alarm. Low temperature is an indication that the continuous spray rate is too small.

5. Safety and Relief Valve Discharge Temperatures

Temperatures in the pressurizer safety and relief valve discharge lines are measured and indicated. An increase in a discharge line temperature is an indication of leakage through the associated valve or the valve being open.

6. Spray Line Temperatures

Temperatures in the spray lines from two loops are measured and indicated. Alarms from these signals are actuated by low spray-water temperature. Alarm conditions indicate insufficient flow in the spray lines.

7. Pressurizer Relief Tank Water Temperature

The temperature of the water in the pressurizer relief tank is indicated, and an alarm actuated by high temperature informs the operator that cooling of the tank contents is required.

8. Reactor Vessel Flange Leakoff Temperature

The temperature in the leakoff line from the reactor vessel flange O-ring seal leakage monitor connections is indicated. An increase in temperature above ambient is an indication of O-ring seal leakage. High temperature actuates an alarm.

9. Reactor Coolant Pump Motor Temperature Instrumentation

a. Thrust Bearing Upper and Lower Shoes Temperature

Resistance temperature detectors are provided, with one located in the shoe of the upper and one in shoe of the lower thrust bearing. These elements provide a signal for a high-temperature alarm and indication.

b. Stator Winding Temperature

The stator windings contain six resistance-type detectors, two per phase, imbedded in the windings. A signal from one of these detectors is monitored by the plant computer, which actuates a high temperature alarm.

c. Upper and Lower Radial Bearing Temperature

Resistance temperature detectors are located one in the upper and one in the lower radial bearings. Signals from these detectors actuate a high-temperature alarm and indication.

B. Flow Indication

1. Reactor Coolant Loop Flow

Flow in each reactor coolant loop is monitored by three differential pressure measurements at a piping elbow tap in each reactor coolant loop. These measurements on a two-out-of-three coincidence circuit provide a low-flow signal to actuate a reactor trip.

C. Pressure Indication

1. Pressurizer Pressure

Pressurizer pressure transmitters provide signals for individual indicators in the control room for actuation of both a low-pressure trip and a high-pressure trip.

One of the signals may be selected by the operator for indication on a pressure recorder.

Three transmitters provide low-pressure signals for safety injection initiation and for safety injection signal unblock during plant startup.

In addition, one transmitter is used, along with a reference pressure signal, to develop a demand signal for a three-mode controller. The lower portion of the controller's output range operates the pressurizer heaters. For normal operation, a small group of heaters is controlled by variable power to maintain the pressurizer operating pressure. If the pressure-error signal falls toward the bottom of the variable heater control range, all pressurizer heaters are turned on. The upper portion of the controller's output range operates the pressurizer spray valves and one power relief valve. The spray valves are proportionally controlled in a range above normal operating pressure with spray flow increasing as pressure rises. If the pressure rises significantly above the proportional range of the spray valves, a power relief valve (interlocked with P-11 to prevent spurious operation) is opened. A further increase in pressure will actuate a high-pressure reactor trip. A separate transmitter (interlocked also with P-11 to prevent spurious operation) provides power relief valve operation for a second valve upon high-pressurizer pressure.

2. Reactor Coolant Reference Pressure (Deadweight Test)

A differential pressure transmitter provides a signal for indication of the difference between the pressurizer pressure and a pressure generated by a deadweight tester located outside the reactor containment. The indication is used for online calibration checks of the pressurizer pressure signals.

3. Reactor Coolant Loop Pressures

Two transmitting channels are provided. Each transmitting channel provides an indication of reactor coolant pressure on one of the hot legs. This is a wide-range transmitter which provides pressure indication over the full operating range. The wide range channel indicators serve as guides to the operator for manual pressurizer heater and spray control and letdown to the chemical and volume control system (CVCS) during plant startup and shutdown. Amplified signals from the lower portion of the range provide improved readability at the lower pressures.

The two wide-range channels provide the permissive signals for the residual heat removal loop suction line isolation valve interlock circuit. In addition, the two channels each provide an input to both trains of the core subcooling monitors.

There are also two local pressure gauges for operator reference during the shutdown condition located in two of the hot loops. These gauges are equipped with auxiliary pointers which remain at the maximum pressure measured until reset locally.

4. Pressurizer Relief Tank Pressure

The pressurizer relief tank pressure transmitter provides a signal to a pressure indicator and an annunciator on the main control board.

5. Reactor Coolant Pump Motor Pressure

a. Oil Lift Switch

A dual-purpose switch is provided on the high-pressure oil lift system. Upon low oil pressure, the switch will actuate an alarm on the main control board. In addition, the switch is part of an interlock system that will prevent starting of the pump until the oil lift pump is started manually prior to starting the reactor coolant pump motor. A local pressure gauge is also provided.

b. Lower Oil Reservoir Liquid Level

A level switch is provided in the motor lower radial bearing oil reservoir. The switch will actuate a high- and low-level alarm on the main control board.

c. Upper Oil Reservoir Liquid Level

A level switch is provided in the motor upper radial bearing and thrust bearing oil reservoir. The switch will actuate a high- or low-level alarm on the main control board.

D. Liquid Level Indication

1. Pressurizer Level

Three pressurizer liquid transmitters provide signals for use in the reactor control and protection system, the emergency core cooling system (ECCS), and the chemical and volume control system. Each transmitter provides an independent high-water-level signal that is used to actuate an alarm and a reactor trip. The transmitters also provide independent low-water-level signals that will activate an alarm. Each transmitter also provides a signal for a level indicator that is located on the main control board.

In addition to the above, signals may be selected for specific functions as follows:

a. Any one of the three level transmitters may be selected by the operator for display on a level recorder located on the main control board. This same recorder is used to display a pressurizer reference liquid level.

b. Two of the three transmitters perform the following functions. (A selector switch allows the third transmitter to replace either of these two.)

    (1) One transmitter provides a signal which will actuate an alarm when the liquid level falls to a fixed level setpoint. The same signal will trip the pressurizer heaters "off" and close the letdown line isolation valves.

    (2) One transmitter supplies a signal to the liquid level controller for charging flow control and also initiation of a low-flow (high-demand) alarm. This signal is also compared to the reference level and actuates a high-level alarm and turns on all pressurizer backup heaters if the actual level exceeds the reference level. If the actual level is lower than the reference level, a low alarm is actuated.

    A fourth independent pressurizer level transmitter that is calibrated for low-temperature conditions provides water level indication during startup, shutdown, and refueling operations.

2. Pressurizer Relief Tank Level

The pressurizer relief tank level transmitter supplies a signal for an indicator and high- and low-level alarms.

E. Vibration Indication

Each of the reactor coolant pump assemblies is equipped for continuous monitoring of reactor coolant pump shaft and frame vibration levels. Shaft vibration is measured by two relative shaft probes mounted on top of the pump seal housing; the probes are located 90° apart in the same horizontal plane and mounted near the pump shaft. Frame vibration is measured by two velocity seismoprobes located 90° apart and mounted at the top of the motor support stand. Proximeters and converters provide output of the probe signals, which are displayed on meters in the electrical penetration room and annunciated in the control room. These meters automatically indicate the highest output from the relative shaft probes and the frame seismoprobes. Manual selection allows monitoring of individual probes. Indicator lights display caution and danger limits of vibration, and are adjustable over the full range of the meter scale.

Process control instrumentation for the residual heat removal system is provided for the following purposes:

A. Furnish input signals for monitoring and/or alarming purposes for:

1. Temperature indications.

2. Pressure indications.

3. Flow indications.

B. Furnish input signals for control purposes of such processes as follows:

1. Control valve in the residual heat removal pump bypass line so that it opens at flows below a preset limit and closes at flows above a preset limit.

2. Residual heat removal inlet valves control circuitry. See section 7.6 for the description of the interlocks and requirements for automatic closure.

3. Control valve in the residual heat removal heat exchanger bypass line to control temperature of reactor coolant returning to reactor coolant loops during plant cooldown.

4. Residual heat removal pump circuitry for starting residual heat removal pumps on "S" signal.