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.

Primary Side Hydrostatic Test Before Initial Startup

The pressure tests covered by this section include both shop and field hydrostatic tests which occur as a result of component or system testing. This 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.

Secondary Side Hydrostatic Test Before Initial Startup

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 Design Criteria for Vessels and Piping

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:

$Pm \leq 1.2$ Sm or tabulated yield (Sy) whichever is greater $Pl + Pb \leq 1.8$ Sm or 1.5 Sy 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 Compliance with Reactor Vessel Material Surveillance Program Requirements

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 1.77×10^{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 charging/boron injection tank line is isolated from the RCS by two check valves and normally closed remote manual valves. Leakage from the RCS would be detected by pressure changes in the line.

5. The Residual Heat Removal System and 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 10CFR50.55(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

1. Wiesemann, R. A.; Tome, R. E.; and Salvatori, R., "Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels Under Earthquake Loading," WCAP-5890, Revision 1, October 1967.

2. Gibbon, et al., "Fatigue Properties of Irradiated Pressure Vessels," ASTM STP 426, pages 408 to 437.

3. Pellini, W. S. and Puzak, P. P., "Fracture Analysis Diagram Procedures for the Fracture Safe Engineering Design of Steel Structures." NRL Report 5920, March 15, 1963, Welding Research Council Bulletin No. 88, 1963.

4. Pellini, W. S. and Puzak, P. P., "Practical Considerations in Applying Laboratory Fracture Test Criteria to the Fracture - Safe Design of Pressure Vessels," NRL Report 6630, November 5, 1963.

5. Pellini, W. S. and Loss, F. J., "Integration of Metallurgical and Fracture Mechanics Concepts of Transition Temperature Factors Relating to Fracture-Safe Design for Structural Steels," Welding Research Council Bulletin No. 141, 1969.

6. Magar, T. R. and Hazelton, W. S., "Evaluation of Linear Elastic Fracture Mechanics of Radiation Damage to Pressure Vessel Steels," Vienna I AEA-Meeting, June 2-9, 1969.

7. Porse, L., "Reactor Vessel Design Considering Radiation Effects," ASME 6 WA 100.

8. Deleted.

9. Deleted.

10. Deleted.

11. O'Donnell, W. J., "A Study of Perforated Plates with Square Penetration Patterns," Welding Research Council Bulletin No. 124, September 1967.

12. Mahoney, J. B. and Salerno, V. L., "Stress Analysis of a Circular Plate Containing a Rectangular Array of Holes," Welding Research Council Bulletin No. 106, July 1965.

13. Lemcoe, M. M., "Feasibility Studies of Stresses in Ligaments," Welding Research Council Bulletin No. 65, November 1960.

14. O'Donnell, W. J. and Purdy, C. M., "The Fatigue Strength of Members Containing Cracks," ASME Transactions, Journal of Engineering for Industry, Vol. 86-B, pp. 205-213, 1964.

15. Cooper, K., et al., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972.

16. Hazelton, W. S., "Topical Report - Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary) March 1970 and WCAP-7735 (Non-Proprietary), August 1971.

17. Smithells, C. J., "Metals Reference Book," Vol. II, p. 621, Plenum Press, 1967.

18. Scharfstein, L. R., "Effects of Residual Elements on the General Corrosion Resistance of Austenitic Stainless Steels," Effects of Residual Elements on Properties of Austenitic Stainless Steels, ASTM, STP 418, 1967.

19. Gunia, R. B. and Woodrow, G. R., "Nitrogen Improves Engineering Properties of Chromium-Nickel Stainless Steels," Journal of Metals 5, No. 2, p. 413, June 1970.

20. Jones and Laughlin Data Sheet - Type 304-N Stainless Steel, Jones and Laughlin Steel Corporation, Stainless and Strip Division, Warren, Michigan.

21. Robinson, Earnest L., "Bursting Tests of Steam-Turbine Disk Wheels," Transactions of the ASME, July 1944.

22. Winne, D. H. and Wundt, B. M., "Application of the Griffith-Irwin Theory of Crack Propagation to the Bursting Behavior of Disks, Including Analytical and Experimental Studies," ASME, December 1, 1957.

23. Terek, E. and Haas, C. M., "Evaluation of Pressurized Thermal Shock for Salem Unit 1," WCAP-15567, Revision 0, November 2000.

24. Terek, E. and Haas, C. M., "Evaluation of Pressurized Thermal Shock for Salem Unit 2," WCAP-15568, Revision 0, October 2000.

25. Terek, E., "Salem Unit 1 Heatup and Cooldown Curves for Normal Operation," WCAP-15565, Revision 1, February 2001.

26. Terek, E., "Salem Unit 2 Heatup and Cooldown Curves for Normal Operation," WCAP-15566, Revision 1, February 2001.

27. Letter from Mr. James C. Stone, NRC, to Mr. Steven E. Miltenberger, PSE&G, dated May 25, 1994, "Leak-Before-Break Evaluation of Primary Loop Piping, Salem Nuclear Generating Station, Units 1 and 2".

28. Bhomick, D.C. and Prager, D.E., "Technical Justification for Design Basis for the Salem Generating Station Units 1 and 2, "WCAP-13659 and WCAP-13660, Revision 0, May 1993.

29. PSEG VTD #328392, AREVA NP Document 77-5050070-00, "Salem Unit 2 Overpressure Protection Report Addendum," February 2005.

30. Lohmeier A., Small N. C., Bonin H. J., Gonnet B., "Collapse Of Ductile Heat Exchanger Tubes With Ovality Under External Pressure," 2nd SMIRT, Vol. II, Part E-F, Berlin, September 1973.

31. PSEG VTD 900013 Sheet 1, AREVA NP SAS Document BUCRSA/NGV 2001 Rev. B, Salem Unit 2 RSG - Design Report.

32. Specification S-2-RC-MDS-0397, Revision 4, Replacement Steam Generators for Salem Generating Station Unit 2 Detailed Design Specification.