SECTION 5

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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SECTION 5

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

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

Reactor Coolant System design data are listed in Table 5.1-1.

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

The RCS provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values its release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor behavior. During transient operation the system's heat capacity attenuates thermal transients generated by the core or steam generators. The RCS accommodates coolant volume changes within the protection system criteria presented in Section 7.

Reactor Vessel

The reactor vessel (Figure 5.1-1) is cylindrical with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the core, core support structures, control rods, thermal shield, and other parts directly associated with the core. The reactor vessel closure head contains head adaptors. These head adaptors are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adaptors contains acme threads for the assembly of the control rod drive mechanisms. The seal arrangement at the upper end of these adaptors consists of a welded flexible canopy seal.

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

The five (5) reactor vessel closure head Core Exit Thermocouple (CET) column penetrations on the original Salem 1 and 2 reactor vessel closure heads were cut below the threaded canopy seal weld joint and were capped using full penetration butt welded caps. The replacement reactor vessel closure heads on Unit 1 and Unit 2 do not have any CET column penetrations.

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

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

Pressurizer

The pressurizer (Figure 5.1-2) provides a point in the RCS where liquid and vapor can be maintained in equilibrium under saturated conditions for control purposes.

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. Electrical heaters are installed through the bottom head of the vessel while the spray nozzle, relief and safety valve connections are located in the top head of the vessel. The heaters are removable for maintenance or replacement.

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

Pressurizer Relief Tank

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

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

Steam enters the tank through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature. The tank is equipped with an internal spray and a drain which are used to cool the tank

following a discharge. The tank is protected against overpressurization by two rupture discs that discharge into the reactor containment. The tank is carbon steel with a corrosion-resistant coating on the wetted surfaces. A flanged nozzle is provided on the tank for the pressurizer discharge line connection. This nozzle and the discharge piping and sparger within the vessel are austenitic stainless steel.

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. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. Manways are provided for access to both sides of the divided head. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The units are primarily carbon steel. For Unit 1 the heat transfer tubes are Inconel, the primary side of the tube sheets is clad with Inconel, and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. The Unit 2 steam generator heat transfer tubes are Inconel 690 thermally treated and all primary side ferritic steel surfaces (primary side of the tubesheet and inside surfaces of the primary head) are clad with austenitic stainless steel (Type 308L/Type 309L) or Inconel 600 to prevent corrosion. A Unit 2 replacement steam generator (Areva NP Model 61/19T) is shown on Figure 5.1-3, and a Unit 1 replacement steam generator (Model F) is shown in Figure 5.1-3A.

Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage mixed flow pump that employs a controlled leakage seal assembly. The pump is shown on Figure 5.1-4 and net positive suction head characteristics are shown on Figure 5.1-5.

Reactor coolant is drawn up through the primary pump impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The rotor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the

pump in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

Reactor Coolant Piping

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. All smaller piping which comprise part of the RCS boundary, such as the pressurizer surge line, spray and relief lines, loop drains and connecting lines to other systems are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank 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 due to rapid changes in fluid temperature during normal operational transients.

Valves

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

All RCS valves, (except as listed below), which contain radioactive fluid and which normally operate above 212°F, are provided with double-packed stuffing boxes and stem intermediate lantern gland leakoff connections. All throttling control valves, regardless of size, (except as listed below), are provided with double-packed stuffing boxes and with stem leakoff connections. All leakoff connections are piped to the reactor coolant drain tank or pressurizer relief tank.

The valves listed below have a single set packing configuration and inactive leakoff lines.

Salem #1			Salem #2		
1PS1	1PS25	1PR6	2PS1	2PS25	2PR6
1PS3	1PS28	1PR7	2PS3	2PS28	2PR7
1PS24	1PS29	1PS59	2PS24	2PS29	2PS59

Reactor Coolant System High Point Venting

Venting of the RCS during abnormal conditions permits removal of noncondensable gases, thereby aiding natural circulation flow.

There are three principal high points in the RCS: the pressurizer, the reactor vessel head, and the steam generator tube bundle invert.

The pressurizer power-operated relief valve serves as a vent and provides remote venting capability from the control room. This vent is safety grade and meets the single failure criterion.

The high points created by the tube bundle in the steam generator cannot be vented at that location. A Westinghouse study (1) has concluded, however, that only a small amount of noncondensables would be present during any transient which would depend significantly on the steam generators for decay heat removal. It further concluded that the presence of a small amount of noncondensables would not significantly impact natural circulation in the system.

The replacement reactor vessel heads on both units contain a dedicated vent tap adjacent to the center head penetration near the top of the dome. As shown in the RCS flow diagram (Plant Drawings 205201 and 205301), the vessel head can be vented either to the pressurizer relief tank or to the containment.

The Units 1 and 2 replacement reactor vessel head vent piping is a ¾″ diameter Schedule 160 pipe and includes a 3/8″ diameter restricting orifice close to the reactor vessel, in the first horizontal piping run from the center to the periphery of the RVCH. The vent for each unit then runs to the pressurizer relief tank through a redundant grouping of solenoid valves. Break flanges are provided to allow the reactor vessel head removal and to provide room for the manipulator crane movement. The piping is supported to Seismic Category I requirements to ensure that allowable loadings on the part-length CRDM housing are not exceeded. The piping and valves are stainless steel.

The reactor vessel head vent is designed to meet the requirements of NUREG-0737 (2). The vent can be remote-manually actuated from the control room utilizing a key lock switch. The solenoid operated vent valves are powered from two redundant vital dc buses. Open/close indications for the solenoid valves are provided in the control room with both visual and audible alarms. Valve operating logic is shown on Figure 5.1-6C.

Piping, valves, and components for the reactor vessel vent are classified as Seismic Category I and Nuclear Safety Class 2. Design pressure and temperature of the piping valves and components are 2485 psi and 650°F, respectively.

Maximum conditions (pressure and temperature) for the vent piping are as specified in Public Service Electric & Gas' specifications which meet the intent of Standard Review Plan Section 5.2.3 requirements.

The reactor vessel vent system size is kept within the loss-of-coolant-accident (LOCA) definition size (3/8 inch) with a 3/8 inch diameter orifice, which permits venting 1/2 gas volume of the RCS in 1 hour. This minimizes the challenges to the Emergency Core Cooling System (ECCS) since inadvertent vent opening would not require ECCS actuation.

A pipe break in either the pressurizer or reactor vessel head vent lines is an infrequent fault and is covered in Section 15 as a loss of reactor coolant accident resulting from a small bore ruptured pipe. The analysis presented in Section 15 shows that the high head portion of the ECCS together with the accumulators provide sufficient core flooding to keep the calculated peak clad temperature below the required limits of 10CFR50.46. The 3/4-inch reactor head vent line was analyzed not as a source but as a target of High Energy Line Break. The normally pressurized portion of the Pressurizer Vent System is located within the pressurizer missile shield. The effects of internal missiles on

these lines have been analyzed and found acceptable. The head vent and pressurizer vent have been analyzed for the following failures and found not to prevent essential operation of safety-related systems required for safe reactor shutdown or mitigation of the consequences of a design basis accident.

1. Seismic failure of any pressurizer vent components that are not designed to withstand the safe shutdown earthquake.

2. Postulated missiles generated by failure of pressurizer vent components.

3. Dynamic effects associated with the postulated rupture of pressurizer vent piping greater than 1-inch nominal size.

Operability testing of the reactor vessel head vent valves will be performed in accordance with Subsection IWV of Section XI of the ASME Code.

5.1.1 Piping and Instrumentation Diagram

The RCS is shown on Plant Drawings 205201 and 205301, and the system design and operating parameters are given in Table 5.1-1.

5.1.2 Arrangement Drawing

Figure 5.1-12 and Plant Drawings 204803, 204804, 204805, 204806, 204807 and 204808 are plan and elevation drawings providing principal dimensions of the RCS.

5.1.3 References for Section 5.1

1. "Report on Small Break Accidents for Westinghouse Nuclear Steam Supply System," WCAP-9600 (Proprietary) and WCAP-9601 (Nonproprietary), June 1979.

2. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, November 1980.

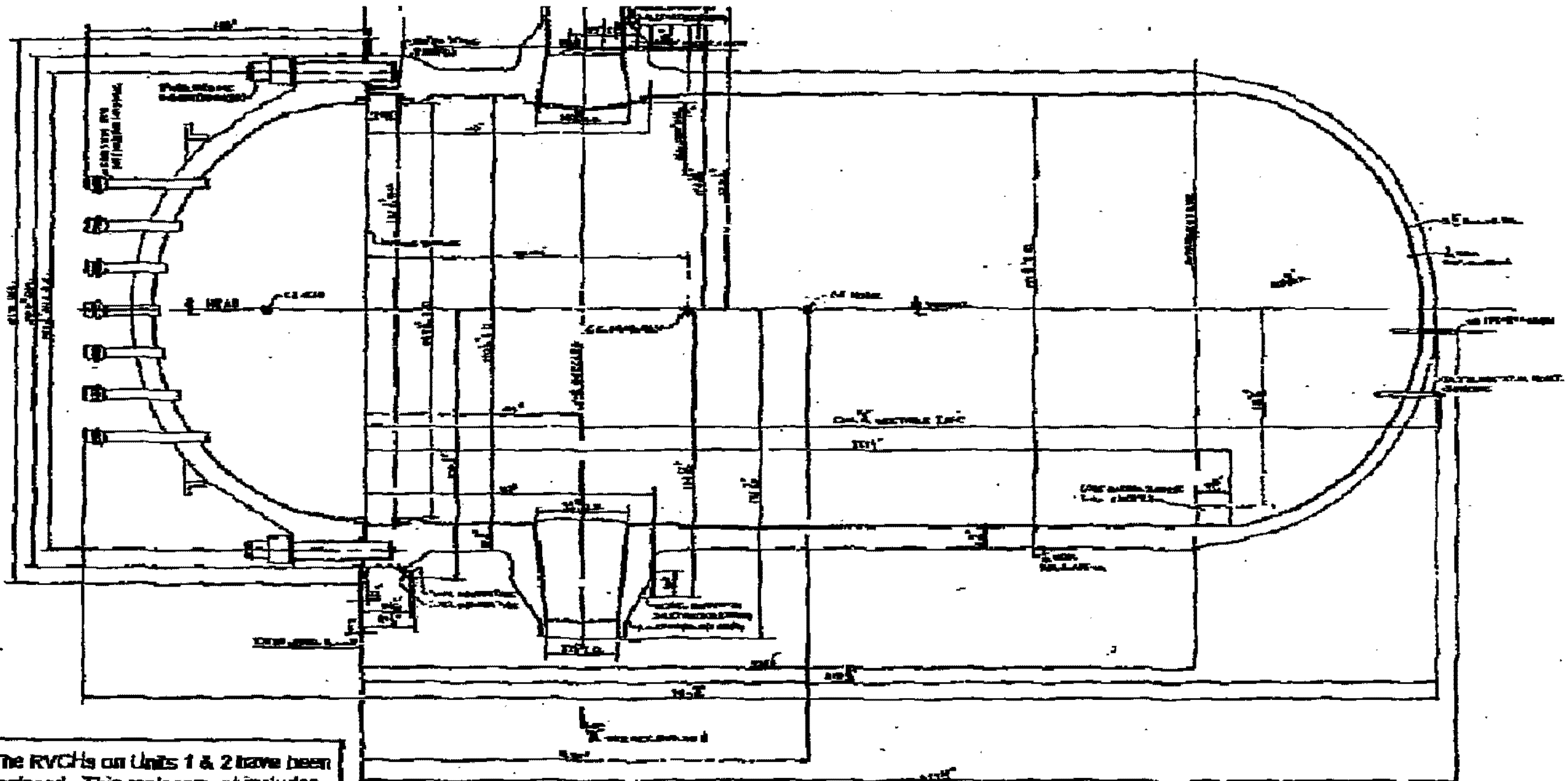
TABLE 5.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS *

Plant design life, years	40
Number of heat transfer loops	4
Design pressure, psig	2485
Nominal operating pressure, psig	2235
Total system volume including pressurizer and surge line (ambient conditions), ft^3	12,071 (Unit 1) 13,011 (Unit 2)
System liquid volume, including pressurizer and surge line (ambient conditions), ft^3	11,351 (Unit 1) 12,291 (Unit 2)
Total heat output (100 percent power), Btu/hr	$11,844 \times 10^6$

Reactor vessel coolant temperature at full power:		High T_{avg}	Low T_{avg}
Inlet, nominal, °F	Unit 1	542.7	530.2
	Unit 2	542.8	530.3
Outlet, °F	Unit 1	613.1	601.8
	Unit 2	613.1	601.7
Coolant temperature rise in vessel at full power, avg, °F	Unit 1	70.4	71.6
	Unit 2	70.3	71.4
Total design coolant flow rate, lb/hr	Unit 1	125.3×10^6	127.3×10^6
	Unit 2	125.8×10^6	127.9×10^6
Design steam pressure at full power, psia (at 0% steam generator tube plugging)	Unit 1	869	778
	Unit 2	900	805

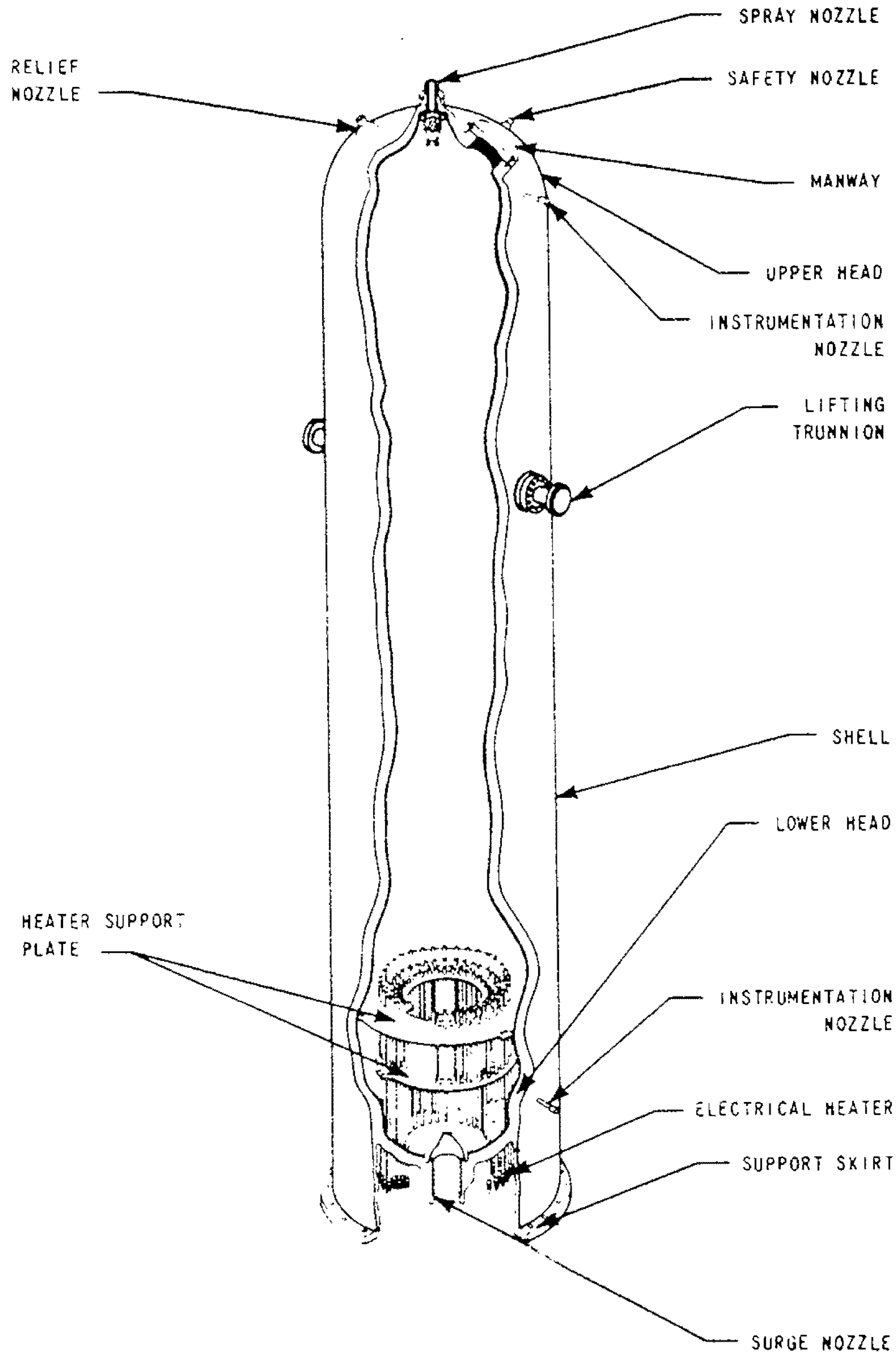
* Values are based on thermal design flow.



The RVCHs on Units 1 & 2 have been replaced. This replacement includes the installation of the IHAs.

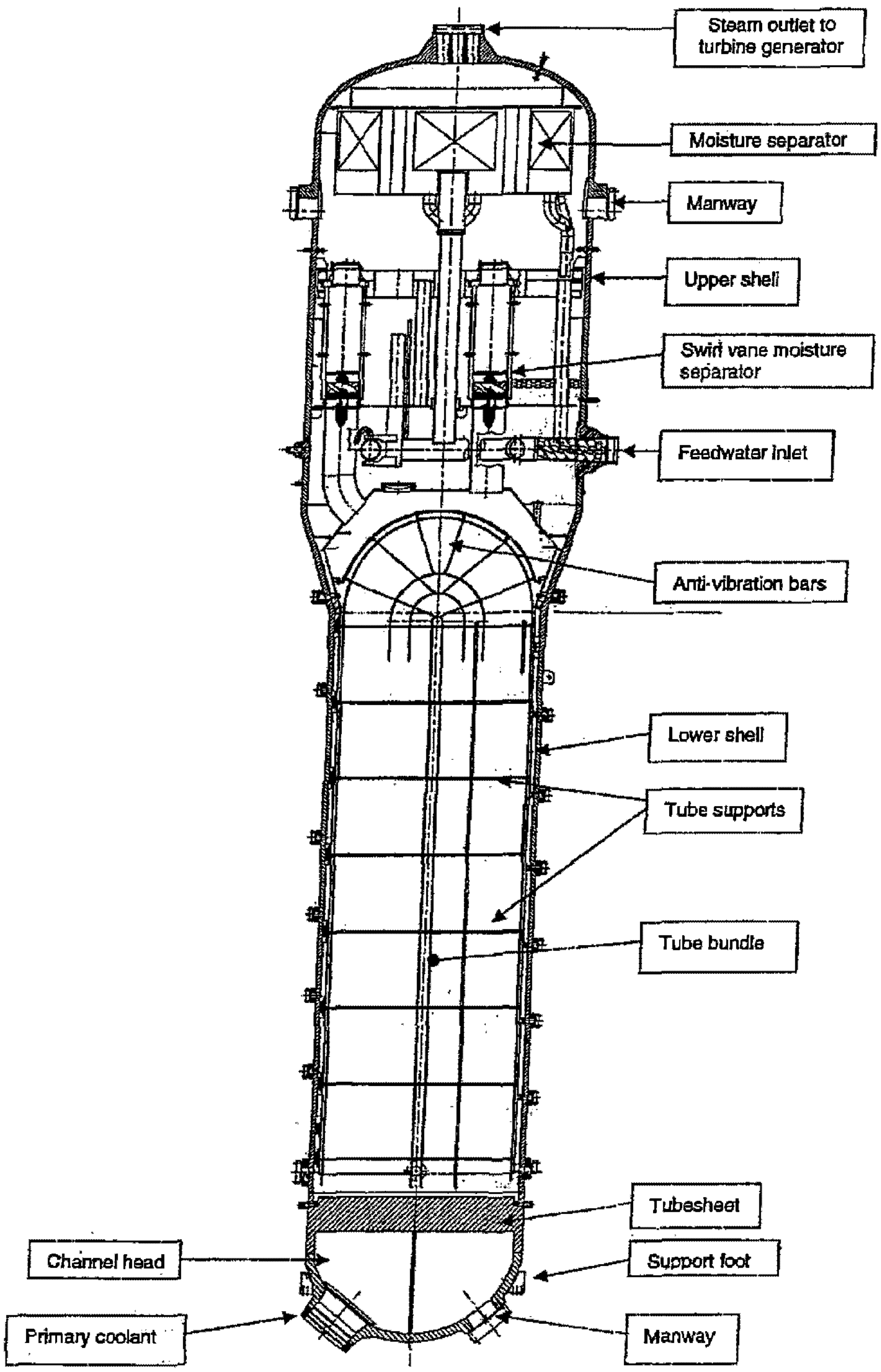
Revision 22, May 5, 2006

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station REACTOR VESSEL SCHEMATIC
	Updated FSAR — Figure 5.1-1



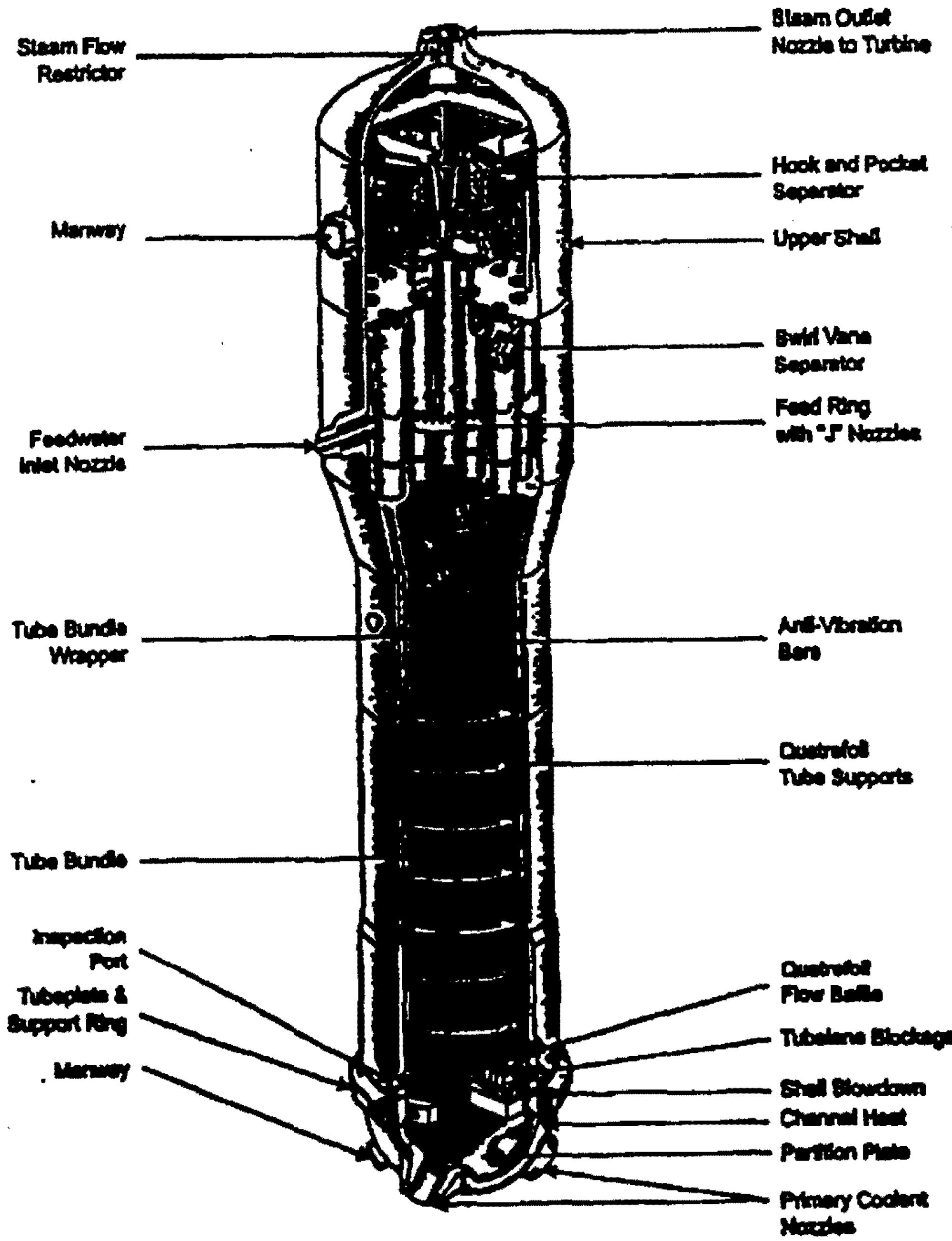
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Cutaway View of Pressurizer	
	Updated FSAR	Figure 5.1-2



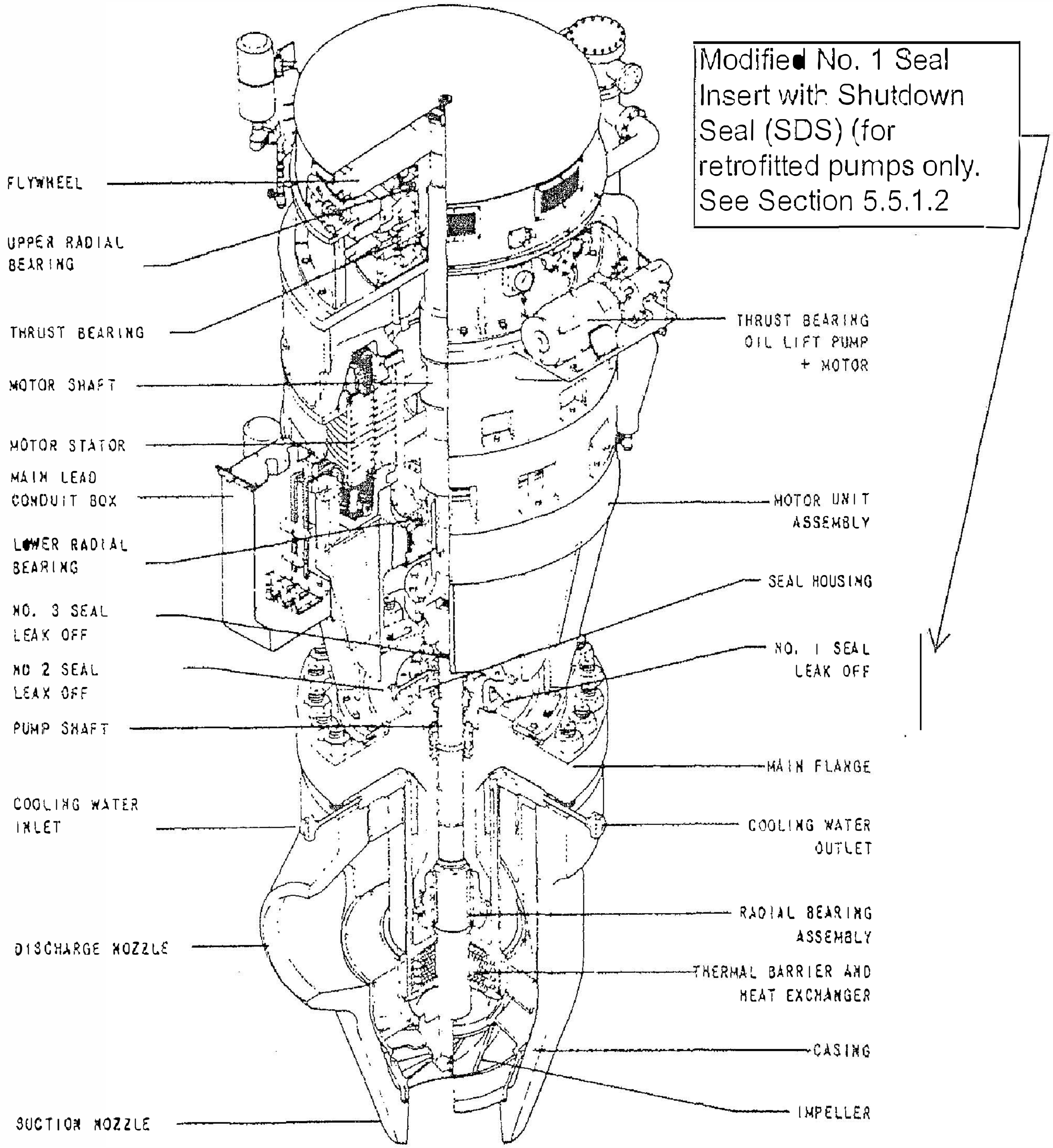
Revision 24
May 11, 2009

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station STEAM GENERATOR UNIT 2 ONLY
	Updated FSAR Figure 5.1-3



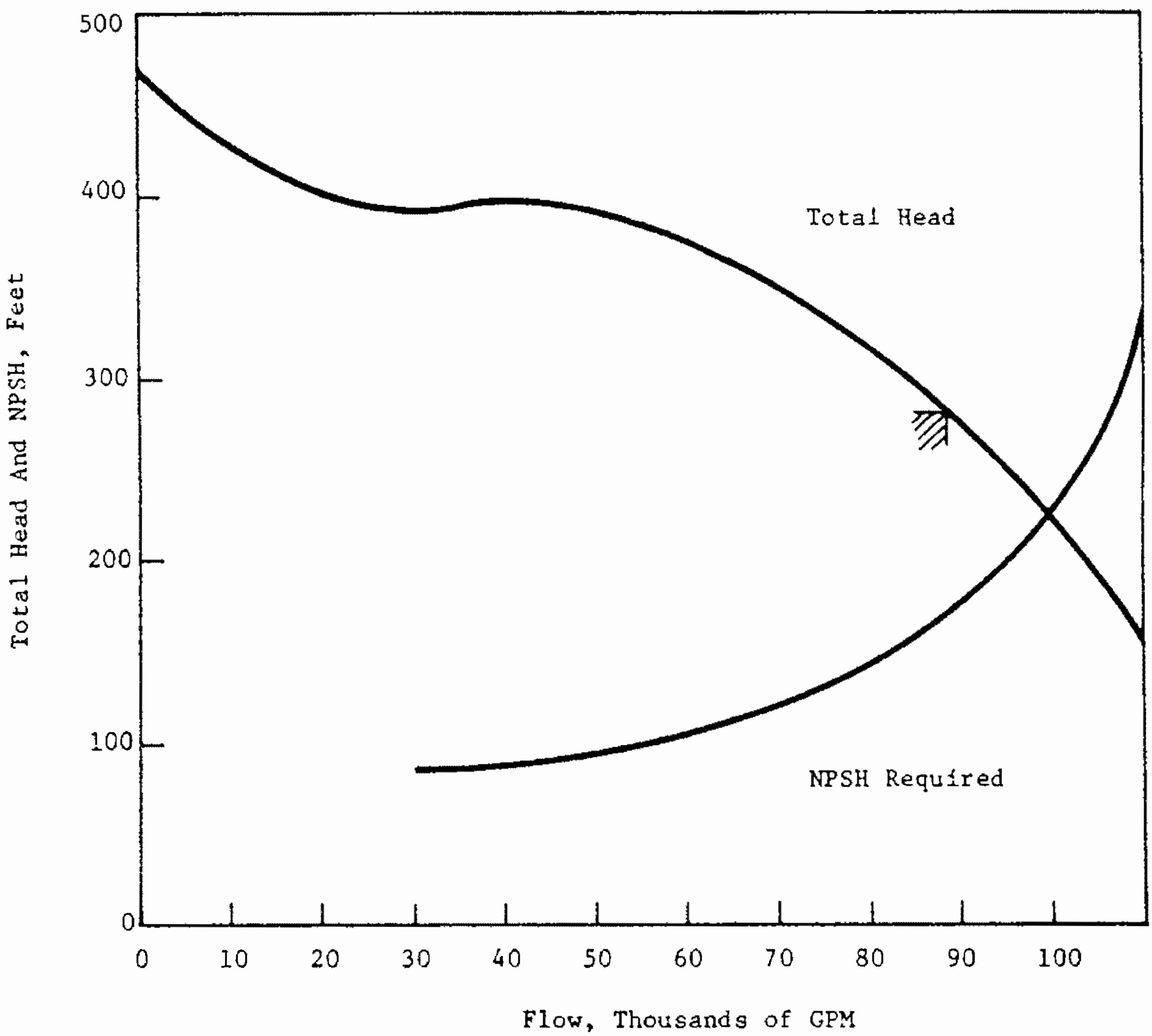
Revision 18, April 26, 2000

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station STEAM GENERATOR (UNIT 1)
	Updated FSAR Figure 5.1-3A



Revision 30, MAY 11, 2018

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CUTAWAY VIEW OF REACTOR COOLANT PUMP
	Updated FSAR — Figure 5.1-4



REVISION 6
FEBRUARY 15, 1987

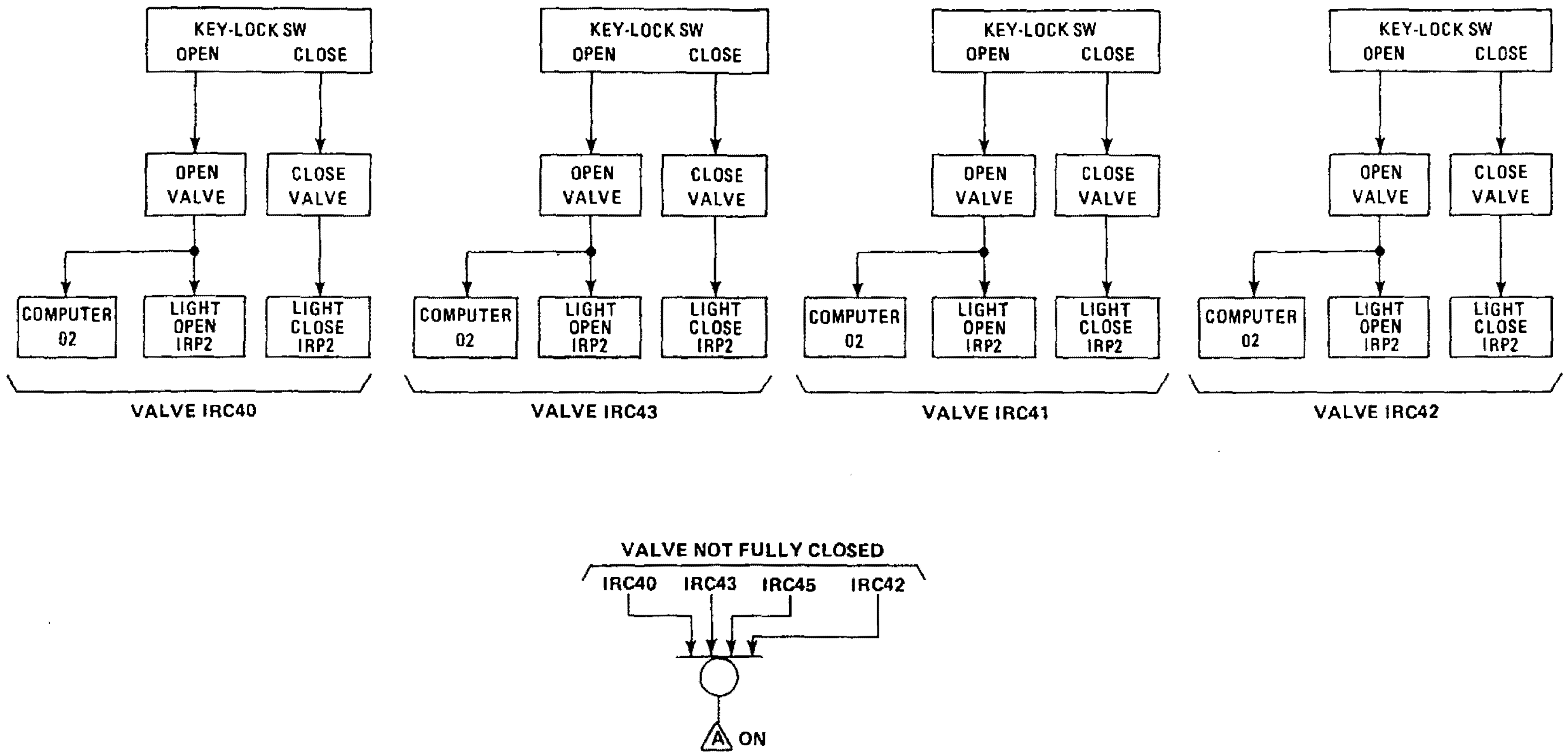
PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Reactor Coolant Pump Performance Characteristics	
	Updated FSAR	Figure 5.1-5

Figure F5.1-6A Sheets 1, 2 & 3 of 3 intentionally deleted.

Refer to plant drawing 205201 in DCRMS

Figure F5.1-6B Sheets 1, 2 & 3 of 3 intentionally deleted.

Refer to plant drawing 205301 in DCRMS



REVISION 6
FEBRUARY 15, 1987

Ref. Dwg. N/A

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	LOGIC DIAGRAM, HEAD VENT SYSTEM UNIT 1 & 2
	Updated FSAR Fig 5.1-6C

Figure F5.1-7 intentionally deleted.

Refer to plant drawing 204803 in DCRMS

Figure F5.1-8 intentionally deleted.

Refer to plant drawing 204804 in DCRMS

Figure F5.1-9 intentionally deleted.

Refer to plant drawing 204805 in DCRMS

Figure F5.1-10 intentionally deleted.

Refer to plant drawing 204806 in DCRMS

Figure F5.1-11 intentionally deleted.

Refer to plant drawing 204807 in DCRMS

SECURITY-RELATED
INFORMATION-WITHHELD
UNDER 10 CFR 2.390

REVISION 6, FEBUARY 15, 1987

PSEG NUCLEAR, L.L.C. SALEM NUCLEAR GENERATING STATION	AUXILIARY BUILDING AND REACTOR CONTAINMENT ELEVATION
	Updated FSAR Figure 5.1-12

Figure F5.1-13 intentionally deleted.

Refer to plant drawing 204808 in DCRMS

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

The Reactor Coolant System (RCS) is of primary importance with respect to its safety function in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication, and inspection conform to the applicable provisions of recognized codes and good nuclear practice.

The materials of construction of the pressure retaining boundary of the RCS are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions so that continued safe operation is possible.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

This section discusses the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB).

5.2.1 Design of Reactor Coolant Pressure Boundary

5.2.1.1 Performance Objectives

The RCS transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Demineralized light water is circulated at the flow rate 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 RCS provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values its release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor behavior. During transient operation the system's heat capacity attenuates thermal transients generated by the core or steam generators. The RCS accommodates coolant volume changes within the Protection System criteria.

The inertia of the reactor coolant pumps reduces the thermal-hydraulic effects to a safe level during the pump coastdown, which would result from a loss-of-flow situation. The layout of the system assures the natural circulation capability following a loss-of-flow to permit decay heat removal without overheating the core. Part of the system piping serves as part of the Emergency Core Cooling System (ECCS) to deliver cooling water to the core during a loss-of-coolant accident (LOCA).

5.2.1.2 Design Parameters

Design Pressure

The RCS design and operating pressure together with the safety, power relief and pressurizer spray valves set points, and the Protection System set point pressures are listed in Table 5.2-1. The selected design margin includes operating transient pressure changes from core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. Table 5.2-2 gives the design pressure drop of the RCS components.

Design Temperature

The design temperature for each component was selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. The design and operating temperatures of the respective system components are listed in Tables 5.2-3 through 5.2-8.

Seismic Loads

The seismic loading conditions were established by the operational basis earthquake (OBE) and design basis earthquake (DBE). The former was selected to be typical of the largest probable ground motion based on the site seismic history. The latter was selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties.

For the OBE loading condition, the Nuclear Steam Supply System (NSSS) is designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment needed for this purpose are required to remain operable. The seismic design for the DBE is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the RCS components do not lose their capability to perform their safety function. This has come to be referred to as the "no-loss-of-function" criteria and the loading condition as the "design basis earthquake" loading condition.

The criteria adopted for allowable stresses and stress intensities in vessels and piping subjected to normal loads plus seismic loads are defined in Section 5.2.1.8.

Design and construction practices in accordance with these criteria assure the integrity of the RCS under seismic loading. The combination of seismic loads with operating and pipe rupture

loads for the design of the RCS support structures and their respective allowable stresses are given in Table 5.5-3.

5.2.1.3 Compliance with 10CFR50.55a

All pressure-containing components of the RCS were designed, fabricated, inspected, and tested in conformance with the applicable codes listed in Table 5.2-9.

The RCS is classified as Class I for seismic design, requiring that there will be no loss-of-function of such equipment in the event of the assumed DBE ground acceleration acting in the horizontal and vertical directions simultaneously, when combined with the RCS steady-state stresses.

5.2.1.4 Applicable Code Cases

Specific Code Cases used prior to original plant startup may be identified in various system descriptions. Code Cases applied in the RCS design are listed in Table 5.2-9. ASME Code Cases which have been used subsequent to original plant startup are associated with the following four areas: 1) Inservice Inspections, 2) Inservice Testing, 3)Repair and Replacement activities, and 4) the initiation or revision of component design specifications resulting from plant modifications. Specific Code Case usage is identified in the ISI Program, the IST Program, the NBU Repair Program, or the individual design specifications of components affected by plant modifications. New Code Cases must be reviewed for acceptability against the current revision of Regulatory Guides 1.84, 1.85, or 1.147; as applicable.

5.2.1.5 Design Transients

The RCS and its components are designed to accommodate 10-percent of full power step changes in plant load and 5-percent of full power per minute ramp changes over the range from 15-percent full power up to and including but not exceeding 100-percent of full power without reactor trip. The RCS can accept a complete loss-of-load from full power with reactor trip. In addition, the steam dump system makes it possible to accept a 50-percent loss of external load from full power without reactor trip.

All components in the RCS are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal power changes, reactor trip, and startup and shutdown operations. The number of thermal and loading cycles used for design purposes and their bases are given in Table 5.2-10. During unit startup and

shutdown, the rates of temperature and pressure changes are limited as indicated below.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation were based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients, and accident conditions. To a large extent, the specific transient operating conditions were considered for equipment; fatigue analyses were based upon engineering judgment and experience. Those transients were chosen which were representative of transients to be expected during plant operation, sufficiently severe or frequent enough to be of possible significance to component cyclic behavior.

Clearly it is difficult to discuss in absolute terms the transients that the plant will actually experience during the 40-year operating life. For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients.

5.2.1.5.1 Heatup and Cooldown

The normal heatup or cooldown cases are conservatively represented by a continuous operation performed at a uniform temperature rate of 100°F per hour.

For these cases, 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 per hour may not be attainable because of other limitations such as:

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

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

The number of such complete heatup and cooldown operations is specified at 200 times each, which corresponds to five such occurrences per year for the 40-year plant design life. For the ideal plant, only one heatup and one cooldown would occur per 100-percent full power year (i.e., the period between refueling). In practice, experience to date indicates that during the first year or so of operation, additional unscheduled plant cooldowns may be necessary for plant maintenance; the frequency of maintenance shutdowns reduce as the plant matures. As experience was gained with Yankee-Rowe, the number of shutdowns decreased; for example Core II ran for a year from 1962 to 1963 with no cooldowns. Table 5.2-11 is a summary of the Yankee-Rowe plant outage for the period 1964 to 1969.

5.2.1.5.2 Unit Loading and Unloading

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5 percent per minute between 15-percent load and full load. This load swing is the maximum possible consistent with operation with automatic reactor control. The reactor coolant temperature will vary with load as prescribed by the Temperature Control System. The number of each operation is specified at 18,300 times or one time per day with approximately 40-percent margin for plants with a 40-year design life.

5.2.1.5.3 Step Increase and Decrease of 10 Percent

The ± 10 percent step change in load demand is a control transient which is assumed to be a change in turbine control valve opening which might be occasioned by disturbances in the electrical network into which the plant output is tied. The RCS is designed to restore plant equilibrium without reactor trip following a ± 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 RCS attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and 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 RCS automatically inserts the control rods to reduce core power. With load decrease, the reactor coolant temperature will be 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 turbine steamline inlet pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow 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 RCS 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 will be raised to a value above its initial equilibrium value at the beginning of the transient. The number of each operation is specified at 2000 times or 50 per year for the 40-year plant design life.

5.2.1.5.4 50-Percent Step Decrease in Load

This transient applies to a 50-percent step decrease in turbine load of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side Steam Dump System that will prevent a reactor shutdown or lifting of steam generator safety valves. If a Steam Dump System was not provided to cope with this transient, there would be such a strong mismatch between what the turbine is asking for and what the reactor is furnishing, that a reactor trip and lifting of steam generator safety valves would occur.

The number of occurrences of this transient is specified at 200 times or five per year for the 40-year plant design life. Reference to the Yankee-Rowe record indicates that this basis is adequately conservative.

5.2.1.5.5 Loss of Load

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. In this assumed case, the reactor and turbine eventually trip as a consequence of a high pressurizer level trip initiated by the Reactor Protection System (RPS).

The number of occurrences of this transient is specified at 80 times or two per year for the 40-year plant design life. Since redundant means of tripping the reactor upon turbine trip are provided as part of the RPS, transients of this nature are not expected.

5.2.1.5.6 Loss of Power

This transient applies to a blackout situation involving the loss of outside electrical power to the station and a reactor and turbine trip, on low reactor coolant flow, culminating in a complete loss of plant electrical power. Under these circumstances, the reactor coolant pumps are de-energized 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 Feedwater (AFW) System operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric pilot-operated relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times or one per year for the 40-year plant design life.

5.2.1.5.7 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 that pump. The consequences of such an accident at a high power level 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 sizeable reduction in the hot leg coolant temperature of the affected loop.

The number of occurrences of this transient is specified at 80 times or two per year for the 40-year plant design life.

5.2.1.5.8 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 RPS causes the control rods to drop into the core.

The number of occurrences of this transient is specified at 400 times or ten per year for the 40-year plant design life.

5.2.1.5.9 Turbine Roll Test

This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power was used to heat the reactor coolant to operating temperature and the steam generated used to perform a turbine roll

test. However, the plant cooldown during this test exceeded the 100°F per hour maximum rate specified in Section 5.2.1.5.1.

The number of such test cycles is specified at ten times to be performed at the beginning of plant operating life prior to irradiation.

5.2.1.5.10 Hydrostatic Test Conditions

The pressure tests are outlined below.

<u>Primary Side Hydrostatic Test Before Initial Startup</u>

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 hydrostatic test was performed at a water temperature which was compatible with reactor vessel material ductility transition temperature (DTT) requirements and a minimum test pressure of 3107 psig. In this test, the primary side of the steam generator was pressurized to 3107 psig coincident with the secondary side pressure of 0 psig. The RCS is designed for five cycles of this hydrostatic test, except that the Unit 2 RSGs are designed for ten cycles of this hydrostatic test.

<u>Secondary Side Hydrostatic Test Before Initial Startup</u>

The secondary side of the steam generator was pressurized to 1356 psig (1482 psig - Unit 2) with a minimum water temperature of 70°F coincident with the primary side at 0 psig.

The steam generator may experience five cycles (ten cycles - Unit 2) of this test.

5.2.1.5.11 Primary Side Leak Test

This type of test is performed to test the integrity of the RCS after a maintenance procedure has been completed in which the RCS boundary has been opened. To account for the shift in DTT on the

reactor vessel due to irradiation effects later in life, this leak test is analyzed at a minimum water temperature above NDT and an assumed system pressure of 2485 psig. The design heatup rate is limited to 100°F per hour. Since pumping power is used to heat the water, the actual heatup rate is considerably below 100°F per hour. The number of these tests is specified at 50 for the 40-year plant design life. The normal requirement is that which follows a refueling operation.

5.2.1.5.12 Pressurizer Surge and Spray Line Connections

The surge and spray nozzle connections at the pressurizer vessel are subject to cyclic temperature changes resulting from the transient conditions described previously. The various transients are characterized by variations in reactor coolant temperature which in turn result in water surges into or out of the pressurizer. The surges manifest themselves as changes in system pressure, which, depending upon whether an increase or decrease in pressure occurs, result in introducing spray water into the pressurizer to reduce pressure or actuating the pressurizer heaters to increase pressure to the equilibrium value. To illustrate a load change cycle as it affects the pressurizer, consider a design step increase in load. The pressurizer initially experiences an outsurge with a drop in system pressure which actuates the pressurizer heaters to restore system pressure. As the RCS reacts, the reactor coolant temperature is increased which causes an insurge into the pressurizer raising system pressure. As pressure is increased, the heaters go off and at some pressure setpoint, the spray valves open to limit the pressure rise and restore system pressure. Thus the pressurizer surge nozzle is subjected to a temperature increase on the outsurge followed by a temperature decrease on the insurge during this load transient. The pressurizer spray nozzle is subjected to a temperature decrease when the spray valve opens to admit reactor coolant cold leg water into the pressurizer. The pressurizer experiences a reverse situation during a load decrease transient,

i.e., an insurge followed by an outsurge. It is assumed that the spray valve opens to admit spray water into the pressurizer once at the design flowrate for each design step change in plant load. Thus the number of occurrences for the spray nozzle corresponds to that shown for the other components listed in Table 5.2-10.

During plant cooldown, spray water is introduced into the pressurizer to cool down the pressurizer. The maximum pressurizer cooldown rate is specified at 200°F per hour which is twice the rate specified for the other RCS components.

5.2.1.5.13 Accident Conditions

The effect of the accident loading was evaluated in combination with normal loads to demonstrate the adequacy to meet the stated plant safety criteria.

A brief description of each accident transient that was considered follows. In each case, one occurrence was evaluated.

Reactor Coolant Pipe Break

This accident involves the rupture of a RCS pipe resulting in a loss of primary coolant. It is conservatively assumed that the system pressure and temperature are reduced rapidly and the Safety Injection System (SIS) is initiated to introduce 70°F water into the RCS. The safety injection signal also results in a turbine and reactor trip. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal is still at no-load temperature conditions when the 70°F safety injection water is introduced into the system.

Steam Line Break

For component evaluation, the following conservative conditions were considered:

1. The reactor is initially in a hot, zero-load, just critical 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. Subsequent to the break, there is no return to power and the reactor coolant temperature cools down to 212°F.

4. The ECCS pumps restore the reactor coolant pressure to 2500 psia.

The above conditions result in the most severe temperature and pressure variations that the component will encounter during a steam break accident. Both Areva NP Model 61/19T (Unit 2) and Model F (Unit 1) are qualified for the given conditions.

Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will typically occur due to a safety injection signal on low pressurizer pressure. When the accident occurs, some of the reactor coolant blows down into the affected steam generator causing the level to rise. If the level rises sufficiently, a high level alarm will occur and the feedwater regulating valve will close. After the ruptured steam generator is identified, the planned procedure

to recovery from this accident calls for isolation of the steam line leading from the affected steam generator with the subsequent cooldown and depressurization of the RCS below the faulted steam generator pressure. This accident will result in a transient which is no more severe than that associated with a reactor trip. For this reason, it requires no special treatment in so far as fatigue evaluation is concerned.

5.2.1.6 Protection Against Environmental Factors

Essential equipment has either been designed to withstand a credible tornado including a single large missile generated thereby, or has been placed in a structure which will withstand the tornado and missile. Where sufficient redundancy exists, equipment may be physically separated without protection against tornado missiles.

Engineered safety features are protected against dynamic effects and missiles resulting from equipment failures. The means for accomplishing this protection are described in Section 3.5.

5.2.1.7 Protection Against Proliferation of Dynamic Effects

5.2.1.7.1 Criteria

Protection, in the form of barriers, restraints, supports, and physical separation has been provided to assure that in the unlikely event of an accident the following criteria will be met:

1. Containment integrity will be protected throughout the accident.

2. A second accident will not occur as a result of the original accident.

3. For a steam system rupture, no more than one steam generator will blow down.

For the purpose of the above criteria, an accident is defined as the rupture of a pipe in any one of the following systems:

1. Reactor Coolant System (LOCA), as limited by the NRC's approval of Leak-Before-Break (Reference 27 in Section 5.2.9).

2. Main Steam System, from each steam generator up to and including the main steam stop outside the containment.

3. Feedwater System, from each steam generator up to and including the non-return valve outside the containment.

5.2.1.7.2 Dynamic Effects

Protection has been provided against the following effects:

1. Jet forces resulting from the release of high pressure steam or water from a ruptured line.

2. Pipe whip caused by the formation of a plastic hinge in a pipe due to a rupture somewhere else in the same pipe.

3. Missiles which can be generated in coincidence with an accident.

5.2.1.7.3 Barriers

The polar crane wall serves as a barrier between the reactor coolant loops and the containment liner. In addition, the refueling cavity walls, various structural beams, the operating floor, and the crane wall provide some separation of the reactor coolant loops, thereby minimizing the effects of an accident occurring in any one loop on another loop or the containment.

The Class I portion of the steam and feedwater lines from each steam generator have been routed behind barriers which separate

these lines from the steam and feedwater lines from the other steam generators, as well as from the reactor coolant piping.

The barriers described above will withstand loadings caused by jet forces, pipe whip impact forces, or the generation of all credible missiles coincident with an accident.

All equipment inside the containment, required for safe shutdown in the event of an accident, is located between the crane wall and the containment wall and is thereby protected from all dynamic effects of an accident occurring within the loop compartment.

5.2.1.7.4 Restraints

All lines connected to the reactor coolant loop, which penetrate the containment wall are anchored to the crane wall. Each anchor is designed to be stronger than the pipe. Should a reactor coolant loop rupture occur, the resulting jet force will therefore not be transferred through to the containment wall through any branch lines.

Main steam and feedwater lines are anchored outside the containment so that a rupture anywhere in the line will not affect containment integrity. These lines are also restrained inside the containment to prevent whipping and to maintain containment integrity.

5.2.1.7.5 Supports

Major components of the RCS (reactor vessel, steam generators, and pumps) are supported to isolate the effects of an initial rupture so that a second accident cannot occur.

5.2.1.7.6 Physical Separation

Physical separation is accomplished primarily by placing redundant essential equipment on either side of a barrier so that one, but not both items, may be vulnerable to missiles, jet forces, and pipe whip.

Safeguard lines serving the RCS are routed so that main headers are located outside the crane wall and are not vulnerable to any dynamic effects. Branch lines serving an individual loop penetrate the crane wall as close to the loop as possible. In this manner, branch lines serving unaffected loops will not be damaged by the loop in which the accident may have occurred.

5.2.1.8 <u>Design Criteria for Vessels and Piping</u>

5.2.1.8.1 Load Combinations and Stress Criteria

This section deals with the loads imposed on RCS components and supports during normal conditions as well as during seismic events and pipe rupture. Stress criteria are presented as a function of the various load combinations. Two types of seismic loading are considered: OBE and DBE.

For the OBE loading condition, the NSSS is designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment needed for this purpose are required to operate within design limits. The seismic design for the DBE is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that required critical structures and components do not lose their capability to perform their safety function. This has come to be referred to as the "no-loss-of-function" criteria and the loading condition as the "design basis earthquake" loading condition.

Not all critical components have the same functional requirements for safety. For example, the reactor containment must retain capability to restrict leakage to an acceptable level. Therefore, general elastic behavior of this structure under the "design basis earthquake" loading condition was ensured. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain their contents and allow fluid flow.

The normal, as well as abnormal loads are considered singly and in combination (see Table 5.2-12), and the allowable stress limits for each of the possible combinations are limited to those specified in Table 5.2-13. The Unit 2 RSG allowable stresses for faulted conditions are provided by the ASME code. For other NSSS components, the design limit curves that give the allowable stresses for faulted conditions were developed by using the approach presented in Reference 1. This report develops limit curves by using 50 percent of the ultimate strain as the maximum allowable membrane strain. Subsequent to the submission of Reference 1, the allowable membrane strain was limited to 20 percent of the uniform strain. Design limit curves were developed by using the following procedure:

1. Use material data to develop stress-strain curves.

   Stress-strain curves of Type 304 stainless steel, Inconel 600, and SA302b low alloy steel at 600°F have been generated from tests using graphs of applied load versus cross-head displacement as automatically plotted by the recorder of the tensile test apparatus. The scale and sensitivity of the test apparatus recorder assure accurate measurement of the uniform strain.

   For other materials, stress-strain curves are developed by conservative use of pertinent available material data (i.e., lowest values of uniform strain and initial strain hardening). Should the available data not be

sufficient to develop a reliable stress-strain curve, three standard American Society for Testing and Materials' (ASTM) tensile tests of the material in question will be performed at design temperature. These data could conservatively apply in developing a stress-strain curve as described above.

2. Normalize the ordinate (stress) of the stress-strain curves to the measured yield strength (Figure 5.2-1).

3. Use 20 percent of the uniform strain as defined on the curve developed under Item 1 as the allowable membrane strain.

4. Establish the normalized stress ratio at 20 percent of uniform strain on the normalized stress ratio-strain curves developed under Item 2.

5. Establish the value of membrane stress limit.

   Multiply the normalized stress ratio in Item 4 by the applicable code yield strength at the design temperature to get the membrane stress limit. As an alternate, the actual physical properties as determined for standard ASTM tensile tests on specimens from the same heats may be used to determine the membrane stress limit. If such an approach is adopted, sufficient documentation will be provided to support the actual material properties used.

6. Develop limit curves for the combination of local membrane and bending stresses.

   The limit curves are developed by using the analytical approach presented in Reference 1 and the stress-strain curve up to the membrane stress limit as developed under

Item 5. Stress and stability analysis results are to be compared with these limits.

Examples of design limit curves are developed by using the above procedure and are given on Figures 5.2-2 and 5.2-3.

5.2.1.8.2 Stress Analysis for Structural Adequacy

Reactor Vessel

The following components of the reactor pressure vessel were analyzed in detail through systematic analytical procedures:

1. Control rod housings

2. Closure head flange and shell

3. Main closure studs

4. Inlet nozzle (and vessel support)

5. Outlet nozzle (and vessel support)

6. Vessel wall transition

7. Core-barrel support pads

8. Bottom head to shell juncture

9. Bottom head instrument penetrations, etc.

   a. An interaction analysis was performed on the control rod drive mechanism (CRDM) housing. The flange was assumed to be a ring and the tube a long cylinder. The different values of Young's Modulus

and coefficients of thermal expansion of the tubes were taken into account in the analysis. Local flexibility was considered at appropriate locations. The closure headway was treated as a perforated spherical shell with modified elastic constants. The effects of redundants on the closure head were assumed to be local only. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation was made for the J-weld.

b. The closure head, closure head flange, vessel flange, vessel shell, and closure studs were all evaluated in the same analysis. An analytical model was developed by dividing the actual structure into different elements such as sphere, ring, long cylinder, and cantilever beam, etc. An interaction analysis was performed to determine the stresses due to mechanical and thermal loads. These stresses were evaluated in light of the strength and fatigue requirements of the ASME Boiler and Pressure Vessel Code, Section III.

c. An analysis similar to Item b was performed for the vessel flange to vessel shell juncture and main closure studs.

d. For the analysis of nozzle and nozzle-to-shell juncture, the loads considered were internal pressure, operating transients, thermally induced and seismic pipe reactions, static weight of vessel, earthquake loading and expansion and contraction, etc. A combination of methods was used to evaluate the stresses due to mechanical and thermal loads and external loads resulting from

seismic pipe reactions, earthquake, pipe break, etc.

For fatigue evaluation, peak stresses resulting from external loads and thermal transients were determined by concentrating the stresses as calculated by the above-described methods. Combining these stresses enables the fatigue evaluation to be performed.

e. The method of analysis for outlet nozzle and vessel supports was the same as described above for Item d.

f. Vessel wall transition was analyzed by means of a standard interaction analysis. The thermal stresses were determined by the skin stress method where it was assumed that the inside surface of the vessel is at the same temperature as the reactor coolant and the mean temperature of the shell remains at the steady state temperature. This method is considered conservative.

g. Thermal, mechanical, and pressure stresses were calculated at various locations on the pad and at the vessel wall. Mechanical stresses were calculated by the flexure formula for bending stress in a beam. Pressure stresses were taken from the analysis of the vessel to bottom head juncture and thermal stresses were determined by the conservative method of skin stresses. The stresses due to the cyclic loads were multiplied by a stress concentration factor where applicable and used in the fatigue evaluation.

h. Standard interaction analysis and skin stress methods were employed to evaluate the stresses due to mechanical and thermal stresses, respectively. The fatigue evaluation was made on cumulative basis where superposition of all transients was taken into consideration.

i. An interaction analysis was performed by dividing the actual structure into an analytical model composed of different structural elements. The effects of the redundants on the bottom head were assumed to be local only. It was also assumed that for any condition where there is interference between the tube and the head, no bendings at the weld can exist. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation was made for the J-weld.

The location and geometry of the areas of discontinuity and/or stress concentration are shown on Figures 5.2-4, 5.2-5, and 5.2-6.

A summary of the estimated primary plus secondary stress intensity for components of the reactor vessel and the estimated cumulative fatigue usage factors for the components of the reactor vessel is given in Tables 5.2-14 and 5.2-15.

The cycles specified for the fatigue analysis are the results of an evaluation of the expected plant operation coupled with experience from nuclear power plants then in service, such as Yankee-Rowe.

The conservatism of the design fatigue curves used in the fatigue analysis has been demonstrated by the Pressure Vessel Research Committee (PVRC) in a series of cyclic pressurization tests of model vessels fabricated to the Code. The results of the PVRC tests showed that no crack initiation was detected at any stress

level below the code allowable fatigue curve and that no crack progressed through a vessel wall in less than three times the allowable number of cycles. Similarly, fatigue tests have been performed on irradiated pressure vessel steels with comparable results (2).

The vessel design pressure is 2485 psig while the normal operating pressure will be 2235 psig. The resulting operating membrane stress is therefore amply below the code allowable membrane stress to account for operating pressure transients.

The stress allowed in the vessel in relation to operation below NDT temperature and DTT (NDT temperature plus 60°F) to preclude the possibility of brittle failure are:

1. At DTT, a maximum stress of 20 percent yield

2. From DTT to DTT minus 200°F, a maximum stress decreasing from 20 to 10 percent yield

3. Below DTT minus 200°F, a maximum stress of 10 percent yield

These limits are based on a conservative interpretation of the Fracture Analysis Diagram developed at the Naval Research Laboratory (3, 4, 5) after many years of research and confined by extensive correlations with service failures. There have been no known service failures under conditions permitted by these limits. The Fracture Analysis Diagram is the most widely known and generally accepted criterion for brittle fracture prevention and includes linear elastic fracture mechanics concepts. The limits established by the Fracture Analysis Diagram have been correlated with linear elastic fracture mechanics insofar as possible (6) and are conservative in providing protection against brittle fractures. The stress limits are maintained by operating

procedures which prescribe pressure and temperature control limits during heatup and cooldown (7, 31, 32).

The actual shift in NDT temperature is established periodically during plant life by testing of vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for any increase in the NDT temperature caused by irradiation, the pressure and temperature limits are periodically changed to stay within the stress limits.

The vessel closure contains 54 7-inch studs. The stud material is ASTM A-540 which has a minimum yield strength of 104,400 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is less than half this value. This means that about half of the 54 studs have the capability of withstanding the hydrostatic end load on vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

In establishing the initial temperature-pressure limits, emphasis is placed on heatup and cooldown because the normal operating temperature always exceeds even the highest anticipated DTT during the life of the plant. Conservatism is emphasized during heatup and cooldown because long-term irradiation of the vessel raises the DTT and thereby limits the heatup or cooldown rates. The following conservative limits are applied:

1. Use of a stress concentration factor of four on assumed flaws in calculating the stress.

2. Use of nominal yield of material instead of actual yield.

3. Neglecting the increase in yield strength resulting from radiation effects.

The factor of four is not an actual stress concentration factor such as is described in Article 4, Design, of Section III, but is a margin of conservatism based on the Fracture Analysis Diagram in ASTM E-208 as well as the stress limits maintained by the prescribed operating procedures which rely upon administrative pressure and temperature control during heatup and cooldown (6). At the DTT, the stress is 20 percent of the yield strength versus a prescribed upper limit of 80 percent of the yield strength; therefore, at this point there is a margin of four (80/20).

Since the Fracture Analysis Diagram is based on a plot of nominal stress versus temperature and different size flaws (cracks) are assumed, the use of actual stress concentration factors does not apply.

As part of the plant operator training program, supervisory and operating personnel were instructed in reactor vessel design, fabrication, and testing, as well as precautions necessary for pressure testing and operating modes. The need for record keeping was stressed; such records being helpful for future summation of time at power level and temperature which tends to influence the irradiated properties of the material in the core region. These items are incorporated into the operating instructions.

Piping

The analysis of the Reactor Coolant Loop/Supports System is described in Section 3.9.

Steam Generators

Calculations confirm that the steam generator tube sheet will withstand the loading (which is quasi-static rather than a shock loading) caused by loss of reactor coolant. The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 36.78 ksi for the Unit 2 steam generator. This is well below ASME Section III criterion 1.05 Su = 94.5 ksi at 620°F. Because the pressure in the primary channel head would drop to zero under the condition postulated, no damage will result to the channel head.

The rupture of primary or secondary piping was assumed to impose a maximum pressure differential of 2485 psi across the tubes and tube sheet from the primary side or a maximum pressure differential of 1007 psi across the tubes and tube sheet from the secondary side, respectively. Under these conditions there is no rupture of the primary to secondary boundary, including tubes and tube sheet. This criterion prevents any violation of the containment boundary.

To meet this criterion, it was established that under the postulated accident conditions, where a primary to secondary side differential pressure of 2,485 psi exists, the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, do not exceed 0.7Su at 620°F. An examination of stresses under these conditions shows that for the case of a 2,485 psi maximum tubesheet pressure differential; the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Reference 31.

A complete tube sheet analysis was performed to verify the structural integrity of the primary-secondary boundary under blowdown plus seismic conditions. In the case of a primary pressure loss accident, the secondary-primary pressure differential can reach 1,007 psi. Reference 31 shows that the criteria are met for the tubesheet.

The tubes were designed to the requirements (including stress limitations) of Section III for normal operation, assuming 2485 psig as the normal operating pressure differential. Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

For Unit 2, no significant corrosion of the Inconel 690 tubing is expected during the lifetime of the plant. Operating experience has shown that Inconel 600 tubing can be susceptible to several degradation mechanisms and as such these active and potential corrosion mechanisms are monitored by periodic inspections, still applicable to Unit 1 but no longer applicable to Unit 2.

For Unit 2 RSGs, PWSCC and other potential tube degradation mechanisms such as denting and IGA/SCC are expected to be precluded due to:

- The use of Inconel 690 thermally treated tubing
- The stress relief heat treatment of the small U-bends
- The full depth high pressure hydraulic expansion of the tubes inserted in the tubesheet holes with a low clearance, and
- The use of stainless steel broached tube support plates with flat lands

For the Unit 2 tubes, the risk of elastoplastic instability has been verified under an external pressure, i.e., the secondary/primary differential pressure. The criterion, in this case, is the allowable external pressure calculated according to the paragraph NB-3133 of the ASME Boiler and Pressure Vessel Code Section III and document entitled "Collapse of Ductile Heat Exchange Tubes with Ovality Under External Pressure," Reference 30. For design conditions, the allowable external pressure is equal to 979 psi, which is higher than the maximum secondary/primary differential pressure of 670 psi. For hydrotest conditions, the maximum external pressure is equal to 1,482 psi, which does not exceed 80% of the lowest bound collapse pressure of 2,289 psi.

In faulted conditions, the following loadings, which conform to the ASME Boiler and Pressure Vessel Code Section III, have been considered for the tube bundle:

- seismic loads,
- transients pressure load differentials.

The structural analysis and evaluation of the lower assembly (composed of the channel head, the tubesheet, the lower secondary shell, the partition plate and the support pads) have been performed in accordance with the requirements of the Design Specification (Reference 32) for the loads, and in accordance with the criteria of the ASME Boiler and Pressure Vessel Code Section III.

To perform the primary stress analysis (design, faulted, test conditions), the primary and secondary stress analysis (3 S_m analysis), and the fatigue analysis in the perforated tubesheet, appropriate stress correction factors, which take into account the tubesheet hole pattern, have been applied to the stresses.

The tube bundle is analysed in accordance with the following paragraphs of the ASME Boiler and Pressure Vessel Code Section III:

- NB-3221 for design conditions
- NB-3222, NB-3223 for the normal and upset conditions
- F-1331.1 for faulted conditions
- NB-3226 for test conditions

To address the entire tube bundle, the straight part of the tube, the curved part of the tube, and the tube/tubesheet connection have been analysed. Ovality tolerances have been taken into account for both the straight and curved parts of the tube.

The loads considered for all service conditions are:

- primary/secondary differential pressure
- secondary/primary differential pressure,
- hot leg temperature and primary pressure,
- secondary temperature and secondary pressure,
- steam temperature,
- displacement induced by the tubesheet,
- bending stresses due the interaction between the tubes and the tube support plates.

Tabulations of significant results of the tubesheet assembly are provided by Reference 31 and Figures 5.2-7 through 5.2-9. Figure 5.2-10 denotes the primary - secondary boundary component locations.

In all evaluated cases, the tubesheet assembly met the stress limitations and fatigue criteria specified in the ASME Boiler and Pressure Vessel Code Section III.

The Westinghouse analysis of the Unit 1 steam generator tube sheets was included as part of the Stress Report requirement for Class A nuclear pressure vessels. The evaluation was based on the stress and fatigue limitations outlined in Article 4, Design, of Section III. The stress analysis techniques utilized included all factors considered appropriate to conservative determination of the stress levels utilized in evaluation of the tube sheet complex. The analysis of the tube sheet complex included the effect of all appurtenances attached to the perforated region of the tube sheet considered appropriate to conservative analysis of stress for evaluation on the basis of Section III stress limitations. The evaluation involved the heat conduction and stress analysis of the tube sheet, channel head, secondary shell structure for particular steady design conditions for which Code stress limitations were to be satisfied and for discrete points during transient operation for which the temperature/pressure conditions must be known to evaluate stress maxima-minima for fatigue life usage. In addition, limit analyses were performed to determine tube sheet capability to sustain emergency operating conditions for which elastic analysis does not suffice. The analytic techniques utilized were computerized and significant stress problems were verified experimentally to justify the techniques where possible.

Generally, the analytic treatment of the tube-tube sheet complex included determination of elastic equivalent plate stress within the perforated region from an interaction analysis utilizing effective elastic constants appropriate to the nature of the perforation array. For the perforated region of the tube sheet, the flexural rigidity was based on studies of behavior of plates

with square hole arrays utilizing techniques such as those reported by O'Donnell (11), Mahoney (12), Lemcoe (13), and others. Similarly, stress intensity factors were determined for square hole arrays using the combined equivalent plate interaction forces and moments applied to results of photo-elastic tests of model coupons of such arrays as well as verification using computer analysis techniques such as "point matching" or "collocation". The stress analysis considered stress due to symmetric temperature and pressure distribution as well as asymmetric temperature distribution due to temperature drop across the tube sheet divider lane.

The fatigue analysis of the complex was performed at potentially critical regions in the complex such as the junction between tube sheet and channel head or secondary shell as well as at many locations throughout the perforated region of the tube sheet. For the holes for which fatigue evaluation was done, several points around the hole periphery were considered to assure that the maximum stress excursion has been considered. The fatigue evaluation was computerized to include stress maxima-minima excursions considered on the intra-transient basis.

The evaluation of the tube-to-tube sheet juncture was based on a stress analysis of the interaction between tube and tube sheet hole for the significant thermal and pressure transients that are applied to the steam generator in its predicted histogram of cyclic operation. The evaluation was based on the numerical limits specified in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

Of importance in the analysis of the interaction system is the behavior of the tube hole, where it is recognized that the hole behavior is a function of the behavior of the entire tube sheet complex with attached head and shell. Hence, the output of the tube sheet analysis giving equivalent plate stresses in the

perforated region was utilized in determining the free boundary displacements of the perforation to which the tube is attached.

Analysis of the juncture for the fillet-type weld was made with consideration of the effect of the rolled-in joint in the weld region as well as with the conservative assumption that the tube flexure relative to the perforation is not inhibited with the rolled-in effect.

The major concern in fatigue evaluation of the tube weld was the fatigue strength reduction factor to be assigned to the weld root notch. For this reason, Westinghouse conducted low-cycle fatigue tests of tube material samples to determine the fatigue strength reduction factor and applied them to the analytic interaction analysis results in accordance with the accepted techniques in the Nuclear Pressure Vessel Code for Experimental Stress Analysis. The fatigue strength reduction factor determined therefrom was not different from that reported in the well known paper on the subject by O'Donnell and Purdy (14). An actual tube sheet joint contained in a tube sheet was successfully tested under thermal transient conditions much more severe than that achieved in anticipated power plant operation.

A wide range of computational tools were utilized in these solutions including finite element, heat conduction, and thin shell computer solutions. In addition, analysis techniques were verified by photo-elastic model tests and strain gage measurement of prototype models of an actual steam generator tube sheet.

Finally, in order to evaluate the ultimate safety of structural complex, a computer program for determining a lower-bound pressure limit for the complex based on elastic-plastic analysis was developed and applied to the structure. This was verified by a strain gage steel model of the complex tested to failure.

In all cases evaluated, the steam generator tube sheet complex met the stress limitations and fatigue criteria specified in Article 4 of the Code as well as emergency condition limitations specified in the Equipment Specifications.

In this way, the tube-tube sheet integrity was demonstrated under the most adverse conditions resulting from a major breach in either the primary or secondary system piping.

Tabulations of significant results of the Unit 1 tube sheet complex are in Table 5.2-33. Table 5.2-34 presents significant results from the Unit 1 secondary shell and transition cone analysis. Stress results from the tube analysis are tabulated in Tables 5.2-35 and 5.2-36. Figures 5.2-22, 5.2-23 and 5.2-24 denote important stress locations.

Pressurizer

The pressurizer was analyzed for fatigue conditions in accordance with Section III of the ASME Boiler and Pressure Vessel Code using the thermal and pressure transient conditions listed elsewhere in this Section.

The pressurizer vessel was analyzed for the following loading conditions:

1. Normal operation loadings which included:

   a. Weight of water based on the vessel filled with cold water, and including insulation

   b. Normal loadings exerted by connecting piping

2. Seismic loadings which included:

   a. For the OBE, the pressurizer vessel is designed to resist earthquake loadings simultaneously in the horizontal and vertical directions and to transmit

such loadings through the vessel supports to the foundation. The OBE results in mechanical loadings and their combination with the normal operational loads is to be considered an upset condition. The components of loadings exerted by the external piping due to the OBE were included in this evaluation.

b. For the DBE, pressurizer vessel function is not impaired so as to prevent a safe and orderly shutdown of the reactor plant when the DBE loadings both horizontal and vertical acting simultaneously are imposed on the vessel. These loadings and the centers of gravity involved were determined on the basis of the vessel at normal operating pressure, temperature, and water level.

The DBE was considered a faulted condition with the following exceptions:

(1) The combination of all primary stress intensities in the vessel support skirt was required to be within the support skirt material yield strength specified in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The stress intensity limits of the vessel associated with the DBE in combination with normal operation were as follows:

$P_m \leq 1.2\ S_m$ or tabulated yield (S_y) whichever is greater $P_l + P_b \leq 1.8\ S_m$ or $1.5\ S_y$ whichever is greater

The components of loadings exerted by the external piping due to the DBE were included in this evaluation.

3. The pressurizer vessel, nozzles, and vessel supports were designed to resist pipe break loadings in combination with the normal operational loads. The moment and forces were considered as acting in combination with each force separately. The pipe break accident was considered to be a faulted condition with the exception of the stress intensity limits being those specified under the DBE condition.

4. The pressurizer vessel, nozzles, and vessel supports were analyzed for the combination of normal operating loads plus the DBE loads plus the pipe break loads. The resulting stress intensities did not exceed the stress intensity limits of Paragraph N17.11 (faulted conditions) in Section III of the Code with the following exception. The combination of all primary stress intensities in the vessel supports were within the support material yield strength specified in the Code. If necessary, higher stress intensity values are adopted in the vessel supports where plastic instability analyses of the support and supported component system are performed in accordance with Paragraph N417.11 of ASME Code, Section III.

   A plastic instability analysis of the support and supported system was not needed since the adequacy was proved by elastic analysis.

Reactor Coolant Pump

All the pressure bearing parts of the reactor coolant pump were analyzed in accordance with Article 4 of the ASME Code,

Section III. This included the casing, the main flange, and the main flange bolts. The analysis included pressure, thermal, and cyclic stresses; and these were compared with the allowable stresses in the Code.

Mathematical methods of the reactor coolant pump parts were prepared and used in the analysis which proceeded in two phases.

1. In the first phase, the design was checked against the design criteria of the ASME Code with pressure stress calculations, although thermal effects were included implicitly with the experience factors. By this procedure, the shells were profiled to attain optimum metal distribution with stress levels adequate to meet the more limiting requirements of the second phase.

2. In the second phase the interactive forces needed to maintain geometric capability between the various components were determined at design pressure and temperature and applied to the components along with the external loads to determine the final stress state of the components. They were finally compared with the Code allowable values.

There were no other sections of the Code which were specified as areas of compliance, but where Code methods, allowable stresses, fabrication methods, etc., were applicable to a particular component, these were used to give a rigorous analysis and conservative design.

5.2.2 Overpressurization Protection

5.2.2.1 Pressure-Relieving Devices

The RCS is protected against overpressure by control and protective circuits such as the high-pressure trip and by relief

and safety valves connected to the top head of the pressurizer. The relief and safety valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10 percent, in accordance with Section III of the ASME Boiler and Pressure Vessel Code. Their capacity is determined from considerations of the RPS and accident or transient conditions which may potentially cause overpressure.

The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve setting. Safety and relief valve design parameters are shown in Table 5.2-8.

5.2.2.2 Report on Overpressure Protection

The "Report on Overpressure Protection" is not a part of the Code requirement for the Salem Station. Applicable codes for this station are listed in Tables 5.2-9A and 5.2-9B.

However, the overpressure protection capability of Westinghouse PWRs, including Salem Station, is discussed in Reference 15 and in Reference 29 for Unit 2.

5.2.2.3 RCS Pressure Control During Low Temperature Operation

Refer to Section 7 for a discussion of the Overpressure Protection System for low temperature operation.

5.2.3 General Material Considerations

Table 5.2-26 summarizes the quality assurance program with regard to inspections performed on RCS components. In addition to the inspections shown in Table 5.2-26, there were those which the equipment supplier performed to confirm the adequacy of material received and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME Code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication were governed by USAS B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels.

Procedures for performing the examinations were consistent with those established in the ASME Code Section III and were reviewed by qualified engineers. These procedures have been developed to provide the highest assurance of quality material and fabrication. They consider not only the size of the flaws, but equally as important, how the material was fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the accessible external surfaces of the primary RCS pressure containing segments receive a 100-percent surface inspection by magnetic particle or liquid penetrant testing after hydrostatic test. All reactor vessel plate material was subjected to angle beam gas well as straight beam ultrasonic testing to give maximum assurance of quality. All reactor vessel forgings received the same inspection. In addition, 100 percent of the material volume was covered in these tests as an added assurance over the grid basis required in the Code.

Quality control engineers monitored the supplier's work, witnessing key inspections not only in the supplier's shop but in the shops of subvendors of the major forgings and plate material. Normal surveillance included verification of records of material, physical and chemical properties, review of radiographs,

performance of required tests, and qualification of supplier personnel.

Section III of the ASME Code requires that nozzles carrying significant external loads are attached to the shell by full penetration welds. This requirement was carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop were made using full penetration welds.

The RCS components were welded under procedures which required the use of both preheat and post-heat. Preheat requirements, (not mandatory for Unit 1 only) under Code rules, were performed on all weldments including P1 and P3 materials which are the materials of construction in the reactor vessel, pressurizer, and steam generators. Preheat and post-heat of weldments both served a common purpose: the production of tough, ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones whereas post-heating achieves this by tempering any hard zones which may have formed due to rapid cooling.

5.2.3.1 Material Specifications

Each of the materials used in the RCS is selected for the expected environment and service conditions. The major component materials are listed in Table 5.2-27.

All RCS materials which are exposed to the coolant are corrosion-resistant. They consist of stainless steels and Inconel, and were chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 5.2-28. Reactor coolant chemistry is further discussed in Section 5.2.3.4.

5.2.3.2 Compatibility with Reactor Coolant

The water in the secondary side of the steam generators is held within the chemistry specification given in Section 10.

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

It is a characteristic of stress-corrosion that combinations of alloy and environment which result in cracking are usually quite specific. Environments which have been shown to cause stress-corrosion cracking of stainless steels are free of alkalinity in the presence of chlorides, fluorides, and free oxygen. However, the reactor coolant chemistry is controlled to avoid the occurrence of these species in any significant contribution. The steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress-corrosion in caustic and chloride aqueous solutions has indicated that Inconel 690 TT, used for Unit 2, has better resistance to general and pitting-type corrosion in severe operating water conditions than Inconel 600 TT, used for Unit 1. Extensive operating experience with Inconel units has confirmed this conclusion.

5.2.3.3 Compatibility with External Insulation

All external insulation of RCS components is compatible with the component materials. The cylindrical shell exterior and closure flanges and bottom head of the reactor vessel are insulated with stainless steel metallic reflective insulation. The closure head is insulated with stainless metallic reflective insulation. All

other external corrosion-resistant surfaces in the RCS are insulated with low or halide-free insulating material as required.

5.2.3.4 Chemistry of Reactor Coolant

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

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to the reactor coolant water quality listed in Table 5.2-28. The limitations on RCS chemistry ensure that corrosion of the RCS is minimized and reduces the potential for RCS leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State limits for dissolved oxygen, chloride and fluoride (Table 5.2-28) provides adequate corrosion protection to ensure the structural integrity of the RCS over the life of the plant. The associated effects of exceeding the dissolved oxygen, chloride and fluoride limits are time and temperature dependent. Dissolved oxygen limits in Table 5.2-28 apply whenever temperature of reactor coolant exposed to a metal surface is greater than 250°F, including the pressurizer. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State limits, up to the Transient limits, for a limited time interval without having a significant effect on the structural integrity of the RCS. A specified time interval permitting continued operation within the restrictions of the Transient limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State limits. Plant procedures define required actions if the Steady State limits or the Transient limits are exceeded. The sample and analysis frequency contained in chemistry procedures provides adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System (CVCS) and Sampling System which are described in Section 9.

5.2.3.5 Electroslag Weld Quality Assurance

The Salem 90° elbows were electroslag welded. The following efforts were performed for quality assurance of these components:

1. The electroslag welding procedure employing one wire techniques was qualified in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section IX and Code Case 1355 plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from a 5-inch thick weldment and successfully tested:

   a. 6 transverse tensile bars - as welded

   b. 6 transverse tensile bars - 2050°F, H_2O quench

   c. 6 transverse tensile bars - 2050°F, H_2O quench plus 750°F stress relief heat treatment

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d. 6 transverse tensile bars - 2050°F, H_2O quench, tested at 650°F

e. 12 guided side bend test bars

2. The casting segments were surface conditioned for 100-percent radiographic and penetrant inspections. The acceptance standards were ASTM E-186 severity level 2 except no Category D or E defectiveness was permitted and USAS Code Case N-10, respectively.

3. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were USAS Code Case N-10.

4. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100-percent radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with USAS Code Case N-10.

5. Weld metal and base metal chemical and physical analysis were determined and certified.

6. Heat treatment furnace charts were recorded and certified.

The Salem reactor coolant pump casings were electroslag welded. The following efforts were performed for quality assurance of the components.

The electroslag welding procedure employing two or three wire techniques was qualified in accordance with the requirements of

the ASME Boiler and Pressure Vessel Code Section IX and Code Case 1355 plus supplemental evaluations as requested by Westinghouse. The following test specimens were removed from an 8-inch thick and from a 12-inch thick weldment and successfully tested for both the two-wire and the three-wire techniques, respectively:

1. Two-wire electroslag process - 8-inch thick weldment

   a. 6 transverse tensile bars - 750°F post-weld stress relief

   b. 12 guided side bend test bars

2. Three-wire electroslag process - 12-inch thick weldment

   a. 6 transverse tensile bars - 750°F post-weld stress relief

   b. 17 guided side bend test bars

   c. 21 Charpy V-notch specimens

   d. Full section macroexamination of weld and heat affected zone

   e. Numerous microscopic examinations of specimens removed from the weld and heat affected zone regions

   f. Hardness survey across weld and heat affected zone

3. A separate weld test was made using the two-wire electroslag technique to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and

techniques as such an occurrence was anticipated during production applications due to equipment malfunction, power outages, etc. The following test specimens were removed from an 8-inch thick weldment in the stop-restart-repaired region and successfully tested:

a. 2 transverse tensile bars - as welded

b. 4 guided side bend test bars

c. Full section macroexamination of weld and heat affected zone

4. All of the weld test blocks in Items 1, 2, and 3 were radiographed using a 24 Mev Betatron. The radiographic quality level as defined by ASTM E-94 obtained was between one-half of 1 percent to 1 percent. There were no discontinuities evident in any of the electroslag welds.

   a. The casting segments were surface conditioned for 100-percent radiographic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2 except no Category D or E defectiveness was permitted for section thickness up to 4 1/2 inches and ASTM E-280 severity level 2 for section thicknesses greater than 4 1/2 inches. The penetrant acceptance standards were ASME Boiler and Pressure Vessel Code Section III, Paragraph N-627.

   b. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME Boiler and Pressure Vessel Code Section III, Paragraph N-627.

c. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100-percent radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code Section III, Paragraph N-627.

d. Weld metal and base metal chemical and physical analyses were determined and certified.

e. Heat treatment furnace charts were recorded and certified.

5.2.4 Fracture Toughness

5.2.4.1 Compliance with Code Requirements

Assurance of adequate fracture toughness of the RCS is provided by compliance with the requirements for fracture toughness included in the Summer 1996 Addenda to Section XI of the ASME Boiler and Pressure Vessel Code and by Code Case N-640.

5.2.4.2 Acceptable Fracture Energy Levels

Allowable pressures as a function of the rate of temperature change and the actual temperature relative to the vessel RT_{NDT} will be established according to the methods given in 10CFR50 Appendix G, Appendix G included in the Summer 1996 Addenda of Section XI of the ASME Boiler and Pressure Vessel Code, and by Code Case N-640. Typical Pressure - Temperature limit curves incorporating allowances for instrument error in measurement of temperature and pressure are given on Figures 5.2-11 and 5.2-12.

The results of the radiation surveillance program will be used to verify that the RT_{NDT} and Charpy Upper Shelf Energy (USE) predicted from Regulatory Guide 1.99, Revision 2, are appropriate.

The use of RT_{NDT} from Regulatory Guide 1.99, Rev.2 includes a ΔRT_{NDT} to account for radiation effects on the core region material and also includes margin added to obtain conservative, upper bound values of RT_{NDT}. Tables 5.2-37 and 5.2-38 summarize End of License (EOL, 32 EFPY) values for RT_{NDT} used in the Pressure-Temperature limit curve analysis. (References 31 and 32)

The Pressure-Temperature limit curves, or heatup and cooldown curves, are incorporated into Technical Specifications for Salem Units 1 and 2 and are based on Regulatory Guide 1.99, Rev. 2 methodology, in compliance with NRC Generic Letter 88-11.

5.2.4.3 Operating Limitations During Starting and Shutdown

Operating limits for the RCS with respect to heatup and cooldown rates are defined in the Technical Specifications.

The heatup and cooldown curves for the plant are based on the actual measured fracture toughness properties of the vessel materials, determined in accordance with the above mentioned new fracture toughness requirements.

5.2.4.3.1 Maximum Heating and Cooling Rates

The RCS operating cycles used for design purposes are given in Table 5.2-10 and described in Section 5.2.1.5. The maximum system heating and cooling rate is 100°F per hour. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level of 55°F per hour. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant.

5.2.4.3.2 Maximum Pressure

The RCS serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of

a fuel cladding failure, the RCS is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the RCS is assured. Thus, the safety limit of 2735 psig (110 percent of design pressure) has been established. This represents the maximum transient pressure allowable in the RCS under the ASME Code, Section III. The RCS pressure settings are given in Table 5.2-1.

5.2.4.3.3 System Minimum Operating Conditions

Minimum operating conditions for the RCS for all phases of operation are given in the Technical Specifications.

5.2.4.4 <u>Compliance with Reactor Vessel Material Surveillance Program Requirements</u>

5.2.4.4.1 Surveillance Capsule

In the surveillance program, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading (WOL) fracture mechanics test specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and is in accordance with ASTM-E-185-70, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." The surveillance program does not include thermal control specimens. These specimens are not required since the surveillance specimens will be exposed to the combined neutron irradiation and temperature effects, and the test results will provide the maximum transition temperature shift. Thermal control specimens are considered in ASTM E-185-70 and would not provide any additional information on which the operational limits for the reactor vessel are set.

The reactor vessel surveillance program uses eight specimen capsules. The capsules are located about 3 inches from the vessel wall directly opposite the center portion of the core. Sketches of an elevation and plan view showing the location and dimensional spacing of the capsules with relation to the core, thermal shield and vessel and weld seams are shown on Figures 5.2-14 and 5.2-15, respectively. The capsules can be removed when the vessel head is removed, and can be replaced when the internals are removed. The capsules contain reactor vessel steel specimens from the limiting shell plate or plates located in the core region of the reactor and associated weld metal and heat affected zone metal. (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 base metals and as deposited weld metal.) In addition, correlation monitors made from full documented specimens of SA-533, Grade B, Class 1 material obtained through Subcommittee II of ASTM Committee E10, Radioisotopes and Radiation Effects, are inserted in the capsules of Unit 1 only. The eight capsules contain tensile specimens, Charpy V-notch specimens (which include weld metal and heat affected zone material) and WOL specimens. Dosimeters including Ni, Cu, Fe (Unit 2 only) Co-Al, Cu shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238 are placed in filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low-melting alloys are included to monitor 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.

Each of three capsules (S, V and Y) for Unit 1 contains the following specimens:

Material	No. Charpy	No. Tensile	No. WOL
Plate	8	2	2
Weld Metal	8	2	2
Heat Affected Zone Metal	8	-	-
ASTM Reference	8	-	-

Note: Each capsule contains baseplate material from a different plate.

Capsule S - Plate 1
Capsule V - Plate 2
Capsule Y - Plate 3

Dosimeters

Pure Cu
Pure Ni
CoAl (0.15 percent Co)
CoAl (Cadmium Shielded)
U-238 (Cadmium Shielded)

Np-237 (Cadmium Shielded)

Thermal Monitors

97.5 percent Pb, 2.5 percent Ag (579°F MP)
97.5 percent Pb, 1.75 percent Ag, 0.75 percent Sn (590°F MP)
(MP = Melting Point)

Each of five additional capsules (T, U, W, X, and Z) for Unit 1 contains the following specimens:

Material	No. Charpy	No. Tensile	No. WOL
Plate No. 1	8	1	2
Plate No. 2	8	1	2
Plate No. 3	8	1	2
ASTM Reference	8	-	-

Dosimeters

Pure Cu

Pure Ni

CoAl (0.15 percent Co)

CoAl (Cadmium Shielded)

Thermal Monitors

97.5 percent Pb, 2.5 percent Ag (579°F MP)

97.5 percent Pb, 1.75 percent Ag, 0.75 percent Sn (590°F MP)

Each of four capsules S, V, W and X for Unit 2 contains the following specimens:

Material	No. Charpy	No. Tensile	No. WOL
Limiting Plate*	8	-	-
Limiting Plate**	12	2	4
Weld Metal	12	2	-
Heat Affected Zone Metal	12	-	-

Each of four additional capsules (T, U, Y and Z) for Unit 2 contains the following specimens:

Material	No. Charpy	No. Tensile	No. WOL
Limiting Plate*	8	-	-
Limiting Plate**	12	2	-
Weld Metal	12	2	4
Heat Affected Zone Metal	12	-	-

Dosimeters

Pure Cu

Pure Fe

Pure Ni

CoAl (0.15 percent Co)

CoAl (Cadmium shielded)

U-238 (Cadmium shielded)

Np-237 (Cadmium shielded)

*Specimens oriented parallel to the principal rolling direction.

**Specimens oriented normal (transverse) to the principal rolling direction.

Thermal Monitors

97.5 percent Pb, 2.5 percent Ag (579°F MP)

97.5 percent Pb, 1.75 percent Ag, 0.75 percent Sn (590°F MP)

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the adjacent vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the nil ductility transition temperature (NDTT) measurements are representative of the vessel at a later time in life. Data from fracture toughness samples (WOL) are expected to provide additional information for use in determining allowable stresses for irradiated material.

The calculated maximum fast neutron exposure (E > 1 Mev) at the vessel wall is computed to be $1.64 \times 10^{19} \times n/cm^2$ for Unit 1 and $1.77 \times 10^{19}\ n/cm^2$ for Unit 2 at the end of 32 EFPY (Section 5.4.3.5). The reactor vessel surveillance capsules are located at 4° and 40° as shown on Figure 5.2-15. The relative exposures of the capsules and the adjacent vessel wall, and the vessel maximum are listed below:

Capsules at			Lead Vessel Maximum by a Multiplying Factor of:	
	Unit 1	Unit 2	Unit 1	Unit 2
4°	(V, X, U and W)	(S, V, W and Z)	1.28	1.31
40°	(S, Y, T and Z)	(T, U, X and Y)	3.47	3.39

Correlations between the calculations and the measurements on the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Section 5.4.3 and have indicated good agreement.

The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution will be considered in

the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure will be made by use of data on all capsules withdrawn.

The surveillance program for Unit 1 was prepared to meet ASTM E-185-70, Section 3.3. The test materials were procured and machined in 1969 prior to publication of ASTM E-185-70 and prior to issuance of "Reactor Vessel Material Surveillance Program Requirements," 10CFR50, Appendix H. Therefore, the eight capsules to be provided do not include five capsules which contain specimens from base metal, weld metal, and heat affected zone metal as required in 10CFR50, Appendix H.

For Unit 2, the surveillance program has been revised to meet proposed ASTM and NRC requirements reflecting ASME Code Case 1514 and will include eight capsules, each of which will contain the most limiting base plate, weld metal, and heat affected zone material.

The tentative schedule for capsule removal outlined hereunder is based on the ASTM E-185 removal requirement and the projected fluence based on the current low leakage fuel design.

The tentative schedule for Unit 1 capsule removal is as follows:

Capsule T	Removed in 1979
Capsule Y	Removed in 1984
Capsule Z	Removed in 1987
Capsule S	Removed in 1996
Capsule V	Standby
Capsule U	Standby
Capsule X	Standby
Capsule W	Standby

The tentative schedule for Unit 2 capsule removal is as follows:

Capsule T	Removed in 1983
Capsule U	Removed in 1986
Capsule X	Removed in 1992
Capsule Y	Removed in 2000
Capsule S	At 32 EFPY
Capsule V	Standby
Capsule W	Standby
Capsule Z	Standby

5.2.4.4.2 Material Properties of Salem 1 and 2 Reactor Pressure Vessels

A good set of RPV material data is very important in accurately assessing the irradiation effects on fracture toughness properties of the vessel beltline materials. The chemical and mechanical properties of the Salem weld data, used in this study, were the result of a thorough investigation performed by Combustion Engineering Owners Group. The RPV plate data in the beltline region were obtained from Westinghouse documents.

Figure 5.2-18 shows the assembly of the Salem Reactor Pressure Vessel. The Salem vessels were fabricated by Combustion Engineering. The Reactor Vessel beltline materials that directly surround the effective height of the core and the adjacent region that are predicted to experience sufficient neutron irradiation damage consist of the intermediate and lower shell course and their associated welds. Figures 5.2-19 and 5.2-20 identify and schematically locate the Reactor Vessel plates and welds in the beltline region for Salem Units 1 and 2, respectively. Figure 5.2-21 shows the locations of surveillance capsules.

The chemical and mechanical properties of the beltline region welds and plates of Salem Units 1 and 2 which are necessary to calculate the fracture toughness of the vessel are tabulated in Tables 5.2-30, 5.2-31 and 5.2-32.

5.2.4.5 Pressurized Thermal Shock (PTS)

RPV fracture toughness calculations for Salem Units 1 and 2 have been performed in response to the NRC final rule on protection against PTS events (10CFR50.61).

Two key inputs are required to calculate the fracture toughness of the reactor pressure vessel which is characterized by the quantity RT_{pts}. These are neutron fluence and the chemical/mechanical properties of RPV beltline region materials. The neutron fluence calculations have been performed using the industry accepted transport theory method, see Section 5.4.3.5. The chemical/ mechanical data of beltline region welds are discussed in the previous section.

The fracture toughness state of the vessel with respect to PTS is characterized by the quantity RT_{PTS} (Reference Temperature for PTS). The RT_{PTS} equation in the final PTS rule (10CFR50.61) is used. The equation is:

$$RT_{PTS} = RT_{NDT(u)} + M + \Delta RT_{PTS}$$

Where $RT_{NDT(u)}$ = Reference Temperature for Nil Ductility Transition for the unirradiated material

M	=	Margin added to account for uncertainties in the values of $RT_{NDT(u)}$, copper and nickel contents, fluence and the calculational procedures
ΔRT_{PTS}	=	Change in RT_{NDT} due to fluence and is dependent on chemical/material properties
	=	FF * CF
FF	=	Fluence factor
	=	$f^{(0.28 - 0.10 \log f)}$
f	=	Best estimate fluence, in units of 10^{19} n/cm^2 (for energies greater than or equal to 1.0 MeV) at the clad-base metal interface on the inside surface of the vessel at the material location.
CF	=	Chemistry factor, which is a function of copper and nickel content.

The PTS rule requires the licensee to have projected values of RT_{PTS} for each reactor vessel beltline material for End of Operating License (EOL). 32 effective full power years (EFPY) was used to determine EOL fluence. This equates to 80% capacity factor for the 40-year operating license of the Salem units. The PTS rule also includes PTS screening criteria or values of RT_{PTS} for beltline materials above which the plant cannot operate without justification.

Tables 5.2-39 and 5.2-40 provide results of the PTS evaluation for Salem units 1 and 2 for the EOL fluence. These tables also provide the appropriate PTS screening criteria.

The key results are as follows: (References 29 and 30)

1. For Salem Unit 1, the limiting beltline region materials are weld seams 3-042A/B/C. The RT_{pts} of weld seams 3-042A/B/C are projected to be 264°F at EOL. This RT_{pts} value is below the screening criterion of 270°F.

2. For Salem Unit 2, the limiting beltline region materials are weld numbers 3-442A/B/C. The RT_{pts} of these welds are projected to be 229°F at EOL. This RT_{pts} value is below the screening criterion of 270°F.

5.2.4.6 Charpy Upper Shelf Energy

The Charpy Upper Shelf Energy (USE) of Reactor beltline materials decreases with irradiation. Regulatory Guide 1.99, Rev. 2 specifies the methodology for predicting USE decrease for fluence and copper content. End of License (EOL, 32 EFPY) USE predictions were made using this methodology. The results are presented in Tables 5.2-41 and 5.2-42. All beltline materials are expected to have USE greater than 50 ft-lb through EOL as required by 10CFR50 Appendix G.

5.2.5 Austenitic Stainless Steel

The core support structural load bearing members and the stainless

steel reactor coolant pressure boundary components were welded in accordance with the Westinghouse criteria, which are as follows.

Type 308 weld filler material is used for all welding applications to avoid microfissuring. As an option, Type 308L weld filler metal analysis is substituted for consumable inserts when this technique is used for the weld root closure. 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

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combination to be capable of providing not less than 5 percent delta ferrite in the deposit. Electrodes conforming to SFA-5.4 are of the -15 or -16 (lime type) current characteristics.

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 III, NB-2430 and in addition, shall include delta ferrite determinations. These determinations are made by calculation using the "Schaeffler Constitution Diagram for Stainless Steel Weld Metal." Subsequent in-process delta ferrite determinations are not required. Other methods of ferrite determinations are useable on the basis of the developmental data and recommendations concurrently existing from the Advisory Subcommittee for Welding Stainless Steel of the High Alloy Committee in the Welding Research Council.

Methods used in manufacturing components of the reactor coolant pressure boundary and core structural load bearing members to minimize possible problems with severely sensitized stainless steel are as follows.

Reactor Vessel (Unit 1)

Primary nozzle safe ends are wrought austenitic stainless steel attached to the nozzles prior to final post-weld heat treatment and therefore, are sensitized. Other safe ends were installed after post-weld heat treatments.

There are no part length CRDMs installed on the Unit 1 replacement RVCH.

Reactor Vessel (Unit 2)

The primary nozzle safe ends, and other safe ends, are fabricated the same as Unit 1. There are no part length CRDMs installed on Unit 2.

Steam Generators (Unit 1)

The nozzle safe ends are prepared by buttering with austenitic stainless steel weld metal.

Pressurizers

Safe ends are of Type 316 stainless steel. Safe end post-weld heat treatment consisted of heating to 1125 - 25°F for 9 hours on Unit 1 and 5 hours on Unit 2 with heating and cooling rates in accordance with the ASME Section III code rules. Testing to determine the degree of sensitization that could have occurred as a result of the post-weld heat treatment cycle was not performed.

Internals (Both Units)

For internals where austenitic stainless steel must be given a stress-relieving treatment above 800°F, a high temperature stabilizing procedure is used. This is performed in the temperature range of 1600°F to 1900°F with holding times sufficient to achieve chromium diffusion to the grain boundary regions and would be expected to pass ASTM-A-393. No tests were performed on the core structural components to determine whether or not desensitization was accomplished by the elevated temperature stabilization treatment.

Internals (Unit 1)

The Unit 1 austenitic stainless steel core structural components were weld fabricated using the manual gas shielded tungsten arc, manual shielded metal arc, and semi-automatic submerged arc

welding processes. All of these welding processes and welders were previously qualified to 1965 ASME Section IX code rules. All of the welds were limited to a 350°F maximum interpass temperature. The heat input in kilojoules/inch were as follows, using the formula:

$$H = \frac{EI \times 60}{S}$$

where:

H = joules/inch energy input
E = volts
I = current in amperes
S = travel speed in in./min

GTAW Energy Input = 16.5 to 36.4 kj
SMAW Energy Input = 27.0 to 94.5 kj
SAW Energy Input = 45.4 to 62.0 kj

All full-penetration welds in the core structural components were penetrant tested at the root level and in the final finished condition on the "nearside" surfaces. The welds were radiographically examined through 100 percent of the volume using 2-2T sensitivity techniques with the acceptance standards conforming to 1968 ASME Section III code rules, Paragraph N624.3. All continuous partial penetration welds used in attaching accessory internal parts to the core structural components were progressively penetrant tested at the root level, each additional 1/2 inch of deposit thickness, and on the final finished surface. All non-continuous partial penetration fillet welds used in attaching accessory internal parts and locking devices to the core structural components were visually examined using 5x magnification to determine freedom from any type of linear discontinuity.

Internals (Unit 2)

The Unit 2 austenitic stainless steel core structural components were weld fabricated using the manual gas shielded tungsten arc, manual shielded metal arc, automatic gas shielded hot-wire tungsten arc, and the automatic submerged arc welding processes. The qualifications, interpass temperature control, and nondestructive testing of these components was the same as for the Unit 1 components.

The heat input in kilojoules/inch were as follows for each of the applied welding processes:

Manual GTAW Energy Input = 22.5 to 43.2 kj
Manual SMAW Energy Input = 18.0 to 120 kj
Automatic GTAW-HW Energy Input = 11.0 to 35 kj
Automatic SAW Energy Input = 63.4 to 138 kj (dc and ac)

For core support structural load bearing members and stainless steel RCPB welds, all welding on stainless steel was conducted by procedures that limited the interpass temperature to 350°F maximum.

The pressure or strength bearing stainless steel components or parts in the reactor vessel and associated RCS that may have become "furnace sensitized"* during the fabrication sequence include:

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

1. Reactor Vessels

Primary nozzle safe ends - Type 316 stainless steel forgings

2. Steam Generators (Unit 1)

Primary nozzle safe ends - weld metal buttered ends

3. Pressurizers

	Unit 1	Unit 2
Surge nozzle safe end	Type 316 forging	Type 316L forging
Spray nozzle safe end	Type 316 forging	Type 316L forging
Relief nozzle safe end	Type 316 forging	Type 316L forging
Safety (3) nozzle safe end	Type 316 forging	Type 316L forging

Westinghouse has evaluated the use of sensitized stainless steel and reactor components in pressurized water reactors (PWRs). The results of this evaluation are summarized in Reference 16 which covers the nature of sensitization, conditions leading to stress corrosion, and associated problems with both sensitized and non-sensitized stainless steel. The results of extensive testing and service experience that justify the use of stainless steel in the sensitized condition for components in Westinghouse systems is presented in Reference 16. References 16 through 20 provide evidence that the addition of nitrogen does not adversely affect the corrosion resistance of sensitized stainless steels.

A program has been established to monitor systems in which stainless steel piping contains stagnant, oxygenated, borated water as defined in IE Bulletin 79-17. The affected systems are: Residual Heat Removal, SIS, Containment Spray and CVCS. The program complies with IE Circular 76-06.

5.2.6 Pump Flywheels

A flywheel on the shaft above the motor provides additional inertia to extend flow coastdown. Each pump contains a ratchet mechanism to prevent reverse rotation. The reactor coolant pump flywheel is shown on Figure 5.2-16.

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

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.

The most adverse operating condition of the flywheels is visualized to be the loss-of-load situation. The following conservative design operation conditions preclude missile production by the pump flywheels. The wheels are fabricated from rolled, vacuum-degassed, ASTM A-533 steel plates. Flywheel blanks are flame-cut from the plate, with allowance for exclusion of flame-affected metal. A minimum of three Charpy tests are made from each plate parallel and normal to the rolling direction; they determine that each blank satisfies design requirements. An NDTT less than +10°F is specified. The finished flywheels are subjected to 100-percent volumetric ultrasonic inspection. The

finished machined bores are also subjected to magnetic particle or liquid penetrant examination.

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown on Figure 5.2-17) less than 50 percent of the minimum specified material yield strength at room temperature (100°F to 150°F). Bursting speed of the flywheels has been calculated on the basis of Griffith-Irwin's results (21,22) to be 3900 rpm, more than three times the operating speed.

A fracture mechanics evaluation was made on the reactor coolant pump flywheel. This evaluation considered the following assumptions:

1. Maximum tangential stress at an assumed overspeed of 125 percent

2. A crack through the thickness of the flywheel at the bore

3. 400 cycles of startup operation in 40 years

Using critical stress intensity factors and crack growth data obtained on flywheel material, the critical crack size for failure was greater than 17 inches radially and the crack growth data was 0.030 inch to 0.060 inch per 1000 cycles.

5.2.7 Reactor Coolant Pressure Boundary Leakage Detection Systems

RCS components were manufactured to exacting specifications which exceed normal code requirements. In addition, per use of the welded construction of the RCS and the extensive non-destructive testing to which it is subjected, it is considered that leakage through metal surface or welded joints is very unlikely.

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

5.2.7.1 Leakage Detection Methods

The existence of leakage from the RCS to the containment regardless of the source of leakage, is detected by one or more of the following:

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

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

3. An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer, or an increase in containment sump level.

5.2.7.1.1 Containment Air Particulate and Containment Radiogas Monitors

The containment air particulate monitor is the most sensitive instrument of those available for detection of reactor coolant leakage into the containment. This instrument is capable of detecting particulate radioactivity in concentrations as low as 1.0^{-9} μc/cc of containment air.

The sensitivity of the air particulate monitor to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal baseline leakage into the containment. The sensitivity is greatest where baseline leakage is low as has been demonstrated by the experience of Indian Point Unit 2, Yankee Rowe, and Dresden Unit 1. Where containment air particulate activity is below the threshhold of detectability, operation of the monitor with stationary filter paper would increase leak sensitivity to a few cubic centimeters per minute. Assuming a low background of containment air particulate radioactivity, a reactor coolant corrosion product radioactivity (Fe, Mn, Co, Cr) of 0.2 μc/cc (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking radioactivity into the containment air, sensitivity calculations indicate the air particulate monitor to be capable of detecting leaks as small as approximately 0.13 gpm (50 cc/min) within 30 minutes after they occur. If only 10 percent of the particulate activity is actually dispersed in the air, the threshhold of detectable leakage is raised to approximately 1.3 gpm (500 cc/min).

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

The containment radiogas monitor is inherently less sensitive (threshold at 10^{-6} μc/cc) than the containment air particulate

monitor, and would function only in the event that significant reactor coolant gaseous activity exists due to fuel cladding defects. Assuming a reactor coolant gas activity of 0.3 μc/cc, the occurrence of a leak of 5 gpm would be detected within an hour. In these circumstances this instrument would be useful as a backup to the air particulate monitor.

The air particulate and radiogas monitors are calibrated using a pulse generator to drive the counting circuits and using a check source to check detectors and input circuitry to the instruments. The alarm setpoints were verified at calibration. The system operability is checked during shutdown of the reactor.

5.2.7.1.2 Humidity Detector

The humidity detection instrumentation offers another means of detection of leakage into the containment. This instrumentation has not nearly the sensitivity of the air particulate monitor, but has the advantage of being sensitive to vapor originating from all sources: the Reactor Coolant, the Steam, and the Feedwater Systems. Plots of containment air dew point variations above a baseline maximum established by the cooling water temperature to the air coolers should be sensitive to incremental leakage equivalent to 0.2 to 1.0 gpm.

The sensitivity of this method is dependent on cooling water temperature, containment air temperature variation, and condensation on internal surfaces. With the least sensitivity, based on peak summer cooling water temperatures, it is estimated that an increase of 0.2 gpm in leak rate will cause a rise in containment dew point temperature of 1°F.

The dew point measuring equipment is checked for accuracy by using calibrated check coils. The system operability is checked during shutdown of the reactor.

5.2.7.1.3 Liquid Inventory in the Process Systems and in the Containment Sump

An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer is indicated by an increase in charging flow.

Gross leakage is indicated by a rise in normal containment sump level and periodic operation of containment sump pumps.

5.2.7.1.4 Condensate Measuring System

The Condensate Measuring System permits measurements of the flow rate of liquid run-off from the drain pans under each containment fan cooler unit. It consists of a vertical standpipe, valves, and instrumentation installed in the drain piping of the fan cooler unit.

Depending on the number of fan cooler units in operation, the drainage flow rate from each unit due to normal condensation is calculated. With the initiation of a leak, the containment humidity and condensate runoff rate both increase, the water level rises in the vertical pipe, and the high condensate flow alarm is actuated.

The containment specific humidity increases proportionately to time and leakage until the dew point is reached at the fan cooler cooling coils. With the increasing specific humidity, the heat removal capacity needed to cool the steam-air mixture to its dew point decreases. Therefore, increases in specific humidity and available heat removal capacity from the cooling coils result in added condensate flow. The condensate flow rate is then a function of specific humidity. Through accurate measurements of condensate flow variation, a reliable estimate of the reactor coolant leakage rate can be made.

A preliminary estimate of the leakage can be obtained from the rate of condensate flow increase during the transient; a better estimate can be made from the steady state condensate flow at equilibrium conditions. The device alarms on a 0.06 gpm condensate flow rate, which indicates that a 1 gpm or larger leak has been developing for about 5 minutes.

The system can be checked during reactor shutdown.

5.2.7.1.5 Intersystem Leakage Detection

The following provisions are available for the detection of intersystem leakage from the RCS:

1. Radiation monitors are provided for the Steam Generator Blowdown System, each Main Steam Line and condenser air removal effluent line which alert the operator to reactor coolant leakage into the Main Steam and Feedwater Systems from steam generator tube leaks.

2. Radiation monitors are provided for the Component Cooling System to detect reactor coolant leakage into the system from the Residual Heat Removal System. Surge tank level is also an indicator for leakage detection.

3. The accumulators are isolated from the RCS by two check valves. They are also provided with a remote manual valve. Leakage would be detected by level and pressure changes in the accumulators.

4. The high-head SIS line is isolated from the RCS by two check valves and normally closed remote manual valves. Leakage from the RCS, that would pass the normally closed SJ12/SJ13 gate valves, would be detected by pressure changes in the line.

5. The Residual Heat Removal System and the Intermediate Head SIS are isolated from the RCS by two check valves and normally closed remote manual valves. Leakage would cause operation of the relief valves which discharge to the containment sump.

RCS leakage can also be detected by level changes in the volume control tank, as well as by RCS water inventory balances, which are performed periodically. The indications identified above are provided, with appropriate alarms, in the control room.

5.2.7.2 Indication in Control Room

Positive indications in the control room of leakage of coolant from the RCS to the lower containment compartment are provided by equipment which permits continuous monitoring of the lower containment compartment air activity and humidity, and condensate run-off from the fan coolers. This equipment provides indication of normal background which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters are an indication of change within the lower containment compartment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate, and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

5.2.8 Inservice Inspection Program

Preservice and inservice inspection for Class 1, 2, and 3 components are in accordance with the rules of 1OCFR50.55(a), Paragraph (g) to the extent practical. Relief from the applicable ASME Section XI inspection requirements have been transmitted to the NRC through the Inservice Inspection Program Long Term Plans and Testing Programs.

5.2.8.1 Provisions for Access to Reactor Coolant Pressure Boundary

Provisions have been made in the design and arrangement of the RCS, Engineered Safety Systems and certain associated auxiliary systems to allow access for inservice inspection.

Public Service Electric & Gas has considered problems associated with inservice inspection during the design of the Station. These considerations have provided increased access such as the main coolant nozzle-to-pipe welds.

5.2.8.2 Equipment for Inservice Inspections

The reactor vessels are inspected using mechanized remote ultrasonic examination equipment for preservice baseline examination and subsequent inservice examinations. This mechanized remote ultrasonic equipment examines from the inside diameter of the nozzles and vessel welds to the extent practical. Extent of examination coverage and Relief Requests are described in the Inservice Examination Program Long Term Plan.

5.2.8.3 Recording and Comparing Data

Vendors who perform the mechanized remote ultrasonic examinations, have developed special forms and procedures for manual and mechanized inspections. Results of manual and mechanized inspections are recorded and can be compared with preservice and previous inservice data. Data is stored for correlation and subsequent inspections.

5.2.8.4 Reactor Vessel Acceptance Standards

Examination results are evaluated in accordance with the applicable edition of ASME, Section XI.

5.2.8.5 Coordination of Inspection Equipment with Access Provisions

Liaisons are maintained within the industry to discuss and resolve matters related to access for future inspection of components. Of consideration, during plant erection were the items to be inspected, as defined by the applicable editions of Section XI of the ASME Boiler and Pressure Vessel Code, and the capabilities of mechanized equipment either in use or in development. This information is constantly under review since additional experience is gained at other plants using new and improved equipment during Inservice Inspections. Those areas where examinations are limited or prevented entirely by access restrictions, have been transmitted to the NRC through the Inservice Inspection Program Long Term Plan.

5.2.9 Reference for Section 5.2

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TABLE 5.2-1

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	Pressure, Psig
Design Pressure	2485
Operating Pressure	2235
Safety Valves	2485
Power Relief Valves	2335
Pressurizer Spray Valves (Begin to Open)	2260
Pressurizer Spray Valves (Full Open)	2310
High Pressure Trip	2385
High Pressure Alarm	2385
Low Pressure Trip	1865
Low Pressure Alarm	1865
Hydrostatic Test Pressure	3107
Backup Heaters On	2210
Proportional Heaters (Begin to Operate)	2250
Proportional Heaters (Full Operation)	2220

TABLE 5.2-2

REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP

Unit 2	Pressure Drop, psi with DFBN (estimated)	Pressure Drop, psi with SDFBN (estimated) [2]
Across Pump Discharge Leg	3.1 [1]	3.1
Across Vessel, Including Nozzles	49.9 [1]	49.3
Across Hot Leg	1.2 [1]	1.2
Across Steam Generator	33.8 [1]	34.4
Across Pump Suction Leg	2.9 [1]	2.9
Total Pressure Drop	90.9 [1]	90.9
Unit 1		
Across Pump Discharge Leg	1.5	3.1
Across Vessel, Including Nozzles	52.0	49.3
Across Hot Leg	1.9	1.1
Across Steam Generator	35.67	35.8
Across Pump Suction Leg	1.8	2.9
Total Pressure Drop	92.87	92.2

NOTES:

1) Based on Best Estimate Flow (BEF) with Replacement Steam Generators RSGs at 0% tube plugging and Tave = 566° F.

2) Based on Best Estimate Flow (BEF) of 94,200 gpm/loop for Salem 1 and 94,800 gpm/loop for Salem 2 with Replacement Steam Generators RSGs at 0% tube plugging and Tave = 566° F.

TABLE 5.2-3

REACTOR VESSEL DESIGN DATA

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, psig	3107
Design Temperature, °F	650
Overall Height of Vessel and Closure Heat, ft-in. (bottom head OD to top of control rod mechanism adapter)	43-10
Thickness of Insulation, min., in.	3
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head Studs, in.	7
ID of Flange, in.	172.5
OD of Flange, in.	205
ID at Shell, in.	173
Inlet Nozzle ID, in	27-1/2
Outlet Nozzle ID, in.	29
Clad Thickness, min., in.	5/32

TABLE 5.2-3 (Cont.)

Lower Head Thickness, min., in. (base metal)	5-3/8	
Vessel Belt-Line Thickness, min., in. (base metal)	8.5	
Closure Heat Thickness, in.	7	
	Low T_{avg}	High T_{avg}
Reactor Coolant Inlet Temperature, °F	530.2	542.7 (Unit 1)
	530.3	542.8 (Unit 2)
Reactor Coolant Outlet Temperature, °F	601.8	613.1 (Unit 1)
	601.7	613.1 (Unit 2)
Reactor Coolant Flow, lb/hr	127.3×10^6	125.3×10^6 (Unit 1)
	127.9×10^6	125.8×10^6 (Unit 2)
Total Water Volume Below Core, ft^3	1050	
Water Volume in Active Core Region, ft^3	665	
Total Water Volume to Top of Core, ft^3	2164	
Total Water Volume to Coolant Piping Nozzles Centerline, ft^3	2959	
Total Reactor Vessel Water Volume, (with core and internals in place), ft^3	4945	
Total Reactor Coolant System Volume, ft^3	12,076 (Unit 1)	
Total Reactor Coolant System Volume, ft^3	13,011 (Unit 2)	

TABLE 5.2-4

PRESSURIZER AND PRESSURIZER RELIEF TANK DESIGN DATA

Pressurizer	
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design/Operating Temperature, °F	680/653
Water Volume, Full Power, ft^3*	1080
Steam Volume, Full Power, ft^3	720
Surge Line Nozzle Diameter, in.	14
Shell ID, in.	84
Electric Heaters Capacity, kW	1800
Heatup Rate of Pressurizer (using heaters only)°F/hr	55 (approximately)
Maximum spray rate, gpm	800
Pressurizer Relief Tank	
Design Pressure, psig	100
Rupture Disc Release Pressure, psig	100
Design Temperature, °F	340
Normal Water Temperature, °F	Containment Ambient (120°F max.)
Total Volume, ft^3	1800
Total Rupture Disc Relief Capacity, lb/hr	1.60×10^6

*60 percent of net internal volume (maximum calculated power)

TABLE 5.2-5

STEAM GENERATOR DESIGN DATA* #
(AREVA NP Model 61/19T) - Unit 2 Only

Number of Steam Generators	4	
Design Pressure (Reactor coolant/steam), psig	2485/1185	
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig	3107	
Design Temperature (reactor coolant/steam), °F	650/600	
	Low T_{avg}	High T_{avg}
Reactor Coolant Flow, lb/hr	31.97×10^6	31.46×10^6
Total Heat Transfer Surface Area, ft^2	66,236	
Heat Transferred, Btu/hr	2961×10^6	2961×10^6
Steam Conditions at Full Load, Outlet Nozzle:		
Steam Flow, lb/hr	3.76×10^6	3.77×10^6
Steam Pressure, PSIA	805	900
Maximum Moisture Carryover, wt percent	<0.1	
Feedwater, °F	432.8	432.8
Shell OD (upper/lower)Maximum, in.	175.75 / 135.0	
Number of U-tubes	5048	
U-tube OD, in.	0.750	
Tube Wall Thickness (minimum), in.	0.043	
Number of Manways/ID, in.	4/16	
Number of handholes/ID, in.	4/6	
Number of Inspection Ports/ID, in.	13 / 2.5	

*Quantities are for each steam generator

#Values are based on thermal design flow

TABLE 5.2-5 (Cont)

STEAM GENERATOR DESIGN DATA*#
(AREVA NP Model 61/19T) - Unit 2 Only

	Rated Load	HISTORICAL INFORMATION No Load
Reactor Coolant Water Volume, ft^3	1201	1080
Primary Side Fluid Heat Content, Btu	42.34×10^6	27.7×10^6
Secondary Side Water Volume, ft^3	2101	3524
Secondary Side Steam Volume, ft^3	3496	2344
Secondary Side Steam Fluid Heat Content, Btu	6.113×10^7	9.628×10^7

*Quantities are for each steam generator

#Values are based on thermal design flow

TABLE 5.2-5a

STEAM GENERATOR DESIGN DATA*
(Model F) - Unit 1 Only

Number of Steam Generators	4	
Design Pressure (Reactor coolant/steam), psig	2485/1185	
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig	3107	
Design Temperature (reactor coolant/steam), °F	650/600	
	Low T_{avg}	High T_{avg}
Reactor Coolant Flow, lb/hr	31.83×10^6	31.33×10^6
Total Heat Transfer Surface Area, ft^2	55,050	
Heat Transferred, Btu/hr	2961×10^6	2961×10^6
Steam Conditions at Full Load, Outlet Nozzle:		
Steam Flow, lb/hr	3.76×10^6	3.78×10^6
Steam Temperature, °F	515.0	527.8
Steam Pressure, PSIA	778	869
Maximum Moisture Carryover, wt percent	0.25	
Feedwater, °F	432.8	432.8
Overall Height, ft-in.	67-8	
Shell OD (upper/lower), in.	176.25 / 135.42	
Number of U-tubes	5626	
U-tube OD, in.	0.688	
Tube Wall Thickness (minimum), in.	0.041	
Number of Inspection Openings/ID, in.	4/2.7	
Number of Manways/ID, in.	4/16	
Number of handholes/ID, in.	6/6	
Reactor coolant Volume, ft^3 (Rated Load)	966.1	
Reactor coolant Volume, ft^3 (No Load)	966.1	

*Quantities are for each steam generator

TABLE 5.2-6

REACTOR COOLANT PUMPS DESIGN DATA
(Model 93A)

Number of Pumps	4
Design Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design Temperature (casing), °F	650
RPM at Nameplate Rating	1180
Suction Temperature, °F	559
Developed Head, ft	277
Capacity, gpm	88,500
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump Discharge Nozzle ID, in.	27 1/2
Pump Suction Nozzle ID, in.	31
Overall Unit Height, ft-in	25-5 1/4
Water Volume, ft^3	56
Pump-Motor Moment of Inertia, $lb\text{-}ft^2$	82,000
Motor Data:	
Type	AC Induction Single Speed, Air Cooled
Voltage	4160
Insulation Class	B Thermalastic Epoxy
Phase	3
Frequency, cps	60
Starting	
Current, amp	4800
Input (hot reactor coolant), kW	4260
Input (cold reactor coolant), kW	5690
Power, Hp (nameplate)	6000
Pump Weight, lb (dry)	169,200

TABLE 5.2-7

REACTOR COOLANT PIPING DESIGN PARAMETERS

Reactor Inlet Piping ID, in.	27.5
Reactor Inlet Piping Nominal Thickness, in.	2.38
Reactor Outlet Piping ID, in.	29
Reactor Outlet Nominal Thickness, in.	2.50
Coolant Pump Suction Piping ID, in.	31
Coolant Pump Suction Piping Nominal Thickness, in.	2.66
Pressurizer Surge Line Piping ID, in.	(1)
Pressurizer Surge Line Piping Nominal Thickness, in.	(2)
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (Cold), psig	3107
Design Temperature, °F	650
Design Temperature (pressurizer surge line), °F	680

Water Volume, (all 4 loops including surge line), ft^3	1455
Design Pressure (pressurizer relief lines), psig	(3)
Design Temperature (pressurizer relief lines), °F	(3)

(1) Unit 1 11.500″, Unit 2 11.188″

(2) Unit 1 1.25″, Unit 2 1.406″

(3) From pressurizer to safety valve 2485 psig 650°F.
From safety valve to pressurizer relief tank 600 psig 600°F.

TABLE 5.2-8

PRESSURIZER VALVES DESIGN PARAMETERS

PRESSURIZER SPRAY CONTROL VALVES

Number of Valves	2/Unit
Design Pressure	2485 psig
Design Temperature	650°F
Design Flow (valves full open, each)	400 gpm
Fluid Temperature	545°F
Position (after failure of actuating force)	Closed

SAFETY VALVES

1. VALVE PARAMETERS

Number of Valves	3/Unit
Manufacturer	Crosby Valve and Gage Co.
Type	Crosby HB-BP-86 6M6 Safety Valve (Steam Internals)
Point	2485 psig
Size	6" Inlet x 6" Outlet Orifice Size = 2.154 (3.644 sq. in.^{2})
Rated Capacity (Saturated Steam)	420,000 lb/hr each
Design Pressure and Temp.	2485 psig and 650°F
Constant Back Pressure	
Normal	3-5 psig
Developed	350 psig
Inlet Flange Rating	1500 #ASA
Discharge Flange Rating	600 #ASA

2. INLET PIPING PARAMETERS

Diameter	6" Sch 160	
Length	Unit 1	Unit 2
Loop 3	14.553'	12.054'
Loop 4	12.873'	12.241'
Loop 5	12.309'	11.719'

POWER OPERATED RELIEF VALVES

Number of Valves	2/Unit
Manufacturer	Copes-Vulcan Division
Type	Diaphragm Operated Relief Valve
Set Point	*2335 psig
Size	2" Valve with 3" inlet and outlet BW connection Orifice 2"
Rated Capacity (Saturated Steam)	210,000 lb/hr at 2335 psig
Design Pressure and Temp.	2485 psig and 680° F
Valve	1500 #ASA

TABLE 5.2-8 (Cont.)

PORV BLOCK VALVES

Number of Valves	2/Unit
Valve Manufacturer	Velan Engineering Co.
Operator Manufacturer	Limitorque
Type	3" Motor Operated Gate Valve 3GM58FN with BW ends and SMB-00-15 motor operator
Valve Rating	1500 #ASA

* Pressurizer Relief Valves lift at 2335 psig and reset at 2315 psig.

TABLE 5.2-9A

UNIT 1 REACTOR COOLANT SYSTEM - CODES

Component	Code	Date & Addenda	Code Cases
Reactor Vessel	ASME III	1965 & all thru Winter 1965	All applicable in effect prior to 4/26/66
Replacement Reactor Vessel Closure Head	ASME III	1998 & all thru Summer 2000	--
Steam Generator*	ASME III	1971 & all thru Summer 1973	All applicable in effect prior to 1971, 1484-3, 1528-3 & N474-1
F/L CRDMs	ASME III	1965 & all thru Summer 1966	--
RC Pump	No Code	(Design per ASME III, Article 4)	--
Pressurizer	ASME III	1965 & all thru Winter 1966	All applicable in effect at the time
Przr Relief Tank	ASME III	1968 & all thru Summer 1968	All applicable in effect at the time
Przr Safety Valves	ASME III	1968 & all thru Summer 1968	--
RC Piping	ASA B31.1	1955	Applicable portions of ASA N-7 and N-10

TABLE 5.2-9A

UNIT 1 REACTOR COOLANT SYSTEM - CODES

Component	Code	Date & Addenda	Code Cases
Sys Ppg & Fittings	ASA B31.1	1955	Applicable portions of ASA N-7 and N-10
System Valves	ASA B16.5, or	1964	Applicable portions of N-10
	MSS-SP-66, or	1964	--
	ASME III	1968	--

* The steam generators were procured and installed in accordance with NRC GL 89-09 to meet ASME III Section III Class 1 requirements. Lower narrow range level taps conform to 1989 ASME Section III Class 1 reconciled to the original construction code. The tube side and the shell side conform to the requirements of ASME Section III for Class 1 vessels. The steam generators were NPT stamped by the manufacturer prior to hydrostatic testing. The tube side and the shell side were subsequently hydrostatic pressure tested prior to installation at Unit 1. The primary piping to steam generator primary inlet and outlet welds conform to the requirements of the 1989 Edition of the ASME Code Section III for Class 1 piping. Applicable Code Cases are N-416-1 and N-389.

TABLE 5.2-9B

REACTOR COOLANT SYSTEM - CODES
UNIT 2

Component	Code	Date and Addenda	Code Cases
Reactor Vessel	ASME III	1965 and all addenda through Winter 1966	All applicable in effect prior to 4/3/67
Reactor Vessel Closure Head	ASME III	1998 with addenda through 2000	--
Steam Generator	ASME III	1995 and all addenda through 1996	N-20-4
F/L CRDMs	ASME III	1998 with addenda through 2000	--
Reactor Coolant Pump Casing	ASME III		
Pressurizer	ASME III	1965 and all addenda through Winter 1966	All applicable in effect at the time
Pressurizer Relief Tank	ASME III	1968 and all addenda through Summer 1968	All applicable in effect at the time
Pressurizer Safety Valves	ASME III	1968 and all addenda through Summer 1968	--
Reactor Coolant Piping	USAS B31.1.0	1967#	Applicable portions of ASA N-7 and N-10
System Piping and Fittings	USAS B31.1.0	1967#	Applicable portions of ASA N-7 and N-10
System Valves	B16.5, or MSS-SP-66, or ASME III	1964 1964 1968	Applicable portions of N-10 -- --

#RCS piping fabrication, installation, welding, and examination involved in installing the Unit 2 Replacement Steam Generators utilized ASME Section XI (1998 Edition with 2000 Addenda) and ASME Section III, Subsection NB (1995 Edition with 1996 Addenda). Both of these later codes are NRC-endorsed per 10CFR 50.55a and were reconciled to the original construction codes.

TABLE 5.2-10

DESIGN THERMAL AND LOADING CYCLES*(1)
AREVA NP Model 61/19T SG - Unit 2

		Design Cycles**
1.	Heatup at 100°F/hr	200
	Cooldown at 100°F/hr (Pressurizer 200°F/hr)	200
2.	Unit Loading at 5 Percent of Full Power/Min	18,300
	Unit Unloading at 5 Percent of Full Power/Min	18,300
2a.	Unit Loading at 5 Percent of Full Power/Min	14,500 (Unit 2 RRVCH)
	Unit Unloading at 5 Percent of Full Power/Min	14,500 (Unit 2 RRVCH)
3.	Step Load Increase of 10 Percent of Full Power	2,000
	Step Load Decrease of 10 Percent of Full Power	2,000
4.	50 Percent Step Decrease in Load (with steam dump)	200
5.	Loss of Load (without immediate turbine or reactor trip)	80
6.	Loss of Power (blackout with natural circulation in the RCS)	40
7.	Loss of Flow (partial loss of flow one pump only)	80
8.	Reactor Trip From Full Power	400
9.	Turbine Roll Test	10
10.	Hydrostatic Test Conditions	
	a. Primary Side Hydrostatic Test Shop and Field	10
	b. Secondary Side Hydrostatic Test Before Initial Startup	10
11.	Primary Side Leak Test	50
12.	Accident Conditions	
	a. Reactor Coolant Pipe Break	1
	b. Steam Pipe Break	1
	c. Steam Generator Tube Rupture	1

TABLE 5.2-10 (Cont)
DESIGN THERMAL AND LOADING CYCLES*[(1)]
AREVA NP Model 61/19T SG - Unit 2

		Design Cycles**
13.	Steady State Fluctuations - the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psi. It is assumed that an infinite number of such fluctuations will occur.	
14.	Design Earthquake Cycles	
	a. Operating Basis Earthquake	50
	b. Design Basis Earthquake	10

* The ASME Section III Nuclear Power Plant Components Code is inapplicable to the Salem Station; hence, the normal, upset, emergency, and faulted conditions terminology does not apply to the transients identified in this table. However, since the RCS vessels (reactor vessel, pressurizer, and steam generators) are basically standard components, analysis on these vessels with the more recent ASME Code conditions (normal, upset, emergency, and faulted) have been performed as discussed in Sections 5.1.2.8.1 and 5.1.2.8.2.

** Estimated for equipment design purposes (40-year life) and not intended to be an accurate representation of actual transients or to reflect actual operating experience.

(1) Note that the actual transient definition for design purposes may be more stringent than listed in this table.

TABLE 5.2-10a

DESIGN THERMAL AND LOADING CYCLES*
Model F SG - Unit 1

		Design Cycles**
1.	Heatup at 100°F/hr	200
	Cooldown at 100°F/hr (Pressurizer 200°F/hr)	200
2.	Unit Loading at 5 Percent of Full Power/Min	13,200***
	Unit Unloading at 5 Percent of Full Power/Min	13,200***
2a.	Unit Loading at 5 Percent of Full Power/Min	14,500 (Unit 1 RRVCH)
	Unit Unloading at 5 Percent of Full Power/Min	14,500 (Unit 1 RRVCH)
3.	Step Load Increase of 10 Percent of Full Power	2,000
	Step Load Decrease of 10 Percent of Full Power	2,000
4.	50 Percent Step Decrease in Load (with steam dump)	200
5.	Loss of Load (without immediate turbine or reactor trip)	80
6.	Loss of Power (blackout with natural circulation in the RCS)	40
7.	Loss of Flow (partial loss of flow one pump only)	80
8.	Reactor Trip From Full Power	400
9.	Turbine Roll Test	10
10.	Hydrostatic Test Conditions	
	a. Primary Side Hydrostatic Test Before Initial Startup	5
	b. Secondary Side Hydrostatic Test Before Initial Startup	5
11.	Primary Side Leak Test	50
12.	Accident Conditions	
	a. Reactor Coolant Pipe Break	1
	b. Steam Pipe Break	1
	c. Steam Generator Tube Rupture	1

DESIGN THERMAL AND LOADING CYCLES*
Model F SG - Unit 1

		Design Cycles**
13.	Steady State Fluctuations - the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psi. It is assumed that an infinite number of such fluctuations will occur.	
14.	Design Earthquake Cycles	
	a. Operating Basis Earthquake	50
	b. Design Basis Earthquake	10

* The ASME Section III Nuclear Power Plant Components Code is inapplicable to the Salem Station; hence, the normal, upset, emergency, and faulted conditions terminology does not apply to the transients identified in this table. However, since the RCS vessels (reactor vessel, pressurizer, and steam generators) are basically standard components, analysis on these vessels with the more recent ASME Code conditions (normal, upset, emergency, and faulted) have been performed as discussed in Sections 5.1.2.8.1 and 5.1.2.8.2.

** Estimated for equipment design purposes (40-year life) and not intended to be an accurate representation of actual transients or to reflect actual operating experience.

*** Model F steam generators on Unit 1 are designed to 13,200 cycles.

TABLE 5.2-11

SUMMARY OF PLANT OUTAGE FOR YANKEE ROW (1964 to 1969)

Starting Date	Duration Days	Duration Hours	Outage Type	Case Equipment/System
1/17/64	-	3.1	Forced	Turbine Trip
2/12/64	-	21.8	Scheduled	Control Rod Drop Testing
3/11/64	-	4.5	Forced	Moisture Separator Level Switch Tripped due to Vibration
3/26/64	-	4	Forced	Control Valves Sticking
5/18/64	-	5.4	Forced	Low Condensate Pump Discharge pressure
8/2/64	35	-	Scheduled	Refueling and General Maintenance
9/9/64	-	2.4	Scheduled	Check of Overspeed Trip
9/11/64	-	14.7	Forced	Spurious Reactor Trip
10/18/64	-	12.2	Forced	Condenser Noise
10/22/64	-	22.4	Forced	Neutron Counter Gain Control
2/12/65	-	15.2	Forced	Switchyard Electric
3/5/65	-	-	Scheduled	Switchyard Electric
8/9/65	93	6	Scheduled	Refueling
11/26/65	2	20	Scheduled	Turbine Repair-Physics Testing
2/4/66	-	3.12	Forced	Reactor Scram
4/4/66	-	89.5	Scheduled	Leaking Pressurizer Safety Valves
7/10/66	-	3.68	Forced	Reactor Scram
8/25/66	-	2.40	Forced	Reactor Scram

TABLE 5.2-11 (Cont)

Starting Date	Duration Days	Duration Hours	Outage Type	Case Equipment/System
10/4/66	34	10.23	Scheduled	Refueling
12/24/66	-	2.88	Forced	Reactor Scram
12/28/66	-	2.12	Forced	Reactor Scram
3/8/67	11	21	Scheduled	Steam Generator Leak Repair
5/12/67	-	16.87	Scheduled	Condenser Cleaning
7/9/67	17	1.5	Scheduled	Steam Generator Leak Repairs
10/28/67		9	Scheduled	AEC Operator Examinations
10/13/67	-	2.6	Forced	Reactor Scram
3/23/68	38	-	Scheduled	Core VI-VII Refueling and Maintenance
7/20/68	1	10	Scheduled	Repair Leak from No. 1 Main Coolant Pump Stator Cap
11/8/68	6	16.42	Scheduled	Repair No. 4 Main Coolant Pump Thermal Barrier Leak and other Maintenance
1/18/69	1	2.1	Scheduled	Operator Training
2/15/69	1	1.8	Scheduled	Operator Training
3/1/69	-	11	Scheduled	AEC Operator Examination
4/11/69	4	18	Forced	Repair Reactor Instrument Leak
7/17/69	-	4.8	Forced	Reactor Scram
8/2/69	53	18.5	Scheduled	Refueling Maintenance
10/16/69	-	6.1	Forced	Reactor Scram
10/29/69	-	12	Scheduled	Turbine Valve Flange Steam Leak Repair

TABLE 5.2-12

LOAD COMBINATIONS AND STRESS LIMITS

	Load Combination	Stress Limit*
1.	Normal (deadweight, thermal and pressure)	Normal Conditions
2.	Normal and Operation Basis Earthquake	Upset Condition
3.	Normal and Design Basis Earthquake	Faulted Condition
4.	Normal and Pipe Rupture	Faulted Condition
5.	Normal and Design Basis Earthquake and Pipe Rupture	Faulted Condition

1. Normal Condition - Any condition in the course of system startup, operation in the design power range and system shutdown, in the absence of Upset, Emergency, or Faulted Conditions.

2. Upset Condition - 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 Condition includes those transients caused by a fault in a system component requiring its isolation from the system, transients due to a loss of load or power and any system upset not resulting in a forced outage. The estimated duration of an Upset Condition shall be included in the Design Specifications. The Upset Conditions include the effect of the specified earthquake for which the system must remain operational or must regain its operational status.

3. Emergency Condition - Any deviations from Normal Conditions which 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 shall not exceed twenty-five.

4. Faulted Condition - 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 where considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be a specified by jurisdictional authorities. Among the Faulted Conditions may be a specified earthquake for which safe shutdown is required.

*Definition of Operating Condition categories from Summer 1968 Addenda to the ASME Boiler and Pressure Vessel Code, Section III.

TABLE 5.2-13

LOADING CONDITIONS AND STRESS LIMITS: PRESSURE VESSELS

Loading Conditions	Stress Intensity Limits	Note
1. Normal Condition	(a) $P_m \leq S_m$	
	(b) P_m (or P_L) + $P_B \leq 1.5S_m$	1
	(c) P_m (or P_L) = $P_B + Q$ $\leq 3.0S_m$	2
2. Upset Condition	(a) $P_m \leq S_m$	
	(b) P_m (or P_L) + $P_B \leq 1.5S_m$	1
	(c) P_m (or P_L) = $P_B + Q$ $\leq 3.0S_m$	2
3. Emergency Condition	(a) $P_m \leq 1.25S_m$ or S_y, whichever is larger	
	(b) P_m (or P_L) + $P_B \leq 1.5(1.2S_m)$ or $1.5S_y$, whichever is larger	
4. Faulted Condition	Design Limit Curves as discussed in the text and attached. For the Unit 2 RSG, ASME criteria were Applied.	4,5

where:

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 for ASME B and PV Code, Section III, Nuclear Vessels

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

TABLE 5.2-13 (Cont.)

LOADING COMBINATIONS AND STRESS LIMITS: PRESSURE PIPING

Loading Conditions	Stress Intensity Limits
1. Normal Conditions	(a) $P_m \leq S$
	(b) P_m (or P_L) + $P_B \leq S$
2. Upset Conditions	(a) $P_m \leq 1.25$
	(b) P_m (or P_L) + $P_B \leq 1.2S$
3. Emergency Conditions	(a) $P_m \leq 1.2S$
	(b) P_m (or P_L) + $P_B \leq (1.5)(1.2)\ S$
4. Faulted Conditions	Design Limit Curves as discussed in the text and attached. OR Maximum stress $\leq$ 2.4 S

where:

P_m = primary general membrane stress intensity

P_L = primary local membrane stress intensity

P_B = primary bending stress intensity

S = allowable stress from USASI B31.1 Code for Pressure Piping

LOADING CONDITIONS AND STRESS LIMITS: EQUIPMENT SUPPORTS

Loading Conditions	Stress Intensity Limits
1. Normal Condition	Working Stresses or Applicable Factored Load Design Values
2. Upset Condition	Working Stress or Applicable Factored Load Design Values
3. Emergency Condition	Within yield after load redistribution
4. Faulted Condition	Permanent Deflection of Supports Limited To Maintain Supported Equipment Within Design Limit Curves as Discussed in the Text.

Note 1: The limits on local membrane stress intensity ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_L (or P_L) + $P_B \leq 1.5S_m$) 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 of the lower bound collapse load as per paragraph N-417.6 (b) of the ASME B and PV Code, Section III, Nuclear Vessels.

Note 2: In lieu of satisfying the specific requirements for the local membrane ($P_L \leq 1.5S$) or the primary plus secondary stress intensity ($P_L + P_B + 0 \leq 3S_m$) 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

LOADING CONDITIONS AND STRESS LIMITS: EQUIPMENT SUPPORTS

paragraph N-417.6(a) (2) of the ASME B and PV Code, Section III, Nuclear Vessels.

Note 3: The limits on local membrane stress intensity ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_M (or P_L) + P_B $\leq 1.5S_m$) 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 120 percent of 2/3 of the lower bound collapse load as per paragraph N-417.10 (c) of the ASME B and PV Code, Section III, Nuclear Vessels.

Note 4: As an alternate to the design limit curves which represent a pseudo plastic instability analysis, a plastic instability analysis may be performed in some specific cases considering the actual strain-hardening characteristics of the material, but with yield strength adjusted to correspond to the tabulated value at the appropriate temperature in Table N-424 or N-425, as per paragraph N-417.11 (c) of the ASME B and PV Code, Section III, Nuclear Vessels. These specific cases will be justified on an individual basis.

Note 5: For the Unit 2 RSG, the faulted condition criteria is as provided in ASME Code Section III, Division 1, Sub-section NB, Appendix F, Sub-paragraph 1331.1.

TABLE 5.2-14

SUMMARY OF PRIMARY PLUS SECONDARY STRESS INTENSITY RANGES
FOR COMPONENTS OF THE REACTOR VESSEL

Area	Range of Stress Intensity (psi)	Allowable Stress (psi) (at Operating Temperatures)
Control Rod Housing	77,760 *	69,900
Head Flange	65,260	80,100
Vessel Flange	60,040	80,100
Primary Nozzles	57,090	80,100
Stud Bolts	109,400	110,400
Vessel Support	**	80,100
Core Support Pads	47,240	69,900
Bottom Head to Shell	34,690	80,100
Bottom Instrumentation Penetrations	67,110	69,900
Vessel Wall Transition	33,730	80,100

* Justified by simplified elastic-plastic analysis

** Lower than primary nozzle stress

TABLE 5.2-15

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR COMPONENTS OF THE REACTOR VESSEL

Item	Usage Factor*,**
Control Rod Housing	0.71
Replacement Head Flange	0.263
RVLIS/RVHV Penetrations	0.75
Vessel Flange	0.183
Stud Bolts	0.89
Primary Nozzles	0.1510
Vessel Support Pads	0.05
Core Support Pads (lateral)	0.012
Bottom Head to Shell	0.0118
Bottom Instrumentation Penetrations	0.1002

*Covers all transients

**As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

TABLE 5.2-16

STRESS DUE TO MAXIMUM STEAM GENERATOR TUBE
SHEET PRESSURE DIFFERENTIAL (2485 PSIG)
Model 61/19T - Unit 2

Note - Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for the required stresses and stress factors for the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-17

RATIO OF ALLOWABLE STRESS TO COMPUTED STRESSES
FOR A STEAM GENERATOR TUBE
SHEET PRESSURE DIFFERENTIAL OF 2485 PSIG
Model 61/19T - Unit 2

Note - Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for the stress analyses associated with the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-18

STEAM GENERATOR PRIMARY-SECONDARY BOUNDARY COMPONENTS
Model 61/19T - Unit 2

Note - Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for tabulations associated with the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-19

STEAM GENERATOR PRIMARY-SECONDARY COMPONENTS
Model 61/19T - Unit 2

Note - Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for tabulations associated with the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-20

STEAM GENERATOR PRIMARY-SECONDARY BOUNDARY COMPONENTS
Model 61/19T - Unit 2

Note - Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for tabulations associated with the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-21

STEAM GENERATOR PRIMARY - SECONDARY BOUNDARY COMPONENTS
Model 61/19T - Unit 2

Note - Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for tabulations associated with the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-22

MODEL 61/19T STEAM GENERATOR
USAGE FACTORS (INDIVIDUAL TRANSIENTS)
PRIMARY AND SECONDARY BOUNDARY COMPONENTS

UNIT 2

Note - Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for tabulations associated with the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-23

MODEL 61/19T STEAM GENERATOR
USAGE FACTORS (INDIVIDUAL TRANSIENT)

UNIT 2

Note - Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for tabulations associated with the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-24

TUBE SHEET STRESS ANALYSIS RESULTS
FOR MODEL 61/19T STEAM GENERATORS
UNIT 2

Note - Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for tabulations associated with the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-25

LIMIT ANALYSIS CALCULATION RESULTS
TABLE OF STRAINS, LIMIT PRESSURES, AND FATIGUE EVALUATIONS FOR MODEL 61/19T STEAM GENERATORS
UNIT 2

Note – Refer to VTD 900013 Sheet 1, "Salem Unit 2 RSG - Design Report," (Reference 31) for tabulations associated with the Salem Unit 2 Model 61/19T steam generators.

TABLE 5.2-26
REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component		RT*	UT*	PT*	MT*	ET*	LT*
1.	Steam Generator						
1.1	Tube Sheet						
1.1.1	Forging		yes		yes		
1.1.2	Cladding		yes(1)	yes(2)			
1.2	Channel Head						
1.2.1a	Casting (Unit 1)	yes			yes		
1.2.1b	Forging (Unit 2)		yes		yes		
1.2.2a	Cladding (Unit 1)			yes			
1.2.2b	Cladding (Unit 2)		yes	yes			
1.3	Secondary Shell and Head						
1.3.1a	Plates (Unit 1)		yes				
1.3.1b	Forging (Unit 2)		yes		yes		
1.4	Tubes		yes			yes	
1.5	Nozzles (forgings)		yes		yes		
1.6	Weldments						
1.6.1	Shell, longitudinal (Unit 1)	yes			yes		
1.6.2a	Shell, circumferential (Unit 1)	yes			yes		
1.6.2b	Shell, circumferential (Unit 2)	yes	yes	yes			
1.6.3a	Cladding (channel head-tube sheet joint cladding restoration)(Unit 1)			yes			
1.6.3b	Cladding (channel head-tube sheet joint cladding restoration) (Unit 2)		yes	yes			
1.6.4a	Steam and Feedwater Nozzle to Shell (Unit 1)	yes			yes		
1.6.4b	Steam and Feedwater Nozzle to Shell (Unit 2)	yes	yes	yes			
1.6.5a	Support brackets (Unit 1)				yes		
1.6.5b	Support brackets (Unit 2)			yes			
1.6.6a	Tube to Tube Sheet (Unit 1)			yes			
1.6.6b	Tube to Tube Sheet (Unit 2)	yes		yes			yes
1.6.7a	Instrument Connections (Unit 1) (secondary) for lower NR level taps. No primary connections.	yes		yes	yes		
1.6.7b	Instrument Connections (Unit 2) (primary and secondary).			yes			

TABLE 5.2-26 (Cont)

Component	RT*	UT*	PT*	MT*	ET*	LT*
1.6.8a Temporary Attachments After Removal (Unit 1)				yes		
1.6.8b Temporary Attachments After Removal (Unit 2)			yes			
1.6.9a After Hydrostatic Test (all welds and complete channel head - where accessible) (Unit 1)				yes		
1.6.9b After Hydrostatic Test (all welds and complete channel head - where accessible) (Unit 2)			yes			
1.6.10 Nozzle Safe Ends (weld deposit) yes			yes			
2. Pressurizer						
2.1 Heads						
2.1.1 Casting	yes			yes		
2.1.2 Cladding			yes			
2.2 Shell						
2.2.1 Plates		yes		yes		
2.2.2 Cladding			yes			
2.3 Heaters						
2.3.1 Tubing(4)		yes	yes			
2.3.2 Centering of element	yes					
2.4 Nozzle		yes	yes			
2.5 Weldments						
2.5.1 Shell, longitudinal	yes			yes		
2.5.2 Shell, circumferential	yes			yes		
2.5.3 Cladding			yes			
2.5.4 Nozzle Safe End (if forging)	yes		yes			
2.5.5 Nozzle Safe End (if weld deposit)			yes			
2.5.6 Instrument Connections			yes			
2.5.7 Support Skirt				yes		
2.5.8 Temporary Attachments After Removal				yes		
2.5.9 All Welds and Cast Heads After Hydrostatic Test				yes		
2.6 Final Assembly						
2.6.1 All Accessible Surfaces After Hydrostatic Test				yes		

TABLE 5.2-26 (Cont)

Component		RT*	UT*	PT*	MT*	ET*	LT*
3.	Piping						
3.1	Fittings and Pipe (Castings)	yes		yes			
3.2	Fittings and Pipe (Forgings)		yes	yes			
3.3	Weldments						
3.3.1	Circumferential	yes		yes			
3.3.2	Nozzle to Runpipe (No RT for nozzles less than 4 inches)	yes		yes			
3.3.3	Instrument Connections		yes	yes			
4.	Pumps						
4.1	Casting	yes		yes			
4.2	Forgings						
4.2.1	Main Shaft		yes	yes			
4.2.2	Main Studs		yes	yes			
4.2.3	Flywheel (Rolled Plate)		yes				
4.3	Weldments						
4.3.1	Circumferential	yes		yes			
4.3.2	Instrument Connections			yes			
5.	Reactor Vessel						
5.1	Forgings						
5.1.1	Flanges		yes		yes		
5.1.2	Studs		yes		yes		
5.1.3	Head Adapters		yes	yes			
5.1.4	Head Adapter Tube		yes	yes			
5.1.5	Instrumentation Tube		yes	yes			
5.1.6	Main Nozzles		yes		yes		
5.1.7	Nozzle Safe Ends (if forging is employed)		yes	yes			
5.2	Plates		yes		yes		
5.3	Weldments						
5.3.1	Main Steam	yes			yes		
5.3.2	CRD Head Adapter Connection			yes			
5.3.3	Instrumentation Tube Connection			yes			
5.3.4	Main nozzles	yes			yes		
5.3.5	Cladding		yes (3)	yes			
5.3.6	Nozzle-Safe Ends (if forging)	yes		yes			

TABLE 5.2-26 (Cont)

Component		RT*	UT*	PT*	MT*	ET*	LT*
5.3.7	Nozzle Safe Ends (If weld deposit)	yes		yes			
5.3.8	Head Adaptor Forging to Head Adapter Tube	yes		yes			
5.3.9	All Welds After Hydrotest				yes		
6.	Valves						
6.1	Castings	yes		yes			
6.2	Forgings (No UT for valves two inch and smaller)		yes	yes			

* RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current
LT - Leak

(1) Flat Surfaces Only
(2) Weld Deposit Areas Only
(3) UT of Clad Bond-to-Base Metal
(4) Or a UT and ET

TABLE 5.2-27

MATERIALS CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Materials
Reactor Vessel	Pressure Plate - Unit 2	ASTM A-533 Grade B Class 1
	Pressure Plate - Unit 1	SA-302-B Grade B
	Pressure Forgings	ASTM A-508 Class 2
	Cladding, Stainless	Type 304 or equivalent
	Stainless Weld Rods	Type 308, 309, or 312
	O-Ring Head Seals	Inconel - 718
	Studs	SA-540 Grade B-24
	Instrumentation Nozzles	Inconel SB 167
	Insulation	Stainless Steel
Replacement Reactor Vessel Closure Head (Units 1 & 2)	Monoblock forging	SA-508 Gr. 3 Cl. 1
	CRDM Housing	SA-182 Type 304
	Lower Tube	SB-167, A690 (UNS N06690)
Steam Generator Unit 2 - AREVA NP Model 61/19T	Pressure Plate	SA-533, Type B, CL 2
	Pressure Forgings	SA-508, GR 3, CL 2
	Cladding for Heads	Stainless Steel 308L/309L
	Stainless Weld Rod	Stainless Steel 308L/309L
	Cladding for Tube Sheets	Inconel 600
	Tubes	Inconel 690 Thermally Treated
	Channel Head Divider Plate	SB-168 UNSN 06690
Steam Generator Unit 1 - Model F	Shell Material	SA-533 Class 2
	Forgings	SA-508 Class 2a
	Cladding for Heads,	Type 308 or 309 SS
	Stainless Weld Rod	Type 308L or 309L
	Cladding for Tube Sheets	Inconel
	Tubes	Inconel SB-163, Code Case 1484-3
	Channel Head Castings	SA-216 Grade WCC
Pressurizer	Shell	SA-533 Class 1
	Heads	SA-216 Grade WCC
	Support Skirt	SA-516 Grade 70
	Nozzle Weld Ends	SA-182 F316
	Inst. Tube Coupling	SA-182 F316
	Cladding, Stainless	Type 304 or equivalent
	Internal Plate	SA-240 Type 304
	Inst. Tubing	SA-213 Type 304
	Heater Well Tubing	SA-213 Type 316 Seamless
	Heater Well Adaptor	SA-182 F316
Pressurizer Relief Tank	Shell	ASTM A-285 Grade C
	Heads	ASTM A-285 Grade C
	Internal Coating	Amercoat 55
Pipe	Pipes	ASTM A-376 Type 316
	Fittings	ASTM A-351 Grade CF8M
	Nozzles	ASTM A-182 Grade F316
Pump	Shaft	ASTM A-182 Grade F347
	Impeller	ASTM A-351 Grade CF8
	Casing	ASTM A-351 Grade CF8
Valves	Pressure Containing Parts	ASTM A-351 Grade CF8M and ASTM A-182 Grade F316

TABLE 5.2-28

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Conductivity	Determined by the concentration of boric acid and alkali present.
Dissolved Oxygen*, ppm, max.	0.10 (Steady State) 1.00 (Transient)
Chloride, ppm, max	0.15 (Steady State) 1.50 (Transient)
Fluoride, ppm, max.	0.15 (Steady State) 1.50 (Transient)
Sulfate, ppm, max	0.15 (Steady State) 1.50 (Transient)
Hydrogen, cc (STP)/kg H_2O	25-50 **
Suspended Solids, ppm, max.	1.0 ***
pH Control Agent (Li^7OH)	Up to 3.5 ± 0.15 ppm steady state, in accordance with Station Lithium Program
Boric Acid as ppm B	Variable from 0 to ~4000

* Limit not applicable with temperature < 250°F

** Control range during operation. Hydrogen may be reduced to 15 cc/kg 24 hours prior to a planned outage.

*** Limit applies ONLY in Modes 1 & 2. Goal for continuous RCP operation is < 350 ppb.

TABLE 5.2-29

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TABLE 5.2-30

RPV BELTLINE REGION WELD CHEMISTRY FOR SALEM UNITS 1 AND 2

Weld Seam	Weld Wire Heat/Flux Lot	Cu (w/o)	Ni (w/o)	Basis
Salem 1				
2-042 A/B/C	34B009/3692 & 34B196/3692 Ni-200 Wire	0.18	1.04	CE NPSD-1039, Rev 2
9-042	13253/3791	0.22	0.73	CE NPSD-1119, Rev 1
3-042 A/B/C	34B009/3708 + Ni-200 wire	0.19	1.04	CE NPSD-1039, Rev 2
Salem 2				
2-442 A/B/C	13253/3833 & 20291/3833	0.219	0.735	CE NPSD-1119, Rev 1
9-442	90099/3977	0.197	0.060	CE NPSD-1039, Rev 2
3-442 A/B/C	21935/3889 & 12008/3889	0.213	0.867	CE NPSD-1119, Rev 1

TABLE 5.2-31

RPV BELTLINE REGION WELD MECHANICAL PROPERTIES FOR SALEM UNITS 1 AND 2

Unit	Weld Seam	Weld Wire Heat/Flux Lot	Initial RT_{NDT} (1) (°F)	Weld Flux Type	Unirradiated USE (ft-lb)
Salem 1					
	2-042	34B009/3692	-56	1092	96.2
	A/B/C	39B196/3692	-56	1092	
		Ni-200 Wire			
	9-042	13253/3791	-56	1092	112
	3-042	34B009/3708	-56	1092	112
	A/B/C	+ Ni-200 Wire			
Salem 2					
	2-442	13253/3833	-56	1092	96.2
	A/B/C	20291/3833	-56	1092	
	9-442	90099/3977	-56	0091	99.7
	3-442	21935/3889	-56	1092	114
	A/B/C	12008/3889	-56	1092	

(1) Generic RT_{NDT} for C-E's SAW Weld; $\sigma = 17$

TABLE 5.2-32

RPV BELTLINE REGION PLATE MATERIAL CHEMICAL AND MECHANICAL PROPERTIES FOR SALEM UNITS 1 AND 2

Plate No.	Chemical Composition[1]		Mechanical Properties[1]	Unirradiated USE (ft-lb)
	Cu (w/o)	Ni (w/o)	Initial RT_{NDT} (F)	
Salem 1				
B2402-1	0.24	0.53	45	91
B2402-2	0.24	0.53	-5	98
B2402-3	0.22	0.51	-3	104
B2403-1	0.19	0.48	4	93
B2403-2	0.19	0.49	18	83
B2403-3	0.19	0.48	6	85
Salem 2				
B4712-1	0.13	0.56	0	106
B4712-2	0.12	0.62	12	97
B4712-3	0.11	0.57	10	107
B4713-1	0.12	0.60	8	98
B4713-2	0.12	0.57	8	103
B4713-3	0.12	0.58	10	121

(1) Measured data. All tests performed by Westinghouse.

TABLE 5.2-33

STRESS RESULTS OF UNIT 1 TUBESHEET AND SHELL JUNCTIONS ANALYSIS

MODEL F SG

Thin Cast Head Model

	LOCATION[1]				
	1	3	4	6	7
CONDITION:					
Design	0.11[2] 1.0[4][7]	0.41[3] -	0.43[3] -	0.40[3] -	0.28[2] 0.61[4]
Normal and Upset Fatigue Usage	(6) <0.71	0.97[5] <0.38	(6) <0.41	0.71[5] <0.25	0.65[5] <0.25
Emergency	0.05[2] 0.72[4]	0.28[3] -	0.51[3] -	0.23[3] -	0.10[2] 0.25[4]
Faulted	0.03[2] 0.77[4]	0.27[3] -	0.34[3] -	0.33[3] -	0.14[2] 0.61[4]
Test	0.07[2] 0.95[4]	0.37[3] -	0.65[3] -	0.29[3] -	0.08[2] 0.23[4]

Notes:

(1) See Figure 5.2-22

(2) P_M/Allowable

(3) P_L/Allowable

(4) $(P_L + P_b)$/Allowable

(5) $(P_L + P_b + Q)$/Allowable

TABLE 5.2-33 (Cont)

Notes (Cont):

(6) The $3S_M$ limit on $P_L + P_b + Q$ stress intensity range was exceeded. However, the provisions of Paragraph NB-3228.3 (Simplified elastic-plastic analysis) of Reference 1 were satisfied.

(7) Satisfied 2/3 the lower bound collapse load of NB-3228.2 of Reference 1.

TABLE 5.2-34

UNIT 1 SECONDARY SHELL AND TRANSITION CONE STRESS RESULTS

MODEL F SG

CONDITION:		SECTIONS (Figure 5.2-23) A-A	B-B	C-C	D-D	E-E	p + d[1]
Design		0.94[2]	0.33[3]	0.94[2]	0.78[3] 0.76[5]	0.94[2]	0.72[4] 0.64[5]
Normal &[6] Upset	Inside	0.41	0.47	0.68	0.78	0.71	-
	Outside	0.44	0.57	0.85	(7)	0.87	0.87
Fatigue	Inside	<0.01	<0.03	<0.01	<0.03	<0.01	<0.03
	Outside	<0.01	<0.01	<0.01	<0.02	<0.01	-
Emergency		0.48[2]	0.17[3]	0.47[2]	0.40[3]	0.48[2]	-
Faulted		0.47[2]	0.16[3]	0.47[2]	0.39[3] 0.77[5]	0.47[2]	0.88[4] 0.76[5]
Test		0.63[2]	0.22[3]	0.63[2]	0.53[3]	0.63[2]	-

Notes: (1) At Upper Lateral Load Pad location. Not shown in Figure 5.2-23.

(2) P_M/Allowable

(3) P_L/Allowable

(4) $(P_M + P_b)$/Allowable

(5) $(P_L + P_b)$/Allowable

(6) $(P_L + P_b + Q)$/Allowable

(7) The maximum primary + secondary stress intensity range exceeds the allowable stress limit. Therefore, a simplified elastic-plastic analysis was performed. This analysis is reflected in the cumulative usage factor calculations.

TABLE 5.2-35

UNIT 1 STRESS RESULTS OF TUBE ANALYSIS

MODEL F SG

CONDITION:	LOCATION[1] A-A	B-B	C-C	D-D	E-E
Design	0.60[2]	0.62[2]	0.88[3]	0.997[3]	0.60[2]
Normal/Upset	0.96	0.92[4]	0.85[4]	0.67[4]	0.84
Fatigue Usage	0.88	0.53	0.46	0.22	0.22
Emergency	0.67[2]	0.69[2]	0.74[2]	0.80[2]	0.67[2]
Faulted (LOCA + SSE)	0.17[2]	0.17[2]	0.99[3]	0.96[3]	0.17[2]
Faulted (FLB + SSE)	0.47[2]	0.48[2]	0.51[3]	0.55[3]	0.47[2]
Test	0.91[2]	0.94[2]	0.99[2]	0.68[2]	0.91[2]

Notes: (1) See Figure 5.2-24

(2) P_M/Allowable

(3) $(P_M + P_b)$/Allowable

(4) $(P_L + P_b + Q)$/Allowable

Table 5.2-36

UNIT 1 TUBE ANALYSIS FOR EXTERNAL PRESSURE

MODEL F SG

CONDITION	σ_y (ksi)	Actual ΔP (psi)	Criteria Used	Allowable Pressure Differential, psi at these Sections[1] A-A	B-B	C-C	D-D	E-E
Design	-	670	P_a	780	780	780	780	780
Emergency	-	537[2]	$1.2P_a$	936	936	936	936	936
Faulted	35.3	985[3]	$0.9P_c$	2602	2523	2424	1531	2602
Test	38.9	1481[4]	$0.8P_c$	2549	2471	2374	1500	2549

Notes: (1) See Figure 5.2-24

(2) Small LOCA

(3) Large LOCA

(4) Secondary Hydrotest

TABLE 5.2-37

SALEM 1 PREDICTED EOL RT_{NDT} FOR REACTOR VESSEL BELTLINE MATERIALS

Material	Initial RT_{NDT}[1] (°F)	EOL RT_{NDT} ¼T[2] (°F)	EOL RT_{NDT} ¾T[2] (°F)
Intermediate Shell B2402-1	45	215	171[3]
Intermediate Shell B2402-2	-5	151	113
Intermediate Shell B2402-3	-3	119	89
Lower Shell B2403-1	4	166	129
Lower Shell B2403-2	18	181	144
Lower Shell B2403-3	6	168	131
Intermediate to Lower Shell Circumferential Weld Seam 9-042	-56	197	143
Intermediate Shell Longitudinal Weld Seams 2-042 A & B	-56	205	144
Intermediate Shell Longitudinal Weld Seam 2-042 C	-56	172	118
Lower Shell Longitudinal Weld Seams 3-042 A & B	-56	206	144
Lower Shell Longitudinal Weld Seam 3-042 C	-56	232[3]	168

(1) Values from Tables 5.2-31 and 5.2-32
(2) EOL RT_{NDT} is also termed Adjusted Reference Temperature (ART). Values from reference 31. ¼T and ¾T represent 25% and 75% vessel wall thickness. EOL is 32 EFPY.
(3) Limiting material

TABLE 5.2-38

SALEM 2 PREDICTED EOL RT_{NDT} FOR REACTOR VESSEL BELTLINE MATERIALS

Material	Initial RT_{NDT} [1] (°F)	EOL RT_{NDT} ¼T [2] (°F)	EOL RT_{NDT} ¾T [2] (°F)
Intermediate Shell B4712-1	0	125	100
Intermediate Shell B4712-2	12	145	117
Intermediate Shell B4712-3	10	119	98
Lower Shell B4713-1	8	126	103
Lower Shell B4713-2	8	126	102
Lower Shell B4713-3	10	128	104
Intermediate to Lower Shell Circumferential Weld Seam 9-442	-56	102	76
Intermediate Shell Longitudinal Weld Seam 2-442 A	-56	153	104
Intermediate Shell Longitudinal Weld Seams 2-442 B & C	-56	181	128
Lower Shell Longitudinal Weld Seams 3-442 A & C	-56	199[3]	140[3]
Lower Shell Longitudinal Weld Seam 3-442 B	-56	168	114

(1) Values from Tables 5.2-31 and 5.2-32

(2) EOL RT_{NDT} is also termed Adjusted Reference Temperature (ART). Values from reference 32. ¼T and ¾T represent 25% and 75% vessel wall thickness. EOL is 32 EFPY.

(3) Limiting material

TABLE 5.2-39

SALEM 1 PREDICTED RT_{PTS} FOR REACTOR VESSEL BELTLINE MATERIALS[1]

Material	Fluence[2] (10^{19}n/cm^2, E>1.0 MeV)	ΔRT_{PTS} (°F)	Margin (°F)	$RT_{NDT(u)}$[3] (°F)	EOL RT_{PTS}[4] (°F)	RT_{PTS} Screening Criteria (°F)
Intermediate Shell B2402-1	1.64	175.1	17	45	237	270
Intermediate Shell B2402-2	1.64	160.1	17	-5	172	270
Intermediate Shell B2402-3	1.64	121.0	17	-3	135	270
Lower Shell B2403-1	1.64	146.8	34	4	185	270
Lower Shell B2403-2	1.64	148.1	34	18	200	270
Lower Shell B2403-3	1.64	146.8	34	6	187	270
Intermediate to Lower Shell Circumferential Weld Seam 9-042	1.64	214.9	65.5	-56	224	300
Intermediate Shell Longitudinal Weld Seams 2-042 A, B & C	1.18	228.1	65.5	-56	238	270
Lower Shell Longitudinal Weld Seams 3-042 A, B & C	1.64	254.9	65.5	-56	264[5]	270

(1) Values from reference 29 unless otherwise noted.
(2) Values from Table 5.4-7 for clad-metal interface and are predicted EOL values. All welds assumed to have maximum fluence for the weld seam group.
(3) Values from Tables 5.2-31 and 5.2-32.
(4) EOL is 32 EFPY.
(5) Limiting material.

TABLE 5.2-40

SALEM 2 PREDICTED RT_{PTS} FOR REACTOR VESSEL BELTLINE MATERIALS[1]

Material	Fluence[2] (10^{19}n/cm^2, E>1.0 MeV)	ΔRT_{PTS} (°F)	Margin (°F)	$RT_{NDT(u)}$[3] (°F)	EOL RT_{PTS}[4] (°F)	RT_{PTS} Screening Criteria (°F)
Intermediate Shell B4712-1	1.77	104.2	34	0	138	270
Intermediate Shell B4712-2	1.77	112.8	34	12	159	270
Intermediate Shell B4712-3	1.77	85.5	34	10	130	270
Lower Shell B4713-1	1.77	96.3	34	8	138	270
Lower Shell B4713-2	1.77	95.6	34	8	138	270
Lower Shell B4173-3	1.77	95.8	34	10	140	270
Intermediate to Lower Shell Circumferential Weld Seam 9-442	1.77	106.0	65.5	-56	116	300
Intermediate Shell Longitudinal Weld Seams 2-442 A, B & C	1.20	198.5	65.5	-56	208	270
Lower Shell Longitudinal Weld Seams 3-442 A, B & C	1.20	219.0	65.5	-56	229[5]	270

(1) Values from reference 30 unless otherwise noted.
(2) Values from Table 5.4-8 for clad-metal interface and are predicted EOL values. All welds assumed to have maximum fluence for the weld seam group.
(3) Values from Tables 5.2-31 and 5.2-32.
(4) EOL is 32 EFPY.
(5) Limiting material.

TABLE 5.2-41

SALEM 1 USE PROJECTIONS FOR REACTOR VESSEL BELTLINE MATERIALS[1]

Material	Fluence at ¼T [2] (n/cm^2, E>1 MeV)	Unirradated USE [3] (ft-lb)	Projected USE Decrease (%)	Projected EOL USE (ft-lb)
Intermediate Shell B2402-1	9.8×10^{18}	91	19	74
Intermediate Shell B2402-2	9.8×10^{18}	98	15	83
Intermediate Shell B2402-3	9.8×10^{18}	104	16	87
Lower Shell B2403-1	9.8×10^{18}	93	29	66
Lower Shell B2403-2	9.8×10^{18}	83	29	59
Lower Shell B2403-3	9.8×10^{18}	85	29	60
Intermediate to Lower Shell Circumferential Weld Seam 9-042	9.8×10^{18}	112	36	72
Intermediate Shell Longitudinal Weld Seams 2-042 A, B & C	9.8×10^{18}	96.2	32	65
Lower Shell Longitudinal Weld Seams 3-042 A, B & C	9.8×10^{18}	112	32	76

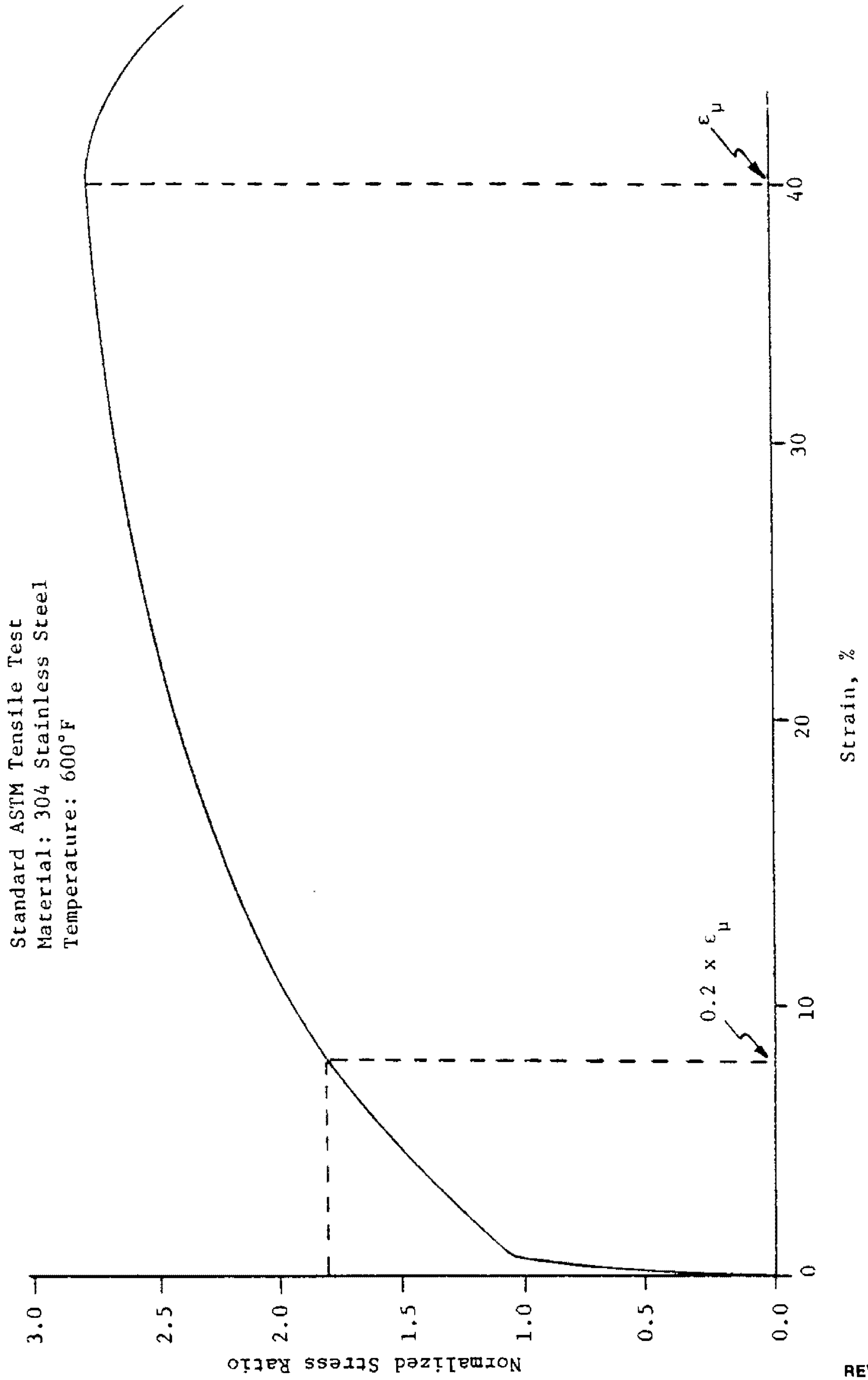
(1) Values from reference 31 unless otherwise noted.
(2) Fluence calculated for EOL (32EFPY) from Table 5.4-7. All welds assumed to have peak fluence.
(3) Values from Tables 5.2-31 and 5.2-32.

TABLE 5.2-42

SALEM 2 EOL USE PROJECTIONS FOR REACTOR VESSEL BELTLINE MATERIALS[1]

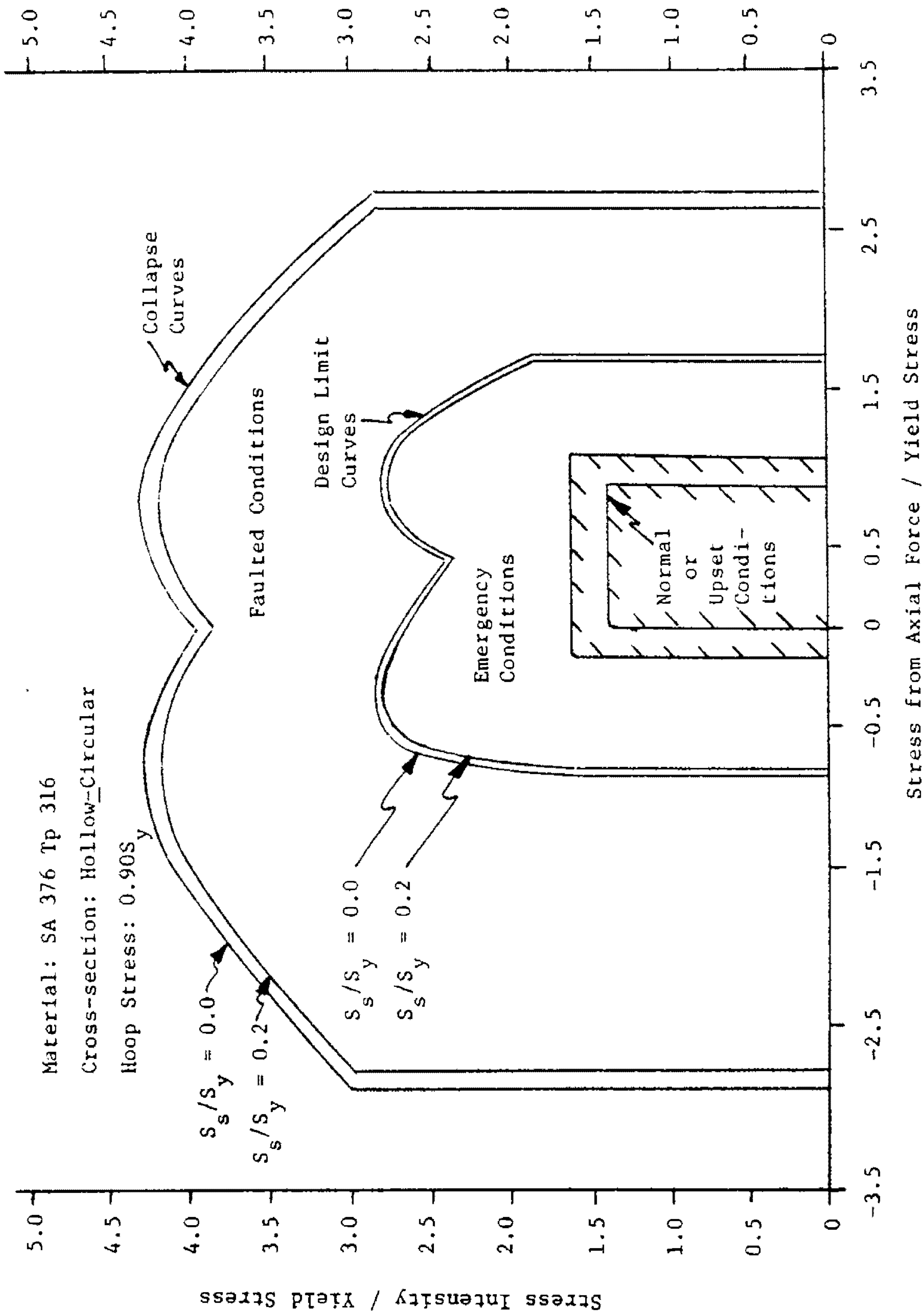
Material	Fluence at ¼T [2] (n/cm^2)	Unirradiated USE [3] (ft-lb)	Projected USE Decrease (%)	Projected EOL USE (ft-lb)
Intermediate Shell B4712-1	1.06×10^{19}	106	22	83
Intermediate Shell B4712-2	1.06×10^{19}	97	14.5	83
Intermediate Shell B4712-3	1.06×10^{19}	107	20	86
Lower Shell B4713-1	1.06×10^{19}	98	21	77
Lower Shell B4713-2	1.06×10^{19}	103	21	81
Lower Shell B4713-3	1.06×10^{19}	121	21	96
Intermediate to Lower Shell Circumferential Weld Seam 9-442	1.06×10^{19}	99.7	35	65
Intermediate Shell Longitudinal Weld Seams 2-442 A, B & C	1.06×10^{19}	96.2	37	61
Lower Shell Longitudinal Weld Seams 3-442 A, B & C	1.06×10^{19}	114	37	72

(1) Values from reference 32 unless otherwise noted.
(2) Fluence calculated for EOL (32EFPY) from Table 5.4-8. All welds assumed to have peak fluence.
(3) Values from Tables 5.2-31 and 5.2-32.



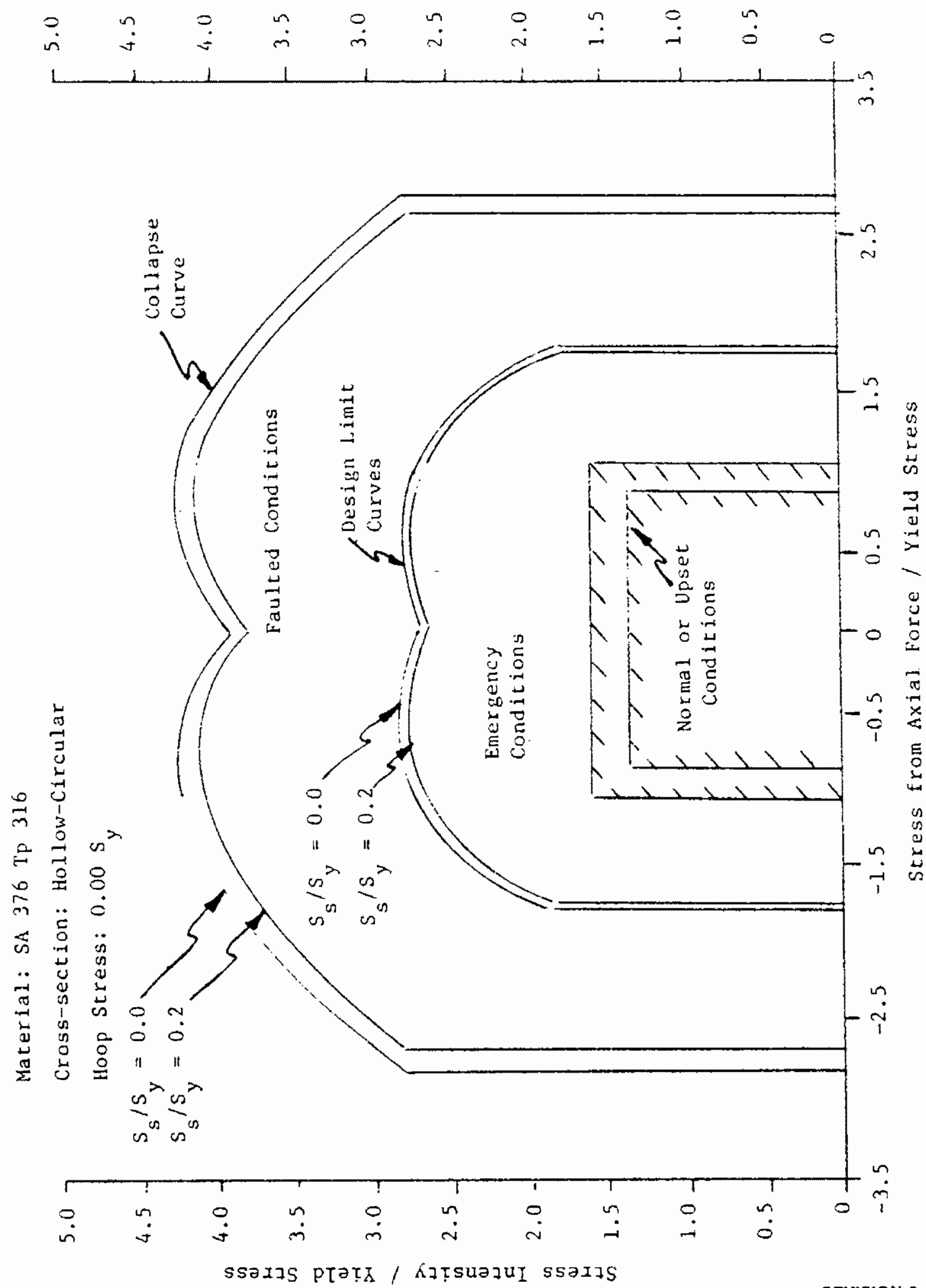
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Typical Stress - Strain Curve	
	Updated FSAR	Figure 5.2-1



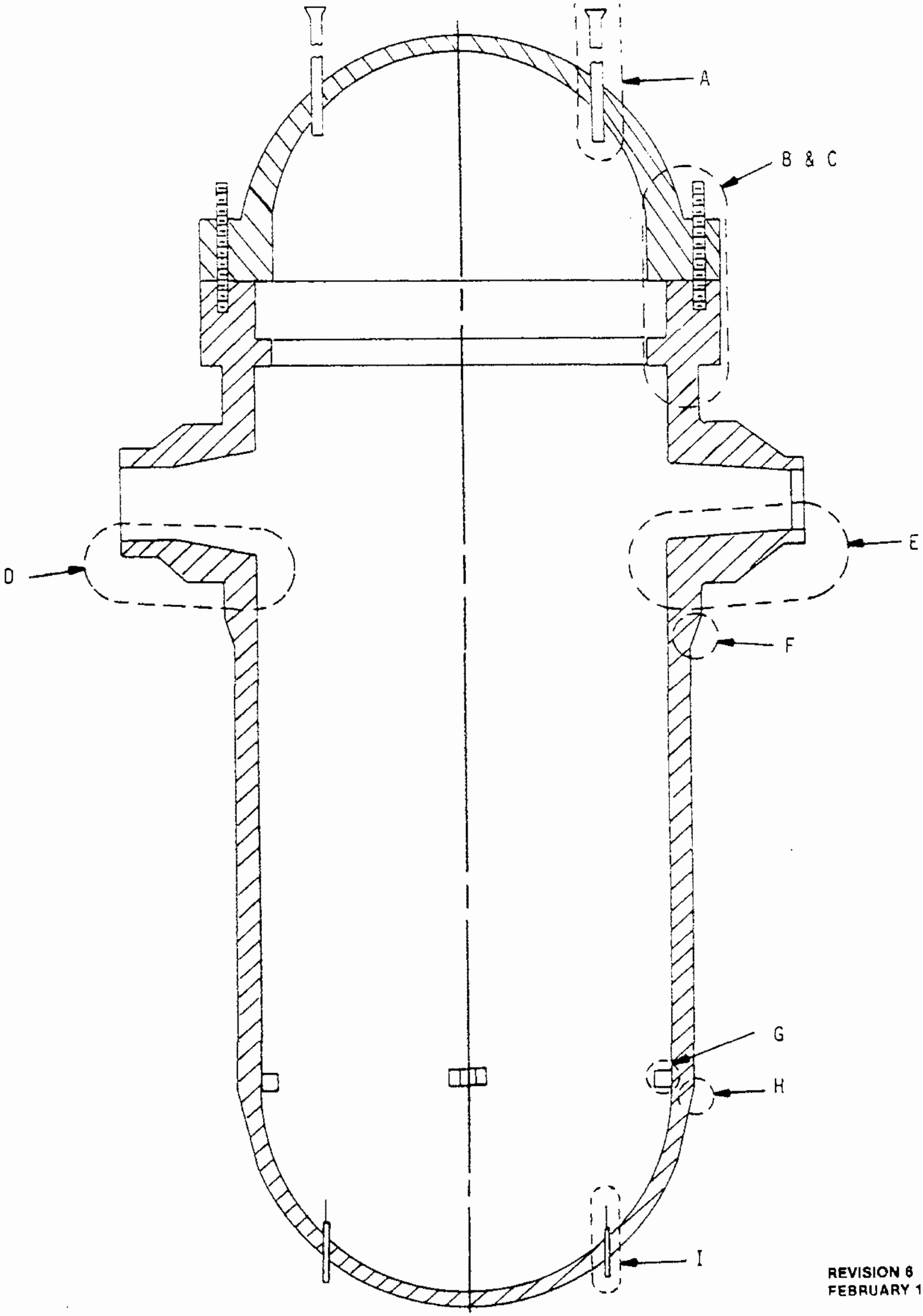
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Comparison Between Design and Collapse Conditions (0.90 S_y)	
	Updated FSAR	Figure 5.2-2



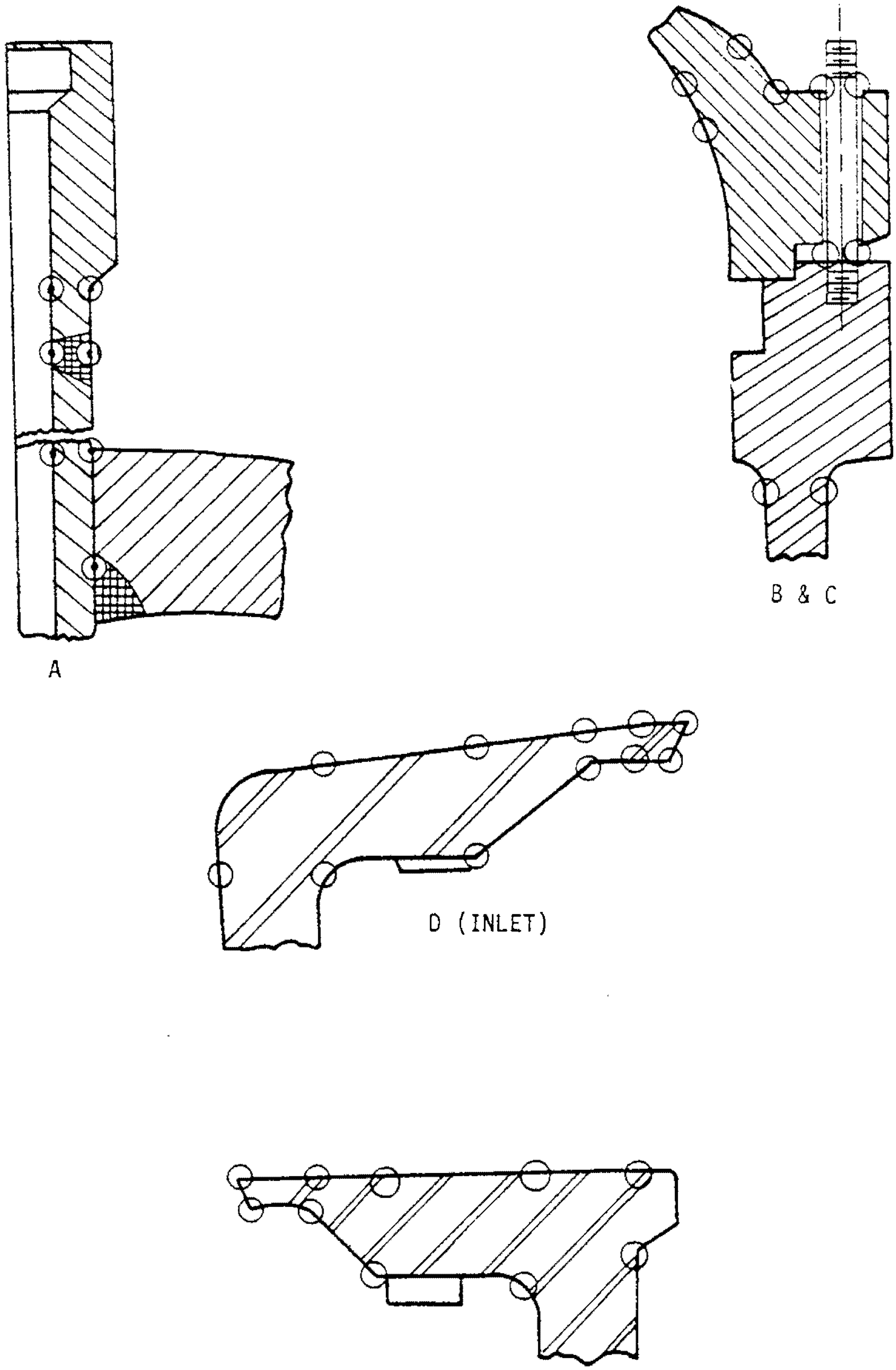
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Comparison Between Design and Collapse Conditions (0.00 S_y)	
	Updated FSAR	Figure 5.2-3



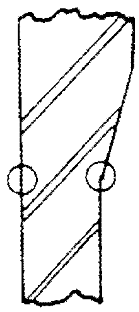
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Reactor Vessel Stress Analysis Areas Examined
	Updated FSAR — Figure 5.2-4



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Reactor Vessel Stress Analysis Details - Upper	
	Updated FSAR	Figure 5.2-5



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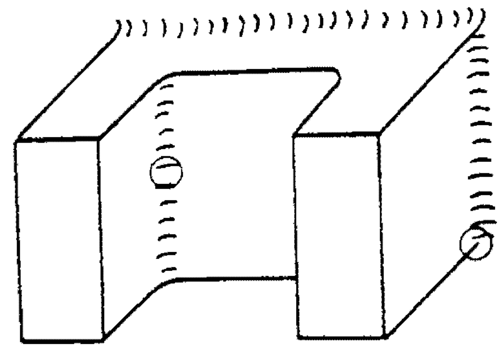

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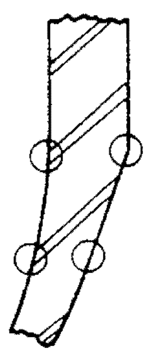
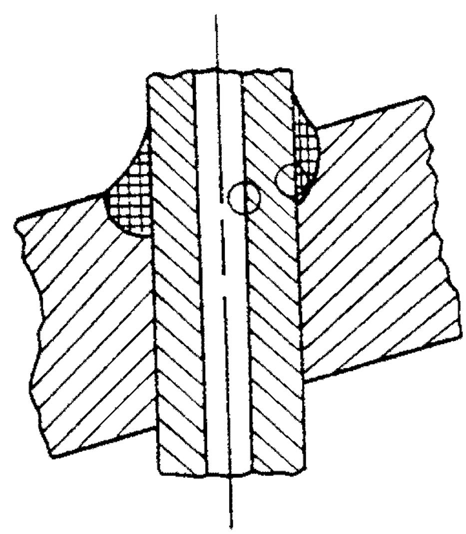


G



H

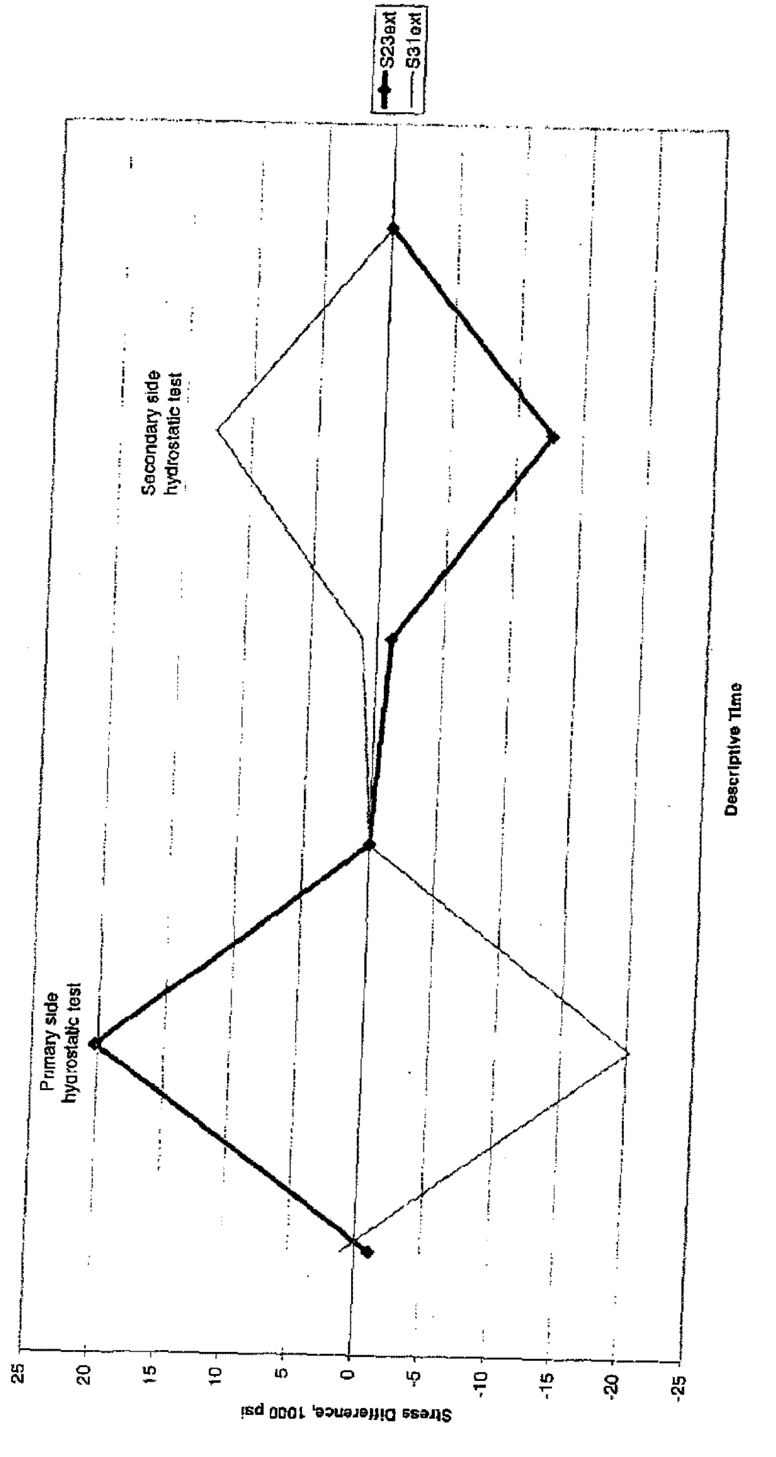
NOTE:

THE POINTS CIRCLED IN THE SKETCHES REPRESENT THE GENERAL LOCATION AND GEOMETRY OF THE AREAS OF DISCONTINUITY AND/OR STRESS CONCENTRATION.

REVISION 6
FEBRUARY 15, 1987

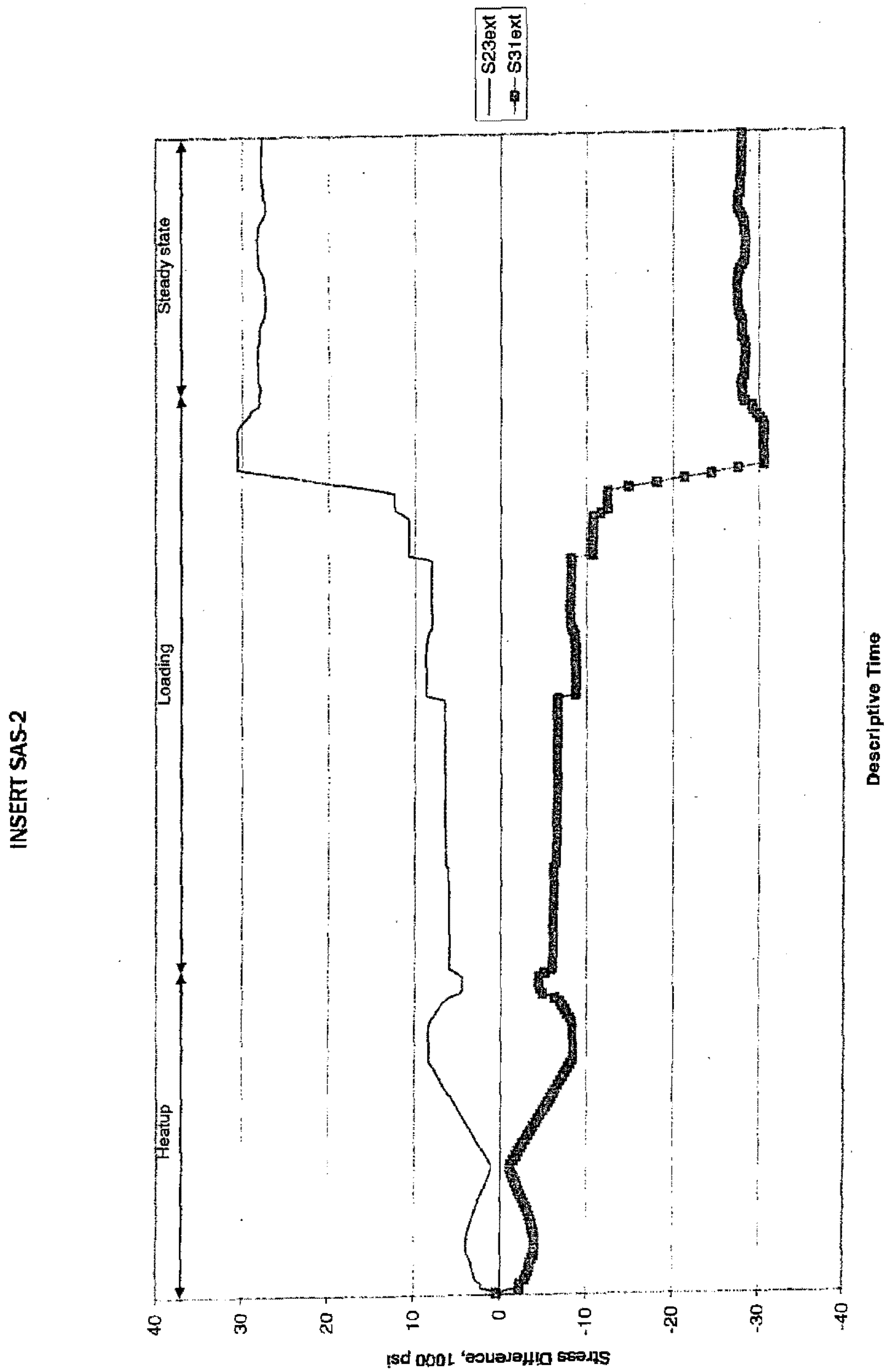
PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Reactor Vessel Stress Analysis Details - Lower	
	Updated FSAR	Figure 5.2-6

INSERT SAS-1



Revision 24
May 11, 2009

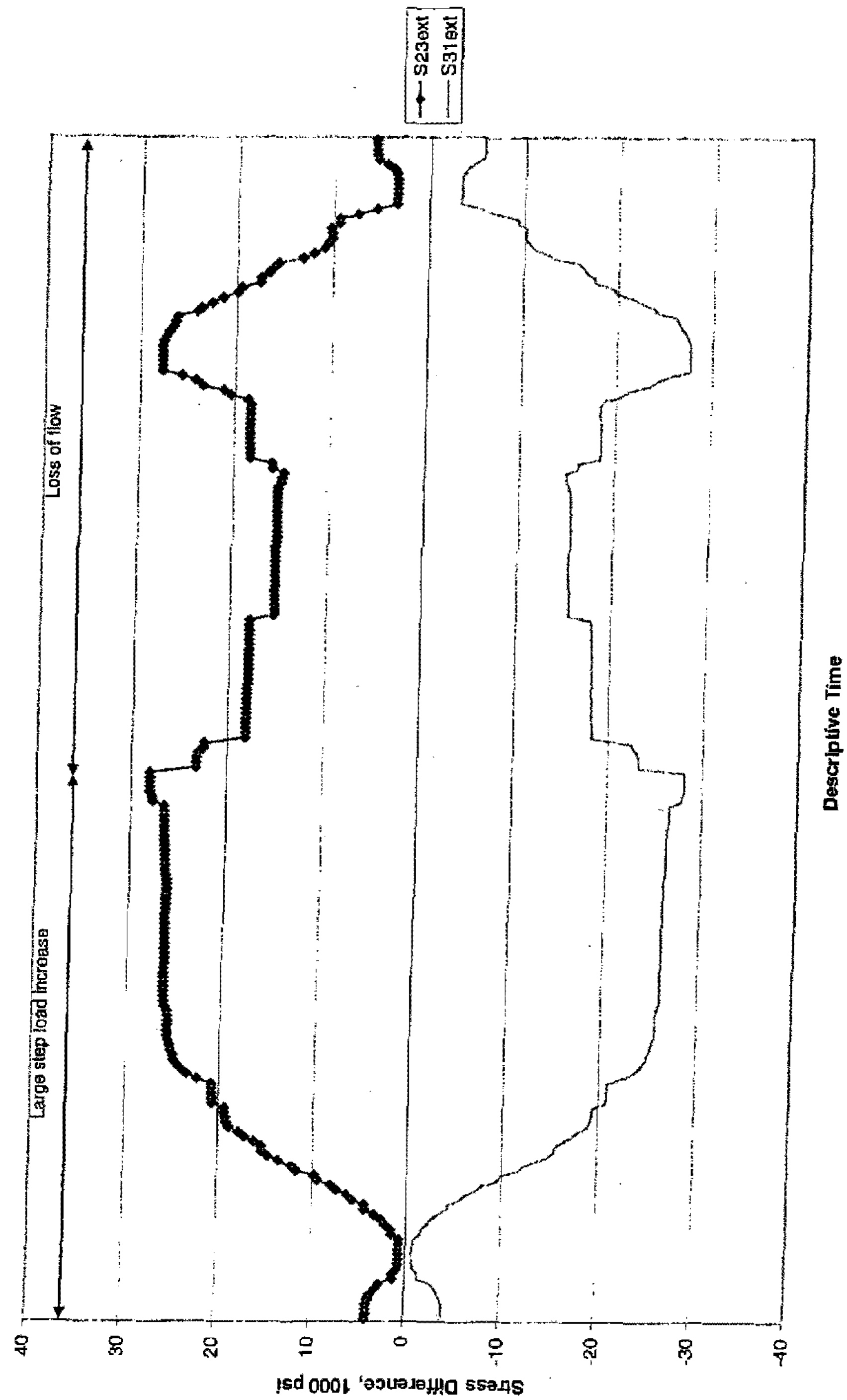
PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station PRIMARY AND SECONDARY HYDROSTATIC TEST STRESS HISTORY FOR THE CENTER HOLE LOCATION AREVA NP Model 61/19T SG (UNIT 2 ONLY)
	Updated FSAR Figure 5.2-7



Revision 24
May 11, 2009

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station PLANT HEATUP AND OPERATIONAL LOADING TRANSIENTS (WITH STEADY-STATE PLATEAU) STRESS HISTORY FOR THE HOT SIDE CENTER HOLE LOCATION AREVA NP Model 61/19T SG (UNIT 2 ONLY)
	Updated FSAR — Figure 5.2-8

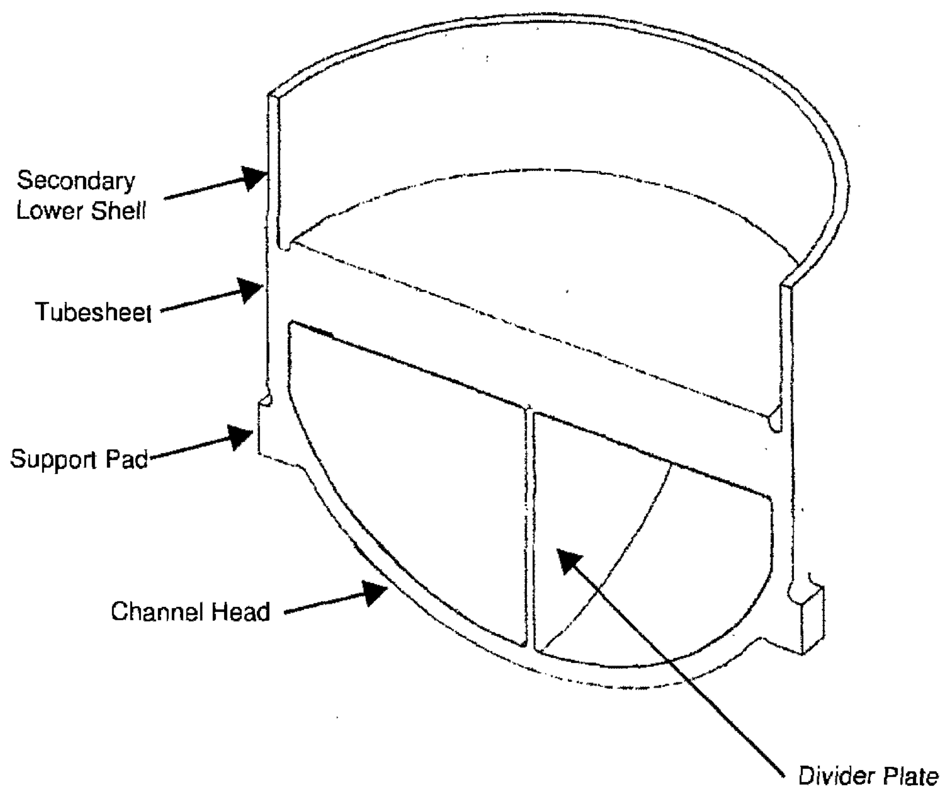
INSERT SAS-3



Revision 24
May 11, 2009

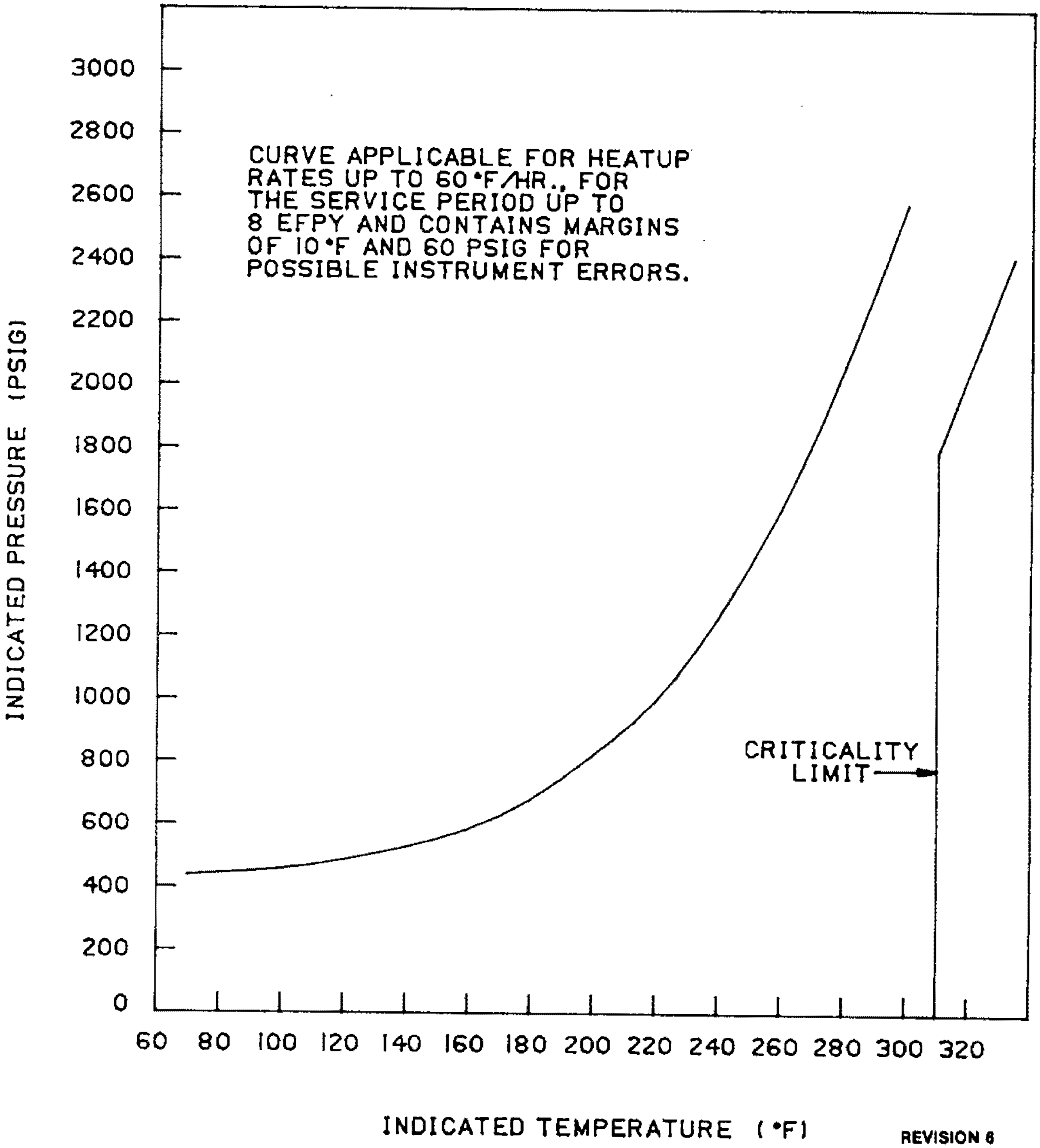
PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station LARGE STEP LOAD DECREASES AND LOSS OF FLOW STRESS HISTORY OF THE HOT SIDE CENTER HOLE LOCATION AREVA NP Model 61/19T SG (UNIT 2 ONLY)
	Updated FSAR Figure 5.2-9

INSERT SAS-4



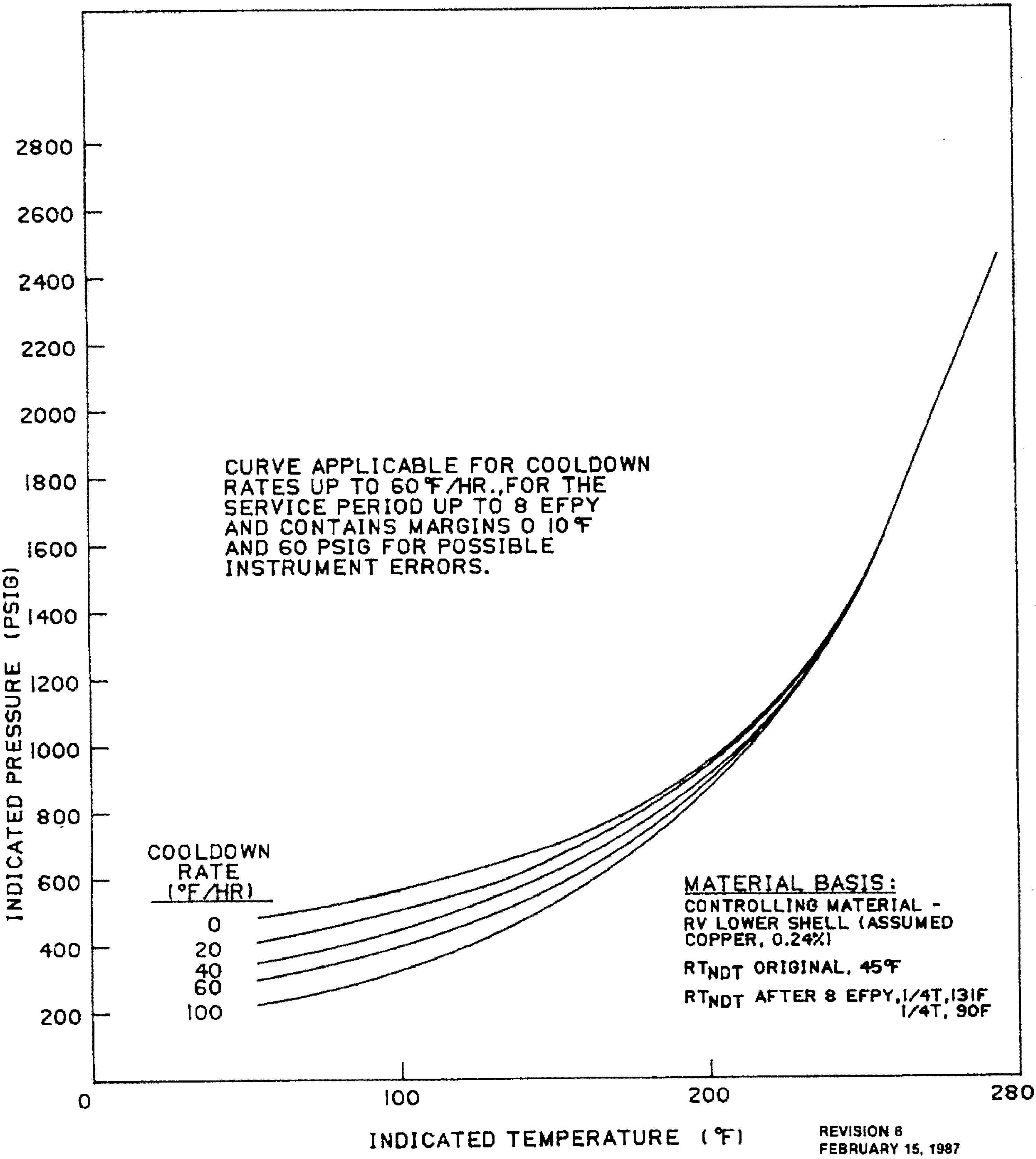
Revision 24
May 11, 2009

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station PRIMARY AND SECONDARY BOUNDRY COMPONENTS SHELL LOCATIONS OF STRESS INVESTIGATIONS **AREVA NP Model 61/19T SG (UNIT 2) ONLY**
	Updated FSAR — Figure 5.2-10



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Reactor Coolant System Heatup Limitations for First 8 EFPY of Operation Updated FSAR — Figure 5.2-11



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Reactor Coolant System Cooldown Limitations for First 8 EFPY of Operation
	Updated FSAR — Figure 5.2-12

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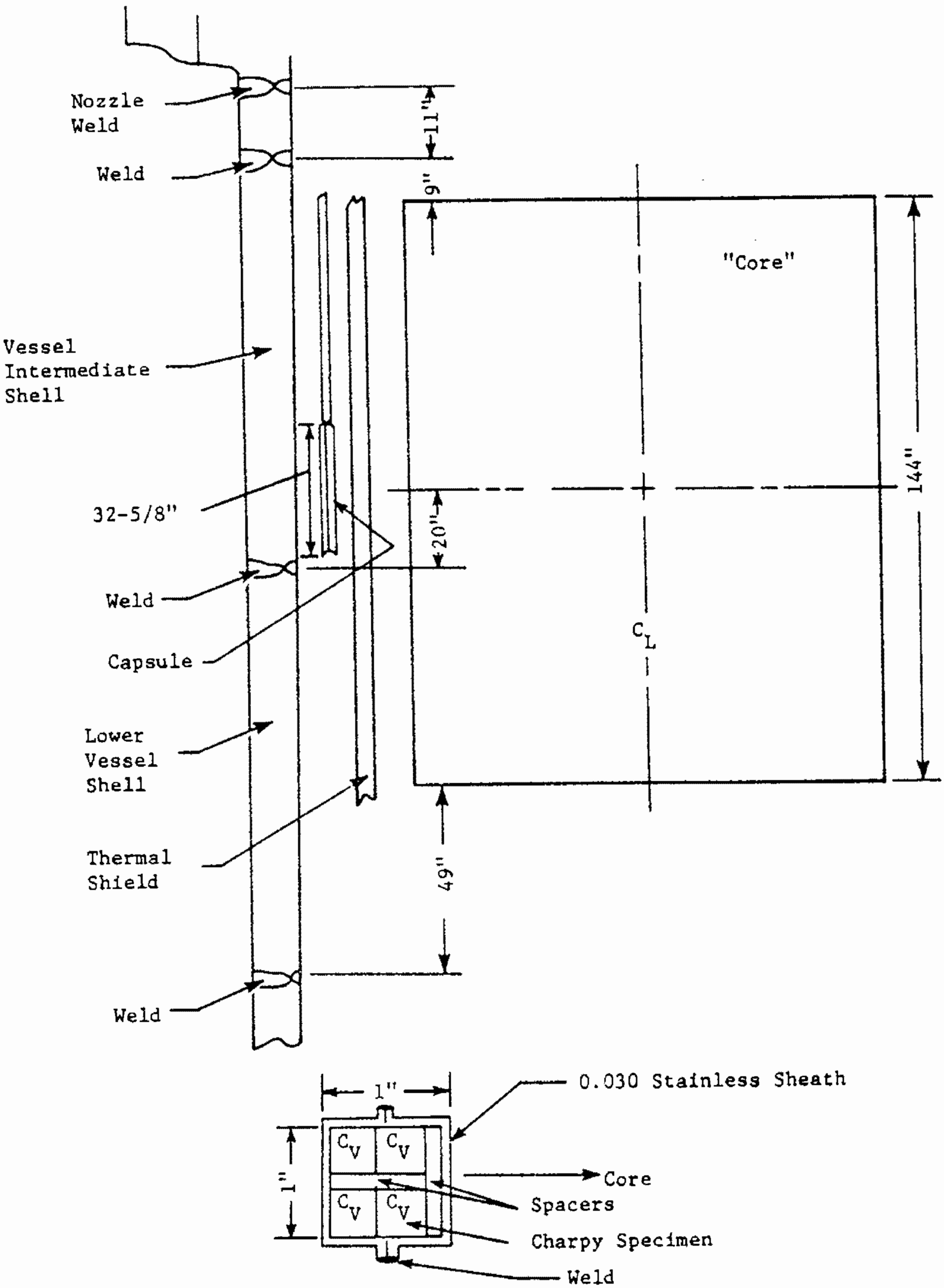
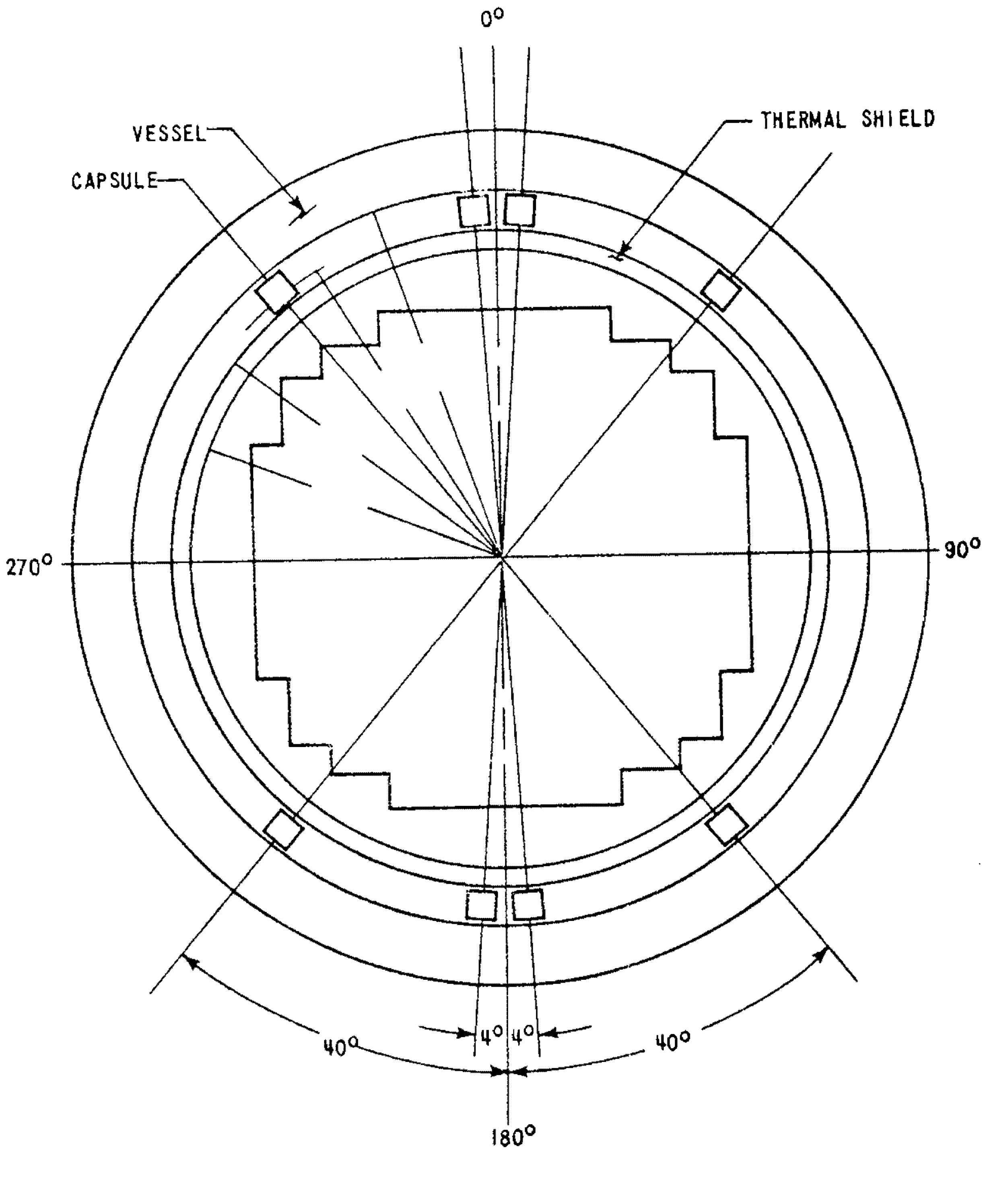
PSEG NUCLEAR L.L.C. SALEM GENERATING STATION
SALEM UFSAR - REV 19 SHEET 1 OF 1 November 19, 2001 F5.2-13



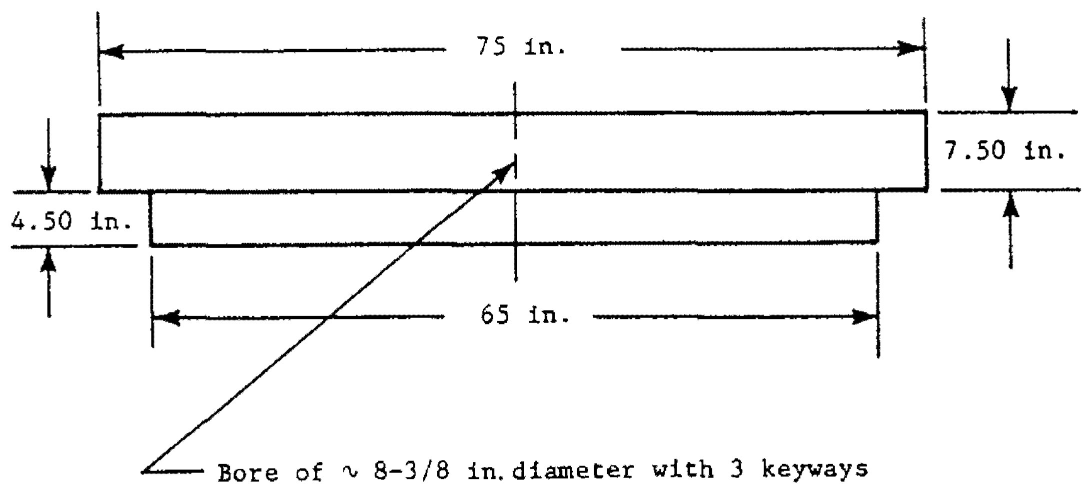
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Typical Surveillance Capsule Elevation View
	Updated FSAR Figure 5.2-14

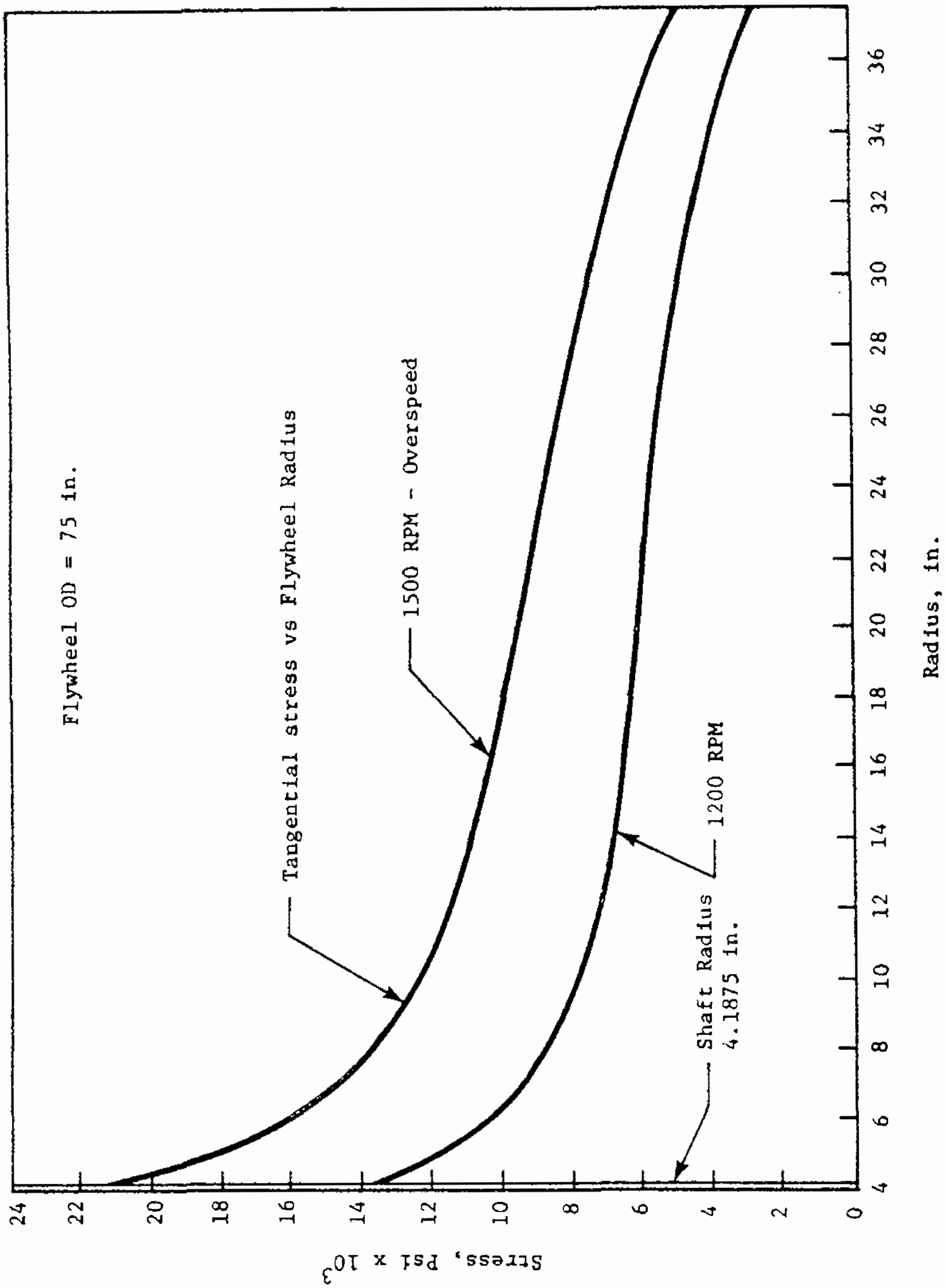


REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Surveillance Capsule Plan View
	Updated FSAR — Figure 5.2-15

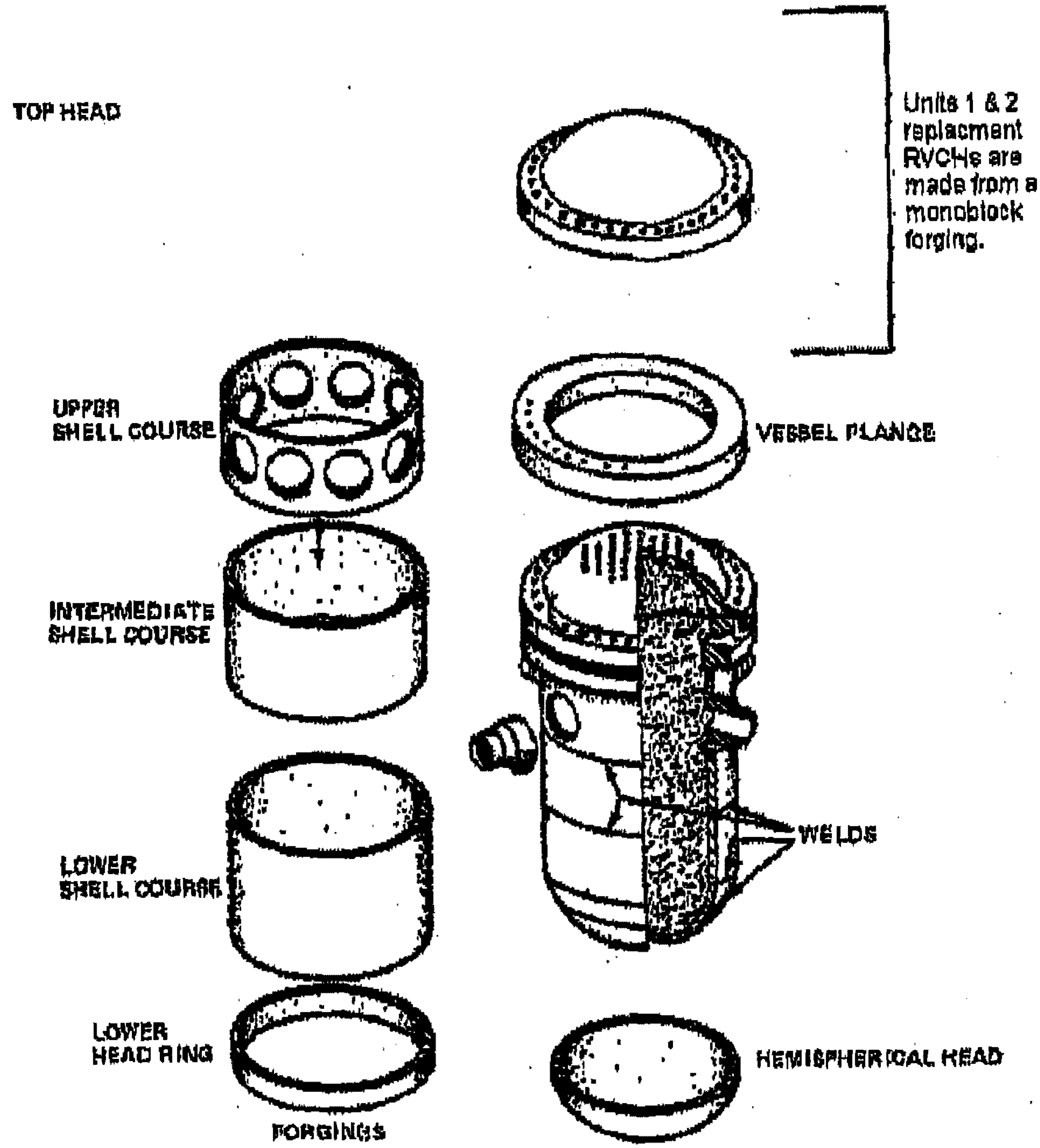


NOTE: The plates are bolted together with the bolts aligned perpendicular to the planes of the plates.



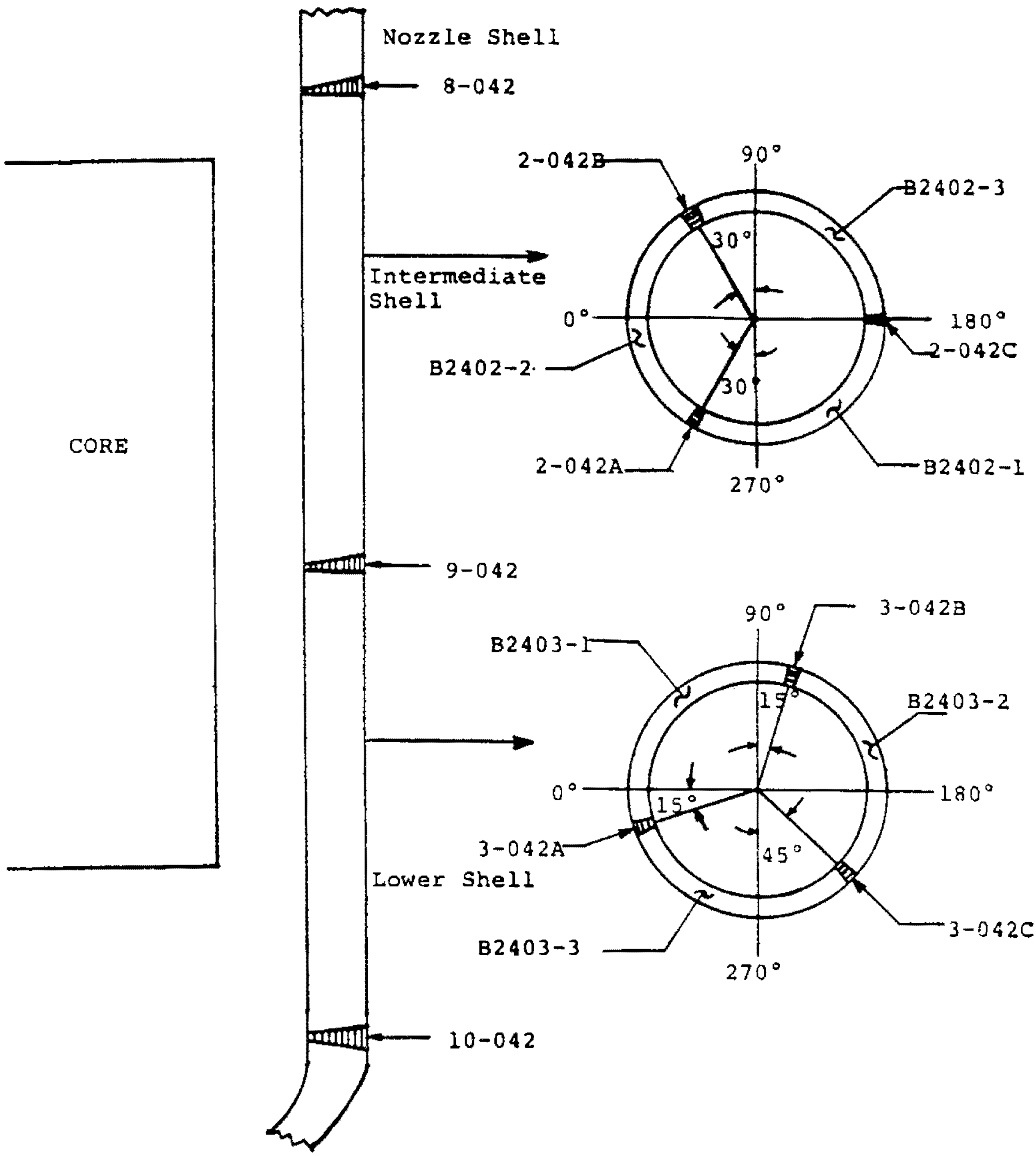
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Flywheel Characteristics Curve	
	Updated FSAR	Figure 5.2-17



Revision 22, May 5, 2006

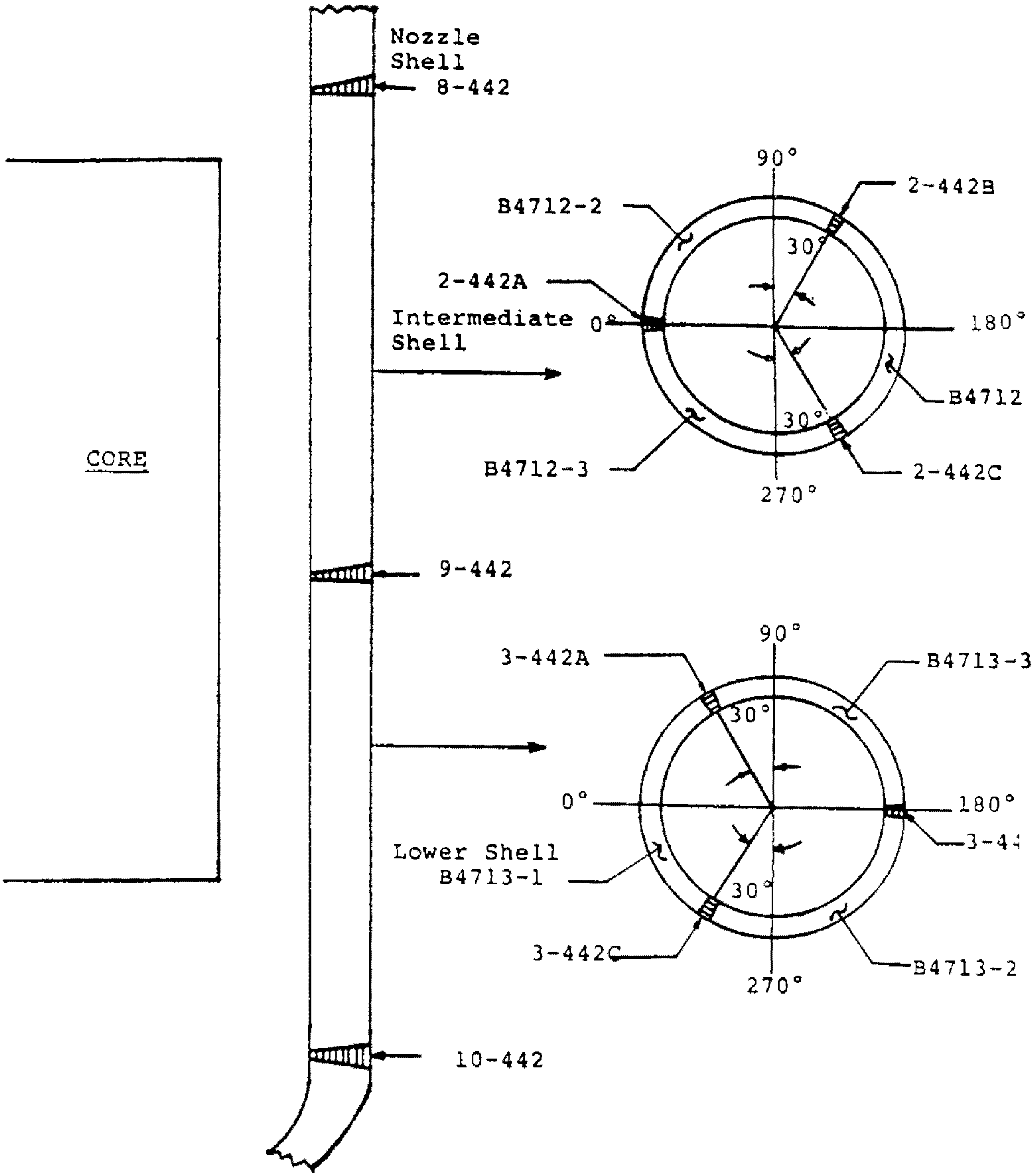
PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station REACTOR PRESSURE VESSEL ASSEMBLY
	Updated FSAR — Figure 5.2-18



NOTE: Relative to the midplane of the active core, the circumferential weld 8-042 is about 91.5" above the core midplane and circ. welds 9-042 and 10-042 are 16.7" and 123.2" below the core midplane respectively.

REVISION 7
JULY 22, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Identification and Location of Beltline Region Material for the Salem Unit No. 1 Reactor Vessel
	Updated FSAR — FIG. 5.2-19

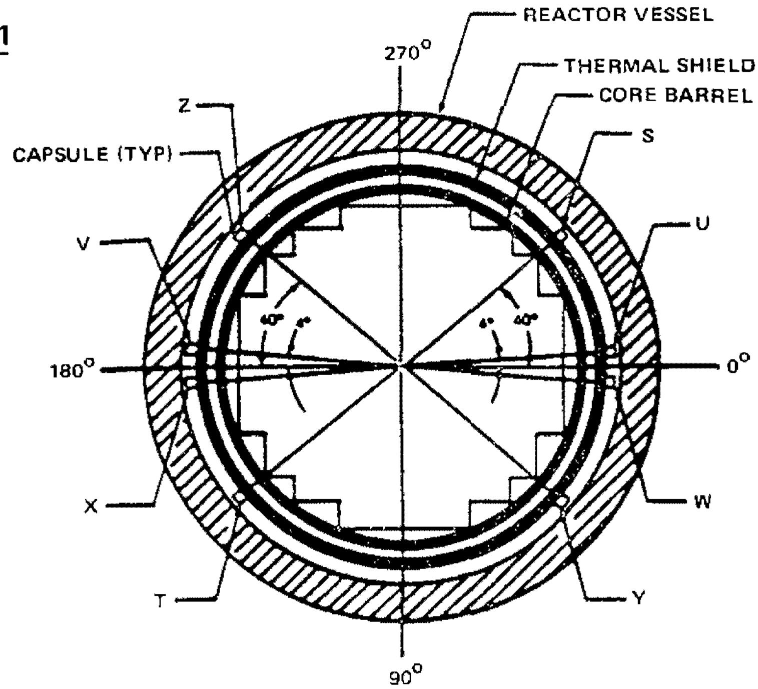


NOTE: Relative to the midplane of the active core, the circumferential weld 8-442 is about 92.1" above the core midplane and circ. welds 9-442 and 10-442 are 16.7" and 123.2" below the core midplane respectively.

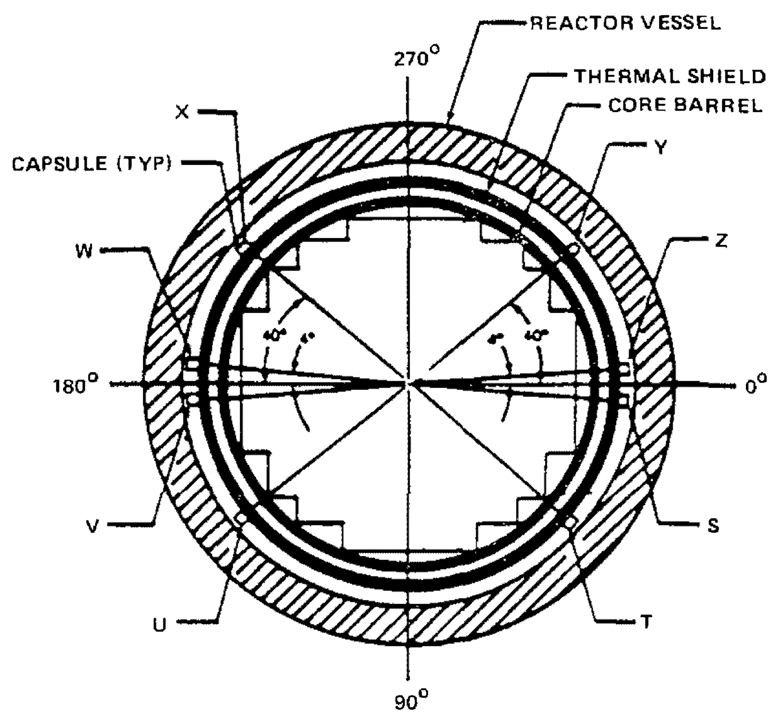
REVISION 7
JULY 22, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Identification and Location of Beltline Region Material for the Salem Unit No. 2 Reactor Vessel
	Updated FSAR FIG. 5.2-20

SALEM UNIT 1

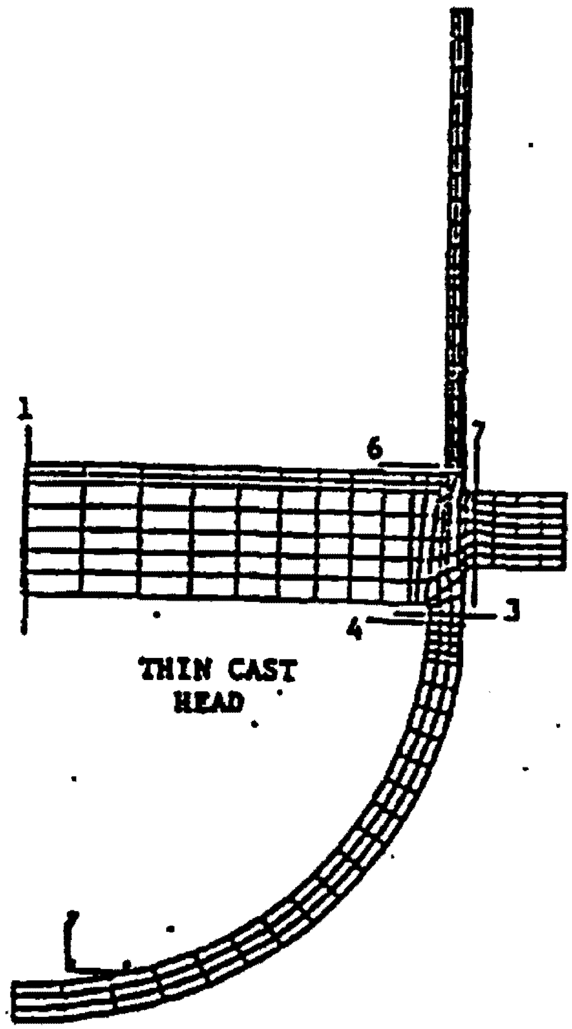


SALEM UNIT 2



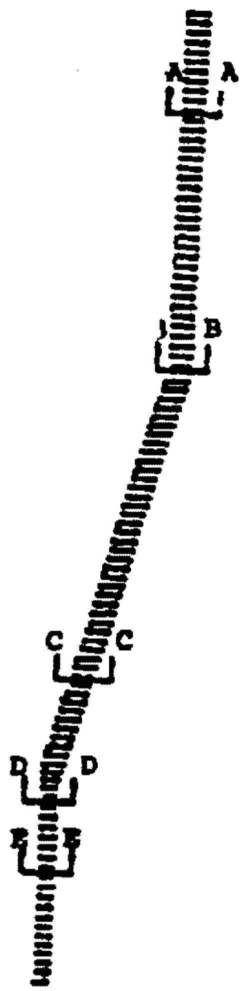
REVISION 7
JULY 22, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Arrangement of Surveillance Capsules in the Reactor Pressure Vessel	
	Updated FSAR	FIG. 5.2-21



Revision 18, April 26, 2000

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station TUBSHEET AND SHELL JUNCTIONS IMPORTANT STRESS LOCATIONS UNIT 1 MODEL FSG
	Updated FSAR — Figure 5.2-22



Revision 18, April 26, 2000

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station SECONDARY SHELL AND TRANSITION CONE IMPORTANT LOCATIONS UNIT 1 MODEL FSG
	Updated FSAR — Figure 5.2-23

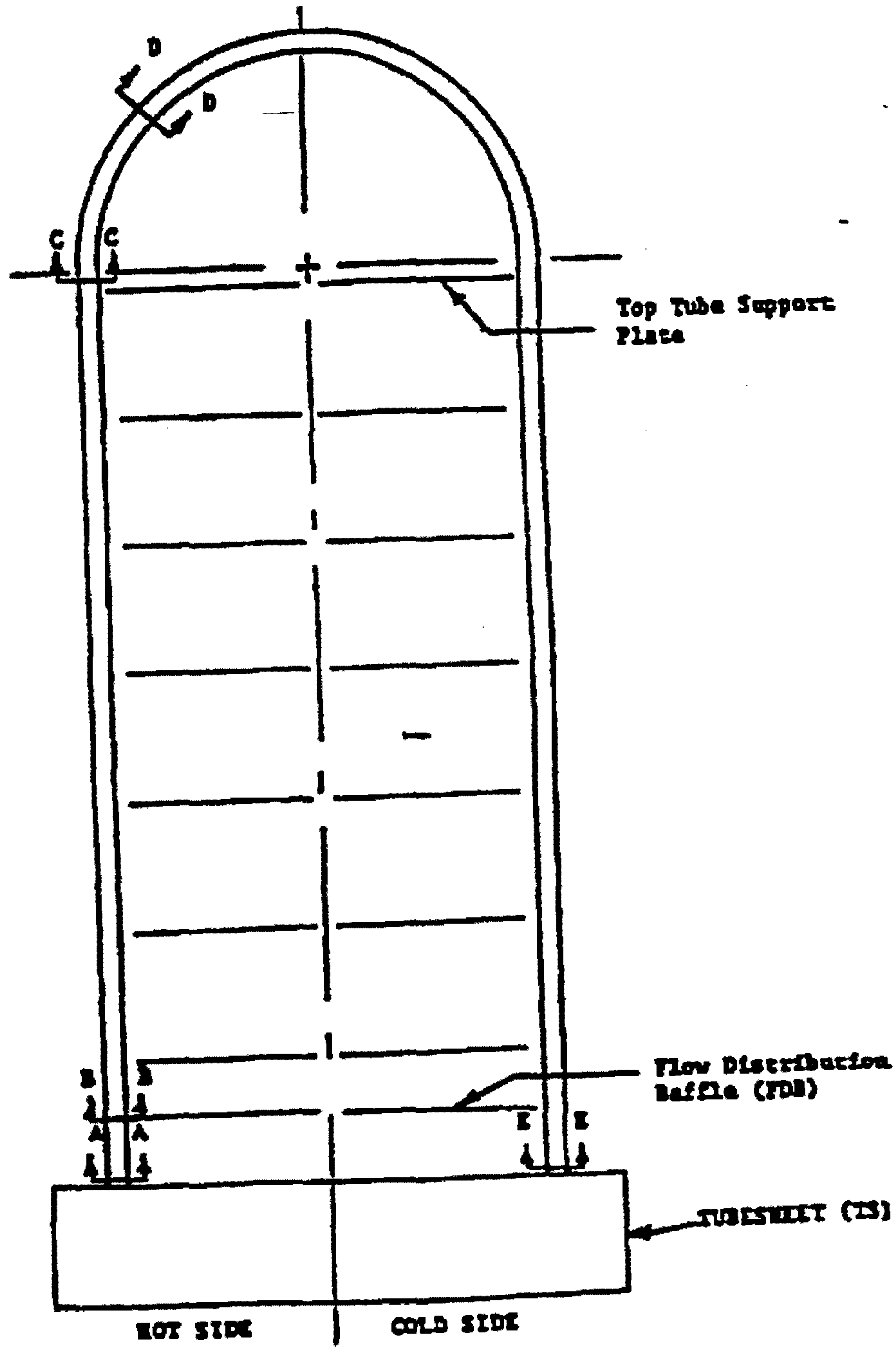


FIGURE 5 2-24 Tubes, Important Stress Locations - Unit 1

Model F SG

Revision 18, April 26, 2000

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station TUBES, IMPORTANT STRESS LOCATIONS UNIT 1 MODEL FSG
	Updated FSAR — Figure 5.2-24

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 Section 4.

5.3.2 Operating Restrictions on Pumps

The minimum net position suction head (NPSH) and minimum seal injection flow rate must be established before operating the reactor coolant pumps. Requirements are set forth in the pump operating instructions.

5.3.3 Temperature Power Operating Map

Reactor power is controlled to maintain average coolant temperature at a value which is a linear function of load.

5.3.4 Load Following Characteristics

Load following is discussed in Section 5.2.1.5.2.

5.3.5 Transient Effects

Transient effects on the RCS are evaluated in Section 15.

5.3.6 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.

5.3.7 Natural Circulation Capability

The capability to perform natural circulation cooldown has been analyzed following a transient in a Pressurized Water Reactor of different design in which significant void formation occurred in the reactor vessel head. The analysis took into account such factors as the amount of bypass flow to the upper head region, RCS cooldown/depressurization rate, heat removal via the Control Rod Drive Mechanism (CRDM) cooling fans and ambient losses. Salem is a "T-Hot" plant, i.e., one in which the upper head water temperature is assumed equal to the hot leg temperature. Analysis results are summarized below.

The average cooldown rate of the upper head fluid due to the 25°F per hour natural circulation cooldown rate is about 10°F per hour. The total upper head cooldown rate due to both the natural circulation cooldown and the CRDM fans varies from 31°F per hour initially to around 21°F per hour when the upper head temperature is cooled to 350°F. Thus, with the CRDM fans operating during the cooldown with no void formation occurring in the upper head area, the operator is required to maintain a minimum of 50°F subcooling during the depressurization.

A Salem-specific analysis has been performed to determine the natural circulation cooldown strategy to prevent void formation without CRDM fans in operation. Without the CRDM fans in operation, the plant can be cooled down to Residual Heat Removal (RHR) System conditions at a natural circulation cooldown rate of 25°F per hour with no void formation occurring in the upper head with appropriate precautions being taken by the operators. The operator is required to maintain a minimum of 50°F subcooling until the primary system pressure reaches 50 psi below the permissive to block SI. After the automatic safety injection signals are blocked, the operator establishes 200°F subcooling (approximately 430°F in the hot leg) and maintains at least 200°F subcooling (or the Technical Specification limit if it is more restrictive) to a primary system pressure of 1200 psig. Depressurization is stopped at 1200 psig and the cooldown is continued until the primary system temperature is less than 350°F. At this point, the operator is required to wait for approximately 8 hours to allow the upper head to cool off, corresponding to a saturation temperature less than the RHR cut-in pressure (325 psig). Finally, the primary system is depressurized to 325 psig and the RHR System used for further cooldown.

Unit 1 has Model-F steam generators which are similar in design to the Series 51 in natural circulation capabilities. Hence, the general conclusions drawn above are also applicable to Unit 1 with the Model-F steam generators. The Unit 2 AREVA NP Model 61/19T steam generators have adequate natural circulation capability to remove decay heat.

5.4 REACTOR VESSEL AND APPURTENANCES

Section 5.4 is divided into four principal subsections: (1) Design Basis, (2) Description, (3) Evaluation, and (4) Tests and Inspections.

5.4.1 Design Basis

The reactor vessel was designed and fabricated to Class A of the ASME Boiler and Pressure Vessel Code, Section III. Material specifications are discussed in Section 5.2.3.1. Fracture toughness of the reactor vessel materials is discussed in Section 5.2.4. Design transients are discussed in Section 5.2.1.

5.4.2 Description

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the core, core support structures, control rods, thermal shield, and other parts directly associated with the core. The reactor vessel closure head contains head adaptors. These head adaptors are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adaptors contain acme threads for the assembly of the control rod drive mechanisms and/or instrumentation adaptors. The seal arrangement at the upper end of these adaptors consists of a welded flexible canopy seal. The vessel has inlet and outlet nozzles located in a horizontal plane just below the vessel flange but above the top of the core. Coolant enters the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom and flows up through the core to the outlet nozzles.

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

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

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

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

1. (Not Used)

2. Vessel flange

3. Eight primary nozzles

The cylindrical portion of the reactor vessel below the refueling seal ledge is permanently insulated with a metallic reflective-type insulation supported from the reactor coolant nozzles. This insulation consists of inner and outer sheets of stainless steel spaced 3 inches apart with multilayers of stainless steel as the insulating agent. Removable panels of the metallic reflective type insulation described above are provided for the reactor vessel head and closure region. These panels are supported on the refueling seal ledge and vent shroud support ring. The rest of the closure head is insulated with removable panels of at least 3 inches of the reflective insulation described

or halide free insulating material. The bottom head is also insulated with reflective insulation, but it is not removable.

A schematic of the reactor vessel is shown on Figure 5.1-1. The materials of construction are given in Table 5.2-27 and the design parameters are given in Table 5.2-3.

The following summarizes those features which preclude wetting of the reactor vessel studs with boric acid.

Refueling procedures include removal of studs before moving the head, and replacement only after the head is sealed. Hole plugs with O-ring seals are placed in the stud holes whenever the head is off. The flange will be dried completely before replacing the head and removing the stud hole plugs.

Spilling of boric acid onto the studs during venting will be precluded by the following precautions:

1. Detailed step-by-step venting procedures exist.

2. Personnel have accurate knowledge of RCS level during venting.

3. Only a small quantity of coolant is released at each venting step.

4. The small quantity of released coolant is piped into portable containers for collection.

5.4.3 Evaluation

5.4.3.1 Compliance With 10CFR50, Appendices G and H

The Unit 2 reactor vessel was built to the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III, and Addenda up to and including the Summer of 1966. Thus the ferritic materials

in the reactor vessel were not tested by the vessel fabricator to meet later editions of Section III of the ASME Code as required by 10CFR50, Appendix G. Westinghouse performed tests, as part of the surveillance program, on the reactor vessel intermediate and lower shell course plates, which surround the effective height of the fuel assemblies. Full Charpy test curves were obtained on these plates from specimens oriented normal to the principal rolling direction. A summary of the results of these tests is shown in Table 5.4-1.

Based on the test results shown in Table 5.4-1, the core region shell plates have a minimum upper shelf energy greater than 75 ft-lb as required by Appendix G.

The stress intensity factors for various reactor vessel locations were not calculated to determine if they are lower than the reference stress intensity factors specified in Appendix G of the Code. Westinghouse has performed these calculations for many older reactor vessels with similar properties and the results have always shown that the calculated stress intensity factors are lower than the reference stress intensity. Thus, based on past experience, Westinghouse is confident that if the calculation was performed, the results would be shown to be acceptable and be lower than the reference stress intensity factors as required by Appendix G of the ASME Code.

Heatup and cooldown limit curves, including preoperational system leakage and hydrostatic pressure tests, were determined in accordance with the method described in Appendix G of the ASME Code.

Reactor vessel bolting material tests were not performed to demonstrate conformance with the minimum requirement of 25 mils lateral expansion, and 45 ft-lb at the preload temperature or at the lowest service temperature, whichever is lower. Tests were performed to meet 35 ft-lb at 10°F. The results of the tests are shown in Table 5.4-2.

Table 5.4-2 shows that all the bolting material met the 45 ft-lb requirement at 10°F except for one end of one bar, which was used for closure head nuts and washers. It is expected that this bar would exhibit at least 45 ft-lb if tested at 50°F which is considered to be the lowest preload or service temperature.

At the time the tests were conducted, lateral expansion measurements were not required. However, it is expected that these materials would exhibit at least 25 mils lateral expansion if tested at 50°F, based on test results from other bolting materials where both impact energy and lateral expansion data were obtained.

The reactor vessel is designed to permit a thermal annealing treatment to recover material toughness properties of ferritic materials in the reactor vessel beltline.

Reactor vessel beltline region materials will be monitored by a surveillance program which includes eight surveillance capsules which will receive a neutron flux at least as high, but not more than three times as high as that received by the vessel inner surface. The surveillance program is in compliance with ASTM E-185-73 with the exception of the surveillance weld. The high flux region of the reactor vessel was fabricated from different combinations of weld wire and lots of welding flux for which sufficient tests and chemical analyses are not available to select surveillance weld metal as required by ASTM E-185-73. The surveillance weld, although fabricated using the same weld wire and flux lot number as used in core region vertical seams, may not be limiting material in the reactor vessel.

Information relative to the changes in fracture toughness due to irradiation are discussed in Section 5.2.4.

5.4.3.2 Radiation Analysis and Neutron Dosimetry Of Surveillance Capsules

Knowledge of the neutron environment within the pressure vessel/surveillance capsule geometry is required as an integral part of LWR pressure vessel surveillance programs for two reasons. First, in the interpretation of radiation-induced properties, changes observed in materials test specimens and the neutron environment (fluence, flux) to which the test specimens are exposed must be known. Second, in relating the changes observed in the test specimens to the present and future condition of the reactor pressure vessel, a relationship must be established between the environment at various positions within the reactor vessel and that experienced by the test specimens. The former requirement is normally met by employing a combination of rigorous analytical techniques and measurements obtained with passive neutron flux monitors contained in each of the surveillance capsules. The latter information is derived solely from analysis.

This section describes a discrete ordinates S_n transport analysis to determine the fast neutron (E >1.0 MeV) flux and fluence as well as the neutron energy spectra within the reactor vessel and surveillance capsules. The analytical data is then used to develop lead factors for use in relating neutron exposure of the pressure vessel to that of the surveillance capsules.

5.4.3.3 Neutron Transport Methodology

Fast neutron exposure calculations for the reactor geometry are carried out using appropriately benchmarked discrete ordinates transport techniques. Plant specific calculations are completed using the DORT[1] two-dimensional discrete ordinates code and a benchmarked ENDF/B-VI based cross-section library. Both the BUGLE-93[2] and BUGLE-96[3] ENDF/B-VI multigroup cross-section libraries provide acceptable results for LWR applications. In the transport analyses, anisotropic scattering is treated with a P_3 Legendre expansion at a minimum; and the angular discretization is modeled with at least an S_8 order of angular quadrature.

In developing an analytical model of the reactor geometry, nominal design dimensions are employed for the various structural components. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor are taken to be representative of nominal full power operating conditions. The reactor core itself is treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc. Sensitivities of the analytical results to tolerances in the internals dimensions as well as to fluctuations in water temperature are used to establish the uncertainties associated with the neutron exposure projections at the pressure vessel wall.

For each operating fuel cycle, the spatial variation of the neutron source is obtained from a burnup weighted average of the power distributions occurring during the course of the fuel cycle. These spatial distributions include pinwise gradients for all fuel assemblies located at the periphery of the core and include a uniform or flat distribution for fuel assemblies interior to the core. The energy distribution of the source is likewise determined by selecting a fuel burnup representative of conditions averaged over the fuel cycle and an initial fuel assembly enrichment characteristic of the cycle specific core loading pattern. From the average burnup and initial enrichment, a fission split by isotope including ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu is derived; and, from that fission split, composite values of energy release per fission, neutron yield per fission, and fission spectrum are determined. These composite values are then combined with the spatial distribution to produce the overall absolute neutron source for use in the transport calculations.

5.4.3.4 Neutron Dosimetry

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.

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 high purity germanium 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 determined from plant power generation 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, sensor reaction rates referenced to full power operation are then determined. This reaction rate calculation also includes the effects of varying fuel cycle dependent neutron flux at the locations of the sensor sets.

Prior to using these measured reaction rates in the determination of fast neutron exposure parameters in terms of neutron fluence (E > 1.0 MeV) and Iron Atom Displacements (dpa), 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 the corrections made for the presence of 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.

These measured reaction rates, including all corrections, along with the results of the plant specific neutron transport calculations are then input to a least squares adjustment procedure to determine a best estimate neutron energy spectrum with associated uncertainties at the surveillance capsule location. Best estimates for key exposure parameters such as neutron fluence (E > 1.0 MeV) or iron atom displacements (dpa) along with their uncertainties are then easily obtained from the adjusted spectrum. The use of measurements in combination with the analytical results reduces the uncertainty in the calculated spectrum and acts to remove biases that may be present in the analytical technique.

5.4.3.5 Vessel Fluence Calculation For Fracture Toughness Determination for Salem Units

Fracture toughness calculations for Salem Units 1 and 2 Reactor Pressure Vessels (RPV) were performed for Pressure-Temperature Limit (P-T) Curve development (section 5.2.4.2), Pressurized Thermal Shock (PTS) evaluations (section 5.2.4.5), and Upper Shelf Energy (USE) projections (section 5.2.4.6). A key input in the calculation of RPV fracture toughness was the neutron fluence (n/cm^2) at the vessel clad-bare metal interface and at depths within the vessel wall corresponding to 25 and 75% of the wall thickness for each of the materials constituting the beltline region. The 25 and 75% wall thickness are commonly referred to as the ¼T and ¾T positions in the vessel wall. The fast neutron fluence at the vessel clad-bare metal interface is used in the PTS evaluations, the ¼T and ¾T fluences are used in the P-T curve development, the USE projections use ¼T fluence. Vessel fluence calculations were performed using the methodology described in section 5.4.3.3 for end of life (EOL) conditions which is defined as 32 EFPY. The results are presented in tables 5.4-7 and 5.4-8.

5.4.3.6 Measurement of the Initial NDT Temperature of the Reactor Pressure Vessel Base Plate and Forging Material

The unirradiated or initial NDT temperature of pressure vessel base plate and forging materials is presently measured by two methods. These methods are the drop weight test per ASTM E208 and the Charpy V-notch impact test (Type A) per ASTM E23. The NDT temperature is defined in ASTM-E208 as "the temperature at which a specimen is broken in a series of tests in which duplicate no break performance occurs at 10°F higher temperature." Using the Charpy V-notch test, the NDT temperature is defined as the temperature at which the energy required to break the specimen is a certain "fixed" value.

For SA 533B Class 1 and A508 Class 2 and Class 3 steel the ASME III Table N-421 specifies an energy value of 30 ft-lb. This value is based on a correlation with the drop weight test and is referred to as the 30 ft-lb "fix." A curve of the temperature versus energy absorbed in breaking the specimen is plotted. To obtain this curve, 15 tests are performed which include three tests at five different temperatures. The intersection of the energy versus temperature curve with the 30 ft-lb ordinate is designated as the NDT temperature.

As part of the Westinghouse surveillance program, Charpy V-impact tests, tensile tests, and fracture mechanics specimens are taken from the core region plates and forgings, and core region weldments including heat-affected zone material. The test locations are similar to those used in the tests by the fabricator at the plate mill.

The uncertainties of measurement of the NDT of base plate are:

1. Differences in Charpy V-notch foot-pound values at a given temperature between specimens.

2. Variation of impact properties through plate thickness.

The fracture toughness technology for pressure vessels and correlation with service failures based on Charpy V-notch impact data are based on the averaging of data. The Charpy V-notch 30 ft-lb "fix" temperature is based on multiple tests by the material supplier, the fabricator, and by Westinghouse as part of the surveillance program. In the review of available data, differences of 0 to approximately 40°F are observed in comparing curves plotted through the minimum and average values respectively. The value of NDT temperature derived from the average curve is judged to be representative of the material because of the averaging of at least 15 data points, consistent with the specified procedures of ASTM E23. In the case of the assessment of NDT temperature shift due to fast neutron flux, the displacement of transition curves is measured. The selection of maximum, minimum of average curves for this assessment is not significant since like curves are used.

There are quantitative differences between the NDT temperature measurement at the surface, 1/4 thickness or the center of a plate. Differences in NDT temperature between 1/4 thickness and the center in heavy plates had been observed to vary from improvement in the NDT temperature to increases up to 85°F. The NDT temperature at the surface had been measured to be as much as 85°F lower than at 1/4 thickness.

The 1/4 thickness location is considered conservative since the enhanced metallurgical properties of the surface are not used for the determination of NDT temperature. In addition, the limiting NDT temperature for the reactor vessel after operation is based on the NDT temperature shift due to irradiation. Since the fast

neutron dose is highest at the inner surface, usage of the 1/4 thickness NDT temperature criterion is conservative.

Data are being accumulated on the variation of NDT across heavy section steels at Westinghouse Nuclear Energy Systems. Similarly, the Pressure Vessel Research Committee sponsors an evaluation of properties of pressure vessel steels in plates and forgings greater than 6 inches thick. Preliminary data show NDT temperature differences between 1/4 thickness and center of less than 20°F. The present criteria of using NDT temperature +60°F at the 1/4 thickness location without taking advantage of the enhanced properties at the surface of reactor vessel plates is conservative.

To assess any possible uncertainties in the consideration of NDT temperature shift for welds, heat affected zone, and base metal, test specimens of these three "material types" are included in the reactor vessel surveillance program.

5.4.4 Tests and Inspections

The inspections of the reactor vessel were governed by the ASME Code requirements. The reactor vessel inspections are summarized in Table 5.2-26.

A preoperational volumetric examination utilizing ultrasonic techniques was performed on both reactor pressure vessels. This preoperational examination established a base line upon which the results of subsequent inservice inspections can be compared.

All of the detailed examinations, as set forth in the Technical Specifications, were performed completely, as part of the preservice inspection program which included where practicable 100 percent of the pressure - retaining welds. Evaluations are made of any indications detected during any of the examinations which exceed the standards for materials and welds specified in the ASME Code, Section III Edition applicable to the construction of the component to determine disposition and/or the need to make repairs.

The inservice inspection program is discussed in Section 5.2.8.

5.4.5 References for Section 5.4

1. RSICC Computer Code Collection CCC-650, "DOORS 3.1, One- Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

2. 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.

3. 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.

4. Fracture Toughness Analysis for Salem Units 1 and 2 Reactor Pressure Vessels to Protect Against Pressurized Thermal Shock Events 10CFR50.61, PSE&G Report NFU-060 Revision 0, dated January 10, 1986.

TABLE 5.4-1

REACTOR VESSEL MATERIAL PROPERTIES

Core Region Shell Plate	Plate No.	T_{NDT} °F	50 Ft-Lb 35 Mil Temp °F	RT_{NDT} °F	USE[1] Ft-lbs
Inter. Shell	B4712-1	0	<60	0	106
Inter. Shell	B4712-2	-20	72	12	97
Inter. Shell	B4712-3	-50	70	10	107
Lower Shell	B4713-1	-10	68	8	98
Lower Shell	B4713-2	-20	68	8	103
Lower Shell	B4713-3	-10	70	10	121

(1) USE (Upper Shelf Energy), unirradiated

TABLE 5.4-2

REACTOR VESSEL MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Mat'l. Spec. No.	Bar No.	0.2 ys Ksi	UTS[1] Ksi	Elong.	RA[2]	Energy At 10°F Ft-Lbs
37677	A540, B24	237-1	150.5	165.0	17.0	57.1	50, 54, 48
37677	A540, B24	237	151.0	164.5	17.0	57.3	54, 54, 50
37677	A540, B24	203-1	155.5	165.0	16.0	57.3	58, 60, 60
37677	A540, B24	203	148.0	164.0	15.0	56.4	48, 48, 47
37677	A540, B24	204-1	151.5	166.0	16.5	57.3	54, 50, 56
37677	A540, B24	204	148.5	162.5	16.0	56.4	54, 54, 55

CLOSURE HEAD NUTS AND WASHERS

46251	A540, B23	5	157.5	170.0	16.0	54.7	42, 46, 46
46251	A540, B23	5-1	149.5	163.0	17.0	55.8	50, 54, 53
46251	A540, B23	8	152.5	164.0	16.5	55.3	50, 50, 49
46251	A540, B23	8-1	151.5	162.2	17.0	56.9	50, 50, 54
46251	A540, B23	11	151.0	163.5	17.0	55.8	50, 50, 50
46251	A540, B23	11-1	151.0	164.0	17.0	56.5	52, 54, 50
46251	A540, B23	10	150.5	161.0	17.0	56.0	50, 50, 53
46251	A540, B23	10-1	148.5	161.0	17.0	56.8	50, 55, 48
46251	A540, B23	13	153.0	164.0	17.0	57.3	54, 50, 53
46251	A540, B23	13-1	147.2	159.0	17.0	56.5	56, 50, 51
46251	A540, B23	17	148.8	161.5	17.0	56.5	52, 50, 55
46251	A540, B23	17-1	155.5	167.5	16.0	54.2	51, 51, 50
46251	A540, B23	21	155.5	168.0	16.5	55.5	48, 50, 48
46251	A540, B23	21-1	150.5	162.0	16.5	56.2	54, 54, 54
46251	A540, B23	22	150.2	162.5	16.0	56.0	54, 52, 50
46251	A540, B23	22-1	147.0	160.0	17.0	57.3	54, 56, 54

(1) UTS (Ultimate Tensile Stress)
(2) RA (Reduction Area)

TABLE 5.4-3

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TABLE 5.4-4

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TABLE 5.4-5

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TABLE 5.4-6

NUCLEAR CONSTANTS FOR NEUTRON FLUX MONITORS CONTAINED IN THE SALEM UNIT 1 SURVEILLANCE CAPSULES

Monitor Material	Reaction of Interest	Target Weight Fraction	Product Half-life	Fission Yield (%)
Copper	Cu^{63} (n,α) Co^{60}	0.6917	5.27 years	
Iron	Fe^{54} (n,p) Mn^{54}	0.0585	314 days	
Nickel	Ni^{58} (n,p) Co^{58}	0.6777	71.4 days	
Uranium-238[a]	U^{238} (n,f) Cs^{137}	1.0	30.2 years	6.3
Neptunium-237[a]	Np^{237} (n,f) Cs^{137}	1.0	30.2 years	6.5
Cobalt-aluminum[a]	Co^{59} (n,γ) Co^{60}	0.0015	5.27 years	
Cobalt-aluminum	Co^{59} (n,γ) Co^{60}	0.0015	5.27 years	

a. Denotes that monitor is cadmium-shielded

TABLE 5.4-7

FAST FLUENCE (>1MeV) FOR SALEM 1 REACTOR VESSEL BELTLINE MATERIALS*

	Fluence at clad-bare metal interface (n/cm^2)	Fluence at ¼T (n/cm^2)	Fluence at ¾T (n/cm^2)
Intermediate Shell B2402-1	1.64×10^{19}	9.77×10^{18}	3.47×10^{18}
Intermediate Shell B2402-2	1.64×10^{19}	9.77×10^{18}	3.47×10^{18}
Intermediate Shell B2402-3	1.64×10^{19}	9.77×10^{18}	3.47×10^{18}
Lower Shell B2403-1	1.64×10^{19}	9.77×10^{18}	3.47×10^{18}
Lower Shell B2403-2	1.64×10^{19}	9.77×10^{18}	3.47×10^{18}
Lower Shell B2403-3	1.64×10^{19}	9.77×10^{18}	3.47×10^{18}
Intermediate to Lower Shell Circumferential Weld Seam 9-042	1.64×10^{19}	9.77×10^{18}	3.47×10^{18}
Intermediate Shell Longitudinal Weld Seams 2-042 A&B	1.18×10^{19}	7.03×10^{18}	2.50×10^{18}
Intermediate Shell Longitudinal Weld Seam 2-042 C	6.85×10^{18}	4.08×10^{18}	1.45×10^{18}
Lower Shell Longitudinal Weld Seams 3-042 A&B	1.08×10^{19}	6.44×10^{18}	2.29×10^{18}
Lower Shell Longitudinal Weld Seam 3-042 C	1.64×10^{19}	9.77×10^{18}	3.47×10^{18}

* Fluence calculated for EOL (32EFPY)

TABLE 5.4-8

FAST FLUENCE (>1MeV) FOR SALEM 2 REACTOR VESSEL BELTLINE MATERIALS*

	Fluence at clad-bare metal interface (n/cm^2)	Fluence at ¼T (n/cm^2)	Fluence at ¾T (n/cm^2)
Intermediate Shell B4712-1	1.77×10^{19}	1.06×10^{19}	3.77×10^{18}
Intermediate Shell B4712-2	1.77×10^{19}	1.06×10^{19}	3.77×10^{18}
Intermediate Shell B4712-3	1.77×10^{19}	1.06×10^{19}	3.77×10^{18}
Lower Shell B4713-1	1.77×10^{19}	1.06×10^{19}	3.77×10^{18}
Lower Shell B4713-2	1.77×10^{19}	1.06×10^{19}	3.77×10^{18}
Lower Shell B4713-3	1.77×10^{19}	1.06×10^{19}	3.77×10^{18}
Intermediate to Lower Shell Circumferential Weld Seam 9-442	1.77×10^{19}	1.06×10^{19}	3.77×10^{18}
Intermediate Shell Longitudinal Weld Seam 2-442 A	6.94×10^{18}	4.14×10^{18}	1.47×10^{18}
Intermediate Shell Longitudinal Weld Seams 2-442 B&C	1.20×10^{19}	7.15×10^{18}	2.54×10^{18}
Lower Shell Longitudinal Weld Seams 3-442 A&C	1.20×10^{19}	7.15×10^{18}	2.54×10^{18}
Lower Shell Longitudinal Weld Seam 3-442 B	6.94×10^{18}	4.14×10^{18}	1.47×10^{18}

* Fluence calculated for EOL (32EFPY)

TABLE 5.4-9

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TABLE 5.4-10

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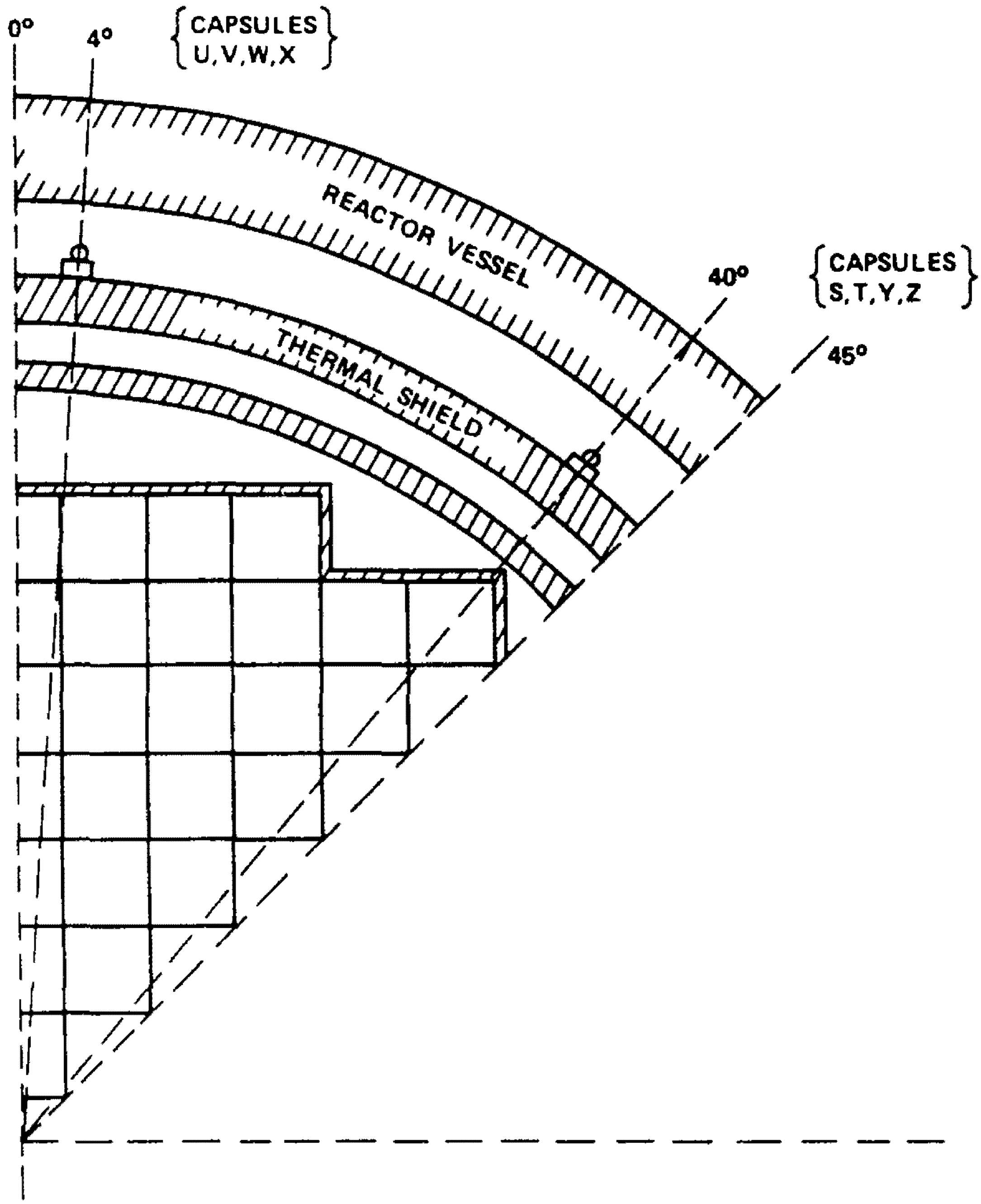
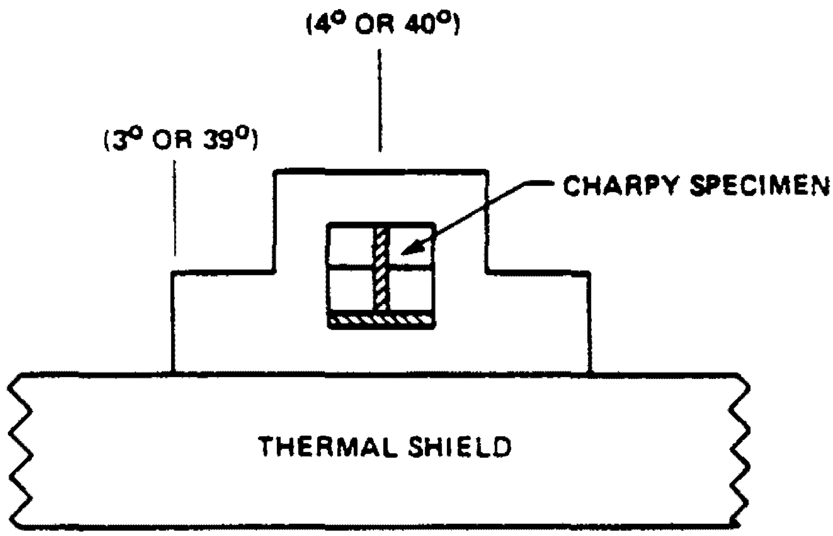


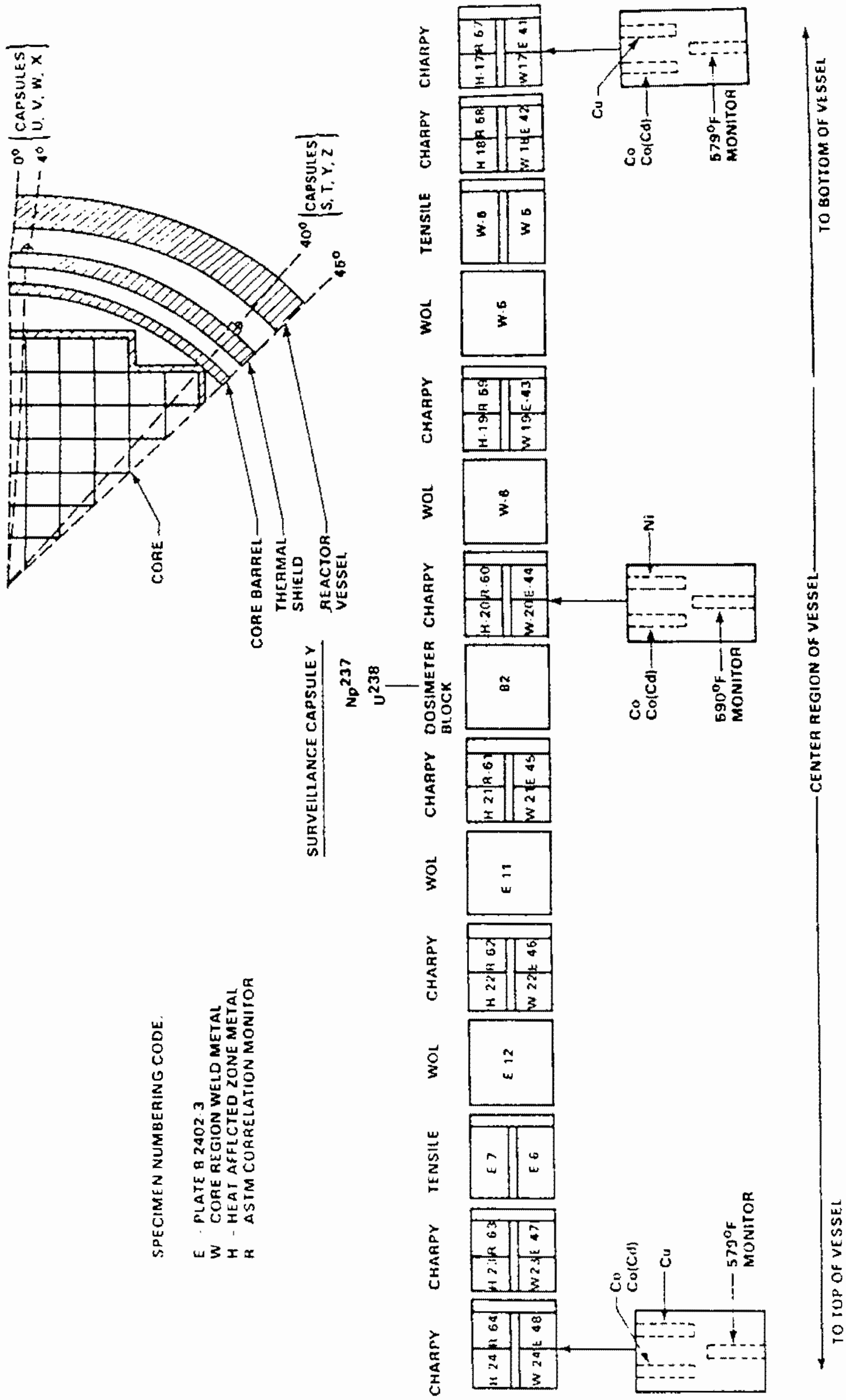
REVISION 7
JULY 22, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Salem Unit 1 Reactor Geometry
	Updated FSAR FIG. 5.4-1



REVISION 7
JULY 22, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Plan View of a Reactor Vessel Surveillance Capsule	
	Updated FSAR	FIG. 5.4-2



REVISION 7
JULY 22, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Capsule Y Diagram Showing Location of Specimens, Thermal Monitors, and Dosimeters
	Updated FSAR — FIG. 5.4-3

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 flow rate for sufficient heat transfer in order to maintain a departure from nucleate boiling ratio greater than 1.3 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 assembly rotational inertia is provided by a flywheel, motor rotor, and pump rotating parts to provide adequate flow during coastdown. This forced flow following an assumed loss of pump power and the subsequent natural circulation effect provides the core with adequate cooling.

The RCP motor is capable of operation without mechanical damage at overspeeds up to and including 125 percent of normal speed.

The RCP is shown on Figure 5.5-1, and its design parameters are given in Table 5.2-6. Code and material requirements are provided in Tables 5.2-9 and 5.2-27.

5.5.1.2 Design Description

The RCP is a vertical, single-stage, centrifugal shaft seal pump designed to pump large volumes of reactor coolant at high temperatures and pressures. The pump consists of three areas from bottom to top: the hydraulics, the shaft seals, and the motor.

1. The hydraulic section consists of an impeller, a diffuser, casing, thermal barrier, heat exchanger, lower radial (pump) bearing, main flange, motor stand, and pump shaft.

2. The shaft seal section consists of three seals. They are the number 1 controlled leakage, film riding face seal and the numbers 2 and 3 rubbing face seals. These seals are contained within the seal housings.

2a. Some RCPs have been retrofitted with the Westinghouse shutdown seal (SDS), as captured in applicable Salem RCP General Assembly Drawings. For those that have been retrofitted, the shaft seal section consists of the No. 1 controlled leakage, film riding face seal, a SDS assembly, and the No. 2 and No. 3 rubbing face seals. The seals are contained within the main flange and seal housing.

3. 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. Component cooling water is supplied to the thermal barrier heat exchanger.

High pressure seal injection water is introduced through a connection on the thermal barrier. A portion of this water flows through the seals; the remainder flows down the shaft through and around the bearing and the thermal barrier (where it acts as a buffer to prevent system water from entering the radial bearing and seal section of the unit) and into the Reactor Coolant System (RCS). The thermal barrier heat exchanger provides a means of cooling reactor coolant to an acceptable level in the event that seal injection flow is lost. The water lubricated journal-type pump 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. Component cooling water is supplied to the two oil coolers on the pump motor.

The motor is an air-cooled, Class B thermelastic 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 anti-reverse rotation device.

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 exhausted to the containment environment.

Each of the RCPs 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, one in line with the pump discharge and the other perpendicular to the pump discharge, are mounted in the same horizontal plane near the pump shaft. Frame vibration is measured by two velocity seismoprobes located 90 degrees apart in the same horizontal plane and mounted at the top of the motor support stand. Proximeters and converters convert the probe signals to linear output, which is displayed in the control room. Both units display caution and danger limits of vibration.

All parts of the pump in contact with the reactor coolant are austenitic stainless steel, except for seals, bearings, and special parts.

The pump internals, motor, 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 flywheel cover.

5.5.1.3 Design Evaluation

5.5.1.3.1 Pump Performance

The RCPs are sized to deliver flow at rates that equal or exceed the required flow rates. Initial RCS tests confirm the total delivery capability, providing assurance of adequate forced circulation coolant flow prior to initial plant operation. The performance characteristics are shown on Figure 5.1-5.

The reactor trip system ensures that pump operation is within the assumptions used for loss-of-coolant flow analyses, and 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 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 Number 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 Number 1 seal entirely removed, which puts full system pressure on the Number 2 seal, shows that relatively small leakage rates would be maintained for a period of time which is sufficient to secure the pump. Even if the Number 1

seal fails entirely during normal operation, the Number 2 seal would maintain these small leakage rates if the proper action is taken by the operator. The plant operator is warned of Number 1 seal damage by the increase in Number 1 seal leakoff rate. Following warning of excessive seal leakage conditions, the plant operator should close the Number 1 seal leakoff line and secure the pump, as specified in the instrumentation manual. Gross leakage from the pump does not occur if the proper operator action is taken subsequent to warning of excessive seal leakage conditions.

Some RCPs have been retrofitted with the Westinghouse SDS. For those that have been retrofitted, the SDS is housed within the No. 1 seal insert and is a passive device actuated by high seal leakoff temperature resulting from a loss of seal injection and CCW cooling to the thermal barrier heat exchanger. Normal LOOP is not expected to cause elevated seal temperatures which would activate the SDS. The SDS is designed to actuate only when exposed to an elevated fluid temperature downstream of the RCP number 1 seal, resulting from an extended loss of seal injection and CCW flow to the thermal barrier heat exchanger. SDS activation limits leakage from the RCS through the RCP seal package.

5.5.1.3.2 Coastdown Capability

It is important to reactor protection that the reactor coolant continues to flow for a short time after reactor trip. In order to provide this flow in a loss of offsite power 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. Some RCPs have been retrofitted with the Westinghouse SDS, as captured in applicable Salem RCP General Assembly Drawings. For those that have been retrofitted, an actuation of the SDS will not have any measureable impact on RCP coastdown or on the pump’s capability to provide sufficient cooling flow to the reactor core. The Pump Motor System is designed for the safe shutdown earthquake (SSE) at the site. Hence, it is concluded that the coastdown capability of the pumps is maintained even under the most adverse case of a blackout coincident with the SSE.

5.5.1.3.3 Flywheel Integrity

Demonstration of integrity of the RCP flywheel is discussed in Section 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, so as to give accurate alignment and smooth operation over long periods of time. 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 lube oil reservoirs signal an alarm in the Control Room and require shutting down of the pump if RCP motor bearing temp and/or vibrations are abnormally high. Each motor bearing contains embedded temperature detectors, and initiation of failure, separate from loss of oil, is indicated and alarmed in the Control Room as a high bearing temperature. This again requires pump shutdown. Even if these indications are ignored, and the bearing proceeded to failure, the low melting point of Babbitt metal on the pad surfaces ensures that no sudden seizure of the bearing occurs. In this event, the motor continues to operate as it has sufficient reserve capacity to drive the pump under such conditions. The high torque required to drive the pump, however, demands high current, which leads to the motor being shut down by the Electrical Protection Systems.

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 would fail in torsion just below the coupling to the motor, thereby disengaging the flywheel and motor from the shaft. This would constitute a loss-of-coolant flow in the loop. Following such a postulated seizure, the motor would continue to run without any overspeed, and the flywheel would maintain its integrity, as it is still supported by the motor with two bearings.

There are no other credible sources of shaft seizure other than impeller rubs. Sudden seizure of the pump bearing is precluded by graphite in the bearing. Any seizure in the seals results in a shearing of the anti-rotation pin (or pins for the RCPs which have installed the upgrade No. 1 RCP SIGMA Seal) in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Protective relays are provided to trip the supply breaker on an overcurrent condition during startup and normal operation. Indication of pump malfunction is provided by the following alarms: bearing water high temperature, excessive Number 1 seal leakoff, and excessive pump vibration. If a pump malfunction is indicated, the affected pump is taken out of service for investigation.

5.5.1.3.6 Critical Speed

The RCP shaft is designed so that its operating speed is below its first critical speed. This shaft design, even under the most severe postulated transient, gives low values of actual stress.

5.5.1.3.7 Missile Generation

Precautionary measures taken to preclude missile formation from RCP components assure that the pumps will not produce missiles under any anticipated accident conditions. 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 best estimate flow is approximately 170 feet (approximately 85 psi). In order for the controlled leakage seal to operate correctly, it is necessary to require a minimum differential pressure of approximately 200 psi across the Number 1 seal. This corresponds to a primary loop pressure at which the minimum 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, the generator breaker disconnects the generator permitting the RCPs to remain connected to the external network for 30 seconds 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. However, 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 of turbine speed. In case a generator trip deenergizes the pump buses, the RCP motors are transferred to offsite power within six to ten cycles.

5.5.1.3.10 Anti-Reverse Rotation Device

Each of the RCPs is provided with an anti-reverse rotation device in the motor. This anti-reverse mechanism consists of 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 slowed and come to a stop, the dropped pawls engage the ratchet plate and, as the motor tends to rotate in the opposite direction, the ratchet plate also rotates until it is stopped by the shock absorbers. The rotor remains in this position until the motor is energized again. When the motor is started, the ratchet plate is returned to its original position by the spring return.

As the motor begins to rotate, the pawls drag over the ratchet plate. When the motor reaches sufficient speed, the pawls are bounced into an elevated position and are held in that position by friction resulting from centrifugal forces acting upon the pawls. Considerable plant experience with the anti-reverse rotation device has 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. Charging flow is directed to each RCP via a seal water injection filter. It enters the pump and is directed to a point between the pump shaft bearing and the pump seals. The flow splits and a portion flows down the shaft through and around the lower radial bearing, down past the thermal barrier heat exchanger and into the RCS; the remainder flows up the pump shaft annulus and provides a back pressure on the Number 1 seal and a controlled flow through the seal. Above the seal, most of the flow leaves the pump via the Number 1 seal leak-off line. Minor flow passes through the Number 2 seal and its leak-off line, and through the Number 3 seal and its leak-off line.

Some RCPs have been retrofitted with the Westinghouse SDS, as captured in applicable Salem RCP General Assembly Drawings. For those that have been retrofitted, in the event of a loss of seal injection and CCW flow to the thermal barrier heat exchanger, reactor coolant begins to travel along the RCP shaft and displace the cooler seal injection water. The SDS, designed to actuate only when exposed to an elevated fluid temperature downstream of the RCP number 1 seal, actuates via retraction of a thermal actuator, which causes the SDS piston ring to constrict around the No. 1 seal sleeve. SDS actuation controls shaft seal leakage and limits the loss of reactor coolant via the RCP seal package.

5.5.1.3.12 Seal Discharge Piping

Discharge pressure from the Number 1 seal is reduced to that of the volume control tank. Water from each pump's Number 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 Number 2 and Number 3 leak-off line dump Number 2 and Number 3 seal leakage to the reactor coolant drain tank.

5.5.1.3.13 Loss of Offsite AC Power

During normal operation, seal injection flow from the Chemical and Volume Control System (CVCS) is provided to cool the RCP seals and the Component Cooling Water System provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the RCP internals. In the event of loss of offsite power, the RCP is deenergized and both of these cooling supplies are terminated; however, the diesel-generators are automatically started and either seal injection flow or component cooling water to the thermal barrier heat exchanger is automatically restored within seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to loss of seal cooling during a loss of offsite power to at least 2 hours.

Some RCPs have been retrofitted with the Westinghouse SDS. For those that have been retrofitted, the SDS is housed within the No. 1 seal insert and is a passive device actuated by high seal leakoff temperature resulting from a loss of seal injection and CCW cooling to the thermal barrier heat exchanger. Normal LOOP is not expected to cause elevated seal temperatures which would activate the SDS. The SDS is designed to actuate only when exposed to elevated fluid temperature downstream of the RCP No. 1 seal, resulting from an extended loss of seal injection and CCW flow to the thermal barrier heat exchanger. SDS activation limits leakage from the RCS through the RCP seal package.

5.5.1.3.14 Loss of Component Cooling Water

Loss of component cooling water and its effects on the RCP are discussed in Section 9.2.

5.5.1.4 Test and Inspections

Pressure boundary parts of the RCPs can be inspected in accordance with the ASME Code for "Inservice Inspection of Nuclear Reactor Coolant Systems," Section XI.

The pump casing is cast in two pieces, and joined by electroslag welding. 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.

The RCP quality assurance program is given in Table 5.2-26.

5.5.2 Steam Generators

5.5.2.1 Design Bases

Steam generator design data are given in Table 5.2-5 for Unit 2 and Table 5.2-5a for Unit 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 Section 11. Rupture of a steam generator tube is discussed in Section 15.

The internal moisture separation equipment is designed to assure that moisture carryover does not exceed 0.25 percent by weight for Unit 1 and 0.1 percent by weight for Unit 2 under the following conditions:

1. Steady-state operation up to 105 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 from 15 percent to 105 percent of full-load steam flow.

3. A step load change of 10 percent of full power in the range from 15 percent to 105 percent full-load steam flow.

The steam generator tube sheet complex meets the stress, limitations and fatigue criteria specified. Code and materials' requirements of the steam generator are given in Tables 5.2-9 and 5.2-27.

The steam generator design maximizes integrity against hydrodynamic excitation and vibration failure of the tubes for plant life.

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.

5.5.2.2 Design Description

The Unit 2 steam generator shown on Figure 5.1-3 (for Salem Unit 2) is an AREVA NP Model 61/19T, vertical shell and U-tube evaporator with integral moisture separating equipment. The Model-F steam generator for Unit 1 is shown in Figure 5.1-3a. The Model-F is very similar to the original Series 51 generator except in tube dimensions, number of tubes and separators. A specific description of the Model-F steam generator is given in Section 5.5.2.2.2.

5.5.2.2.1 Unit 2 AREVA NP Model 61/19T Steam Generators

The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. Manways are provided for access to both sides of the divided head. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The unit is primarily carbon steel. The heat transfer tubes are Inconel 690 thermally treated and the divider plate is Inconel 690.

The interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. The primary side of the tube sheet is weld clad with Inconel 600.

Feedwater flows directly into the annulus formed by the shell and tube bundle wrapper before entering the boiler section of the steam generator. Subsequently, water-steam mixture 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 flow mixes with 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.

5.5.2.2.2 Unit 1 Model-F Steam Generators

The Model-F steam generators are vertical U-tube steam generators that were designed and fabricated in accordance with the ASME Code, 1971 Edition, Summer 1973 Addenda, Class 1, Division 1 for use in a closed cycle pressurizer water reactor system. Unit 1 has Model-F steam generators.

The tube bundle of the Model-F steam generator consists of 5626 thermally-treated U-tubes fabricated from ASME SB-163 (Inconel). The O. D. of each tube is 0.688 inches with a nominal tube wall thickness of 0.040 inches. The ends of the tubes are expanded the full depth of the tube plate, and the tubes are welded to the Inconel cladding on the primary face of the tube plate. Overall height of the Model-F steam generator tube bundle extends approximately 348 inches above the secondary face of the tube plate.

The Model-F steam generators utilize two-stage moisture separators to remove moisture from the wet steam produced by the tube bundle to deliver dry steam to the turbine generator. The first stage separator assembly is located directly above the tube bundle. It is approximately 10 feet high and contains sixteen 20 inch diameter swirl vane assemblies. Steam at the exit end of the first stage separators still contains some entrained moisture and is passed through the second stage separators, containing banks of contoured vanes designed to remove water from steam.

The moisture separator housing consists of two four-sided tiers, one above the other, providing the frames in which the banks of vanes are installed.

The steam outlet nozzle has an I. D. of 29 inches and is located at the apex of the upper elliptical head. The steam outlet nozzle contains a flow limiting device which operates on the venturi principle, to choke flow in the event of a steam line break.

Model-F steam generators are equipped with a blowdown nozzle (2 in. dia.) and a drain nozzle (2 in. dia.). Liquid level connections provide openings for narrow and wide range water level instrumentation used for feedwater control and reactor protection systems. Wide range taps provide a range of 560 inches and narrow range taps provide a range of 128 inches for liquid level measurement. A sampling nozzle (2 inch nominal) is provided in Model-F steam generators used in Unit 1.

5.5.2.3 Design Evaluation

5.5.2.3.1 Natural Circulation Flow

The steam generators (which provide a heat sink) are at a higher elevation than the reactor core (which is the heat source). Thus, natural circulation is assured for the removal of decay heat.

5.5.2.3.2 Secondary System Fluid Flow Instability Prevention

In order to prevent the occurrence of water hammer, the feedwater distribution ring header flow takes place through the top of the headers, rather than out the bottom. This modification has been demonstrated to preclude water hammer as discussed in detail in Section 10.4. The Model-F steam generators in Unit 1 were originally fabricated with this feature to preclude water hammer. The Unit 2 RSG design also incorporates J-tubes and internals that maximize secondary side water inventory above the feed ring with an all-welded thermal sleeve/ring assembly. This eliminates the possibility of steam leakage into the feed ring through a sliding connection.

The limiting case for heat transfer capability is the "Nominal 100 Percent Design" case. The steam generator effective heat transfer coefficient and recirculation ratio are based on the coolant conditions of temperature and flow for this case, and includes a conservative allowance for tube fouling. Adequate tube area is selected to assure that the full design heat removal rate is achieved.

The fouling factor resistance of 0.00005 hr-ft^2-°F/Btu is the value selected to account for the differences in the measured and calculated heat transfer performance as well as provide the margin indicated above. Although margin for tube fouling is available, operating experience to data has not indicated that steam generator performance decreases over a long-term period.

5.5.2.3.3 Tube and Tube Sheet Stress Analyses

Tube and tube sheet stress analyses of the steam generator are given in Section 5.2. Calculations confirm that the steam generator tube sheet will withstand the loading (which is quasi-static rather than a shock loading) caused by loss of reactor coolant.

5.5.2.3.4 Flow-Induced Vibration

In the design of the steam generators, consideration has been given to the possibility of vibratory failure or wear of tubes due to flow-induced excitation. This consideration includes detailed analysis of the tube supporting system based on an extensive research program in both vibration and wear domains. The major cause of tube vibratory failure in heat exchanger components is that due to hydrodynamic excitation by the fluid outside the tube.

Consideration is given to three regions where the possibility of flow-induced vibration may exist:

1. At the entrance of downcomer feed to the tube bundle (cross flow)

2. Along the straight sections of the tube (parallel flow)

3. In the curved tube section of the U-bend (cross flow)

From the description of these regions, it is noted that two types of flow exist, namely, cross flow and parallel flow. For the case of parallel flow, analysis is done 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.

Cross flow-induced vibration analysis are performed to confirm that the tube bundle is adequately supported to avoid significant levels of tube vibration. Vibration Analysis and a Wear Analysis are established to verify that vibrations do not result in excessive wear or fatigue throughout the tube bundle and U-bend regions.

The three pertinent cross Flow-Induced Vibration mechanisms of the steam generator tubes are (1) fluid elastic instability, (2) vortex shedding resonance, and (3) random turbulence. The FIV analysis verifies that excessive tube vibration from these sources is avoided. Particular areas of emphasis are the tube bundle entrance and the U-bend region as indicated above.

The first mechanism, fluid elastic instability, is a mechanism that may cause the fast rupture of the tubes (vibration amplitudes to increase sharply when a certain critical flow velocity is exceeded). The ratio of the effective crossflow velocity to the critical velocity at any point in the bundle is called the stability ratio. Fluid elastic instability occurs when the stability ratio is greater than or equal to 1.0 so for design, the acceptance criterion for the stability ratio is generally some margin less than 1.0.

The second FIV excitation mechanism is vortex shedding resonance. When fluid flows across a circular cylinder, the wake behind the cylinder contains vortices. The vortices detach from the cylinder in a regular manner, i.e. at a certain frequency, and cause the tubes to vibrate at the same frequency in a direction perpendicular to the flow direction. When, at a critical crossflow velocity, the vortex shedding frequency happens to be close to a tube natural frequency, the vibration of the tube can organize the wake, causing it to synchronize (lock-in) with the tube motion at the tube natural frequency. This phenomenon is called vortex shedding resonance.

In a tube bundle, close spacing of tubes greatly suppresses the formation of organized wakes. For flow in tube bundles, vortex shedding resonance has never been observed in Steam Generator tube bundles having pitch/tube OD ratios lower than 1.46. For the steam generators, the pitch/tube OD ratio is 1.43 for Unit 1 and 1.44 for Unit 2 RSGs, which are less than 1.46.

The third mechanism, random turbulence excitation, is the buffeting of the tubes primarily from the turbulence in the flow. It is the "background" mechanism that accounts for tube vibration below fluid elastic instability and outside regions of vortex shedding resonance. It results in relatively low levels of vibration that increase with increasing flow velocity, with amplitudes and mode shapes varying randomly in time and in direction. Vibration analysis shows that amplitudes are small enough to exclude risks of fatigue due to turbulence response.

Three-dimensional analyses are performed to derive detailed flow distributions in the U-bend area. From this analysis, velocity and density profiles are determined along the tubes. Finite element analysis is then used to predict mode shapes for each U-bend supporting structures (i.e., 3 sets of AVBs) and for various mode types and frequencies.

The FIV computer code, and the finite element code are used to determine if the Fluid Elastic Instability (FEI) threshold velocity is avoided and to analyze random turbulent excitation. The potential for fretting is assessed by FIV wear analysis. The FIV analysis is used to confirm that the tube bundle is adequately supported to prevent excessive tube wear due to FIV excitation mechanisms.

The RSG bundle design parameters that are most important for controlling FIV are:

1. Tube and support materials.
2. Tube outside diameter, thickness and pitch/diameter ratio, and diametric clearance at the tube supports.
3. Bundle height.
4. Bend radius of the outermost tube.
5. Number of tube support plates.
6. Number of Anti-Vibration bar sets.
7. Width of Anti-Vibration bars.
8. Steam flow at full power.
9. Circulation ratio.

Thus, the three pertinent cross-flow induced vibration mechanisms have been analyzed. The results show that all the acceptance criteria are met, and it is concluded that the tube bundle is adequately supported for the prevention of detrimental flow-induced vibration.

Summarizing the results of analysis and tests of steam generator tubes for flow-induced vibration, it can be stated that a check of support adequacy has been made using all published techniques appropriate to heat exchanger tube support design. In addition, the Tube Support System is consistent with accepted standards of heat exchanger design utilized throughout the industry (spacing, clearance, etc.). Furthermore, the design techniques are supplemented with a continuing research and development program to understand the complex mechanism of concern. Service experience of steam generators also shows that flow-induced vibration and cavitation effects do not cause tube thinning.

The effects of vibration, erosion, and cavitation have been given consideration and the stress limitations for each category have been met. Analysis of loss-of-coolant accident (LOCA) blowdown forces on as-fabricated U-tubes has shown that the maximum bending load elastic stress intensity is well below the faulted condition limit. The maximum bending load elastic stress intensity (based on the minimum tube wall thickness) would increase only within the range of 5 to 10 percent and would still be below the faulted condition limit. Therefore, as a minimum, at least 2 1/2 mils (per wall) thinning can be tolerated without exceeding the allowable stress limits. Vibration effects are negligible during normal operation by the supporting system. Under LOCA conditions, vibration is of a 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.

Finally, it should be noted that successful operational experience with several steam generator designs, including both the Unit 1 and Unit 2 replacement steam generators, has given confidence in the overall approach to the tube support design problem.

Tests and Inspections

The steam generator quality assurance program is given in Table 5.2-26.

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 tube sheet cladding, channel head cladding, tube-to-tube sheet 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 casting (Unit 1), channel head forging (Unit 2), nozzle forgings, and the following weldments:

1. Nozzle to shell

2. Support brackets

3. Instrument connections (primary and secondary)
   (Note: Unit 1 and Unit 2 replacement steam generators have no primary instrumentation connections)

4. Temporary attachments after removal

5. All accessible pressure containing welds after hydrostatic test.

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 tube sheet forging, tube sheet cladding, secondary shell and heat plate and nozzle forgings.

The heat transfer tubing is subjected to 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 for access to both the primary and secondary sides.

Steam generator tube inspection will be performed in accordance with Technical Specifications. Due to activity in the channel head and the large number of tubes involved, tube testing is done on a per-plant basis. The extent of tube testing planned in any particular plant will depend on tube performance to date, the channel head activity, and the results of tube sample testing. An eddy current testing method is available if the tubes should require inspection.

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 5.2.

Materials of construction are specified to minimize corrosion/erosion and assure compatibility with the operating environment.

RCS pressure boundary piping codes for both units are stated in Table 5.2-9.

5.5.3.2 Design Description

Principal design data for the reactor coolant piping for both units are given in Table 5.2-7. 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 5.2-7. The line between the steam generator and the pump suction is larger to reduce pressure drop and improve flow conditions to the pump suction.

All piping within the reactor coolant pressure boundary is made of austenitic stainless steel with the main piping being seamless forged. Fittings are one-piece castings with the exception of the RCP inlet 90 degree elbow which is two half castings joined by electroslag welding.

All smaller piping which comprise 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 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 due to rapid changes in fluid temperature during normal operational transients. These points include:

1. Charging connections at the primary loop from the CVCS.

2. Both ends of the pressurizer surge line.

3. Pressurizer spray line connection at the pressurizer.

4. Safety Injection/Residual Heat Removal System return.

Thermal sleeves are not provided for the remaining injection connections of the ECCS since these connections are not in normal use. All piping connections from auxiliary systems are made above the horizontal centerline of the reactor coolant piping, with the exception of:

1. Residual heat removal (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 continuing to operate the RHR System, should this be required for maintenance.

2. Loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.

3. The differential pressure taps for flow measurement are downstream of the steam generators on the first 90 degree elbow.

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

1. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force.

2. The Reactor Coolant Sample System taps protrude into the main stream to obtain a representative sample of the reactor coolant.

3. The hot leg temperature detectors are located in resistance temperature detector wells that extend into the reactor coolant pipe.

4. The wide range temperature detectors are located in resistance temperature detector wells that extend into the reactor coolant pipes.

5. The differential pressure transmitters are connected to the RCS sample line to monitor RCS level during Mid-Loop operation.

Each reactor coolant loop is provided with RTDs so that individual temperature signals may be developed for use in the Reactor Control and Protection System.

A description of the installation and operation of the RTDs is provided in Sections 5.6.1 and 7.2.3.2.

Signals from these instruments are used to compute the reactor coolant ΔT (temperature of the hot leg, T_{hot}, minus the temperature of the cold leg, T_{cold},) and an average reactor coolant temperature (T_{avg}). The T_{avg} for each loop is indicated on the main control board.

The RCS pressure boundary piping includes those sections of piping interconnecting the reactor vessel, steam generator, and RCP. It also includes the following:

1. Charging line and alternate charging line from the isolation valve up to the branch connections on the reactor coolant loop.

2. Letdown line and excess letdown line from the branch connections on the reactor coolant loop to the isolation valve.

3. Pressurizer spray lines from the reactor coolant cold legs to the spray nozzle on the pressurizer vessel.

4. RHR lines to or from the reactor coolant loops up to the designated isolation or check valve.

5. Safety injection lines from the designated isolation or check valve to the reactor coolant loops.

6. Accumulator lines from the designated isolation or check valve to the reactor coolant loops.

7. Resistance temperature detector thermowells.

8. Loop fill, loop drain, sample, and instrument lines to or from the designated isolation valve to or from the reactor coolant loops.

9. Pressurizer surge line from one reactor coolant loop hot leg to the pressurizer vessel inlet nozzle.

10. Resistance temperature detector scoop element, pressurizer spray scoop, sample connection with scoop, reactor coolant temperature element installation boss, and the temperature element well itself.

11. All branch connection nozzles attached to reactor coolant loops.

12. 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.

13. Seal injection water and labyrinth differential pressure lines to or from the RCP inside reactor containment.

14. Auxiliary spray line from the isolation valve to the pressurizer spray line header.

15. Sample lines from pressurizer to the isolation valve.

Details of the materials of construction of reactor coolant piping and fittings are listed in Table 5.2-26.

5.5.3.3 <u>Design Evaluation</u>

Piping load and stress evaluation for normal operating loads, seismic loads, blowdown loads, and combined normal, blowdown, and seismic loads is discussed in Section 5.2.

5.5.3.4 <u>Material Corrosion/Erosion Evaluation</u>

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 50 feet per second 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.

5.5.3.5 Tests and Inspections

Inservice inspection is discussed in Section 5.2.8. The RCS piping quality assurance program is given in Table 5.2-26.

A description of the quality assurance inspections of these components is contained in Section 5.2.3.5.

5.5.4 Main Steam Line Flow Restrictors

Each steam line is provided with a flow restrictor to limit the blowdown rate of steam from the steam generators in the event of a main steam line rupture. The flow restrictors are described in detail in Section 10.3.

In addition to the main steam line flow restrictors, the steam generators have flow restrictors in the steam outlet nozzle each with a flow area of 1.4 ft^2.

5.5.5 Main Steam Line Isolation System

Main steam isolation valves are described in Section 10.3.

5.5.6 Reactor Core Isolation Cooling System

This section is not applicable to PWRs.

5.5.7 Residual Heat Removal System

5.5.7.1 Design Bases

The RHR System is designed to remove residual and sensible heat from the core and reduce the temperature of 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 (Section 10).

The RHR System is placed in operation approximately 4 hours after reactor shutdown when the pressure and temperature of the RCS are less than 375 psig and 350°F, respectively. Under normal operating conditions, the RHR System can reduce the temperature of the reactor coolant to 140°F within 22 hours following reactor shutdown. The design residual heat load was based on the residual heat fraction of full core MW (thermal) power level that exists at 20 hours following reactor shutdown from an extended power run near full power (refer to Table 5.5-1).

As a secondary function, the RHR System is used to transfer refueling water between the refueling water storage tank and the refueling cavity at the beginning and end of refueling operations.

In addition, portions of the system are utilized as parts of the ECCS and the Containment Spray System. These functions and the associated analyses are discussed in Section 6.

The RHR System provides sufficient capability in the emergency operational mode to accommodate any single active or passive failure and still function in a manner to avoid risk to the health and safety of the public. Refer to Sections 6 and 15 for a discussion of the operability and capability of the RHR System in an emergency core cooling role.

The system design precludes any significant reduction in the overall design reactor shutdown margin when cooling water is introduced into the core for decay heat removal or during emergency core cooling recirculation mode of operation.

System components whose design pressure and temperature are less than the RCS design limits are provided with redundant isolation means and overpressure protection devices.

All system active components which are relied upon to perform the system functions are redundant and the system design includes provision for hydrostatic testing of system components to applicable code test pressures.

Piping and components of the RHR System are designed to the applicable codes and standards listed in Table 5.5-1. Since the loop contains reactor coolant when it is in operation, austenitic stainless steel piping is employed.

5.5.7.2 System Description

The RHR System (shown on Plant Drawings 205232 and 205332) consists of two residual heat exchangers, two RHR pumps and associated piping valves, and instrumentation.

During system operation, coolant flows from the RCS to the RHR pumps, through the tube side of the residual heat exchangers and back to the RCS. The inlet line to the RHR System loop begins at the hot leg of one reactor coolant loop and the return line is connected to the cold legs of two separate reactor coolant loops. The heat loads are transferred by the residual heat exchangers to the component cooling water.

The cooldown rate of the reactor is controlled by regulating the flow through the tube side of the residual heat exchangers. A bypass line with a remotely-operated control valve around the residual heat exchangers is used to maintain a constant flow through the RHR System.

Coincident with plant cooldown, a portion of the reactor coolant flow may be diverted to the CVCS for cleanup. By regulating diverted flow rate, the RCS pressure may be controlled within the pressure range dictated by the nil-ductility limits of the reactor vessel and the Number 1 seal differential pressure and NPSH requirement of the RCPs.

Design data for the RHR System components described below are listed in Table 5.5-1.

Residual Heat Exchanger

Two residual heat exchangers are installed in the system. Each exchanger is designed to remove one-half of the residual heat load. The installation of two exchangers assures that the heat removal capacity of the RHR System is only partially lost if one exchanger fails or becomes inoperative. Two exchangers also allow maintenance of one exchanger while the other unit is in operation.

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 tube sheet to prevent leakage of reactor coolant.

RHR Pumps

Two identical pumps are installed in the RHR System. Each pump is sized to deliver sufficient reactor coolant flow through the residual heat exchangers to meet the plant cooldown requirements. The use of two pumps, installed in parallel, assures that pumping capacity is only partially lost should one pump become inoperative. This also allows maintenance on one pump while the other pump is in operation. In addition to the RHR duty, the pumps are used for transfer of refueling water before and after a refueling operation.

The two RHR pumps are vertical, centrifugal units with mechanical seals to prevent reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion-resistant material.

RHR Valves

The valves used in the RHR System are constructed of austenitic stainless steel or equivalent corrosion-resistant material.

Manual isolation valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote manual control of residual heat exchanger tube side flow, and for remote manual control of bypass flow. Check valves prevent reverse flow through the RHR pumps.

Isolation of the RHR System is achieved with two remotely-operated series stop valves in the line from the RCS to the RHR pump suction and by two check valves in series in each line from the RHR pump discharge to the RCS, plus a remotely-operated stop valve in each discharge line. Overpressure in the RHR System is relieved through a relief valve to the containment sump.

Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Waste Disposal System.

Manually-operated valves have backseats to facilitate repacking and to limit the steam leakage when the valves are open. Leakoff connections are provided where required by valve size and fluid conditions (with exception of MOVs 1RH2, 2RH2 and 2RH26 that are modified from a double to a single packing arrangement with leakoff ports cut and capped, and MOV 1RH26 that is modified from a double to a single packing arrangement with the leakoff port cut and plugged to reduce packing friction loads and thus helping MOV margin recovery under DCPs 80093314 (for Unit 2 valves) and 80095702 (for Unit 1 valves)).

RHR Piping

RHR piping is austenitic stainless steel. Piping joints and connections are welded except where flanged connections are required to facilitate maintenance.

5.5.7.3 Design Evaluation

For RCS cooldown, the unit is provided with two RHR pumps and two residual heat exchangers. If one of the two pumps, one of the two heat exchangers, or one pump and one heat exchanger is not operable, safe cooldown of the plant is not compromised; however, the time for cooldown is extended.

To assure reliability, the two RHR pumps are connected to two separate buses so that each pump will receive power from a different source.

An emergency power source is required to supply essential electrical equipment if a total loss of power should occur while the system is in service. Each pump is connected to a separate emergency power supply.

5.5.7.3.1 Leakage Provisions

The design operating leakage rate of the RHR System is 50 gpm due to a pump seal failure. The RHR pumps are in separate rooms containing two sump pumps, each adequate to provide the minimum capacity of 50 gpm. The sump pumps discharge to the Waste Disposal System. Sump pump reliability is maximized by using submersible type pumps. In the remote event that no sump pumps are operable, there is adequate volume in the RHR rooms to contain the design percentage while the pump is isolated.

Should a large tube-side-to-shell-side leak develop in an RHR heat exchanger, the water level in the component cooling surge tank would rise or fall depending on which system pressure is higher, and the operator would be alerted by a high or low water alarm. In addition, a leak into the Component Cooling System will be detected by a radiation monitor located in each component cooling header.

If the leaking RHR heat exchanger could not be isolated from the Component Cooling System before an inflow completely filled the surge tank, the overflow-vent line would discharge the excess water to the Waste Disposal System. If the leaking RHR heat exchanger could not be isolated from the Component Cooling System before an outflow completely drained the surge tank, remote motor operated valves in the Component Cooling System header cross-tie lines can be closed from the Control Room to split the two safety headers, thus maintaining at least one available header for safe shutdown.

Since the RHR System is required for long-term post-accident removal of decay heat from the reactor core and containment, independent piping systems are provided for the redundant components so that excessive leakage resulting from the deterioration of, or failure in, some passive element in the system can be identified and isolated without complete system loss of function.

Massive failure of piping is not considered credible because long-term operation of the system occurs only at low pressures and temperatures, and the system is protected from environmental conditions by the Class I (seismic) structures.

5.5.7.3.2 RCS Isolation Provisions

The RHR discharge lines are isolated from the RCS by two check valves in series for each line and a remote-operated valve in each line or common header.

There are two motor-operated isolation valves (RH1 and RH2), in series, in the single letdown line connecting the low-pressure RHR System to the high-pressure RCS. Valve RH1 is the upstream valve (closest to the RCS), and RH2 is the downstream valve. The position indication provided for these valves consists of "OPEN-CLOSED" indication on the main control console and valve "OFF-NORMAL" indication in the Auxiliary Alarm System.

The "OPEN-CLOSED" indication for RH1 and RH2 is powered from separate 125-vdc buses. This power is different than the source control power to the valve operators. Using separate power for control and indication ensures that indication will be maintained when control power is locked out.

The Interlock System consists of the following:

1. Valve RH1 is interlocked with a pressure control signal derived from a pressure transmitter to prevent its opening whenever the RCS pressure is greater than the RHR System

design pressure.

2. The pressure transmitter used in Item 1 is connected to the reactor coolant loop which contains the RHR suction

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line. The pressure transmitter is connected into the RHR suction line inside the containment.

3. The control for valves RH1 and RH2 is administratively locked to prevent inadvertent manual opening.

4. A second pressure channel is provided as a pressure control signal to interlock valve RH2 located adjacent to the RHR System. This will be used to prevent its opening whenever the reactor coolant pressure is greater than the RHR System design pressure.

5. This RH2 associated pressure transmitter is connected by a separate connection into the RHR suction line inside the containment. Therefore, the RHR suction line will contain two separate connections, one for each pressure transmitter.

The interlocks are designed to conform to IEEE Standard 279-1971.

Two overhead alarms are provided in the control room for RH1 and RH2:

1. The overhead alarm for RH1 is activated when the RH1 valve is not fully closed in conjunction with high reactor pressure.

2. The overhead alarm for RH2 is activated when the RH2 valve is not fully closed in conjunction with high reactor pressure.

5.5.7.3.3 Failure Analysis

A failure analysis of RHR pumps, heat exchangers, and valves is presented in Table 5.5-2.

5.5.7.3.4 Compliance with Branch Technical Position RSB 5-1

This section addr ses the items contained in Table II of Branch Technical Position (BTP) RSB 5-1 for PWR Class 2 plants. The numbering corresponds to the numbering in Table II.

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1. Double drop line (or valves in parallel) from the RCS.

A single RHR suction line with two suction isolation valves in series is provided as described in Section 5.5.7.3.2. Compliance is not required since the station can be maintained in a safe hot standby condition while any required manual actions are taken.

2. Safety-grade dump valves, operators, air, and power.

One safety-grade steam generator power-operated relief valve is provided for each of the four steam generators. Safety-grade remote operators and power supplies are not required since hot standby can be achieved and maintained using the safety-grade steam generator safety valves. The steam generator power operated relief valves are provided with handwheels and can be operated locally to permit plant cooldown. See the cold shutdown scenario and single failure evaluation provided below.

3. Capability to cool down to shutdown assuming most limiting single failure in less than 36 hours.

Compliance is not required since the station can be maintained in a safe hot standby condition while any required manual actions are taken. The plant is capable of reaching RHR initiation conditions in approximately 36 to 48 hours, including time required to perform any manual actions.

4. Depressurization with only safety-grade systems assuming single failure.

Compliance is not required since the station can be maintained in a hot standby condition while any required manual actions are taken.

5. Boration with only safety-grade systems assuming single failure.

Compliance is not required since the station can be maintained in a safe hot standby condition while any required manual actions are taken.

6. Provisions for collection and containment of RHR pressure relief discharge.

The RHR relief valves discharge to the containment sump.

7. Additional tests to study mixing of the added borated water and cooldown under natural circulation conditions with and without a single failure of an atmospheric dump valve.

Salem Generating Station is similar to Diablo Canyon Power Station in design, both being Westinghouse PWRs. Due to the similarity of the two stations, no special tests will be conducted by the Salem Unit to establish boron mixing and cooldown capability under natural circulation since Diablo Canyon Station has committed to perform these tests. The results of the tests on Diablo Canyon will be applicable for Salem.

8. Specific operational procedures for cooldown under natural circulation.

Salem Generating Station will generate specific operational procedures that will enable the operators to bring the station from hot standby condition to cold shutdown status using the systems and operating functions given in Item 9 (Cold Shutdown Scenario).

9. Seismic Category I auxiliary feedwater supply for at least 4 hours at hot shutdown plus cooldown to RHR cut-in based on longest time (for only onsite or offsite power and assuming worst single failure).

   A long-term source of auxiliary feedwater is provided by a connection to the Seismic Category I Service Water System.

Cold Shutdown Scenario (Assuming Loss of All Nonseismic Category I Equipment)

The safe shutdown design basis of the Salem Units is hot standby. The station can be maintained in a safe hot standby condition while manual actions are taken to permit achievement of cold shutdown conditions following an SSE with loss of offsite power. Under such conditions the station is capable of achieving RHR initiation conditions (approximately 350°F, 375 psig) in approximately 36 to 48 hours, including the time required for any manual actions. To achieve and maintain cold shutdown, four key functions must be performed. These are: (1) circulation of the reactor coolant, (2) removal of residual heat, (3) boration and makeup, and (4) depressurization.

Circulation of Reactor Coolant

Circulation of the reactor coolant has two stages in a cooldown from hot standby to cold shutdown. The first stage is from hot standby to 350°F. During this stage, circulation of the reactor coolant is provided by natural circulation with the reactor core as the heat source and steam generators as the heat sink. Steam release from the steam generators is initially via the steam generator safety valves and occurs automatically as a result of turbine and reactor trip. Steam release for cooldown is via the steam generator power-operated relief valves which are operated manually with their handwheels. The steam generator power-operated relief valves are accessible for local operation.

The status of each steam generator can be monitored using Class 1E instrumentation located on the console in the control room. Three separate channels of indications for both steam generator pressure and water level are available.

Feedwater to the steam generators is provided from the Auxiliary Feedwater System which has a 220,000 gallon Seismic Category I auxiliary feedwater storage tank as the primary source, and two separate Seismic Category I piping subsystems. The first subsystem is composed of two motor-driven pumps, each powered from a different emergency power train; the second subsystem incorporates a turbine-driven pump which can receive motive steam from either of two steam generators. There are additional sources of feedwater backup which can be manually accessed. Initial backup is provided by the demineralized water storage tank, the domestic water storage tank, and the fire protection water tank. Additional backup is from the Seismic Category I Service Water System. The operation of the Auxiliary Feedwater System can be monitored using Class 1E instrumentation located on the control console in the Control Room. There is a single indication of the flows into each steam generator, pump operating status lights for the motor-driven pumps, and discharge and suction pressure indication for the turbine-driven pump. There are also two separate indications of the level in the auxiliary feedwater storage tank.

The second stage of reactor coolant circulation is from 350°F to cold shutdown. During this stage, circulation of the reactor coolant is provided by the RHR pumps.

Removal of Residual Heat

Removal of residual heat also has two stages in a cooldown from hot standby to cold shutdown. The first stage is from hot standby to 350°F.

During this stage, the steam generators act as the means of heat removal from the RCS. Initially, steam is released from the steam generators' via the steam generator safety valves to maintain hot standby conditions. When the operators are ready to begin the cooldown, the steam generators' power-operated relief valves are slightly opened by local operation with their handwheels. As the cooldown proceeds, the operators will occasionally adjust these valves to increase the amount they are open. This allows a reasonable cooldown rate to be maintained. Feedwater makeup to the steam generators is provided from the Auxiliary Feedwater System. The Auxiliary Feedwater System has the ability to remove decay heat by providing feedwater to all four steam generators for extended periods of operation.

The second stage is from 350°F to cold shutdown. During this stage, the RHR System is brought into operation. The RHR heat exchangers in the RHR System act as the means of heat removal from the RCS. In the heat exchanger, the residual heat is transferred to the Component Cooling System which ultimately transfers the heat to the Service Water System. The Component Cooling and the Service Water Systems are both designed to Seismic Category I. The RHR System includes two RHR pumps and two RHR heat exchangers. Each pump is powered from different emergency power trains and each heat exchanger is cooled by a different component cooling loop. If any component in one loop becomes inoperable, cooldown of the plant is not compromised; however, the time for cooldown would be extended.

The operation of the RHR System can be monitored using Class 1E instrumentation on the control console in the Control Room. For each loop, there is indication of the pump discharge flow, the pump operation status, and the component cooling flow from the discharge of the heat exchanger.

Boration and Makeup

Boration is accomplished using portions of the CVCS. Boric acid, (3.75 to 4.0 weight percent) from the boric acid tanks is supplied to the suction of the centrifugal charging pumps by the boric acid transfer pumps. The centrifugal charging pumps may inject the borated water into the RCS via the normal charging flow path or the high head safety injection, BIT cold leg flow path. The two boric acid tanks, two boric acid transfer pumps, and the associated piping are of Seismic Category I design. There is sufficient boric acid capacity to provide for a cold shutdown with the most reactive rod withdrawn. The boric acid transfer pumps are each powered from different emergency power trains. The boric acid tank level can be monitored to verify the operability of the boration portion of the CVCS. For this, credit is taken for operator action in using a portable differential pressure indicator which can be connected to the level signal lines from the boric acid tanks.

Makeup, in excess of that provided as 3.75 to 4.0 weight percent boric acid, is provided from the refueling water storage tank (RWST) using centrifugal charging pumps and the same injection flow paths as described for boration. Two motor-operated valves, each powered from different emergency power trains and connected in parallel, will transfer the suction of the charging pumps to the RWST. Makeup from the RWST can be monitored using Class 1E instrumentation on the control console in the control room. Two separate channels of RWST level indication exist for Salem Unit 1 and four separate channels for Unit 2.

Depressurization

Depressurization is accomplished using portions of the CVCS. Either 3.75 to 4.0 weight percent boric acid or refueling water can be used as desired for depressurization with the flow path being from the centrifugal charging pumps to the auxiliary spray valve in the pressurizer. The two centrifugal charging pumps of the CVCS are of Seismic Category I, and are powered from different emergency power trains. The pumps can be operated from, and its operating

status monitored in the control room. The depressurization of the RCS can be monitored using Class 1E instrumentation on the control console in the Control Room. Available to the operator are four channels of pressurizer pressure, three channels of pressurizer level, and two channels of reactor coolant pressure.

Maintaining RCS Temperature and Pressure Without Letdown

In performing the cooldown to cold shutdown, the operator can integrate the function of heat removal, boration and makeup, and depressurization so that these functions can be accomplished without letdown from the RCS.

Without letdown available, boration is done concurrently with the RCS makeup required for cooldown contraction. Pressurizer level is maintained at normal shutdown level. The plant need not be taken water solid to accommodate the borated water. The required shutdown margin is maintained throughout the cooldown if the RCS makeup sequence described below is followed:

1. During the initial phase of the cooldown, the makeup is provided from the boric acid tanks. The boric acid tanks should be used as the sole source of makeup until at least the technical specification minimum volume has been charged.

2. Operators can continue using the boric acid tank if additional volume is required, or shift suction of the charging pumps to the RWST. If the boric acid tanks are used, pure boric acid should be charged until the RCS reaches the desired cold shutdown concentration. The cooldown is completed by using blended makeup at the cold shutdown concentration.

Finally the operators use auxiliary spray from the CVCS to depressurize the plant.

A calculation was performed (Reference 3) to demonstrate that the shutdown margin can be maintained throughout the cooldown without letdown and without taking the plant water solid. Worst case conditions of end-of-life and maximum xenon were assumed.

The assumed initial conditions following plant trip are:

RCS Temperature	= 547°F
RCS Pressure	= 2250 psig
Pressurizer Water Volume	= 500 ft^3
Pressurizer Steam Volume	= 1300 ft^3

The depressurization is performed using auxiliary spray with makeup from the Refueling Water Storage Tank.

Single Failure Evaluation

Circulation of the Reactor Coolant

1. From hot standby to 350°F (refer to Figures 5.1-6C and 10.4-7 and Plant Drawings 205201, 205301, 205203 and 205303) - Four reactor coolant loops and steam generators are provided, any one of which can provide sufficient natural circulation flow to provide adequate core cooling. Even with the most limiting single failure (of a steam generator

power-operated relief valve), three of the reactor coolant loops and steam generators remain available.

2. From 350°F to cold shutdown (refer to Plant Drawings 205232 and 205332) - Two RHR pumps are provided, either one of which can provide adequate circulation of the reactor coolant.

Removal of Residual Heat

1. From hot standby to 350°F (refer to Plant Drawings 205242, 205342 and 205312 and Figure 9.2-2A).

   a. Steam generator power-operated relief valves - Four are provided (one per steam generator), any one of which is sufficient for RHR. In the event of a single failure, three power-operated relief valves remain available.

   b. Auxiliary feedwater pumps - Two motor-driven and one steam-driven auxiliary feedwater pumps are provided. In the event of a single failure, two pumps remain available, either of which can provide sufficient feedwater flow.

   c. Flow control valves - Air-operated, fail-open valves. In the event of a single failure of one flow control valve (which affects flow to one steam generator from either a motor-driven pump or the steam-driven pump) auxiliary feed flow can still be provided to all four steam generators from the other pumps.

   d. Backup source - A backup source of auxiliary feedwater can be provided via a spectacle flange from either train of the Seismic Category I Service Water System.

2. From 350°F to 200°F (refer to Plant Drawings 205232, 205332 and 205312 and Figure 9.2-2A).

   a. Suction isolation valves 1RH1 and 1RH2 - These valves are each powered from different emergency power trains. Failure of either power train or of either valve operator could prevent initiation of RHR cooling in the normal manner from the Control Room. In the event of such a failure, operator action could be taken to open the affected valve manually. The mechanical failure of the disc separating from the stem has been investigated (1) and its probability has been found to be in the range of 10^{-4} to 10^{-3} per year. The probability of an earthquake larger than the operating basis earthquake is less than 8×10^{-5} per year. The combined probability of valve stem failure coincident with the earthquake ($<8 \times 10^{-8}$) is so low that it need not be considered in the single failure analysis. In the event of a failure, the station would remain in a safe hot standby condition with heat removal via the steam generators.

   b. Isolation valves 11RH4 and 12RH4 - If either of these normally open motor-operated valves, which are powered from different emergency power trains, were to close spuriously, RHR cooling would be provided by the unaffected RHR pump and heat exchanger. The affected valve could be de-energized and opened with its handwheel.

   c. Pumps 11 and 12 - Each pump is powered from a different emergency power train. In the event of a single failure, either pump provides sufficient RHR flow.

d. Heat exchangers 11 and 12 - If either heat exchanger is unavailable for any reason, the remaining heat exchanger provides sufficient heat removal capability.

e. Unit 1

Flow control valves 11RH18 and 12RH18 - If either of these normally open-fail open valves should close spuriously, sufficient RHR cooling would be provided by the unaffected RHR train.

Unit 2

Flow control valves 21RH18 and 22RH18 - If either of these normally open - fail as-is valves should close spuriously, sufficient RHR cooling would be provided by the unaffected RHR train.

f. RHR/Safety Injection System Cold Leg Isolation Valves 11SJ49 and 12SJ49 are normally open valves with power locked out. A single failure in the control circuitry can not/will not inadvertently close either of these valves. However, an assumed closure of these valves will put the unit outside of its design basis since three cold legs cannot be fed with flow through a single SJ49 valve during the injection phase. Although such operation is outside of unit's design basis, this type of inadvertent operation may not be a safety significant issue provided an associated PCT penalty of 29°F can be accommodated without exceeding the 2200°F limit. (Westinghouse letter NS-OPLS-OPL-II-89-929), dated 12/20/89). Such a situation, if any, will still require a specific 10CFR50.59 evaluation.

g. Component Cooling Water System - Two redundant subsystems are provided for safety-related loads. Either subsystem can provide sufficient heat removal via one of the RHR heat exchangers.

h. Service Water System - Two redundant subsystems are provided for safety-related loads. Either subsystem can provide sufficient heat removal via one of the component cooling water heat exchangers.

Boration and Makeup (refer to Figure 5.1-6C and Plant Drawings 205201, 205301, 205234, 205334, 205228 and 205328).

1. Boric acid tanks 11 and 12 - Two boric acid tanks are provided. Each tank contains sufficient 3.75 to 4.0 percent boric acid to borate the RCS for cold shutdown.

2. Boric acid transfer pumps 11 and 12 - Each pump is powered from a different emergency power train. In the

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event of a single failure, either pump will provide sufficient boric acid flow.

3. Isolation valve 1CV175 - If valve 1CV175, which is supplied from emergency power and is normally closed, cannot be opened due to power train or operator failure, it can be opened locally with its handwheel. If valve 1CV175 cannot be opened with its handwheel, an alternate flow path is available via air operated, fail open valve 1CV172 and normally closed manual valve 1CV174.

   Although this path may be used as an alternate path, it does not have the full capacity of the boration path through 1CV175 and provides protection in depth only. The credited backup path for boration is from the RWST via 1SJ1 and 1SJ2 to the charging pump suctions.

4. Isolation valves 1SJ1 and 1SJ2 - Each valve is powered from a different emergency power train; only one of these normally closed motor-operated valves needs to be opened to provide a makeup flow path from the RWST to the centrifugal charging pumps.

5. Centrifugal charging pumps 11 and 12 - Each pump is powered from a different emergency power train. In the event of a single failure, either pump provides sufficient boration or makeup flow.

6. Flow control valve 1CV55 - This normally open valve fails open on loss of air or power. If 1CV55 closed spuriously, the charging pumps would operate on their miniflow circuits until operator action could open bypass valves 1CV81 and 1CV82.

7. Flow control valve 1CV71 - This normally open valve fails closed on loss of air or power. Use of a portable nitrogen bottle would allow 1CV71 to be reopened. If 1CV71 was stuck closed as a result of a single failure, manual bypass valve 1CV73 could be opened locally.

8. Isolation valves 1CV68 and 1CV69 - If either of these normally open, motor-operated valves, each of which is powered from a different emergency power train, should

close spuriously, operator action could be used to deenergize the valve operator and reopen the valve with its handwheel.

9. Isolation valve 1CV77 - If the normally open valve should close spuriously, alternate charging valve 1CV79, which fails open, could be used.

10. The alternate, high head safety injection, BIT cold leg flow path to the RCS also provides for the postulated failures in the previous items 7 through 9, which all involve failure of the normal charging path to the RCS.

Depressurization

1. Auxiliary spray valve 1CV75 - This normally closed valve fails closed on loss of air or power. Use of a portable nitrogen bottle would allow 1CV75 to be opened. If 1CV75 was stuck closed as a result of a single failure, the redundant Seismic Category I Overpressure Protection System valves can be used to depressurize the RCS by venting the pressurizer to the pressurizer relief tank.

2. Charging valves 1CV77 and 1CV79 - These valves fail open on loss of air or power. Use of portable nitrogen bottles would allow 1CV77 and 1CV79 to be closed. If either was stuck open, the redundant Seismic Category I Overpressure Protection System valves can be used to depressurize the RCS by venting the pressurizer to the pressurizer relief tank.

Environmental Qualification of the RHR Suction Isolation Valves

The RHR suction isolation valves are qualified for the steam line break environment. Therefore, they are qualified for the less severe environment which would result for venting the pressurizer to depressurize the RCS.

5.5.7.3.5 Hydraulic Performance at Run-Out

An RHR pump was tested for the highest runout flow for the worst hydraulic configuration. This configuration is when one RHR pump

is feeding two charging pumps, two safety injection pumps and also discharging directly into two cold legs. The test indicated that the RHR pump flow exceeded the design runout flow.

The system resistance on the discharge side for the RHR pumps was, therefore, increased by changing the orifices on the flow elements (up and downstream of the RHR heat exchanger) on the 8-inch RHR headers. The resized orifices at both the flow elements together provided the required resistance in the RHR System.

Net positive suction head was evaluated for a pump flow of 4800 gpm (greater than the maximum pump flow). Under this condition, available NPSH exceeds the required NPSH.

5.5.7.4 Tests and Inspections

The RHR pump flow instrumentation is calibrated periodically. Periodic visual inspections and preventive maintenance are conducted during plant operation.

5.5.8 Reactor Coolant Cleanup System

The CVCS provides reactor coolant cleanup and is discussed in Section 9. The radwaste considerations are discussed in Section 11.

5.5.9 Main Steam Line and Feedwater Piping

The main steam line and the feedwater piping are discussed in Section 10.

5.5.10 Pressurizer

5.5.10.1 Design Bases

The general configuration of the pressurizer is shown on Figure 5.1-2. Design data are given in Table 5.2-4. Codes and material requirements are provided in Table 5.2-9.

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 the design discharge flow from the safety valves. Overpressure of the RCS does not exceed 110 percent of the design pressure.

The surge line is designed to withstand the thermal stresses that result from volume surges occurring during operation.

5.5.10.1.2 Pressurizer Volume

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

1. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

2. The water volume is sufficient to prevent the heaters from being uncovered during a step load increase of 10 percent of full power.

3. The steam volume is large enough to accommodate the surge resulting from 50-percent reduction of full load with automatic reactor control and steam dump without the water level reaching the high level reactor trip point.

4. The steam volume is large enough to prevent water relief through the safety valves following a loss-of-load with the high water level initiating a reactor trip.

5. The pressurizer does not empty following reactor trip and turbine trip.

6. The safety injection 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 coolant loop hot leg. The line enables continuous volume pressure adjustments between the RCS and the pressurizer.

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

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

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 electric heaters are installed in the bottom head. The heaters can be removed 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 also assists mixing.

The spray line nozzle 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 can also be operated manually by a switch in the control room.

A small, continuous spray flow is provided through a manual bypass valve around the power-operated spray valves to assure that the pressurizer liquid is homogenous with the coolant and to prevent excessive cooling of the spray piping.

During an outsurge from the pressurizer, the flashing of water to steam and the generation 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 for normal design transients. Heaters are energized on high water level during insurge to heat the sub-cooled surge water that enters the pressurizer from the reactor coolant loop.

Power-operated relief valves (PORVs) provide the means for pressurizer venting and a procedure for such an application is included within the Station Emergency Instructions for "natural circulation." Pressurizer vent paths have been evaluated and shown not to result in inadvertent opening or failure to close after initial opening.

The PORVs are set to open before the pressurizer safety valves. Relief through the PORVs can limit the pressurizer pressure to levels below the pressurizer safety valve setpressure, and thereby avoid opening (or challenging) the pressurizer safety valves.

Material specifications for the pressurizer, the pressurizer relief tank, and the surge line are provided in Table 5.2-27.

In the list below, several other aspects of the pressurizer are discussed.

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 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 assure free convection of ambient air past the heater and connector ends for cooling.

Pressurizer Instrumentation

Refer to Section 7 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

Spray Line Temperatures

Temperatures in the spray lines from the cold legs of two loops are measured and indicated. Insufficient flow in the spray lines results in low spray line temperature. Low alarms from these temperature channels are actuated to warn the operator of low bypass spray flow rate.

Safety and Relief Valve Discharge Temperature

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 or relief through the associated valve. High temperature alarms are actuated if the leakage is abnormal.

5.5.10.3 Design Evaluation

5.5.10.3.1 System Pressure Control

Whenever a steam bubble is present within the pressurizer, RCS pressure is controlled by the pressurizer. Analyses indicate that proper control of pressure is maintained for the normal operating conditions. Twenty banks of "backup" heaters can be powered from the Vital Distribution System. This provides assurance that pressure control for natural circulation can be maintained during a loss of offsite power.

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. Thereby, continued integrity of the RCS components is assured. Evaluation of plant conditions of operation indicates that this safety limit is not reached.

During startup and shutdown, the rate of temperature change is controlled by the operator. Heatup rate is controlled by pump energy and by the pressurizer electrical heater capacity.

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 controlled by operation of a charging pump. The appropriate letdown flow is provided via the shutdown path from the RHR System.

5.5.10.3.2 Pressurizer Level Control

The normal operating water volume at full-load conditions is approximately 60 percent of the free internal vessel volume. Under part-load conditions, the water volume in the vessel is reduced for proportional reductions in plant load to approximately 25 percent of free vessel at zero power level.

5.5.10.3.3 Pressure Setpoints

The RCS design and operating pressures are listed in Table 5.2-1 together with the safety, power-operated relief and pressurizer spray valves' setpoints. 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 throttle valve, which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping to the pressurizer forms a water seal, which prevents the steam buildup back to the control valves. The design 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 reactor coolant loop 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 spray line also assists in equalizing

the boron concentration between the reactor coolant loops and the pressurizer.

A flow path from the CVCS to the pressurizer spray line is also provided. This additional facility provides an auxiliary spray flow path to the vapor space of the pressurizer during cooldown if the RCPs are not operating. The thermal sleeve on the pressurizer spray connection and the spray piping is designed to withstand the thermal stresses resulting from the introduction of cold spray water.

5.5.10.4 Tests and Inspections

The pressurizer is designed and fabricated in accordance with the ASME Code, Section III, Safety Class 1 vessels.

The pressurizer quality assurance program is given in Table 5.2-26.

5.5.11 Pressurizer Relief Tank

5.5.11.1 Design Bases

Design data for the pressurizer relief tank (PRT) are given in Table 5.2-4. Codes and materials are given in Tables 5.2-9 and 5.2-27.

The tank design is based on the requirement to condense and cool a discharge of pressurizer steam equal to 110 percent of the volume above the full-power pressurizer water level setpoint. The tank is not designed to accept a continuous discharge from the pressurizer.

5.5.11.2 Design Description

The PRT condenses and cools the discharge from the pressurizer safety and relief valves. Discharges from specific relief valves

located inside the containment are also piped to the relief tank. The tank normally contains water and a predominantly nitrogen atmosphere; however, provision is made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen or oxygen.

By means of its connection to the Waste Processing System, the PRT provides a means for removing any noncondensable gases from the RCS that might collect in the pressurizer vessel.

Steam is discharged through a sparger pipe under the water level. This arrangement provides for condensing and cooling 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 to the sparger pipe.

The PRT has pressure, temperature, and level indications and alarms in the control room.

5.5.11.3 Design Evaluation

The volume of water in the tank is capable of absorbing heat from the assumed discharge, assuming an initial temperature of 120°F and increasing to a final temperature of 200°F. If the temperature in the tank rises above 120°F during plant operation, the tank is cooled by spraying in cool water and draining out the warm mixture to the Waste Disposal System.

The spray rate is designed to cool the tank from 200°F to 120°F in approximately 1 hour following the design discharge of pressurizer steam. The volume of nitrogen gas in the tank is selected to limit the maximum pressure following a design discharge to 50 psig.

The rupture discs on the PRT have a relief capacity equal to or greater than the combined capacity of the pressurizer safety valves. The tank design pressure and the maximum rupture disc

burst point are 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 PRT rupture disc is the vent path for both the reactor vessel head and the pressurizer vent. The annulus area containing the PRT is well ventilated. With three out of five fan coil units running at reduced speed during an accident condition, the annulus area containing the PRT is adequately ventilated with an air change every hour. A review of possible sources of ignition in the immediate vicinity of interest indicates no concern. Venting through the PRT rupture disc will not adversely affect any system or component essential for safe shutdown.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent back-pressure at the safety valves from exceeding 20 percent of the setpoint pressure at full flow.

5.5.12 Valves

Valves in contact with the reactor coolant are primarily constructed of stainless steel. For certain applications, such as hard surfacing and packing, design and functional considerations dictate the use of materials other than stainless steel.

All manual and motor-operated valves of the RCS that are 3 inches and larger, (except as listed below), are provided with double-packed packing boxes and intermediate lantern ring leakoff connections. All throttling control valves, regardless of size, are provided with double-packed stuffing boxes and with stem leakoff connections. Leakage to the atmosphere is essentially zero for these valves. RCS valve codes, materials, and quality assurance measures are summarized in Tables 5.2-9, 5.2-27 and 5.2-26, respectively.

The valves listed below have a single set packing configuration and inactive leakoff lines.

	Salem #1			Salem #2	
1PS1	1PS25	1PR6	2PS1	2PS25	2PR6
1PS3	1PS28	1PR7	2PS3	2PS28	2PR7
1PS24	1PS29	1PS59	2PS24	2PS29	2PS59

5.5.13 Safety and Relief Valves

5.5.13.1 *Design Bases*

The capacity of the pressurizer safety valves accommodates the maximum surge resulting from complete loss of load. By the opening of the steam generator safety valves when steam pressure reaches the steam side safety setting, this objective is met without reactor trip or any operator action.

The RCS uses pressure control equipment in addition to the ASME Code safety valves. Although this pressure control equipment is not required by the ASME Code, it is used to assist in maintaining the RCS within the normal operating pressure.

The pressurizer PORVs are designed to limit pressurizer pressure to a value below the high pressure reactor trip setpoint. They are designed to fail to the closed position on loss of air supply. The PORVs are equipped with air accumulators, and will remain operable for some time following loss of the Control Air System, as long as there is sufficient air pressure in the accumulators.

The pressurizer PORVs are not required to open in order to prevent the overpressurization of the RCS. Failure of the PORVs to open results in higher reactor coolant pressures, but does not result in overpressurization of the system. In fact, the opening of the PORVs is a conservative assumption for the departure-from- nucleate-boiling limited transients by tending to keep the primary system pressure down.

The pressurizer spray control valves are also utilized to control pressurizer pressure variations. During an insurge, the Spray System, which is fed from the cold legs, condenses steam in the pressurizer to prevent the pressure from reaching the setpoint of the PORVs.

5.5.13.2 Design Description

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

The 6-inch pipes connecting the pressurizer nozzles to their respective code safety valves are shaped in the form of a loop seal. Condensate, as a result of normal heat losses to the ambient will drain back to the pressurizer liquid space through the normally open safety valve drain lines. If the pressurizer pressure exceeds the set pressure of the safety valves, they will start lifting, and the water from the seal will discharge during the accumulation period. A temperature indicator in the safety valve discharge manifold alerts the operator to the passage of steam due to leakage or valves lifting.

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 stop 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 percent step load decrease with steam dump but without reactor trip.

Design parameters for the pressurizer spray control, safety, and power relief valves are given in Table 5.2-8.

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. Safety valve position is monitored by limit switches which alarm in the Control Room when any valve is not in the fully closed position.

The pressurizer PORVs prevent actuation of the reactor high pressure trip for all design transients up to and including the design step load decreases with steam dump. The relief valves also limit opening of the spring-loaded safety valves. The opening of any pressurizer PORV is annunciated in the control room.

The Salem PORVs, PORV block valves and associated downstream piping have been evaluated for operation under water-solid conditions, and have been found to be adequate. The PORVs can be relied upon to prevent challenges to the pressurizer safety valves when the pressurizer is water-solid. Administrative controls (procedures) are placed upon the PORV block valves to prevent their closure when the pressurizer is water-solid.

Westinghouse has completed a generic study (2) of PORV reliability and concluded that PORVs are adequately reliable so as not to require automatic block valve closure. Public Service Electric & Gas (PSE&G) has determined that the information provided in the generic report is applicable to the Salem Generating Station. Accordingly, automatic isolation of the PORVs is not provided.

5.5.14 Reactor Coolant System Component Supports

5.5.14.1 Description

Reactor vessel supports are assemblages of plates built up to seat the reactor vessel nozzle shoes. There are four shoe supports for each reactor vessel. The support assemblages are air cooled by negative pressure ducts that draw the air away from the space surrounding the vessel through vent holes drilled in the multiple plates. For support details, see Plant Drawing 201194.

The steam generator supports are shown on Plant Drawing 208903. The weight of the steam generator is transferred through four steel columns at its base to the supporting frame. The steam generator penetrates the operating floor of the Containment Building.

The elevation of the operating floor is approximately at the center of gravity of the steam generator. In the original design, the steam generator was supported at the floor by two sets of snubbers and bumper blocks which resist the horizontal forces and overturning moments generated from pipe rupture or earthquake motion. The snubbers have subsequently been deactivated. The two snubbers on the reactor side of each steam generator have been removed. Each of the two backside snubbers has been converted to a rigid, single-acting compression strut via the addition of a compression collar clamped to the snubber body. These compression struts and the bumper blocks resist the lateral forces and moments from pipe rupture or earthquake motion. The supporting frame has its upper bay braced in both directions. The lower bay consists of two parallel planar trusses that are pin-hinged at the top and bottom to allow for thermal displacement. The horizontal forces at the base of the steam generator are transferred through combined truss and frame action to the lower bay of the support structure. The primary loop piping provides lateral support for the frame in the direction normal to the plane of the trusses. Lateral restraint for blowdown is provided at the top of the support structure by two struts connected to the reactor shield wall. The struts are in two bolted sections with gaps for free thermal travel and adjustment. These struts are active for Unit 2 only. On Unit 1, the struts remain in place but are deactivated with the addition of larger gaps.

The RCP supports as shown on Figure 5.5-5 also consist of an upper and lower section. The upper section is a welded steel assembly and is constructed to accommodate the bolts of the feet of the RCP. The lower section is composed of two parallel planar trusses, pin-hinged at the top and bottom to provide for thermal expansion. Lateral support in the direction normal to the plane of the trusses is provided by the primary loop piping. Blowdown restraint is provided at the top of the supporting structure by struts connected to the shield wall. The struts are in bolted sections with gaps for free thermal travel and adjustment.

The steam generator and RCP supports are anchored to the containment base slab by heavy welded steel frames embedded in the concrete and tied to the base mat by 6 and 4 inch diameter bolts, 18.5 feet long. Typical details for the embedded steel for equipment supports are shown on Plant Drawing 208902.

All the statements above are applicable to both the Model 61/19T and the Model-F steam generators, except that the lower support design was modified to accommodate the Model-F steam generator. The Model 61/19T generator was supported from lugs on the channel head while the Model-F generator is supported from the tube sheet elevation so the existing columns had to be replaced by longer ones.

The lower support design for the Unit 1 Model-F generators is shown in Plant Drawing 208903.

The pressurizer also penetrates the operating floor of the reactor containment. Stop lugs are embedded in the floor slab to provide the lateral support for the pressurizer at its mid-height. The vessel skirt is bolted to a steel plate which is in turn welded to the top of the support structure. The support structure frame is braced in both directions. It is further constrained against lateral movement at its top by four short wide flange struts, two in each perpendicular direction, connected to the polar crane support wall. For pressurizer support details, see Plant Drawing 208907.

The control rod drive mechanisms (CRDMs) are supported by the reactor vessel closure head (RVCH) and the integrated head assembly (IHA). The RVCH supports the deadweight of the CRDMs while the IHA seismic platform provides lateral support at the top of each CRDM. The CRDM missile shield, which is permanently attached to the IHA, provides the necessary missile protection for containment if a CRDM were to fracture. The missile shield over the reactor vessel consists of a 181-inch diameter, 2-inch thick steel plate which is permanently attached to the IHA. It is secured to prevent it from becoming a missile. The CRDM ventilation system and ductwork, radiation shielding, and lifting tripod are also permanently attached to the IHA as well and therefore minimal disassembly and reassembly of the IHA is required during refueling outages.

5.5.14.2 Fabrication

For original fabrication, all shop welding was done in accordance with AWS D2.0, "Specification for Welded Highway and Railway Bridges." Detailed joint procedure specifications were submitted by the fabricator for review and approval by PSE&G engineering personnel. The following preheat requirements were specified to minimize residual stress:

1. Material less than 3/4-inch thick shall be preheated to 100°F if the ambient temperature falls below 40°F.

2. Material 3/4 to 1 1/2-inches thick shall be preheated to 150°F prior to welding.

3. Material 1 1/2 to 2 1/2-inches thick shall be preheated to 225°F before welding.

4. Material over 2 1/2-inches thick shall be preheated to 300°F before welding.

Welding of steam generator support modifications for Unit 1 Steam Generator Replacement was done in accordance with AWS D.1.1, Structural Welding Code - Steel, 1996 Edition.

Welding of steam generator support modifications for the Unit 2 Steam Generator Replacement was performed in accordance with AWS D.1.1, Structural Welding Code - Steel, 1994 Edition.

Most intersecting primary members are connected flange to flange by butt welds or are connected to gusset plates by fillet welds. These types of connections are not susceptible to lamellar tearing.

5.5.14.3 Evaluation

Analysis of the RCS supports is discussed in Section 3.9. Steam generator and RCP support load combinations and allowable stress limits are given in Table 5.5-3. The average operating temperature of these supports is approximately 100°F, with a minimum of 70°F. Material for primary component support structures subject to high-intensity impact loads was required to pass a Charpy impact test of 20 foot-pounds at 20°F to verify its fracture toughness characteristics. This fracture toughness assures that brittle behavior will not be exhibited.

5.5.14.4 Inspection

All welds were subject to visual inspection in accordance with American Welding Society requirements. All full penetration shop welds were subject to magnetic particle inspection at four depths supplemented, where practical, by ultrasonic inspection of the finished weld. After original installation, welds on the supports were subject to another magnetic particle inspection. This inspection revealed only minor surface defects on some welds, none critical to the structural integrity of the supports. Nonetheless, these welds were repaired.

In Unit 1 with the Model-F steam generators, the design basis for the lower supports is PSE&G Detail Specification No. 69-7031. The partial penetration welds between the new support columns and the existing support structure were magnetic particle (MT) examined after the root pass and the final pass. The partial penetration welds for the side plates were magnetic particle (MT) examined after the final pass. For the one side plate with full penetration welds, required because of the column offset of 2.203", the welds were MT examined at four thicknesses (Root, 1/3T, 2/3T, and Final) during the process. The stiffener plates were installed using full penetration welds and were MT examined at four thicknesses (Root, 1/3T, 2/3T, and Final) during the process. All new welds were visually examined (VT3) in accordance with Section IWF-3410 of the ASME Code, Section XI, with Addenda through summer 1983.

DCP 80083663, Steam Generator Supports, implemented changes necessary to perform the replacement of the Unit 2 steam generators in 2008. A replacement upper lateral support (ULS) was fabricated and installed on the Replacement Steam Generator (RSG) components. Temporary and permanent modifications were made to the steam generator lower lateral support (LLS) structure.

For the welding fabrication of the replacement ULS components, all full penetration and partial penetration groove welds received either a liquid penetrant (PT) or magnetic particle (MT) examination and a final visual examination (VT) of the completed weld. All other welds require only a final VT of the completed weld. All new welds were visually examined (VT3) in accordance with Section IWF-3410 of the ASME Code, Section XI, with Addenda through summer 1983.

For the welding modifications of the LLS structure, all full penetration welds received am MT examination of the root pass, 1/3T, 2/3T, and the completed weld and a VT examination of the completed weld. All other welds received only a final VT examination of the completed weld. All new welds were visually examined (VT3) in accordance with Section IWF-3410 of the ASME Code, Section XI, with Addenda through summer 1983.

5.5.15 Partial RCS Loop Operation

During partial drain operations of the RCS, adequate RCS inventory, level control, and Net Positive Suction Head (NPSH) must be maintained. If it is required that the RCS water level be lowered to drain the steam generator tubes, the residual heat removal flow rate through each of the RHRS loops should be throttled back to prevent vortexing and possible air entrainment of the pumps.

Draining is to the point where the indicated level is stable and predetermined point (usually at the elevation of the center of the reactor vessel nozzles). At this point, reactor coolant level is monitored continuously to assure that the RHRS inlet lines do not become uncovered. Inventory makeup, if required, is accomplished via the CVCS centrifugal charging pumps.

Should a RHRS inlet line become uncovered, air may be drawn into the suction piping and entrained in the fluid. Factors that minimize the effects of air entrainment on pump performance are as follows:

1. The location of the residual heat removal pumps provides positive head on the pump inlet, and

2. The circulation flow rate is kept low and unnecessary circulation of fluid is avoided (i.e., minimum flow required for core decay heat removal and boron mixing is maintained).

Provisions have been made to minimize the effects of air entrainment. However, should such an event preclude the continued use of the operating train, actions need to be taken to permit the utilization of the alternate train by providing sufficient refill/makeup from the CVCS/charging pumps. Provisions are incorporated to ensure rapid restoration of the RHRS to service in the event that the RHRS pumps become air bound. On identifying this situation, the affected train would be isolated, the reason for the loss of RHR would be identified and corrected, and heat removal accomplished by the redundant train.

Procedures have been developed to address the provision of alternate sources of cooling should loss of RHR cooling occur during shutdown maintenance evolutions. These provisions consider maintenance evolutions during which more than one cooling system may be unavailable, such as loss of steam generators when the RCS has been partially drained for steam generator inspection or maintenance.

The Outage Equipment Hatch may be used to satisfy the requirement for containment closure during modes 5, 6, or undefined. The Outage Equipment Hatch may remain open during mid-loop operations provided that containment closure can be established prior to the onset of core boiling following a loss of RHR. Operating procedures provide administrative controls for operating conditions, which ensure containment closure is achieved prior to core boiling. This satisfies the requirement of NRC Generic Letter 88-17 to establish containment closure prior to core uncovery.

5.5.16 References for Section 5.5

1. Hill, R. A., et al., "Evaluation of Mispositioned ECCS Valves," WCAP-8966 (Proprietary) and WCAP-9207 (Nonproprietary), September 1977.

2. Westinghouse Electric Corp., "Probabilistic Analysis and Operational Data in Response to NUREG-0737, Item II.K.3.2, for Westinghouse NSSS Plants," WCAP-9804, February 1981.

3. ABB Combustion Engineering Report CEN-606, Revision 00, dated 5/21/93, Boric Acid Concentration Reduction Effort, Technical Basis and Operational Analysis for Salem Generating Station Units 1 and 2

TABLE 5.5-1

RESIDUAL HEAT REMOVAL SYSTEM DESIGN PARAMETERS

Code Requirements

Residual Heat Exchangers (Tube Side)	ASME III, Class C
(Shell Side)	ASME VIII
Residual Heat Removal Piping and Valves	ANSI B31.1.0[(1)]
	ANSI B31.7[(2)]

General

Plant design life, years	40
Component cooling water supply temperature design, °F	95
Reactor coolant temperature at startup of decay heat removal °F	350
Time to cool Reactor Coolant System from 350°F to 140°F, starting at 4 hours after shutdown, hr	16[(3)]

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(1) Used for design.

(2) For piping not supplied by the NSSS supplier, material inspection fabrication and quality control conform to ANSI B31.7. Where not possible to comply with ANSI B31.7, the requirements of ASME III-1971, which incorporated ANSI B31.7, were adhered to.

(3) 16 hours was the original design value. With the 1.4% power uprate, reduction in temperature can be accomplished in 18 hours. To cool down in 16 hours is not a design requirement. The design requirement is to cool down in 72 hours with a single train.

Refueling water storage temperature, °F		Ambient
Decay heat generation at 20 hours after shutdown, Btu/hr		72.1×10^6 *
H_3BO_3 concentration in refueling water storage tank, ppm boron		~2000

COMPONENTS

Residual Heat Exchangers

	Shell	Tube
Number		2 (per unit)
Design heat transfer, Btu/hr		34.15×10^6
Design pressure, psig	150	600
Design temperature, °F	200	400
Design flow rate, lb/hr	2.475×10^6	1.48×10^6
Design outlet temperature, °F	108.8	114
Design inlet temperature, °F	95	137
Fluid	Component cooling water	Reactor coolant (borated demineralized water)

* Original decay heat value used in the initial design

Material of construction	Carbon steel	Austenitic stainless steel

Residual Heat Removal Pumps

Number	2 (per unit)
Type	Vertical centrifugal
Design pressure, psig	600
Design temperature, °F	400
Shutoff head, psi	170
Design flow rate, gpm	3,000
Design head, ft	350
Available NPSH at design flow rate, ft	25
Temperature of pump fluid, °F	40 - 350
Normal fluid	Reactor coolant
Fluid during LOCA recirculation phase	Radioactive borated water with H_2 and NaOH in solution
Material of construction	Austenitic stainless steel

TABLE 5.5-1 (Cont.)

Piping and Valves

	Pump Suction	Pump Discharge
Residual heat removal loop (piping and valves in isolated loop):		
Design pressure, psig	450*	600
Design temperature, °F	400	400
Residual loop isolation valves and piping:		
Design pressure, psig		2,485
Design temperature, °F		650

* Unit 2 piping downstream of 2RH75 & 76 are designed to 600 psig.

TABLE 5.5-2

RESIDUAL HEAT REMOVAL SYSTEM FAILURE ANALYSIS

Component	Malfunction	Comments and Consequences
1. Residual heat removal pumps	Rupture of a pump casing	The casing and shell are designed for 600 psi and 400°F. The pump is protected from overpressurization by two normally closed valves in the pump suction line and by an open relief line, containing a relief valve, back to the containment sump. The pump is inspectable and is located in the Auxiliary Building protected against credible missiles. Rupture is considered unlikely but in any event the pump can be isolated.
2. Residual heat removal pump	Pump fails to start	One operating pump furnishes half of the flow required to meet design cooldown rate. This increases the time necessary for plant cooldown.
3. Residual heat removal pump	Motor operated valve on pump suction is closed	This is prevented by prestartup and startup and operational checks.
4. Residual heat removal pump	Stop valve on discharge line closed or check valve sticks closed	Stop valves are locked open. Prestartup and operational checks confirm position of valves.
5. Remote operated valves inside containment in pump suction line	Valve fails to open	In the improbable event that one of the remote operated valves on the suction line to the residual heat removal pumps is inoperable, an attempt will be made to open it manually. If this is impossible, the plant will be cooled to about 280°F with steam dump from the steam generators, while additional recovery actions could be implemented based on plant's abnormal and emergency operating procedures, equipment availability and resources.

Component	Malfunction	Comments and Consequences
6. Remote operated valves inside containment on pump discharge line	Valve fails to open	Pump discharge pressure gauge shows pump shut-off head indicating no flow. The low head safety injection lines may be opened and utilized to direct flow to the RCS hot legs. A reactor coolant pump must be operated.
7. Residual heat exchanger	Tube or shell rupture	Rupture is considered unlikely, but in any event the faulty heat exchanger may be isolated.
8. Residual heat exchanger vent or drain valve	Left open	This is prevented by prestartup operational checks.

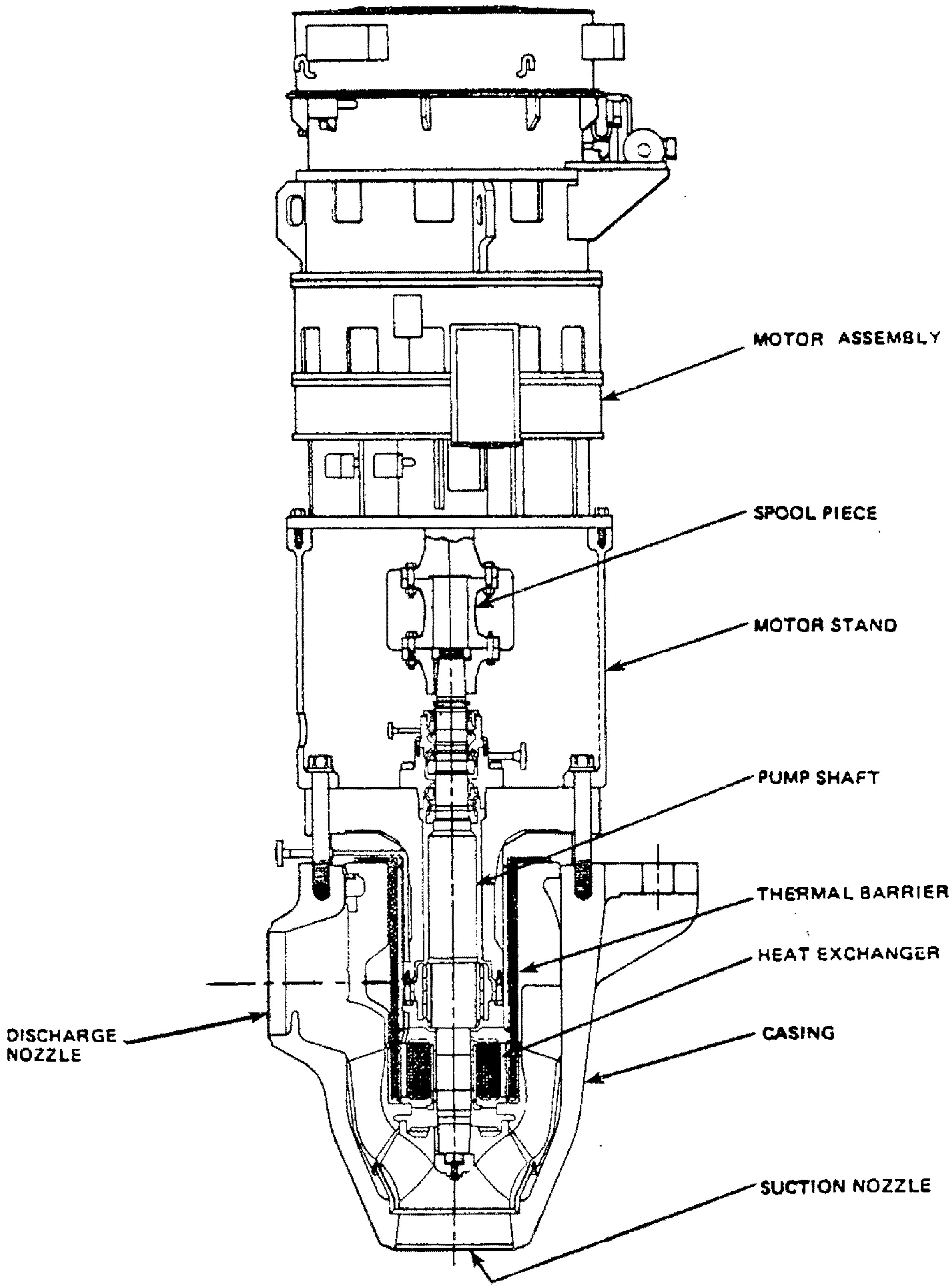
TABLE 5.5-3

SALEM NUCLEAR GENERATING STATIONS
UNIT NOS. 1 AND 2

STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS
LOADING COMBINATION AND ALLOWABLE STRESS LIMITS

LOADING COMBINATIONS	SUPPORTS - ALLOWABLE STRESS LIMIT
1. Normal loads	Working stresses per AISC code
2. Normal loads + operating base earthquake (upset condition)	1-1/3 working stresses AISC code
3. Normal loads + design base earthquake + pipe rupture loads (faulted condition)	Yield stress of material, <u>or</u> AISC Code * <u>or</u> ASME III, Subsection NF and Appendix F

* with increase factors consistent with the guidance of R. G. 1.124



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Reactor Coolant Pump
	Updated FSAR Figure 5.5-1

Figure F5.5-2A Sheets 1 & 2 of 2 intentionally deleted.

Refer to plant drawing 205232 in DCRMS

Figure F5.5-2B Sheets 1 & 2 of 2 intentionally deleted.

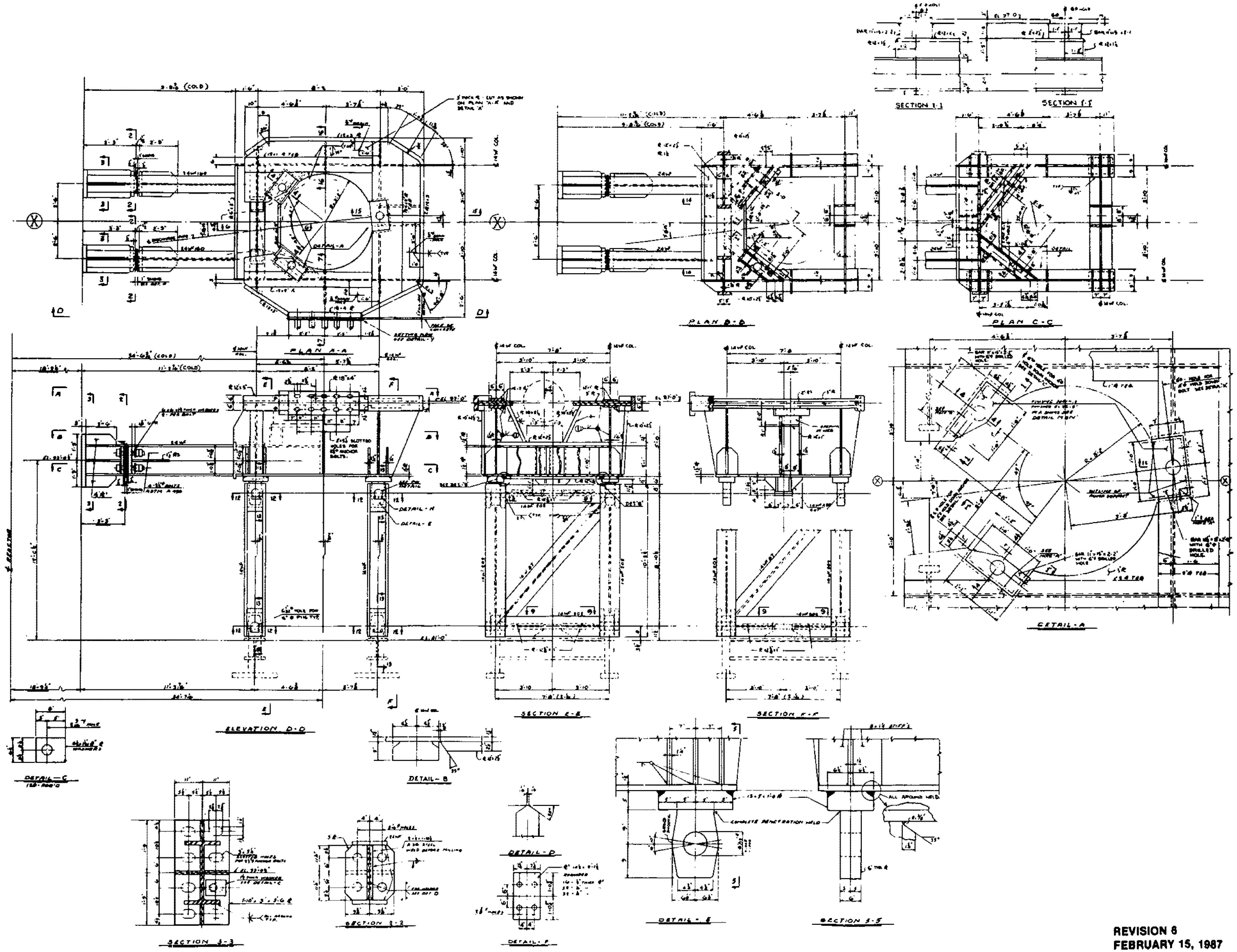
Refer to plant drawing 205332 in DCRMS

Figure F5.5-3 intentionally deleted.

Refer to plant drawing 201194 in DCRMS

Figure F5.5-4 intentionally deleted.

Refer to plant drawing 208903 in DCRMS



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Reactor Coolant Pump Supports
	Updated FSAR — Figure 5.5-5

Figure F5.5-6 intentionally deleted.

Refer to plant drawing 208902 in DCRMS

Figure F5.5-7 intentionally deleted.

Refer to plant drawing 208907 in DCRMS

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 pump motors), as well as for the Residual Heat Removal (RHR) System. The pickoff points for the RCS are shown in the flow diagram (Figure 5.1-6C and Plant Drawings 205201 and 205301); and for the RHR System, on flow diagram Plant Drawings 205232 and 205332.

In general, these input signals are used for the following purposes:

1. Provide input to the Reactor Trip System described in Section 7.

2. Provide input to the Engineered Safety Features Actuation System described in Section 7.

3. Furnish input signals to the nonsafety-related control systems and surveillance circuits.

5.6.1 Loop Temperature

One hot-leg and one cold-leg temperature reading is provided from each coolant loop to use for protection. Narrow-range thermowell resistance temperature detectors (RTDs) are provided for each coolant loop. In the hot legs, sampling scoops are used because the flow is stratified; that is, the fluid temperature is not uniform over a cross section of the hot leg. One dual-element RTD is mounted in each of the three sampling scoops associated with each hot leg. The scoops extend into the flow stream at locations 120 degrees apart in the cross-sectional plane. Each scoop has five orifices which sample the hot-leg flow along the leading edge of the scoop. Outlet ports are provided in the scoops to direct the sampled fluid past the sensing element of the RTDs. One of each RTD's dual elements is used, while the other is an installed spare. Three readings from each hot leg are averaged to provide a hot leg

reading for that loop.

One dual-element RTD is mounted in a thermowell associated with each cold leg. No flow sampling is needed because coolant flow is well mixed by the reactor coolant pumps. One RTD element is used, while the other is an installed spare. The potential for the bulk reactor coolant temperature in the cold leg to be slightly lower or higher than that indicated on the cold leg RTD exists at Salem 1 and 2. This effect, known as cold leg streaming, has been accounted for in the Chapter 15 analyses.

The thermowells are pressure-boundary parts that completely enclose the RTD. They have been shop hydrotested to 1.25 times the RCS design pressure. The external design pressure and temperature are the RCS design temperature and pressure. The RTD is not part of the pressure boundary. The scoop, thermowell, and thermowell/scoop assembly have been analyzed to the ASME Boiler and Pressure Vessel Code, Section III, Class 1. The effects of seismic- and flow-induced loads were considered in the design.

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Separate RTDs, located in the thermowells in the cold- and hot-leg piping of each loop, supply signals to wide-range temperature recorders. This information is used by the operator to control coolant temperature during startup and shutdown.

5.6.2 Pressurizer Temperature

There are two temperature detectors in the pressurizer: one in the steam phase and one in the water phase. Both detectors supply signals to temperature indicators and high-temperature alarms. The steam-phase detector, located near the top of the vessel, alerts the operator if the steam becomes superheated. In addition, it is used during startup to determine water temperature when the pressurizer is completely filled with water. The water-phase detector, located at an elevation near the center of the heaters, is used during cooldown to ensure that the pressurizer temperature is consistent with the RCS.

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. An alarm is actuated on high temperature.

The fluid temperatures in each spray line are measured and indicated. Alarms from these signals are actuated by low spray water temperature. Alarm conditions indicate insufficient flow in the spray lines through the manual throttle valves.

The temperature of the water in the pressurizer relief tank is indicated over a range of 50°F to 350°F, and an alarm, actuated by a high temperature, informs the operator that cooling of the tank contents is required.

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.

5.6.3 Pressure

Four pressurizer pressure transmitters provide signals for individual indicators in the control room, for actuation of a low pressure trip, for high pressure reactor trip, and for alarms. One of the four signals may be selected by the operator for display on a pressure recorder. Three transmitters provide independent low pressure signals for safety injection initiation and for safety injection signals to allow manual block during plant shutdown and automatic unblock during plant startup. In addition, these pressure transmitters provide inputs for pressurizer heater, spray valve, and power-operated relief valve (PORV) control.

Two narrow range differential pressure transmitters connected to the RCS sample line on the No. 1 hot leg and on the No. 3 hot leg are installed to monitor RCS level during Mid-Loop operation. A wide range differential pressure transmitter is connected on the No. 3 hot leg to monitor RCS level during midloop operation, reduced inventory and vacuum fill.

Two wide-range transmitters provide pressure indication over the full operating range. The indicators serve as a guide to the operator during plant startup and shutdown and also provide the open permissive signals and automatic closure signals for the RHR loop isolation valves interlock circuit.

Two local pressure indicators are provided for operator reference during shutdown. They are located in two separate loops and are provided with maximum (drag) pointers to indicate the maximum pressure attained since the last resetting of the pointers.

A pressurizer relief tank (PRT) pressure transmitter provides a signal to close valve PCV-472 on high pressure should it be open when a safety valve lifts discharging steam into the PRT.

5.6.4 Pressurizer Water Level

Three pressurizer liquid level transmitters provide signals for use in the Reactor Control and Protection System, and the Chemical and Volume Control System (CVCS). Each transmitter provides an independent high water level signal that is used to actuate an alarm and, upon two out of the three transmitter signals, will cause a reactor trip. The transmitters may 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, any of the three level transmitters may be selected for display on a level recorder located on the main control board.

Two of the three transmitters may be selected to provide an alarm when the liquid level falls to the fixed low level setpoint. The

same signal will trip the pressurizer heaters "off" and close the letdown line isolation valves. The low pressurizer level signal can be bypassed, in accordance with EOPs, allowing operator control of the Letdown isolation valves when Letdown Bypass Switch is placed in bypass mode. Two transmitters are similarly selected to supply a signal to the liquid level setpoint controller.

A fourth independent pressurizer level transmitter is calibrated for low temperature conditions, provides water level indication during startup, shutdown and refueling operations.

A PRT level transmitter supplies a signal for an indicator and for high and low level alarm.

Two RHR pressure transmitters are installed and connected to the sensing lines of PI631 and PI632 to monitor RHR pumps 1RHE1, 1RHE2, 2RHE1 and 2RHE2 suction pressure during Mid-Loop operation.

5.6.5 Reactor Vessel Water Level

The Reactor Vessel Level Instrumentation System uses three sets of differential pressure (d/p) cells, with two identical cells per set for redundancy, to measure the water level in the vessel. Each of these sets uses cells with different ranges to obtain three different vessel water level measurements.

One set of two d/p cells is installed to sense the fluid pressure differential between the top of the vessel and the loop piping. One side of each cell is connected to a dedicated RVLIS sensing tap in the RVCH and the other sides of the cells are connected to the hot legs of Loops 1 and 4. Each cell's level indicator in the control room shows reactor vessel water level between the hot leg and the top of the vessel. If any reactor coolant pump (RCP) is operating, the associated level indicator will display "INVALID."

Two d/p cells are installed to sense the fluid pressure differential between the bottom and top of the reactor vessel. One side of each cell is connected to the head vent penetration; the other side is connected to an in-core instrumentation conduit at or near the seal table. When no RCP is running, these cells measure the differential pressure between the top and bottom of the reactor vessel to measure the water level above and below the reactor core. The associated level indicator displays "INVALID" if any RCP is operating.

Two d/p cells, with installation similar to that of the two cells used for narrow range measurement, measure reactor core and internal pressure drop for any combination of operating RCPs which, when compared with the normal single phase pressure drop, provides an indication of the relative void content or density of the circulating coolant. These cells may be used on a continuous basis. The RVLIS-86 stores four values of expected reactor coolant void fraction. These expected values correspond to one through four RCPs running. The expected value of void fraction corresponding to the current pump operating status is displayed on the RVLIS-86 remote display panels. When all pumps are off, the indicator displays "INVALID."

All of the d/p cells are located outside of containment to minimize post-accident environmental effects and to facilitate calibration, cell replacement, reference leg checks, and filling and venting. Hydraulic sensors (inside containment) and hydraulic isolators (outside containment), connected by a seal sensing line, are installed between each d/p cell and its connection to the vessel/RCS. These features assure containment isolation in case of a sensing line break and prevent flow of primary coolant to outside containment. To obtain the required accuracy for vessel water level measurement, the d/p cell indications are compensated using measured temperatures of both the d/p cell reference legs and the reactor coolant.

During refueling, the reactor head and associated RVLIS piping are removed. The instrument sensing line normally connected to the RVLIS pressure tap is manually realigned to sense atmospheric pressure. When the refueling mode is selected at the RVLIS-86, the sensors' outputs are re-scaled by the RVLIS-86 software. The upper range transmitters are used to provide the reduced inventory level

indication from 97.3 feet to 106.0 feet. The dynamic range transmitters provide the refueling cavity level indication from 104 feet to 130 feet.

Additional information is presented in Section 7.

5.6.6 Reactor Coolant Flow

Flow in each reactor coolant loop is monitored by three d/p measurements at a piping elbow tap in each reactor coolant loop. These measurements on a two-out-of-three coincidence circuit per loop provide a low flow signal to actuate a reactor trip.

Elbow taps are used in the RCS as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or

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not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap readout has been well established by the following equation:

$$\frac{\Delta P}{\Delta P_o} = \left(\frac{w}{w_o}\right)^2$$

where ΔP_O is the referenced pressure differential with the corresponding referenced flow rate ω_O and ΔP is the pressure differential with the corresponding referenced flow rate ω. The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse pressurized water reactor (PWR) plants. The expected absolute accuracy of the channel is within ±10 percent and field results have shown the repeatability of the trip point to be within ± 1 percent. The analysis of the loss of flow transient presented in Section 14.1 assumed instrumentation error of ±3 percent.

The combined flow from the hot and cold leg RTD manifolds passes through an orifice before discharging back to the RCS at the suction side of the RCP. The flow is indicated locally by a d/p gage and by status lights in the control room. Low flow through either the hot or cold leg warns of possible inaccuracy in the corresponding temperature signals; therefore, an alarm is actuated.

5.6.7 Reactor Coolant Pump Motor Instrumentation

A dual purpose switch is provided on the high pressure oil lift system. Upon low oil pressure the switch actuates an alarm on the main control board. In addition, the switch is part of an interlock system that prevents starting of the pump until the oil

lift system is operating and oil pressure is established. A local pressure gage is also provided.

Level switches are provided in the motor radial bearings and thrust bearing oil reservoirs. The switches actuate high and low level alarms on the main control board.

Thermocouples are located in the upper and lower thrust bearing shoes. These elements provide signals for multi-point recorder on the main control board and actuate an alarm on high temperature.

A RCP trip criterion has been adopted which assures pump trip for all losses of primary coolant for which pump trip is considered necessary, but which also permits pump operation during most non-LOCA events, including steam generator tube rupture events up to the design basis double-ended tube rupture. The controlling parameter selected for pump trip actuation is RCS pressure. The RCS wide-range pressure instrumentation will be monitored.

5.6.8 Loose Parts Monitoring

A Loose Parts Monitoring (LPM) System supplied by Westinghouse Electric Corporation has been installed for each of the two units of Salem Generating Station. (This LPM System has been designated as the Metal Impact Monitoring System by Westinghouse). The LPM System has been designed to enable early detection of the presence of metallic debris, loose parts, or restrained loose parts, inside the Nuclear Steam Supply System (NSSS) during plant startup and commercial operation. Any form of metallic debris, loose parts, or restrained loose parts, when carried or agitated by the reactor coolant flow may attain sufficient velocities to impact and damage the interior of the NSSS pressure boundary.

The LPM system is realized by on-line processing, transmission, and conditioning of the signals from a group of strategically located Piezoelectric accelerometers (a total of 12) mounted externally to the wall of NSSS with proper indication and alarms.

When the insides of the reactor and steam generator walls are struck by metallic debris, loose parts, or restrained loose parts, the structure is shock excited producing local wall acceleration that can be detected in time and frequency domain. These impact signatures can be separated in the frequency domain from the general vessel and background signature. Once the proper frequency band is selected, a threshold Amplitude Detection System and associated rate can be used to activate an Alarm System.