CHAPTER 4 TABLE OF CONTENTS

4.0 REACTOR COOLANT SYSTEM

The Reactor Coolant System shown in the Flow Diagram, Figure 4.2-1 and Figure 4.2-1A, consists of two essentially identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control.

FOREIGN OBLIGATIONS

The reactor vessel closure heads and control rod drive mechanisms for Point Beach Nuclear Plant Units 1 and 2 were manufactured in Japan. Consequently, as stated in letters from the NRC dated December 16 and 17, 2004, use of this equipment (“foreign obligated equipment”) obligates Point Beach Nuclear Plant to comply with certain peaceful use commitments and material tracking obligations specified in the U.S.-Japan Agreement for Peaceful Nuclear Cooperation. This equipment will not be used for any purpose that would result in any nuclear explosive device (e.g., producing tritium for the weapons program). Additionally, export of this equipment will require similar peaceful use assurances from the proposed recipient country. Finally, all nuclear material used in or produced through the use of the reactors with this equipment will also become obligated to Japan so long as that equipment is in use. All nuclear material transaction and status reports must be adjusted accordingly.

4.1 DESIGN BASIS

PERFORMANCE OBJECTIVES

The Reactor Coolant System transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Borated light water, meeting exacting chemical standards, is circulated at the flow rate and temperature consistent with achieving the reactor core thermal hydraulic performance presented in Section 3.2. The water also acts as a neutron moderator and reflector and as a solvent and transport medium for the neutron absorber, boron, used in chemical shim control.

The Reactor Coolant System provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values any release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor operation. During transient operation the system's heat capacity attenuates coolant volume changes within the protection system criteria.

By appropriate selection of the inertia of the reactor coolant pumps, the thermal hydraulic effects which result from a loss-of-flow situation are reduced to a safe level during the pump coastdown. The layout of the system assures the natural circulation capability following a loss-of-flow to permit plant cooldown without overheating the core. The system provides connections for the Safety Injection System to assure cooling water delivery to the core during a loss-of-coolant accident.

GENERAL DESIGN CRITERIA

General design criteria which apply to the Reactor Coolant System are given below.

Quality Standards

CRITERION: Those systems and components of reactor facilities which are essential to the prevention or the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1)

The Reactor Coolant System 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. Details of the quality assurance programs, test procedures, and inspection acceptance levels are given in Section 4.4. Particular emphasis is placed on the assurance of quality of the reactor vessel to obtain material whose properties are uniformly within code specifications.

Performance Standards

CRITERION: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind, or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2)

All piping, components, and supporting structures of the Reactor Coolant System are designed as seismic Class I equipment.

Seismic Design Classification details are given in Appendix A.5.

The Reactor Coolant System is located in the containment building whose design, in addition to being a seismic Class I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Section 5.0.

Records Requirements

CRITERION: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5)

Records of the design, of the major Reactor Coolant System components, and the related engineered safety feature components are maintained at Point Beach and will be retained throughout the life of the plant.

Note: The portion of the following paragraph pertaining to fabrication records is historical. Per the Asset Sale Agreement between WE Energy and FPL Energy Point Beach, FPL Energy Point Beach acquired rights to documents owned by third parties. (Reference 9).

Records of fabrication are maintained in the manufacturer's plants as required by the appropriate code or other requirements pending submittal to Westinghouse or Wisconsin Electric Power Company. They are available at any time to Wisconsin Electric Power Company throughout the life of the plant. Construction records are available at the Point Beach Nuclear Plant where they will be retained for the life of the plant.

Missile Protection

CRITERION: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

This plant-specific General Design Criterion is very similar to 10 CFR 50 Appendix A GDC 4. Under the provisions of that criterion, the dynamic effects associated with postulated pipe ruptures of the RCS may be excluded from the design basis when appropriate analyses approved by the NRC demonstrate that the probability of such ruptures is extremely low (Reference 1). Analyses have been completed for PBNP for the Reactor Coolant Loop piping and the Pressurizer Surge Line (Reference 2 and Reference 6). The NRC has approved the analyses (Reference 3, Reference 7, and Reference 8). As such, the original design features of the facility to accommodate the dynamic effects of a Reactor Coolant pipe or Pressurizer Surge line pipe rupture are no longer applicable. In the balance of this chapter, discussions of these features have been retained for historical information, and to provide continuity in the discussion of related features.

The steam generators are supported, guided, and restrained in a manner which prevents rupture of the steam side of a generator, the steam pipelines, and the feedwater piping as a result of forces created by a Reactor Coolant System pipe rupture. These supports, guides, and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater pipeline rupture. The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the engineered safety features is not impaired.

PRINCIPAL DESIGN CRITERIA

The criteria which apply solely to the Reactor Coolant System are given below.

Reactor Coolant Pressure Boundary

CRITERION: The reactor coolant pressure boundary shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. (GDC 9)

The Reactor Coolant System, in conjunction with its control and protective provisions, is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits. Fabrication of the components which constitute the pressure boundary of the Reactor Coolant System is carried out in accordance with the applicable codes at the time of fabrication. In addition, there are areas where specifications for Reactor Coolant System components go beyond the applicable codes. Details are given in Section 4.4.

The materials of construction of the pressure boundary of the Reactor Coolant System are protected from corrosion phenomena which might otherwise significantly reduce the system structural integrity during its service lifetime by the use of noncorrosive materials (such as stainless steel) and by the maintenance of proper chemistry control.

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 to a safe level.

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. The system is also protected from overpressure at low temperatures by the Low Temperature Overpressure Protection System.

Isolable sections of the system are provided with overpressure relieving devices discharging to closed systems such that the system code allowable relief pressure within the protected section is not exceeded.

Monitoring Reactor Coolant Leakage

CRITERION: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16)

Positive indications in the control room of leakage of coolant from the Reactor Coolant System to the containment are provided by equipment which permits continuous monitoring of containment air activity and humidity, as well as collection of runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units, and from the containment floor drains. 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 is an indication of change within the containment, 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, volume of condensate and floor drain runoff, and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

Further details are supplied in Section 6.0.

Reactor Coolant Pressure Boundary Capability

CRITERION: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition. (GDC 33)

The reactor coolant boundary is shown to be capable of accommodating, without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Section 14.2.6. The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are primarily used to control load variations and boron dilution is used primarily to compensate for core depletion, only the rod cluster control assemblies in the controlling groups are inserted in the core at power, and at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to insure that this condition is met.

By defining control rod groupings, radial locations, and allowed axial position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected control rod accident to a value which precludes excessive pressure surges and any resultant damage to the primary system pressure boundary. The failure of a rod mechanism housing causing a rod cluster to be rapidly ejected from the core is evaluated as a theoretical, though not a credible accident. While limited fuel damage could result from the hypothetical event, any released fission products are confined to the Reactor Coolant System and the reactor containment. The environmental consequences of rod ejection are less severe than from the hypothetical loss-of-coolant for which public health and safety is shown to be adequately protected in Section 14.3.5.

Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

CRITERION: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failures. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes. (GDC 34)

The reactor coolant pressure boundary is designed to reduce to an acceptable level the probability of a rapidly propagating type failure. The fracture toughness of the materials in the beltline region of the reactor vessel will decrease as a result of fast neutron irradiation induced embrittlement. Fracture toughness will decrease with increasing the reference nil ductility temperature (RT_{NDT}), which increases as a function of several factors, including accumulated fast neutron fluence. This

change in material properties is factored into the operating procedures such that the reactor coolant system pressure is limited with respect to RCS temperature during plant heatup, cooldown, and normal operation. These limits are determined in accordance with the methods of analysis and the margins of safety of Appendix G of ASME Code Section XI and are included in the Point Beach Pressure Temperature Limits Report (PTLR). The Low Temperature Overpressure Protection System provides protection during low-temperature operations.

All pressure containing components of the Reactor Coolant System are designed, fabricated, inspected, and tested in conformance with the applicable codes at the time of order placement. Further details are given in Table 4.1-9.

Reactor Coolant Pressure Boundary Surveillance

CRITERION: Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided. (GDC 36)

The design of the reactor vessel and its arrangement in the system permits access during the service life to the entire internal surfaces of the vessel and to the following external zones of the vessel: the flange seal surface, the flange OD down to the cavity seal ring, the closure head and the nozzle to reactor coolant piping welds. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete.

Monitoring of the RT_{NDT} properties of the core region base material, weldments, and associated heat affected zones are performed in accordance with a surveillance program meeting the requirements of 10 CFR 50, Appendix H. Samples of reactor vessel plate and forging materials are retained and catalogued and are available for future testing, as needed.

To define permissible operating conditions heatup and cooldown limit curves are established in accordance with the methods of analysis and the margins of safety of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G. In addition, the Low Temperature Overpressure Protection System using the power-operated relief valves is activated whenever the reactor coolant system is not open to the atmosphere and the coolant temperature is less than criteria established by ASME Section XI.

DESIGN CHARACTERISTICS

Design Pressure and Temperature

The Reactor Coolant System design and operating pressure, together with the safety, power operated relief, and pressurizer spray valves set points, and the protection system set point pressures, are listed in Table 4.1-1. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and assumed system relief valve characteristics. The design pressures and data for the respective system components are

listed in Table 4.1-2 through Table 4.1-6. Table 4.1-7 gives the design pressure drop of the system components. The design temperature for each component is 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
Table 4.1-2 through Table 4.1-6.

Seismic Loads

The seismic loading conditions are established by the "Operating Basis Earthquake" (OBE) and "Safe Shutdown Earthquake" (SSE). The former is selected to be typical of the largest probable ground motion based on the site seismic history. The latter is selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties. For the "Operating Basis Earthquake" loading condition, the systems necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

The seismic design for the "Safe Shutdown Earthquake" is intended to provide a margin in design that assures:

1. The integrity of the reactor coolant pressure boundary.

2. The capability to shutdown the reactor and maintain it in a safe shutdown condition, or

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the exposures of 10 CFR 50.67 (Reference 11).

For the combination of normal plus design earthquake loadings, the stresses in the support structures are kept within the limits of the applicable codes. For the combination of normal plus no-loss-of-function earthquake loadings, the stresses in the support structures are limited to values necessary to ensure their integrity and to keep the stresses in the Reactor Coolant System components within the allowable limits as given in Appendix A.5.

Cyclic Loads

All components in the Reactor Coolant System 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 operation. The number of thermal and loading cycles used for design purposes and their bases are given in Table 4.1-8. During unit heatup and cooldown, pressure and the rates of temperature change are limited. The cycles are estimated to be an accurate representation of actual transients or actual operating experience.

The Reactor Coolant System and its components are designed to accommodate 10% of full power step changes in plant load and 5% of full power per minute ramp changes over the range from 15% full power, up to and including but not exceeding 100% of full power, without reactor trip. The Reactor Coolant System will accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump system make it possible to accept a rapid load decrease of 50% of full power at a rate up to 200%/minute without reactor trip, or a turbine trip from below 50% power without a reactor trip.

To provide the necessary high degree of integrity for the equipment in the Reactor Coolant System, the transient conditions selected for equipment fatigue evaluation are based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, and normal and abnormal load transients. To a large extent, the specific transient operating condition considered for equipment fatigue analyses are based upon engineering judgment and experience. Those transients are chosen which are representative of transients to be expected during plant operation and which are sufficiently severe or frequent 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 60 years operating life. (NRC SE dated 12/2005, NUREG -1839) For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients.

Heatup and Cooldown

The 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 will not be attained because of other limitations such as:

1. Material NDT considerations which may establish maximum permissible temperature rate of change, as a function of plant pressure and temperature, which are below the design rate of 100°F per hour.

2. Slower initial heatup rates attainable from pump energy and pressurizer heaters only.

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

4. Design and operating restrictions associated with reactor critical conditions.

The number of complete heatup and cooldown operations is specified at 200 times for the 60-year plant design life. For the ideal plant, only one heatup and one cooldown would occur per fuel cycle, i.e., the period between refuelings. (NRC SE dated 12/2005, NUREG -1839) 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.

Unit Loading and Unloading

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5% per minute between no load and full load. The reactor coolant temperature will vary with load as prescribed by the temperature control system. The number of each operation is specified in Table 4.1-8 for the 60-year plant life. (NRC SE dated 12/2005, NUREG-1839) In practice, the plant is generally operated at base load conditions with changes in power at a rate much less than 5% per minute.

Step Increase and Decrease of 10%

The ±10% 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 reactor control system is designed to restore plant equilibrium without reactor trip following a ±10% step change in turbine load demand in the range between 15% and 100% full load, the power range for automatic reactor control. In effect, during load change conditions, the reactor control system attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed set point at a sufficiently slow rate to prevent excessive pressurizer pressure change.

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 Reactor Coolant System average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor, the increase in 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 set point change is made as a function of turbine generator load as determined by first stage turbine 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 control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature 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 for the 60-year plant life. (NRC SE dated 12/2005, NUREG -1839)

Large Step Decreases in Load

This transient applies to a 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 condenser steam dump system to avert a reactor shutdown or lifting of steam generator safety valves. The number of occurrences of this transient is specified at 200 times for the 60-year plant life. (NRC SE dated 12/2005, NUREG -1839) The operating experience of Point Beach Nuclear Plant Units 1 and 2 also indicates that this basis is adequately conservative.

Loss-of-Load Transient

The loss-of-load transient is the most severe transient on the Reactor Coolant System. The 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. The reactor and turbine eventually trip as a consequence of a high pressurizer pressure trip initiated by the reactor protection system. See Section 14.1.9 for loss-of-load transient analysis.

Loss-of-Flow

The loss-of-flow 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 are a reactor and turbine trip followed by automatic opening of the steam dump system and flow reversal in the affected loop. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop. See Section 14.1.8 for loss-of-flow transient analysis.

The number of occurrences of the above transients is generally specified at two per year of plant design life. All components in the Reactor Coolant System are designed to withstand the effects of these and other transients that result in system temperature and pressure changes.

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

The number of occurrences of this transient is specified at 400 times for the 60 year plant life. (NRC SE dated 12/2005, NUREG -1839) The tripping history of Point Beach Nuclear Plant Units 1 and 2 indicate that this basis is indeed conservative.

Feedwater Cycling at Hot Standby

Feedwater cycling can occur when the plant is being maintained at hot standby or no-load conditions. This transient assumes the intermittent addition of 32°F feedwater into the steam generator secondary side while it is in a no-load condition at 547°F. For design purposes, it is assumed that the steam generators will experience 25,000 cycles of cold feedwater introduction. Feedwater additions required during plant heatup and cooldown are assumed to by bounded by the feedwater cycling transient, with no increase in the total number of cycles.

Boron Concentration Equalization

Following a large change in boron concentration in the RCS, spray is initiated in order to equalize concentration between the loops and the pressurizer. For design purposes, it is assumed that this operation is performed once after each unit loading or unloading. The number of loading and unloading operations is defined as 11,680 occurrences during the 60-year life of the plant. On this basis, the total number of boron concentration equalization cycles is 23,360.

Loss of Power

This transient applies to a blackout situation involving the loss of outside electrical power to the station with a reactor and turbine trip. Under these circumstances, the reactor coolant pumps are 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 are assumed to receive feedwater from the Auxiliary Feed System (operating from diesel generator power). Steam is removed for reactor cooldown through atmospheric relief valves. The number of occurrences of this transient is assumed to be a total of 40 times in a 60-year plant life.

Inadvertent Actuation of Auxiliary Spray

Inadvertent actuation of auxiliary spray will occur if the auxiliary spray valve is opened inadvertently during normal operation of the plant. This will introduce cold water into the pressurizer with a very sharp pressure decrease within the pressurizer, as a result. The pressure decreases rapidly to the low pressure reactor trip point, at which point it is assumed the trip is actuated. This accentuates the pressure decrease until the pressure is finally limited to the hot leg saturation pressure. At five minutes, spray is stopped and all the pressurizer heaters return the pressure to 2250 psia. For design purposes, it is assumed that there are no temperature changes in the RCS, with the exception of the pressurizer. A total of 10 occurrences of this transient are specified for a 60-year plant life.

It should be noted that the design transient pressurizer pressure and temperature variations are considered only to occur in the pressurizer during Inadvertent Actuation of Auxiliary Spray. The design transient is not applicable to the other RCS components.

Reactor Coolant Pipe Break

This transient involves the postulated rupture of a Reactor Coolant System pipe resulting in a loss of coolant. It is conservatively assumed that the system pressure is reduced rapidly and the emergency core cooling system (ECCS) is initiated to introduce water into the reactor coolant system. Because of the rapid blowdown of coolant from the system and the conservatively large heat capacity of the metal sections of the components, it is likely that the metal will remain at or near the no-load temperature conditions when the ECCS water is introduced into the system.

This hypothetical transient is not expected to occur. The postulated one-time event was included in the transient sets used to evaluate thermal and loading cycles over the 60-year plant life.

Steam Line Break

For component evaluation, the following conservative conditions are considered:

1. The reactor is initially in a hot, zero-power subcritical condition assuming all rods in except the most reactive rod which is assumed to be stuck in its fully withdrawn position.
2. A major steam line rupture occurs and the result is a reactor and turbine trip.
3. Subsequent to the break the reactor coolant temperature cools down to 212°F.
4. The ECCS pumps restore the reactor coolant pressure to 2500 psia.

This hypothetical transient is not expected to occur. The postulated one-time event was included in the transient sets used to evaluate thermal and loading cycles over the 60-year plant life.

Turbine Roll Test

The turbine roll test transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature (no-load conditions), and the steam generated is used to perform a turbine roll test. The number of test cycles is specified as 10 occurrences, to be performed at the beginning of plant operating life prior to irradiation.

Steady-State Fluctuations

The reactor coolant pressure and temperature can vary around the steady state values during operation. For purposes of design, two cases are considered. Initial fluctuations due to control rod cycling during the first 20 months of operation are assumed to result in reactor coolant temperature and pressure variations of +/-3°F and +/-25 psi once during each 2-minute period. The total number of these occurrences is limited to 150,000 cycles. In addition, random fluctuations of reactor coolant temperature (varying by 0.5°F) and pressure (varying up to +/-6 psi) are assumed to occur once during each 6-minute period. The total number of these random occurrences during the plant life is specified at 5,000,000 cycles.

Hydrostatic Test Conditions

The pressure tests outlined below apply to field hydrostatic tests conducted on the erected reactor coolant system. The number of tests given below does not include any allowance for pressure tests conducted on a specific component in the manufacturer's shop in accordance with vessel code requirements.

1. Primary Side Hydrostatic Test Before Initial Startup at 3110 psig

   This hydrostatic pressure test was performed at a minimum water temperature of 100°F imposed by reactor vessel material Crack Arrest Temperature (CAT) of 100°F at beginning of life, and a maximum test pressure of 3110 psig. In this test, the primary side of the steam generator was pressurized to 3110 psig coincident with the secondary side pressure of 0 psig. The Reactor Coolant System was evaluated for up to 5 cycles of this hydrostatic pressure test.

2. Primary Side Post Operation Leak Test at 2485 psig

   The Reactor Coolant System is designed to permit periodic pressure testing to assure the structural and leaktight integrity of its components. All components in the Reactor Coolant System are designed to withstand the effects of transients that result in system temperature and pressure changes.

Stress intensity values at all critical points in the reactor vessel due to these excursions of pressure and temperature are determined for each of these transients through systematic analytical procedures. These stress intensity values S_{ij} (i,j = 1, 2, 3) are plotted against a time interval for each cycle. This plot may represent one or more stress cycles. For each cycle, extreme values of S_{max} and S_{min} are determined. From these values, the largest S_{alt} (alternating stress intensity) is found.

For this value of S_{alt}, an allowable number of cycles (N) is determined through design fatigue curves established for specific materials. The ratio of design cycles (n) to allowable cycles (N) gives the usage factor u_i (i = 1, 2, 3, etc.). Usage factor is determined in this manner for all transients. The cumulative usage factor is determined by summing the individual usage factors. The cumulative usage factor ($U = u_1 + u_2 + u_3 ...$) is never allowed to exceed a value of 1.0. This means that the allowable number of cycles always exceeds the design cycles. This certainty assures safety of the components against fatigue failure.

Service Life

The service life of Reactor Coolant System pressure components depends upon the end of life material radiation damage, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

The reactor vessel is the only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to any appreciable material radiation damage effects. The RT_{NDT} shift of the vessel material and welds during service due to radiation damage effects is monitored by a material surveillance program which conforms with ASTM E185-82 (Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels).

Reactor coolant system pressure and temperature limits, including those for plant heatup and cooldown, are obtained in accordance with 10 CFR 50, Appendix G by following the methods of analysis and the required margins of safety of Appendix G of ASME Code Section XI. Additional discussion of these limits is provided in Section 4.3.

To establish the service life of the Reactor Coolant System components as required by the ASME (Part III) Boiler and Pressure Vessel Code for Class A Vessels, the unit operating conditions have been established for the 60-year life. (NRC SE dated 12/2005, NUREG-1839) These operating conditions include the cyclic application of pressure loadings and thermal transients. The number of thermal and loading cycles used for design purposes is listed in Table 4.1-8 (Reference 10).

CODES AND CLASSIFICATIONS

All pressure containing components of the Reactor Coolant System are designed, fabricated, inspected, and tested in conformance with the applicable codes listed in Table 4.1-9. Unless stated otherwise, the version of the code which was in effect at the time the original component was ordered is applicable. The Reactor Coolant System 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 maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously, when combined with the primary steady state stresses.

REFERENCES

1. G.E. Lear, “Exemption from the requirements of 10 CFR 50 Appendix A, General Design Criterion 4,” dated May 6, 1986.

2. Westinghouse WCAP 14439 P Revision 2, “Technical Justification for Eliminating Large Primary Loop Pipe Units 1 and 2 for the Power Uprate and License Renewal Program.” (Proprietary)

3. H.K. Chernoff, "Point Beach Nuclear Plant, Units 1 and 2, Issuance of Amendments re: Leak Before Break Evaluation for Primary Loop Piping (TAC Nos. MC1279 and MC1280)," dated June 6, 2005.

4. NRC Letter, V. L. Ordaz (NRC) to J. McCarthy (NMC), dated December 16, 2004.

5. NRC Letter, V. L. Ordaz (NRC) to J. McCarthy (NMC), dated December 17, 2004.

6. WCAP-15065-P-A, Rev. 1 "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Point Beach Units 1 and 2 Nuclear Plants," (Proprietary) dated June 1, 2001.

7. NRC SE "Safety Evaluation of the Request to Apply Leak-Before-Break Status to the Pressurizer Surge Line Piping," dated December 15, 2000.

8. NRC SE "PBNP, Units 1 and 2 - Supplement to Safety Evaluation on Leak-Before-Break Regarding Correction of Leak Detection Capability," dated February 7, 2005.

9. NMC Letter, WE Energies and FPL Energy Point Beach to NRC, "Application for Order and Conforming License Amendments to Transfer Facility Operating Licenses," dated January 26, 2007.

10. WCAP-16983-P, Rev. 0, "Point Beach Units 1 and 2 Extended Power Uprate (EPU) Engineering Report," (Proprietary) dated September 2009.

11. NRC Safety Evaluation dated May 3, 2011, "Issuance of License Amendment Regarding Extended Power Uprate (TAC Nos. ME1044 and ME1045)."

12. FPL Energy letter to NRC, NRC 2009-0030, "License Amendment Request 261 Extended Power Uprate," dated April 7, 2009.

Table 4.1-1 REACTOR COOLANT SYSTEM DESIGN PARAMETERS AND PRESSURE SETTINGS

Total Primary Heat Output, MWt (w/RCPs)	1806
Total Primary Heat Output, Btu/hr	6162×10^6
Number of Loops	2
Coolant Volume (liquid), including original pressurizer volume, at full power (60% full), ft^3	6148 (Unit 2) 6000 (Unit 1)
Total Reactor Coolant Flow, lb/hr	$67.6\text{-}69.3 \times 10^6$
	Pressure (psig)
Design Pressure	2485
Operating Pressure (at pressurizer)	2235 ± 100
Safety Valves	2485
Power Operated Relief Valves	2335(1)
Pressurizer Spray Valves (open)	2260
High Pressure Trip	≤2385
Low Pressure Trip	≥ 1855
Hydrostatic Test Pressure (Cold)	3110

(1) ≤ PORV lift setting limits for RCS low temperature operation as defined in TRM 2.2; Pressure Temperature Limits Report.

Table 4.1-2 REACTOR VESSEL DESIGN DATA

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, psig	3110
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, feet inches	39-0
Bottom Head O.D. to top of CRDM Housing	
Water Volume, ft^3 (with core and internals in place),	2473
Thickness of Insulation, min., in.	3
Number of Reactor Closure Head Studs	48
Diameter of Reactor Closure Head Studs, in.	6
Flange, ID, in.	123.8
Flange, OD, in.	157.3
ID at Shell, in.	132
Inlet Nozzle ID, in.	27.47
Outlet Nozzle ID, in.	28.97
Clad Thickness, min., in. (not including closure head)	0.156
Clad Thickness, min., in. (closure head)	0.125
Lower Head Thickness, min., in.	4.125
Vessel Belt Line Thickness, min., in.	6.5
Closure Head Thickness, in.	5.375
Reactor Coolant Inlet Temperature, °F	523.1 (552.5)[1]
Reactor Coolant Outlet Temperature, °F	611.1 (610.1)[1]
Reactor Coolant Flow, lb/hr	67.6 x 10^6
Safety Injection Nozzle, number/size, in.	2/4

(1) Original reactor coolant inlet and outlet temperatures. Reactor coolant temperature operating band was changed subsequent to initial plant operation.

Table 4.1-3 PRESSURIZER AND PRESSURIZER RELIEF TANK DESIGN DATA

<u>Pressurizer</u>	
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design/Operating Temperature, °F	680/653
Water Volume, Full Power, ft^3	472
Steam Volume, Full Power, ft^3	528
Surge Line Nozzle Diameter, in./Pipe Schedule	14/Sch 140
Shell ID, in./Minimum Shell Thickness, in.	84/4.1
Minimum Clad Thickness, in.	0.188
Electric Heaters Capacity, kw (total)	1000(2)
Maximum Heatup rate of Reactor Coolant System using Heaters only, °F/hr	55 (approximately)
<u>Power Relief Valves</u>	
Number	2
Set Pressure (open), psig	2335(1)
Capacity, lb/hr saturated steam/valve	179,000
<u>Safety Valves</u>	
Number	2
Set Pressure, psig	2485
Capacity, lb/hr saturated steam / valve	288,000
<u>Pressurizer Relief Tank</u>	
Design pressure, psig	100
Rupture Disc Release Pressure, psig	100
Design temperature, °F	340
Normal water temperature, °F	Containment Ambient
Total volume, ft^3	800
Rupture Disc Relief Capacity, lb/hr	7.2×10^5

(1) ≤ PORV lift setting limits as defined in TRM 2.2; Pressure Temperature Limits Report.
(2) Design value. Control system analysis supports a minimum value of 670 KW (total).

Table 4.1-4 STEAM GENERATOR DESIGN DATA

Sheet 1 of 2

	Unit 2	Unit 1
Model	Δ47	44F
Number of Steam Generators	2	2
Design Pressure, Reactor Coolant/ Steam, psig	2485/1085	2485/1085
Tube Design Primary-to-Secondary Differential Pressure, psig	1700	1700
Reactor Coolant Hydrostatic Test pressure (tube side-cold), psig	3107	3106
Design Temperature, Reactor Coolant/Steam, °F	650/556	650/556
Reactor Coolant Flow, gpm	89,000	89,000
Total Heat Transfer Surface Area, ft^2	47,500	43,467
Heat Transferred, Btu/hr	3081×10^6	3081×10^6
Steam Conditions at Full Load, Outlet Nozzle:		
Steam Flow, 10^6 lbm/hr	3.68 - 4.06	3.68 - 4.06
Steam Temperature, °F	486.3 - 511.6	486.3 - 511.6
Steam Pressure, psia	601 - 755	601 - 755
Feedwater Temp., at 100% Load, °F	390.0 - 458.0	390.0 - 458.0
Overall Height, ft-in.	62-11	63-1.6
Shell OD, upper/lower, in.	166.4/127.8	166/127
Shell Thickness, upper/lower, in.	3.47/2.61	3.5/2.62
Number of U-Tubes	3499	3214
U-Tube OD, in.	0.875	0.875
Tube Wall Thickness, (nominal), in.	0.050	0.050
Number of Manways/ID, in.	4/16	3/16
Number of Handholes/ID, in.	6/6	6/6
Inspection Ports/ID, in.	2/4	1/3

Table 4.1-4 (cont'd) STEAM GENERATOR DESIGN DATA

Sheet 2 of 2

	---- Unit 2 ----		----Unit 1----	
	1806 MWt	Zero Power	1806 MWt	Zero Power
Reactor Side Coolant Water Volume, ft^3	991	991	925	925
Primary Side Fluid Heat Content, 10^6 Btu	23.6 - 25.8	25.3	22.2 - 24.2	24.42
Secondary Side Water Volume, ft^3	1353-1577	2704	1443 - 1672	2877
Secondary Side Steam Volume, ft^3	3084 - 3309	1970	3026 - 3256	1804
Secondary Side Fluid Heat Content, 10^6 Btu	36.9 - 43.5	73.5	39.7 - 45.6	75.5

Table 4.1-5 REACTOR COOLANT PUMPS DESIGN DATA

Number of Pumps	2
Design Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design Temperature (casing), °F	650
RPM at Nameplate Rating	1189
Suction, Temperature, °F	551.8
Net Positive Suction Head, ft.	172
Developed Head, ft.	252
Capacity, gpm	89,000
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump Discharge Nozzle ID, in.	27.5
Pump Suction Nozzle ID, in.	31
Overall Unit Height, ft.	28.4
Water Volume, ft^3	192
Pump Motor Moment of Inertia, lb ft^2	80,000
Motor Data:	
Type	AC Induction Single Speed, Air Cooled
Voltage	4000
Insulation Class	B Thermalastic Epoxy
Phase	3
Frequency, cps	60
Current, maximum, amp	4800
Input (hot reactor coolant), kw	4000
Input (cold reactor coolant), kw	5300
Power, HP (nameplate)	6000

Table 4.1-6 REACTOR COOLANT PIPING DESIGN DATA

Parameter	Value
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, (cold) psig	3110
Design Temperature, °F	650
Design Temperature, (pressurizer surge line), °F	680
Reactor Inlet Piping, ID, inches	27 1/2
Reactor Inlet Piping, nominal thickness, inches	2.375
Reactor Outlet Piping, ID, inches	29
Reactor Outlet Piping, nominal thickness, inches	2.50
Coolant Pump Suction Piping, ID, inches	31
Coolant Pump Suction Piping, nominal thickness, inches	2.625
Pressurizer Surge Line Piping, ID, inches/Pipe Schedule	10/Sch 140*
Pressurizer Surge Line Piping, nominal thickness, inches	1*
Water Volume, (2 loops) ft^3	552

* Surge line fitted with a 14"/10" adapter at the pressurizer

Table 4.1-7 REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP(1)

	Pressure Drop, psi
Across Pump Discharge Leg	1.3
Across Vessel, including nozzles	44.0
Across Hot Leg	1.5
Across Steam Generator	32.2
Across Pump Suction Leg	3.0
Total Pressure Drop	82.0

(1) These are nominal full power design values provided in the FFDSAR. Subsequent changes, such as the replacement of both units' steam generators, the core barrel upflow modification, and fuel design changes, have changed these values. This information is historical.

Table 4.1-8 THERMAL AND LOADING CYCLES

Transient Condition	Design Cycles*
1. Plant heatup at 100°F per hour	200
2. Plant cooldown at 100°F per hour	200
3. Plant loading at 5% of full power per minute	18,300 (for all components except pressurizer and reactor vessel internal baffle bolts which are 11,600 and 2,485 respectively)
4. Plant unloading at 5% of full power per minute	18,300 (for all components except pressurizer and reactor vessel internal baffle bolts which are 11,600 and 2,485 respectively)
5. Step load increase of 10% of full power (but not to exceed full power)	2,000 (1)
6. Step load decrease of 10% of full power	2,000 (1)
7. Step load decrease of 50% of full power	200 (1)
8. Steady State Fluctuations	
Initial Fluctuations (+3°F and + 25 psi)	1.5×10^5
Random Fluctuations (+0.5°F and + 6 psi)	5×10^6
9. Feedwater cycling at hot standby	2000 Reactor Vessel
	25,000 (Unit 1 - other components)
	10,000 (Unit 2 - other components)
10. Boron concentration equilibrium	23,360
11. Loss of Load	80 (1)
12. Loss of Power	40 (1)
13. Loss of flow in one loop	80 (1)
14. Reactor trip and attendant temperature transients	400 (1)
15. Inadvertent auxiliary spray	10
16. Reactor Coolant Pipe Break	1
17. Steam Line Break	1
18. Turbine roll test	10
19. Hydrostatic test, pressure 3110 psig temperature-cold	5 (preoperational)
20. Hydrostatic test, pressure 2485 psig temperature 400°F	94 (post-operational)
21. Primary to secondary leak test (2250) psig	27
22. Secondary to primary leak test	128

* Estimated for equipment design purposes (60-year life) and not intended to be an accurate representation of actual transients or to reflect actual operating experience. These cycles also assume a power uprate. (NRC SE dated 12/2005, NUREG 1839)

(1) For Reactor Vessel Internal baffle bolts, the total of these 7 transients is 750.

Table 4.1-9 REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

Component	Codes
Reactor Vessel (excluding reactor vessel closure head)	ASME III* Class A
Reactor Vessel Closure Head	ASME III* Class 1; 1998 Edition through 2000 Addenda
Control Rod Drive Mechanism Housing	ASME III* Class 1; 1998 Edition through 2000 Addenda
Steam Generators	
Tube Side	Unit 1: ASME III* Class 1; 1977 Edition through Winter 1978 Addenda.
	Unit 2: ASME III* Division 1, Subsection NB; 1986 Edition, No Addenda.
Shell Side	Unit 1, Upper Shell above Transition Cone: ASME III* Class C; 1965 through 1966 Summer Addenda. NOTE: The shell side of the original Steam Generators conformed to the requirements for Class A vessels and were so stamped.
	Unit 1, Lower Shell and Transition Cone: ASME III* Class 2; 1977 Edition through Winter 1978 Addenda. NOTE: The lower shell and Transition Cone of the replacement Steam Generators were designed to Class 1 requirements.
	Unit 2: ASME III* Division 1, Subsection NB; 1986 Edition, No Addenda.
Reactor Coolant Pump Casing	No Code (Design per ASME III Article H)
Pressurizer	ASME III* Class A
Pressurizer Relief Tank	ASME III* Class C
Pressurizer Safety Valves	ASME III*
Reactor Coolant Piping	USAS B31.1**
System Valves, Fittings, and Piping	USAS B31.1**

Note: The version of the code which was in effect at the time the original component was ordered is applicable.

* ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels
** USAS B31.1 Code for Pressure Piping

4.2 RCS SYSTEM DESIGN AND OPERATION

General Description

The Reactor Coolant Systems of the two nuclear power plant units are essentially identical and do not share any components. The following description applies to either unit.

Each Reactor Coolant System consists of two similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, loop piping, and instrumentation. The pressurizer is connected to one of the loops by the pressurizer surge line. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system is shown in Figure 4.2-1 (Unit 1) and Figure 4.2-1A (Unit 2).

The containment boundary shown on the flow diagram indicates those major components which are to be located inside the containment. The intersection of a process line with this boundary indicates a containment penetration. Reactor Coolant System and components design data are listed in Table 4.1-1 through Table 4.1-7.

Pressure in the system is controlled by the pressurizer, where water and steam pressure are maintained through use of electrical heaters and sprays. Steam can either be formed by the heaters or condensed by a pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressure control system is described in Section 7.0. Spring loaded steam safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank where the discharged steam is condensed and cooled by mixing with water.

COMPONENTS

Reactor Vessel

The reactor vessel is cylindrical in shape with a hemispherical bottom head and a flanged and gasketed removable hemispherical upper head. Figure 4.2-2 is a schematic of the reactor vessel. The materials of construction of the reactor vessel are given in Table 4.2-1.

Coolant enters the reactor vessel through inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction. Approximately 95% of the total coolant flow is effective for heat removal from the core. The core bypass flow provides cooling to parts of the vessel and internal components, including upward flow between the core baffle plates and core barrel to provide cooling of the barrel, the flow deflected into the head of the vessel for cooling and also includes the flow through the RCC guide-tubes and, the leakage across the fuel assembly outlet nozzles. All the coolant is united and mixed in the upper plenum, and the mixed coolant stream then flows out of the vessel through exit nozzles located on the same plane as the inlet nozzles.

A one-piece thermal shield, concentric with the reactor core, is located between the core barrel and the reactor vessel. The shield is bolted and welded to the top of the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the reactor vessel by attenuating much of the gamma radiation and some of the fast neutrons which escape from the core. This

shield minimizes thermal stresses in the reactor vessel which result from heat generated by the absorption of gamma energy. It is illustrated in Figure 3.2-35 and is further described in Section 3.2.3. Thirty-six core instrumentation nozzles penetrate the lower head.

The reactor closure head and the reactor vessel flange are joined by 48 six inch diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two O-rings to monitor leakage across the inner O-ring. In addition, a leak-off connection is also provided beyond the outer O-ring seal.

The reactor vessel insulation is primarily a reflective type, supported from the nozzles and consisting of inner and outer sheets of stainless steel with multi layer stainless steel foil as the reflective (insulating) agent. Metal reflective insulation is also installed on the reactor closure head.

The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, control rod cluster assemblies, surveillance specimens, and in-core instrumentation access thimbles. The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods in the withdrawn position.

Surveillance specimens made from representative reactor vessel steel are located between the reactor vessel wall and the thermal shield. Periodically removed specimens are examined to evaluate reactor vessel material property changes as described in Section 4.4.

The reactor internals are described in detail in Section 3.2.3 and the general arrangement of the reactor vessel and internals is shown in Figure 3.2-35. Reactor vessel design data are listed in Table 4.1-2.

Reactor Vessel - Support Structure

The Reactor Support Structure consists of a six sided structural steel ring supported at each apex by steel columns extending downward to a point below the reactor vessel and, at the center of each segment of the ring, by structural members imbedded in the surrounding concrete.

The reactor vessel has six supports, one at each of four reactor vessel nozzles with pads, and one at each of two reactor vessel support brackets. Each support bears on a support shoe, which is fastened to the support structure. The support shoe is a structural member that transmits the support loads to the supporting structure. The support shoe is designed to restrain vertical, lateral, and rotational movement of the reactor vessel, but allows for thermal growth by permitting radial sliding at each support on bearing plates.

Pressurizer

The general arrangement of the pressurizer is shown in Figure 4.2-3, and the design data are listed in Table 4.1-3. The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure.

The pressurizer vessel contains replaceable direct immersion heaters, multiple safety and relief valves, a spray nozzle, and interconnecting piping, valves, and instrumentation. The electric heaters, located in the lower spherical head of the vessel, maintain the pressure of the Reactor Coolant System by keeping the water and steam in the pressurizer at system saturation temperature. The heaters are capable of raising the temperature of the pressurizer and contents at approximately 55°F/hr during RCS heatup.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line which is attached to the bottom of the pressurizer, connects the pressurizer to the hot leg of a reactor coolant loop. During a positive surge caused by an increase in RCS temperature, the spray system, which is fed from the cold leg of each coolant loop, operates to condense steam in the pressurizer vessel to prevent the pressure from reaching the setpoint of the power-operated relief valves. Though normally automatically controlled, the gas operated spray valves can be operated manually from the control room. A small continuous spray flow is provided to assure that the pressurizer surge line and spray piping do not cool excessively during steady-state conditions.

During a negative pressure surge caused by decreasing RCS temperature, water in the pressurizer flashes to steam to mitigate the pressure drop, and heaters automatically actuate to restore RCS pressure to normal. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer is constructed of carbon steel with internal surfaces clad with austenitic stainless steel. The heaters are sheathed in austenitic stainless steel. All nozzle safe ends (forgings) in the top and bottom heads, and the nozzles of the pressurizer safety valves which could have been furnace sensitized during the fabrication sequence, have received non-destructive examination, which showed no degradation in integrity of the materials.

The pressurizer vessel surge nozzle is protected from thermal shock by a thermal sleeve. A thermal sleeve also protects the pressurizer spray nozzle connection.

PRESSURIZER SAFETY VALVE LIFT INDICATING SWITCH ASSEMBLIES (LISA)

See Section 7.5.1.3 for a description of the LISAs.

Pressurizer - Support Structure

The pressurizer is supported on a heavy concrete slab spanning the concrete shield walls of its compartment. The pressurizer is a bottom-skirt supported vessel.

Steam Generators

Each loop contains a vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 4.2-4. Principal design parameters are listed in Table 4.1-4. Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel, and leaves the generator through another bottom nozzle.

The inlet and outlet channels are separated by a partition. Primary side manways are provided to permit access to the U-tubes. This permits steam generator tubes to be periodically inspected and allows defective tubes to be repaired or plugged in accordance with approved procedures.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus formed by the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly which imparts a centrifugal motion to the mixture, separating the water droplets from the steam. Operation under EPU conditions required modifications to the moisture separation and steam drying components to limit steam moisture content to 0.25%. The mid-deck inlet vent area was reduced, the open top pipe vent design was changed to a flow diverter vent pipe design with vent caps, the formed single pocket vanes in the double tier secondary separators were replaced with double pocket vanes, the mid-deck plate was extended to the S/G shell wall and an inspection hatch was also added. Evaluations identified no predicted vibrational issues for the PBNP Units 1 and 2 SG steam dryer bank assemblies operating at EPU conditions.

The steam generator is constructed primarily of carbon steel. The heat transfer tubes are Inconel. The interior surfaces of the channel heads and nozzles are clad with austenitic stainless steel, and the side of the tubesheet in contact with the reactor coolant is clad with a NiCrFe Alloy. The tube-to-tubesheet joint is welded.

The following discussion of tubesheet stress analysis is retained in the FSAR for historical perspective. (Reference 10)

The evaluation of both units' Westinghouse steam generator tubesheets is performed according to rules of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels, Section III, 1965 through Summer 1966 Addenda Edition Article 4 - Design. The design criteria encompasses steady-state, transient, and emergency operations as specified in the Equipment Specification. Due to the complex nature of the tube-tubesheet shell head structure, the analysis of the tubesheet required the application of results of related research programs (such as the design data on perforated plates resulting from PVRC programs) and the utilization of current techniques in computer analysis, the application of which is verified by comparison of analytical and experimental results for related equipment.

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

Generally, the analytic treatment of the tube-tubesheet complex includes 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 tubesheet, the flexural rigidity is based on studies of behavior of plates with square hole arrays utilizing techniques such as those reported by O'Donnell (Reference 1), Mahoney (Reference 2), Lemcoe (Reference 3), and others. Similarly, stress intensity factors are determined for square hole arrays using the combined equivalent plate interaction forces and moments applied to results of photoelastic tests of model coupons of such arrays as well as verification using computer analysis techniques such as "Point Matching" or "Collocation." The stress analysis considers stress due to symmetric temperature and pressure drop across the tubesheet divider lane.

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

The evaluation of the tube-to-tubesheet juncture of Westinghouse PWR System steam generators is based on a stress analysis of the interaction between tube and tubesheet hole for the significant thermal and pressure transients that are applied to the steam generator in its predicted histogram of cyclic operation. The evaluation is based on the numerical limits specified in the 1968 Edition of 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 tubesheet complex with attached head and shell. Hence, the output of the tubesheet analysis giving equivalent plate stresses in the perforated region is utilized in determining the free boundary displacements of the perforation to which the tube is attached.

Analysis of the juncture for the tube-to-tubesheet fillet-type weld utilized in the Westinghouse steam generator design has been 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 is the fatigue strength reduction factor to be assigned to the weld root notch. For this reason, Westinghouse has conducted low-cycle fatigue tests of tube material samples to determine the fatigue strength reduction factor and 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 is not different from that reported in the well known paper on the subject by O'Donnell and Purdy (Reference 4). An actual tubesheet joint contained in a tubesheet has been successfully tested experimentally under thermal transient conditions much more severe than that achieved in anticipated power plant operation. A wide range of computational tools are utilized in these solutions including finite element, heat conduction, and thin shell computer solutions. In addition, analysis techniques have been verified by photoelastic model tests and strain gaging of prototype models of an actual steam generator tubesheet.

Finally, in order to evaluate the ultimate safety of the structural complex, a computer program for determining a lower-bound pressure limit for the complex based on elastic-plastic analysis has been 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 Westinghouse steam generator tubesheet complex meets the stress limitations and fatigue criteria specified in Article 4 of the Code as well as emergency condition limitations specified in the Equipment Specifications or anticipated otherwise. In this way, the tube-tubesheet integrity of a Westinghouse steam generator is demonstrated under the most adverse conceivable conditions resulting from a major breach in either the primary or secondary system piping.

Steam Generator - Support Structure

Each steam generator is supported on a structural system consisting of four vertical support columns and two (upper and lower) support rings. The vertical columns, which are pin connected to the steam generator support feet, serve as vertical restraint for operating weights, pipe rupture, and seismic considerations while permitting movement in the horizontal plane. The support rings, by using a combination of pins, stops, guides, and snubbers, prevent rotation and excessive movement of the steam generator in any plane. Thermal expansion is permitted in the support rings by a key arrangement.

Unit 1 - Steam Generator Replacement

Both Unit 1 steam generators lower assemblies were replaced during 1984. The performance of the replacement lower assemblies matches the performance of the original lower assemblies. However, several design features that do not alter the performance parameters are included in the design. Design data of the replacement Westinghouse Model 44F steam generators is provided in Table 4.1-4. The design features of the Model 44F steam generator lower assemblies and modifications made to the moisture separator equipment of the upper assemblies provide improved thermal hydraulic performance, provide improved access to the tube bundle, and reduce the potential for secondary side corrosion.

Unit 2 - Steam Generator Replacement

Both Unit 2 steam generators have been replaced. Whereas the Unit 1 replacement project changed out only the lower assemblies, the Unit 2 replacement steam generators (RSGs) consisted of the complete vessel, i.e., both the lower and upper assemblies. The RSGs are Westinghouse Model 47 and are similar in design and functionally the same as the original Westinghouse Model 44 steam generators. Design data of the replacement generators for Unit 2 are provided in Table 4.1-4. The RSGs have design features which provide additional resistance to known degradation mechanisms and which support their reliability and maintainability.

Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage centrifugal pump which employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 4.2-6 and the principal design parameters for the pumps are listed in Table 4.1-5. The reactor coolant pump estimated performance and NPSH characteristic are shown in Figure 4.2-7. The performance characteristic is common to all of the higher specific speed centrifugal pumps and the "knee" at about 45% design flow introduces no operational restrictions since the pumps operate at full flow.

The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pumps in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, as well as a secondary seal which directs the controlled leakage out of the pump, and a third seal which minimizes the leakage of water and vapor from the pump into the containment atmosphere.

The shaft seal section consists of the No. 1 controlled leakage, film riding face seal, a shut down seal (SDS) assembly, and the No. 2 and No. 3 rubbing face seals. The seals are contained within the main flange and seal housing. The SDS is housed within the No. 1 seal area and is a passive device actuated by high temperature resulting from a loss of seal injection and CCW cooling to the thermal barrier heat exchanger. The SDS is designed to function only when exposed to an elevated fluid temperature downstream of the RCP number 1 seal. SDS deployment limits leakage from the RCS through the RCP seal package. Leakage is limited when the SDS thermal actuator retracts due to intrusion of hot reactor coolant water into the seal area, which causes the SDS seal ring to constrict around the pump shaft.

Testing of pumps with the number 1 seal entirely bypassed (full system pressure on the number 2 seal) shows that small (approximately 4 to 12 gpm) leakage rates would be maintained for a period of time sufficient to secure the pump. Even if the number 1 seal were to fail entirely during normal operation, the number 2 seal would maintain these small leakage rates if the proper action is taken by the operator. An increase in number 1 seal leakoff rate will warn the plant operator of number 1 seal damage. Following warning of excessive seal leakage conditions, the plant operator will take corrective actions. Gross leakage from the pump does not occur if these procedures are followed.

A portion of the high pressure water flow from the charging pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal surrounding the lower pump shaft. The labyrinth seal serves as a buffering interface, to limit the exchange of reactor coolant from the seal portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount which leaks through the secondary seal is also collected and removed from the pump. Component cooling water is supplied to the motor bearing cooler and the thermal barrier cooling coil.

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

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. The primary coolant pumps run at 1189 rpm and the motors are designed in accordance with NEMA standards for operation at a maximum speed of 125% of rated speed. Each component of the primary pumps has been analyzed for missile generation. Any fragments would be contained by the heavy stator. The same conclusion applies to the impeller because the small fragments that might be ejected would be contained by the heavy casing.

The primary coolant pump flywheels are shown in Figure 4.2-8. As for the pump motors, the most adverse operating condition of the flywheels is 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, steel plates. The material is ASTM A533 Grade B Class 1. (Reference 11) 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 to determine that each blank satisfies design requirements. An NDTT less than +10°F is specified. The finished flywheels are subjected to 100% 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 in Figure 4.2-9) less than 50% of the minimum specified material yield strength at room temperature (100 to 150°F). Bursting speed of the flywheels has been calculated on the basis of Griffith-Irwin's results (Reference 6), 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%.

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 attained on flywheel material, the critical crack size for failure was greater than 17 inches radially and the crack growth data was 0.030 in. to 0.060 in. per 1000 cycles. Ultrasonic examination techniques which are capable of detecting and sizing flaws smaller than the critical flaw size of the flywheel fracture analysis are utilized for the inspection of the flywheel. Based on the above information and the inspections outlined in the ISI Long-Term Plan, the intent of Regulatory Guide 1.14 is satisfied.

An additional stress and fracture evaluation was completed in November 1996 (WCAP-14535-A). The evaluation assumed a leak before break limitation on the maximum pump speed and 6000 cycles of reactor coolant pump starts and stops for a 60-year service life. The estimated radial crack extension was shown to be negligible even when assuming a large initial crack length. See Section 15.4.3 for further License Renewal information. (NRC SE dated 12/2005, NUREG-1839)

WCAP-15666-A, Revision 1, “Extension of Reactor Coolant Pump Motor Flywheel Examination,” October 2003, builds on the arguments in WCAP-14535-A and provides additional rationale, including a risk assessment of all credible flywheel speeds. The risk assessment

followed the risk-informed methodology and guidelines of Regulatory Guide 1.174 to justify the RCP motor flywheel examination interval extension for all domestic Westinghouse plants from 10 years to 20 years. WCAP-15666-A concludes that the change in risk is below the Regulatory Guide CDF and LERF acceptable guidelines.

The NRC approved the use of the Topical Report in NRC SER "Safety Evaluation of Topical Report WCAP-15666, Extension of Reactor Coolant Pump Motor Flywheel Examination," May 5, 2003. The NRC SER has been incorporated into the "A" revision of the WCAP.

All pressure bearing parts of the reactor coolant pump are analyzed in accordance with Article 4 of the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition. This includes the casing, the main flange, and the main flange bolts. The analysis includes pressure, thermal, and cyclic stresses, and these are compared with the allowable stresses in the Code. Mathematical models of the parts are prepared and used in the analysis which proceeds in two phases.

1. In the first phase, the design is checked against the design criteria of the ASME Code, with stress calculations using the allowable stress at design temperature. By this procedure, the shells are profiled to attain optimum metal distribution with stress levels adequate to meet the more exacting requirements of the second phase.

2. In the second phase, the interacting forces needed to maintain geometric capability between the various components are determined and applied to the components, along with the external load, to determine the final stress state of the components. This stress will also be used in the fatigue analyses. These results are finally compared with the Code allowable values.

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

Stress Analysis Reports are prepared on these components as described in Section 4.3. These reports include the calculation of stress intensities and a summary of fatigue usage factors. These reports are a part of the plant documentation on file with the applicant.

Reactor Coolant Pump Missile Protection

The construction of the loop compartment concrete walls is such that they enclose two sides of the reactor coolant pump area and protect the containment liner from loss-of-coolant accident generated missiles. The third side of the pump area is enclosed by the refueling canal wall. On the fourth side, a partition wall containing reinforcing steel and tension members divides the upper pump area from the steam generator compartment. The minimum compartment wall thickness is 30 inches.

Since there is no assumed mode of failure of the flywheel, no further design calculations were performed on this item as a missile. However, if a missile weight (W) 2500 lbs. (greater than 1/4 of flywheel) and a velocity (V) of 300 ft. per second were to strike the pump cavity walls, the penetration would be less than 20 inches, in accordance with the formula:

$$Penetration = \frac{222 \frac{W}{A} D^{0.215} V^{1.5}}{Y} + \frac{D}{2}$$

where:

Y = A function of the compressive strength of the concrete
A = Impact Area of 2.8 sq. ft.
D = Diameter of 22.7 inches

Pump Support Structure

The reactor coolant pump is supported by a structural system consisting of three vertical columns and a system of stops. The vertical columns are bolted to the pump support feet and permit movement in the horizontal plane to accommodate reactor coolant pipe expansion. Horizontal restraint is accomplished by a combination of tie rods and stops which limit horizontal movement for pipe rupture and seismic effects.

Pressurizer Relief Tank

Principal design parameters of the pressurizer relief tank are given in Table 4.1-3. Steam discharged from the power relief and safety valves passes to the pressurizer relief tank which is partially filled with water at or near ambient containment conditions. The tank normally contains water in a predominantly nitrogen atmosphere. Steam is discharged under the water level to condense and cool by mixing with the water. The tank is equipped with a spray and drain which are operated to cool the tank following a discharge.

The tank size is based on the requirement to condense and cool a discharge equivalent to 110% of the pressurizer steam volume above 60% (original full power) pressurizer level.

The tank is protected against a discharge exceeding the design value by a rupture disc which discharges into the reactor containment. The rupture disc on the relief tank has a relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure (and the rupture disc setting) is twice the calculated pressure resulting from the maximum safety valve discharge described above. This margin is to prevent deformation of the disc. The tank and rupture disc holder are also designed for full vacuum to prevent tank collapse if the tank contents cool without nitrogen being added.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20% of the setpoint pressure at full flow. The pressurizer relief tank, by means of its connection to the Waste Disposal System, provides a means for removing any noncondensable gases from the Reactor Coolant System which might collect in the pressurizer vessel. The tank is constructed of stainless steel.

Piping

The general arrangement of the Reactor Coolant System piping is shown on the plant layout drawings in Section 1. Piping design data are presented in Table 4.1-6. The reactor coolant piping layout is designed on the basis of providing "floating" supports for the steam generator and reactor coolant pump in order to absorb the thermal expansion from the fixed or anchored reactor vessel.

The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 in. I.D. in the hot legs, 27.5 in. I.D. in the cold legs, and 31 in. I.D. between each loop's steam generator outlet and its reactor coolant pump suction. Smaller piping, including the pressurizer surge spray and relief lines, drains, and connections to other systems are austenitic stainless steel. All joints and connections are welded except for stainless steel flange connections to the pressurizer relief tank and the connections at the safety valves.

Thermal sleeves are installed at the following locations where high thermal stresses could otherwise develop due to rapid changes in fluid temperature during normal operational transients:

1. Return line from the residual heat removal loop
2. Both ends of the pressurizer surge line
3. Pressurizer spray line connection to the pressurizer
4. Charging line and auxiliary charging line connections

Valves

Normally operating, outgoing lines connected to the Reactor Coolant System are provided with remote isolation capability. Each line is isolated near its connection to the Reactor Coolant System.

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. Connections to stainless steel piping are welded. Valves that perform a modulating function are equipped with sufficient packing to minimize leakage to the atmosphere.

Applicable Codes

Steel	American Institute of Steel Construction (AISC), "Code of Standard Practice for Steel Buildings and Bridges"
Welding	American Welding Society (AWS) D1.0-66 and (AWS) D12.1, "Standard Specification for Welding Highway and Railway Bridges"
Connections	Bolt Connections Conforming to "Specification for Structural Joints Using ASTM A325 or A490 Bolts" as approved by the Research Council on Riveted and Bolted Structural Joints of the Engineering Foundation, 1964
Concrete	American Concrete Institute (ACI) 318-63

PRESSURE-RELIEVING DEVICES

The Reactor Coolant System is protected against overpressure by control and protective circuits such as the high pressure trip and by code relief valves connected to the top head of the pressurizer. Those relief valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 4.2-1, and the valve design parameters are given in Table 4.1-3. Valve sizes are determined as indicated in Section 4.3.

Power-operated relief valves and code safety valves are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray. Additionally a keyswitch enabled bistable on each of two reactor coolant pressure channels allows the power-operated relief valves to perform as a low temperature overpressure protection system when the RCS temperature is below its minimum pressurization temperature. (Reference 7) The residual heat removal (RH) system relief valves also provide a diverse relief system for the reactor coolant system when the RH system is aligned for decay heat removal operation. (Chapter 9)

The pressurizer relief tank is protected against a steam discharge exceeding the design pressure value by a rupture disc which discharges into the reactor containment. The rupture disc relief conditions are given in Table 4.1-3.

PROTECTION AGAINST PROLIFERATION OF DYNAMIC EFFECTS

Protection against the proliferation of the dynamic effects of a Reactor Coolant System Main Loop or Pressurizer Surge Line pipe rupture is no longer a design or license basis requirement. See the discussion in Section 4.1 under "Missile Protection" for further information and historical context. The following is retained as historical information.

Engineered Safety Features and associated systems are protected from loss of function due to dynamic effects and missiles which might result from a loss-of-coolant accident. Protection is provided by missile shielding and/or segregation of redundant components. This is discussed in detail in Section 6.0.

The Reactor Coolant System is surrounded by concrete shield walls. These walls provide shielding to permit access into the containment during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the containment liner plate. A missile shield is integrated into the design of the reactor vessel head assembly and provides protection from missiles generated by postulated CRDM housing failures.

Steam generator lateral bracing is provided near the tubesheet and feedring elevations to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Missile protection afforded by the arrangement of the Reactor Coolant System is illustrated in the containment structure drawings which are given in Section 5.0.

MATERIALS OF CONSTRUCTION

Each of the materials used in the Reactor Coolant System is selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1. All of the Reactor Coolant System materials which are exposed to the coolant are corrosion resistant. They consist of several types of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in the EPRI PWR Primary Water Chemistry Guidelines (Reference 15). Reactor coolant chemistry is further discussed in Section 4.2.

The phenomena of stress corrosion cracking and corrosion fatigue are not encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, a specific chemical environment, a tensile stress, and time. It is characteristic of stress corrosion that combinations of alloy and environment which result in cracking are usually quite specific. Environments which have been shown to cause stress corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism and the presence of chlorides and free oxygen. With regard to the former, experience has shown that deposition of

chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator. In the presence of this environment, stress corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. However, the steam generators contain Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel alloy has excellent resistance to general and pitting type corrosion in severe operating water conditions.

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

Prior to the initial plant operation, the Nil-Ductility Transition Temperature (NDTT) of the reactor vessel plate or forging material opposite the core was established at a Charpy V-notch test value of 30 ft-lb or greater. The material was tested to verify conformity to specified requirements and to determine the actual NDTT value. In addition, this plate was 100% volumetrically inspected by ultrasonic testing using both longitudinal and shear wave methods.

Subsequently, the NRC issued 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," and Appendix G to Part 50, "Fracture Toughness Requirements." These regulations imposed an additional requirement applicable to Point Beach that the Charpy upper-shelf energy of reactor vessel beltline materials must be maintained no less than 50 ft-lb throughout the life of the vessel, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. Topical reports BAW-2178PA (Reference 8) and BAW-2192PA (Reference 9) were issued by the B&W Owners Group Reactor Vessel Working Group in April, 1994 and were applicable to PBNP Units 1 and 2. These reports demonstrated that the Point Beach Units 1 and 2 reactor vessel beltline welds fabricated by Babcock & Wilcox provided margins of safety against fracture equivalent to those required by Appendix G of the ASME Code through the end of their respective original Operating Licenses.

Additional reactor vessel fracture mechanics analyses for PBNP Units 1 and 2 were performed to satisfy the reactor pressure vessel (RPV) Charpy upper-shelf energy (USE) requirements of 10 CFR 50, Appendix G, Section IV.A.1.c through the end of the unit's extended operating licenses. See Section 15.4.1 for a description of theses analyses.

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

The reactor vessel material was heat treated specifically to obtain good notch ductility, which ensures a low NDTT and thereby gives assurance that the finished vessel can be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. A reactor cavity neutron measurement program has been instituted at Point Beach to provide a continuous monitoring of the reactor pressure vessel and reactor vessel support structure. The use of the cavity measurement program coupled with available surveillance capsule measurements provides a plant specific data base that enables the evaluation of the vessel neutron exposure and the uncertainty associated with that exposure over the service life of the units.

The cavity neutron measurement program also establishes three-dimensional fluence profiles and enables the true effects of three-dimensional and potentially non-symmetric flux reduction measures to be accurately accounted for in a manner that would be difficult using analysis alone. All calculations and dosimetry evaluations are performed based on nuclear cross-section data derived from ENDF/B-VI. The calculational method used to obtain the maximum neutron exposure of the reactor vessel is identical to that for the Point Beach surveillance capsules.

To evaluate the RT_{NDT} shift of welds, heat affected zones, and base material for the vessel, test coupons of these material types have been included in the reactor vessel material surveillance program, which is described in Section 4.4.

MATERIALS OF CONSTRUCTION - COMPARISON TO USAS B31.7

In response to an Atomic Energy Commission question regarding the degree to which the reactor coolant system valves, fittings and piping met the requirements of the USAS B31.7 code, the following response was provided.

The valves, fittings, and piping are designed to the ASA B31.1 (1955) Code for Power Piping using the allowable stresses found in the Nuclear Code, Cases N-7 and N-10 for pipe and fittings, respectively. Nuclear piping, Class I, is defined as the Reactor Coolant System out to the second normally closed isolation valve. For those valves which are normally open, the system extends to the first valve outside containment capable of external actuation.

The quality assurance requirements of Westinghouse WAPD in the purchase and examination of the reactor coolant piping assured that the quality level of the Westinghouse plant is comparable to that delineated for USAS B31.7 1967 Edition nuclear piping, Class I, to the extent described below.

1. All materials for fabrication conform to ASTM specifications listed for Class I nuclear piping. In addition, all materials are certified and identified for conformance to governing ASTM requirements.

2. Piping base materials are examined by methods to quality acceptance criteria and to the extent that meets requirements described in USAS B31.7 for Class I nuclear piping.

3. All welding procedures, welders, and welding operators are qualified to the requirements of ASME IX, Welding Qualifications.

4. All welds are examined by NDE methods and to the extent prescribed in USAS B31.7 for Class I nuclear piping.

5. All branch connection nozzle welds of nominal sizes 3 in. and larger are 100% radiographed. This exceeds USAS B31.7 requirements which requires radiographing nozzle welds of nominal sizes 6 in. and larger.

6. All finished welds are liquid penetrant examined on both the outside and inside (if accessible) surfaces as required by USAS B31.7, Class I.

7. Hydrostatic testing is performed on the erected and installed piping. This requirement is the same as in USAS B31.7, Class I.

A thermal expansion flexibility stress analysis is performed in accordance with the criteria set forth in USAS B31.1 to assure that the stress range and number of thermal cycles are safely within the limits prescribed in B31.1. In addition, seismic analyses are performed on the composite piping, including the combined stress effects of all steady-state (pressure and weight) loadings plus seismic vertical/horizontal loading components. The resultant reactions of the piping due to the separate and combined effects of thermal, sustained, and seismic loadings are factored into the checking of the final design of the equipment nozzles to which the piping is interconnected. In turn, the equipment supporting structures are checked for adequate design, including the added effects of these same loadings. Thus, the total design, including pipe, equipment, and structures include the effects of thermal expansion and sustained and seismic loadings.

Thermally induced stresses arising from temperature gradients are limited to a safe and low order of magnitude in assigning a maximum permissible time rate of temperature change on plant heatup, cooldown, and incremental loadings. Thermal sleeves are utilized at nozzles wherein a cold fluid is introduced into a pipe conveying a significantly hotter fluid or vice-versa. Typical examples are the charging line, pressurizer surge, and residual heat return nozzle connections to the primary coolant loop piping.

Shop and field fabrication requirements, documentation, and quality assurance examinations all comply with those found in USAS B31.7 for Class I nuclear piping except that chemical and physical certifications are documented by pipe lot. The above criteria for Reactor Coolant System valves, fittings, and piping apply to the pressurizer surge line and the remainder of the piping between the 27.5 in., 29 in., and 31 in. pipe to the second isolation stop valve, with the following exceptions:

1. Pipe/fittings of nominal sizes 2 in. and smaller will not be subject to volumetric inspection of the base material.

2. A complete flexibility/seismic stress analysis is not necessarily performed on all of the branch piping to the extent performed on the 27.5 in. and larger primary loop piping.

Piping Code Class I pipe and fittings in the balance of plant conform to USAS B31.1 Code - 1967 Edition.

MAXIMUM HEATING AND COOLING RATES

The reactor system operating cycles used for design purposes are given in Table 4.1-8 and described in Section 4.1. The maximum allowable normal system heatup and cooldown rate is 100°F/hr. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate,

starting with a minimum water level, of 55°F/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant. The fastest cooldown rates which result from the hypothetical case of a break of a main steam line are discussed in Section 14.2.5.

WATER CHEMISTRY

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of reactor coolant system surfaces. All of the materials exposed to reactor coolant are corrosion resistant. Periodic analyses of the coolant chemical composition are performed to monitor the adherence of the system to the reactor coolant water quality as stated in EPRI PWR Primary Water Chemistry Guidelines (Reference 15). Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and Sampling System which are described in Section 9.0.

REACTOR COOLANT FLOW MEASUREMENTS

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

$$\frac{WP}{WP_0} = \left(\frac{1}{1_0}\right)^2$$

where:

WP_0 = the referenced pressure differential with the corresponding referenced flow rate 1_0

WP = the pressure differential with the corresponding referenced flow rate 1

The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within ± 10% and field results have shown the repeatability of the trip point to be within ± 1%. The analysis of the loss-of-flow transient presented in Section 14.1.8 assumes instrumentation error of ± 3%.

RCS GAS VENT SYSTEM

The RCS Gas Vent System is designed to permit the operator to vent non-condensable gases from the reactor vessel head and/or pressurizer steam space remotely from the control room during post-accident situations when large quantities of non-condensable gases may collect. The purpose of venting is to prevent possible interference from accumulated gases with core cooling. Small amounts of gas can be vented to the pressurizer relief tank (PRT) and thus not enter the containment atmosphere. Use of the PRT provides a discharge location which can be used to store small quantities of gas without influencing containment hydrogen concentration levels. Larger volumes will require venting directly to the containment.

The vent path from either the pressurizer or reactor vessel head is single active failure proof with regards to either establishing or isolating a flow path. Parallel valves powered from independent 125 V DC emergency power supplies are provided at both vent sources to ensure a vent path exists to a common header in the event of a single failure of either a valve or a power source. Vent paths from the common header to the PRT and from the common header to the containment atmosphere are provided by separate solenoid valves powered from independent 125 V DC emergency power supplies. All solenoid valves close upon de-energization. The venting rate from either source is controlled by an in-line flow-restricting orifice which limits the flow so that, in the event of a pipe break or isolation valve failure, makeup water for the leakage can be provided by a single charging pump. Covers are installed over the solenoid valve switches to minimize the possibility of inadvertent operation. Open and Closed valve position indication lights are provided in the control room. Pressure instrumentation is used to monitor the system for leakage during normal plant operation. A flow diagram of the system is shown in Figure 4.2-1 (Unit 1) and Figure 4.2-1A (Unit 2). Vent path operability and system testing requirements are discussed in TRM 3.4.4, "Reactor Coolant Gas Vent System."

The design parameters for the Reactor Coolant Gas Vent System are listed below:

Flow	-	> 100 scfm H2, dependent upon RCS pressure and temperature
Temperature	-	700°F
Pressure	-	2500 psia
Line	-	1 inch
Orifice Size	-	7/32 inch

The NRC determined the RCS Gas Vent System design to be acceptable and in conformance to the requirements of 10 CFR 50.44 (c)(3)(iii) and the guidelines of NUREG-0737 Item II.B.1 and NUREG-0800 Section 5.4.12 (Reference 12 and Reference 13). The RCS gas vent requirements of 10 CFR 50.44 (c)(3)(iii) were subsequently revised and relocated to 10 CFR 50.46a.

In addition to its primary, post-accident function, the system may be used to aid in the draining or fill and venting of the reactor coolant system. The system can also be used to reduce primary pressure at hot shutdown allowing boration of the RCS using high head safety injection pumps. Large flow rates can be achieved by opening two normally closed, series connected, one-inch manual valves which bypass the orifice.

REACTOR VESSEL LEVEL INDICATION SYSTEM (RVLIS)

Four channels of reactor vessel level indication (two wide range, two narrow range) were installed by modification, to provide core level indication for all reactor coolant pump combinations, whether operating or secured. (MR IC-244)

RESISTANCE TEMPERATURE DETECTOR BYPASS LOOPS

See Section 7.2.3.2 for a description of the resistance temperature detector bypass loops.

THERMAL RELIEF PROTECTION

All reactor coolant system piping inside containment which is isolated as a result of normal operating alignment, or which could become isolated as a result of automatic action from a containment isolation signal (including in-series containment isolation valves) are protected from the thermal expansion effect of accident conditions by thermal relief valves. (MR 97-132, MR 97-102).

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8. K. K. Yoon, “Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C & D Service Loads,” BAW-2178PA, April 1994.

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12. NUREG-0800, Standard Review Plan Section 5.4.12, "Reactor Coolant System High Point Vents," dated July 1981.

13. NRC Safety Evaluation, Point Beach Nuclear Plant Units 1 and 2, Wisconsin Electric Power Company, dated September 22. 1983.

14. NRC Safety Evaluation dated May 3, 2011, "Issuance of License Amendment Regarding Extended Power Uprate (TAC Nos. ME1044 and ME 1045)."

15. EPRI 1014986, "Pressurized Water Reactor Primary Water Chemistry Guidelines."

Table 4.2-1 MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS

Sheet 1 of 2

Component	Section	Materials
Reactor Vessel	Shell Plate (Unit 1)	SA 302, Gr. B
	Shell Forging (Unit 2)	A 508 Class II
	Nozzle Shell & Nozzle Forgings	A 508 Class II
	Cladding, Stainless Weld Rod	Type 304 Equivalent
	Thermal Shield and Internals	A 240, Type 304
	Insulation	SS SS Foil SS
	Closure Head	SA 508 Grade 3 Class 1
Steam Generators, Unit 1	Upper Shell Barrel	SA 302, Gr. B
	Lower Shell Barrels	SA-533 Gr A, CL. 2
	Channel Head Casting	SA-216 WCC
	Channel Head Cladding Weld Rod	SFA-5.9, CL. ER 308L and 309L
	Tube Sheet Forging	SA-508, CL. 2A
	Cladding for Tubesheet (Primary Side)	NiCrFe Alloy
	Tubes	SB-163, Alloy 600 TT
	Primary Nozzle Safe-Ends	Type 308L Weld Buildup
Steam Generators, Unit 2	Upper and Lower Shell Barrels	SA-533 Type B, CL. 2
	Channel Head Forging	SA-508, CL. 3
	Channel Head Cladding Weld Rod	SFA-5.4 CL. E308L and E309L
	Tube Sheet Forging	SA-508, CL. 3A
	Cladding for Tubesheet (Primary Side)	NiCrFe Alloy
	Tubes	SB-163, Alloy 690 TT
	Primary Nozzle Safe-Ends	SA-336, CL. F316LN
Pressurizer	Shell	SA 302, Gr. B
	Heads	SA 216 WCC
	External Plate	SA 302, Gr. B
	Cladding, Stainless	Type 304 equivalent
	Internal Plate	SA 240 Type 304
	Internal Piping	SA 376 Type 316

'able 4.2-1 MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS

Sheet 2 of 2

Component	Section	Material
Pressurizer Relief Tank	Shell	A 285 Gr. C
	Heads	A 285 Gr. C
Piping	Pipes	A 376 Type 316
	Fittings	A 351, CF8M
	Nozzles	A 182 F316
Pump	Shaft	Type 304
	Impeller	A 351, CF8
	Casing	A 351, CF8M
Valves	Pressure Containing Parts	A 351, CF8M and A 182 F316

Figure 4.2-1 UNIT 1 REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM (Sheet 1)

[redacted]

Figure 4.2-1 UNIT 1 REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM (Sheet 2)

[redacted]

Figure 4.2-1 UNIT 1 REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM (Sheet 3)

[redacted]

Figure 4.2-1A UNIT 2 REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM (Sheet 1)

[redacted]

Figure 4.2-1A UNIT 2 REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM (Sheet 2)

[redacted]

Figure 4.2-1A UNIT 2 REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM (Sheet 3)

[redacted]

Figure 4.2-2 REACTOR VESSEL SCHEMATIC

[redacted]

Figure 4.2-3 PRESSURIZER

[redacted]

Figure 4.2-4 UNIT 1 STEAM GENERATOR

[redacted]

Figure 4.2-5 UNIT 2 STEAM GENERATOR

[redacted]

Figure 4.2-6 REACTOR COOLANT PUMP

[redacted]

U

Figure 4.2-7 REACTOR COOLANT PUMP ESTIMATED PERFORMANCE CHARACTERISTICS

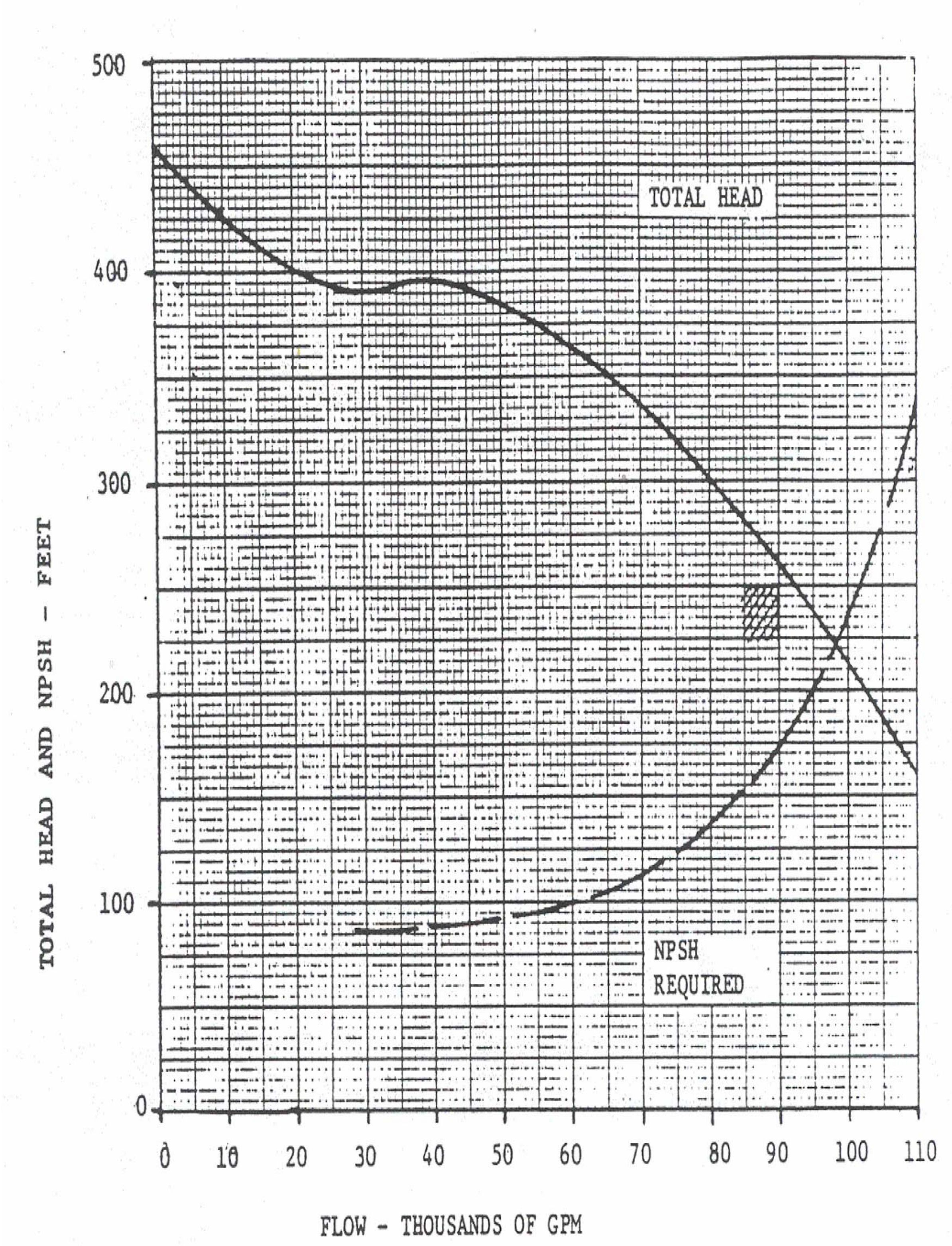
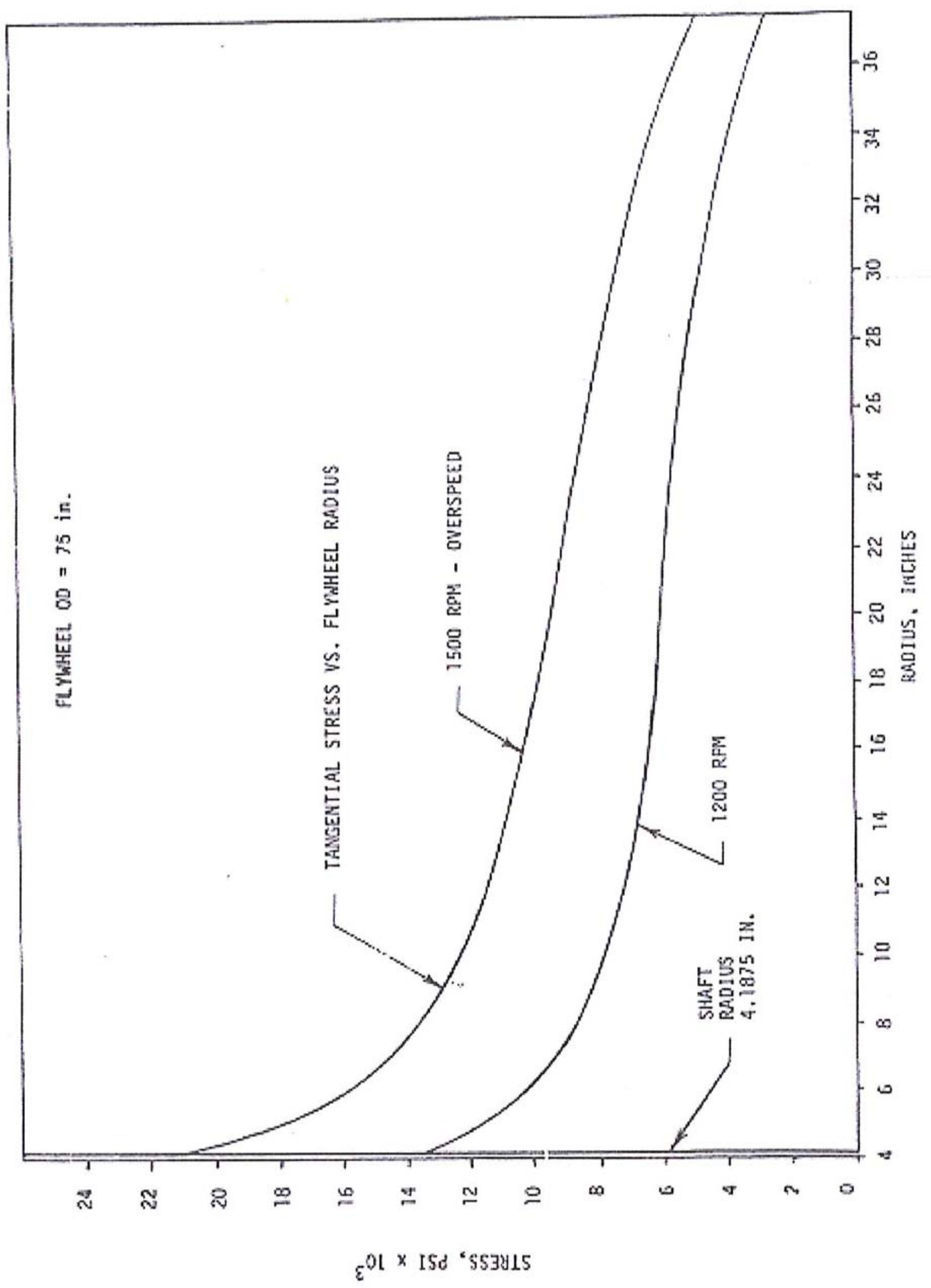


Figure 4.2-8 REACTOR COOLANT PUMP FLYWHEEL

[redacted]

Figure 4.2-9 FLYWHEEL STRESS



4.3 SYSTEM DESIGN EVALUATION

SAFETY FACTORS

The safety of the reactor vessel and all other Reactor Coolant System pressure containing components and piping is dependent on several major factors including design and stress analysis, material selection and fabrication, quality control, and operations control.

Reactor Vessel

The reactor vessel has a 132 in. ID and is within size limits for which good experience exists. A stress evaluation of the reactor vessel has been carried out in accordance with the rules of the applicable Edition of Section III of the ASME Code. The evaluation demonstrates that stress levels are within the stress limits of the Code. Table 4.3-1 presents a summary of the results of the stress evaluation. A summary of fatigue usage factors for components of the reactor vessel is given in Table 4.3-2.

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 such as Yankee Rowe. These cycles include five heatup and cooldown cycles per year, a conservative selection when the vessel may not complete more than one cycle per year during normal operation.

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

Appendix G to 10 CFR 50 establishes requirements for the fracture toughness of the reactor vessel pressure boundary which provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences, to which the pressure boundary may be subjected over its service lifetime. Section IV.A.2 of Appendix G requires that the reactor vessel be operated with pressure temperature limits at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of ASME Code Section XI.

See Section 15.4 for the discussion of the fracture toughness methodology evaluation reviewed and approved by the NRC for License Renewal for Unit 2. (NRC SE dated 12/2005, NUREG-1839)

Appendix G of ASME Code Section XI requires that pressure temperature limits be calculated: (a) using a safety factor of two on the principal membrane (pressure) stresses; (b) assuming a flaw at the surface with a depth of one quarter of the vessel wall thickness and a length of six times its depth; (c) using a conservative fracture toughness curve that is based on the lower bound of static, dynamic, and crack arrest fracture toughness tests on material similar to the Point Beach reactor vessel material; and (d) applying a 2 sigma margin in the determination the adjusted reference temperature (RT_{NDT}). The irradiation induced shift in RT_{NDT} is determined using the guidance of Regulatory Guide 1.99, Rev. 2 (Radiation Embrittlement of Reactor Vessel Materials) which is a conservative measure of material embrittlement.

Limits on the reactor coolant system pressure with respect to temperature during plant heatup, cooldown, and normal operation are determined in accordance with the methods of analysis and the margins of safety of Appendix G of the ASME Code Section XI and are included in the Point Beach Pressure Temperature Limits Report (PTLR).

The vessel closure contains 48 six inch studs. The stud material is ASTM A 540 and Code Case 1335.2, which has a minimum yield strength of 104,000 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is approximately 37,500 psi.

Steam Generators

Calculations confirm that the steam generator tubesheet will withstand the loading (which is a quasi static rather than a shock loading) by loss of reactor coolant. The maximum primary membrane plus primary bending stress in the tubesheet under these conditions is 23,600 psi. This is well below ASME Section III yield strength of 41,400 psi at 650°F. Because the pressure in the primary channel head would drop to zero under the condition postulated, no damage will result to the tubesheet.

The rupture of primary or secondary piping has been assumed to impose a maximum pressure differential of 2250 psi across the tubes and tubesheet from the primary side or a maximum pressure differential of 1100 psi across the tubes and tubesheet from the secondary side, respectively. A criterion is established from these conditions under which there is no rupture of the primary-to-secondary boundary (tubes and tubesheet). This criterion prevents any violation of the containment boundary.

To meet this criterion, it has been established that, under the postulated accident conditions where a primary-to-secondary side differential pressure of 2250 psi exists, the primary membrane stresses in the tubesheet ligaments, averaged across the ligament and through the tubesheet thickness, do not exceed 90% of the material yield stress at the operating temperature. Furthermore, the primary membrane plus primary bending stress in the tubesheet ligaments, averaged across the ligament width at the tubesheet surface location giving maximum stress, do not exceed 135% of the material yield stress at the operating temperature. This criterion is felt to be applicable to abnormal operating circumstances in that it is consistent with the ASME, Nuclear Pressure Vessel Code, Section III rules, Paragraph N 714, 2 for hydrotest limitations. An examination of stresses under these conditions shows that for the case of a 2485 psi maximum tubesheet pressure differential, the stresses are within acceptable limits. These stresses, together with the corresponding stress limits, are given in Table 4.3-3.

The tubes have been designed to the requirements (including stress limitations) of Section III for normal operation, assuming 2485 psi 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. In the case of a primary pressure loss accident, the secondary-to-primary pressure differential can reach 1100 psi. This pressure differential is less than the primary-to-secondary design pressure differential (1700 psi) for normal operating conditions. Hence, no stresses in excess of those covered in Section III rules for normal operation are experienced on the tubesheet for this accident case.

ASME Section VIII design curves for iron chromium nickel steel cylinders under external pressure indicate a collapse pressure of 2310 psi for tubes having the minimum properties required by the ASTM specification. This indicates a minimum factor of safety of 2.4 against collapse. Collapse tests of 7/8 0.050 wall straight tubes at room temperature indicate actual tube strengths are significantly higher than specification and a collapse pressure of 6,000 psi was recorded for the straight tube. The difference is attributed to the fact that the yield strength of the tube tested was 44,000 psi and the Code charts are based on a yield strength of approximately 29,000 psi at room temperature.

Consideration has been given to the superimposed effects of secondary side pressure loss and the maximum potential earthquake loading. The fluid dynamic forces on the internal components affecting the primary-to-secondary boundary (tubes) has been considered as well. For this condition, the criterion is that no rupture of primary-to-secondary boundary (tubes and tubesheet) occurs.

For the case of the tubesheet, the maximum hypothetical earthquake loading will contribute an equivalent static pressure loading over the tubesheet of less than 10 psi (for vertical shock). Such an increase is small when compared to the pressure differentials (up to 2485 psi) for which the tubesheet is designed. Under horizontal shock loading of the maximum hypothetical earthquake, the stresses are less than those for 1.0 g gravity loading experienced in a horizontal position, which the design can readily accept.

The fluid dynamic forces on the internals under secondary steam break accident conditions indicate, in the more severe case, that the tubes are adequate to constrain the motion of the baffle plates with some plastic deformation, but boundary integrity is maintained. The ratios of the allowable stresses (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses, are summarized in Table 4.3-4.

RELIANCE ON INTERCONNECTED SYSTEMS

The principal heat removal systems which are interconnected with the Reactor Coolant System are the steam and feedwater systems and the safety injection and residual heat removal systems. The Reactor Coolant System is dependent upon the steam generators and the steam, feedwater, and condensate systems for decay heat removal from normal operating conditions to a reactor coolant temperature of approximately 350°F. The layout of the system ensures the natural circulation capability to permit plant cooldown following a loss of both reactor coolant pumps.

The flow diagram of the Steam and Power Conversion System is shown in Figure 10.1-1 through Figure 10.1-4A. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the feedwater system may be pumped into the steam generators and the resultant steam vented to the atmosphere. The auxiliary feedwater system (AF) will supply water to the steam generators in the event that the main feedwater pumps are inoperative. The system is described in Section 10.0. The Safety Injection System is described in Section 6.0. The Residual Heat Removal System is described in Section 10.0.

SYSTEM INTEGRITY

A complete stress analysis which reflects consideration of all design loadings detailed in the design specification has been prepared by the manufacturer. The analysis shows that the reactor vessel, steam generator, pump casing, and pressurizer comply with the stress limits of Section III

of the ASME Code. A similar analysis of the piping shows that it complies with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V notch toughness test curves were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator, and pressurizer to provide assurance for hydrostatic testing and initial operation in the ductile region. In addition, drop weight tests were performed on the reactor vessel plate material. Following initial plant operation, additional testing of reactor vessel materials is performed as part of the reactor vessel surveillance program to obtain information on the effects of neutron irradiation embrittlement of reactor vessel materials under operating conditions. This program is described in Section. 4.4.

As an assurance of system integrity, all components in the system were hydrostatically tested at 3110 psig prior to initial operation. In addition, to assure primary system integrity, the system is leak tested at normal operating pressure following each refueling outage, as required by ASME Section XI.

PRESSURE RELIEF

The Reactor Coolant System is protected against overpressure by safety valves located on the top of the pressurizer. The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with the applicable Edition of Section III of the ASME Boiler and Pressure Vessel Code. The capacity of the pressurizer safety valves is determined from considerations of; (1) the reactor protective system, and (2) 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 valves' setpoints.

SYSTEM INCIDENT POTENTIAL

The potential of the Reactor Coolant System as a cause of accidents is evaluated by investigating the consequences of certain credible types of components and control failures as discussed in Section 14.1.1 and Section. 14.2. Reactor coolant pipe rupture is evaluated in Section. 14.3.

REFERENCES

1. NRC Safety Evaluation dated May 3, 2011, "Issuance of License Amendments Regarding Extended Power Uprate (TAC Nos. ME1044 and ME1045)."

2. Westinghouse Calculation CN-MRCDA-08-43, Revision 1, "Reactor Vessel Evaluation for Point Beach Units 1 and 2 17 Percent Power Extended Power Uprate Program," dated April 2, 2009.

3. WCAP-16983-P, Revision 0, "Point Beach Units 1 and 2 Extended Power Uprate (EPU) Engineering Report," (Proprietary) dated September 2009.

Table 4.3-1 SUMMARY OF PRIMARY PLUS SECONDARY STRESS INTENSITY FOR COMPONENTS OF THE REACTOR VESSEL

Area	Stress Intensity (psi)	Allowable Stress 3 Sm (psi)
CRDM Nozzle	45,300	60,000
Closure Head at Flange	69,200	80,100
Vessel at Flange	71,100	80,100
Closure Studs	117,600	118,800
Primary Nozzles	48,800[a]	80,100
External Support Brackets	41,200	80,100
Core Support Pad	57,500	69,900
Bottom Head to Shell Juncture	28,600	80,100
Bottom Instrumentation	57,800	69,900
Safety Injection Nozzle	46,800	80,100
Vent Nozzle	53,600	60,000
Vessel Wall Transition	32, 200	80,100
Instrumentation Port Head Adapter for Core Exit Thermocouple Nozzle Assembly	25,600	50,100

(NRC SE dated 12/2005, NUREG-1839)

a. Limiting value considering both the inlet and outlet nozzles.

Table 4.3-2 SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR COMPONENTS OF THE REACTOR VESSEL

Item	Usage Factor*[a]
CRDM Nozzle	0.672
Closure Head at Flange	0.248
Vessel at Flange	0.992
Closure Studs	0.991
Primary Nozzles	0.155[b]
External Support Brackets	0.842
Core Support Pad	0.960
Bottom Head to Shell Juncture	0.004
Bottom Instrumentation	0.384
Safety Injection Nozzle	0.465
Vent Nozzle	0.023
Vessel Wall Transition	0.006
Instrumentation Port Head Adapter for Core Exit Thermocouple Nozzle Assembly	0.029

(NRC SE dated 12/2005, NUREG-1839)

* Covers all transients

a As defined in the applicable Edition of Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels

b Limiting value considering both the inlet and outlet nozzles.

Table 4.3-3 STRESSES DUE TO MAXIMUM STEAM GENERATOR TUBESHEET PRESSURE DIFFERENTIAL (2485 PSI)

Stress	Computed Value	(668° F) Allowable Value
Primary Membrane Stress	23,300 psi	37,000 psi (0.9 Sy)
Primary Membrane plus Primary Bending Stress	53,000 psi	55,600 psi (1.35 Sy)

In addition to the foregoing evaluation, elasto plastic limit analysis of the tubesheet head shell combination indicates a limit pressure of 3400 psi at operating conditions, giving a safety factor of 1.36 for the abnormal condition.

Table 4.3-4 RATIO OF ALLOWABLE STRESSES TO COMPUTED STRESSES FOR A STEAM GENERATOR TUBESHEET PRESSURE DIFFERENTIAL OF 2485 PSI

Component Part	Stress Ratio
Channel Head	1.35
Channel Head Tubesheet Joint	1.63
Tubes	1.20
Tubesheet	
Maximum Average Ligament	1.04
Effective Ligament	1.58

4.4 TESTS AND INSPECTIONS

REACTOR COOLANT SYSTEM INSPECTION

Nondestructive Inspection of Material and Components Prior to Operation

Table 4.4-1 summarizes the nondestructive examination program for all Reactor Coolant System components. In this table, all of the nondestructive examinations which were required by the Westinghouse specifications on Reactor Coolant System components and materials are specified for each component. All examinations required at the time of manufacture and installation by the applicable codes are included in this table. Westinghouse requirements, which were more stringent in some areas than those requirements specified in the applicable codes, are also included.

Westinghouse required, as part of its reactor vessel specification, that certain special tests which are not specified by the applicable codes be performed. These tests are listed below:

1. Ultrasonic Testing - Westinghouse required that a 100% volumetric ultrasonic test of reactor vessel plate by both shear wave and longitudinal wave be performed. Section III Class A vessel plates are required by code to receive only a longitudinal wave ultrasonic test on a 9 in. x 9 in. grid. The 100% volumetric ultrasonic test is a severe requirement, but it assured that the plate used for Westinghouse reactor vessels is of the highest quality.

2. Material Surveillance Program - The beltline region of the reactor pressure vessel is the most critical region because it is subjected to significant neutron irradiation. The overall effects of neutron irradiation on the mechanical properties of low alloy ferritic materials is known as neutron embrittlement and encompasses an increase in hardness and tensile properties and a decrease in ductility and toughness with cumulative neutron irradiation.

A reactor pressure vessel surveillance program in accordance with the requirements of 10 CFR Part 50, Appendix H (Reactor Vessel Material Surveillance Program Requirements) and ASTM E 185-82 (Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels) has been implemented for the Point Beach Nuclear Plant to obtain information on the effects of irradiation on the reactor pressure vessel material under operating conditions. The program consists of periodically testing irradiated reactor vessel material specimens at intervals defined in E 185-82 and comparing the data with pre-irradiation data to establish the shift in RT_{NDT}. This information may be used in the development of reactor coolant system pressure temperature limits and to demonstrate compliance with 10 CFR 50.60 (Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation) and 50.61 (Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events).

See Section 15 for the discussion of the fracture toughness methodology evaluation reviewed and approved by the NRC for License Renewal for Unit 2 (NRC SE dated 12/2005, NUREG 1839).

Six material surveillance capsules were located in the reactor vessel between the thermal shield and the vessel wall prior to initial startup. The capsules contain Charpy V-notch impact specimens, tensile specimens, Wedge Opening Loading (WOL) specimens from the shell plate or ring forgings of the reactor vessel and representative weld metal, and Charpy V-notch impact specimens of heat affected zone (HAZ) metal and the ASTM correlation monitor material. Dosimeters to measure the integrated neutron flux (fluence) and thermal monitors to measure temperature are also included in each of the six material test capsules. The removal schedules for the Unit 1 and 2 reactor vessel surveillance capsules are contained in TRM 2.2, Pressure Temperature Limits Report.

Pre-irradiation tests consisted of Charpy V-notch impact tests on the vessel shell plate or ring forgings, weld materials, HAZ metal, and on the correlation monitor material, and tensile tests performed on the vessel shell plate or ring forging and weld metal. The data established the nil ductility transition temperature, NDTT, for the materials. As a supplement to the plant specific material surveillance program for Point Beach, additional surveillance data is available through participation in the Babcock & Wilcox Owners Group Master Integrated Reactor Vessel Surveillance Program. This integrated program includes weld metal heats used in the construction of the Point Beach reactor vessels that are not included in the plant specific surveillance program for Point Beach.

Following establishment of the pre-irradiation mechanical properties of the subject materials, the ASME Boiler and Pressure Vessel Code adopted new fracture toughness requirements for ferritic components of nuclear reactor systems. The new Code provisions utilize fracture mechanics concepts as a method of analysis to prevent brittle fracture in reactor pressure vessels.

The method of fracture mechanics is based on the RT_{NDT} (reference nil-ductility temperature), which is defined as the greater of the drop weight nil ductility transition temperature (NDTT per ASTM E-208) or the temperature, which is 60 F less than the 50 ft-lb (and 35 mils lateral expansion) temperature as determined from Charpy specimens oriented normal to the rolling direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) as presented in Appendix G of ASME Boiler and Pressure Vessel Code Section XI. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits are then determined utilizing the allowable stress intensity factors and methodology of ASME Appendix G.

RT_{NDT}, and thus the operating limits of Point Beach Nuclear Plant, are adjusted to account for the effects of radiation on the reactor vessel material properties through the information provided by the reactor pressure vessel surveillance program or by utilizing embrittlement trend correlations prepared by the NRC or others. Details of the development and use of the surveillance program are found in WCAP-9513, June 1978;

WCAP-7712, June 1971; WCAP-7924, July 1972; WCAP-8738, and WCAP-8743, January 1977.

Non-Destructive Examination of Materials

Table 4.4-1 summarizes the non destructive examinations performed on primary system components. In addition to the inspections shown in Table 4.4-1, there are those which the equipment supplier performs to confirm the adequacy of material received, and those performed by the material manufacturer in producing the basic material. The examinations of the reactor vessel, pressurizer, and steam generator are governed by ASME Code requirements. The examination procedures and acceptance standards required on pipe materials and piping fabrication are governed by USAS B31.1 and Westinghouse requirements and are equivalent to those performed on ASME Code vessels.

Procedures for performing the examinations are consistent with those established in the ASME Code, Section III and were reviewed by qualified Westinghouse 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 is fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming, and fabricating processes, received a 100% surface inspection by magnetic particle or liquid penetrant testing after all these operations are completed. All reactor coolant plate materials are also subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. All forgings receive the same inspection. In addition, 100% of the material volume is covered in these tests as an added assurance over the grid basis required in the Code.

Westinghouse quality control engineers and Wisconsin Electric's 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.

Field erection and field welding of the Reactor Coolant System were performed such as to permit exact fit up of the 31 in. ID closure pipe subassemblies between the steam generator and the reactor coolant pump. After installation of the pump casing and the steam generator, measurements were taken of the pipe length required to close the loop. Based on these measurements, the 31 in. ID closure pipe subassembly was properly machined and then erected and field welded to the pump suction nozzle and to the steam generator exit nozzle.

Cleaning of RCS piping and equipment was accomplished before and/or during erection of various equipment. Stainless steel piping was cleaned in sections as specific portions of the systems were erected. Pipe and units large enough to permit entry by personnel were cleaned by locally applying approved solvents (Stoddard solvent, acetone, and alcohol) and demineralized water, and by using a rotary disc sander or 18-8 wire brush to remove

all trapped foreign particles. Standards for final physical and chemical cleanliness are defined in Section 13.

Equipment specifications for fabrication required that suppliers submit the manufacturing procedures (welding, heat treating, etc.) to Westinghouse where they were reviewed by qualified Westinghouse engineers. This also was done on the field fabrication procedures to assure that installation welds were of equal quality.

Section III of the ASME Boiler and Pressure Vessel Code required that nozzles carrying significant external loads be attached to the shell by full penetration welds. This requirement has been carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop were made using full penetration welds.

The Reactor Coolant System components were welded under procedures which require the use of both preheat and post heat. Preheat requirements, nonmandatory under Code rules, were performed on all weldments, including P1 and P3 materials, which were the materials of construction in the reactor vessel, pressurizer, and steam generators. Preheat and post heat of weldments both serve 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, post heating achieves this by tempering any hard zones which may have formed due to rapid cooling. Thus, the Reactor Coolant System components were welded under procedures which require the use of both preheat and post-heat.

Inservice Inspection

During the design phase of the Reactor Coolant System, careful consideration was given to provide access for both visual and nondestructive inservice inspection of primary loop components. If necessary, the following components and areas can be made available for 100% visual and 100% nondestructive inspection (except as noted):

1. Reactor Vessel - The entire inside surface
2. Reactor Vessel Nozzles - The entire inside surface
3. Closure Head - The entire inside and outside surface
4. Reactor Vessel Studs, Nuts, and Washers
5. Field Welds between the Reactor Vessel, Steam Generators, and Reactor Coolant Pumps and the Reactor Coolant Piping
6. Reactor Internals
7. Reactor Vessel Flange Seal Surface
8. Fuel Assemblies (External visual only)
9. Rod Cluster Control Assemblies

10. Control Rod Drive Shafts

11. Control Rod Drive Mechanism Assemblies

12. Reactor Coolant Pipe External Surfaces (except for the five foot penetration of the primary shield)

13. Steam Generator   The external surface, the internal surfaces of the Steam Drum, and the Channel Head

14. Pressurizer - The Internal and External Surfaces

15. Reactor Coolant Pump - The External Surfaces, Motor, Impeller, and Flywheel

The design considerations which have been incorporated into the primary system design to permit the above inspections are as follows:

1. All reactor internals are completely removable. The tools and storage space required to permit these inspections are provided.

2. The closure head is stored dry on an operating deck during refueling to facilitate direct visual inspection.

3. All reactor vessel studs, nuts, and washers are removed to dry storage during refueling.

4. Removable plugs are provided in the primary shield just above the coolant nozzles, and the insulation covering the nozzle welds is readily removable.

5. Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.

6. A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.

7. The storage stands provided for storage of the internals allow for inspection access to both the inside and outside of the structures.

8. The station provided for change out of control rod clusters from one fuel assembly to another is specially designed to allow inspection of both fuel assemblies and control rod clusters. The control rod mechanism is specially designed to allow removal of the mechanism assembly from the reactor vessel head.

9. Manways are provided in the steam generator steam drum and channel head to allow access for internal inspection.

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

11. All insulation on primary system components (except the reactor vessel) and piping (except for the penetration in the primary shield) is removable.

The metal reflective insulation on the closure head may be removed as desired to perform inspection.

The use of conventional nondestructive, direct visual, and remote visual examination techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures.

1. Shop ultrasonic examinations were performed on all internally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed is 3/4 inch.

2. The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit positioning of test equipment without obstruction.

3. Reactor Vessel Postoperational Ultrasonic Testing - Following hydrostatic testing of the vessel, selected areas of the reactor vessel were ultrasonic tested and mapped to facilitate the inservice inspection program. The area selected for ultrasonic testing mapping included:

   a. Vessel flange radius, including the vessel flange to upper shell weld
   b. Middle shell course
   c. Lower shell course above the radial core supports
   d. Nozzle to upper shell weld
   e. Middle shell to lower shell weld
   f. Upper shell to middle shell weld

Various tests have been conducted to determine the effect of cladding surface finish on ultrasonic inspectability of vessel material.

Detailed procedures for inservice inspection are specified in the PBNP Inservice Inspection Program, including the use of visual inspections, ultrasonic, magnetic particle, and dye penetrant testing of selected parts during refueling periods.

The internal surface of the reactor vessel is inspected periodically using optical devices over the accessible areas. During refueling, the vessel cladding can be inspected in certain areas between the closure flange and the primary coolant inlet nozzles. If deemed necessary by this inspection, the core barrel could be removed, making the entire inside vessel surface accessible. The reactor vessel welds are periodically examined by means of ultrasonic testing. In order to facilitate this test program, critical areas of the reactor vessel were mapped during the fabrication phase to serve as a reference base for subsequent ultrasonic tests.

Externally, the control rod drive mechanism nozzles on the closure head, the instrument nozzles on the bottom of the vessel, and the extension spool pieces on the primary coolant outlet nozzles are accessible for visual, magnetic particle, or dye penetrant inspection during refuelings.

The closure head is examined visually during each refueling. Optical devices permit a selective visual inspection of the cladding, control rod drive mechanism nozzles, and the gasket seating surface. The knuckle transition piece, which is the area of highest stress of the closure head, also is accessible on the outer surface for inspection by visual and dye penetrant means.

The closure studs are inspected periodically using either magnetic particle tests and/or ultrasonic tests. Additionally, it is possible to perform strain tests during the tensioning, which assists in verifying the material properties.

These areas are subjected to periodic inservice inspection. A complete program dealing with the frequency of inspection and the methods for such inspections is defined in the PBNP Inservice Inspection Program.

The preservice inspection of the Reactor Coolant System, which established a base line for later inservice inspection, included all the initial tests necessary to evaluate the inservice inspection program. The preservice and initial inservice inspection programs were based on the October 1968 Draft ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems (N-45). Several differences exist between the base line inspections and those outlined in the October 1969 Draft ASME Code. N-45 calls for the preparation of specific patches on the cladding surface of the reactor vessel, pressurizer, and steam generator primary head. No specific patches were prepared, but a complete base line visual and surface inspection was performed on all cladding and a general visual inservice inspection was made of all accessible areas of cladding; not limited to specific patches. The inner radii of integrally cast nozzles of the pressurizer were not subjected to baseline volumetric inspection. These areas require extremely high personnel radiation exposure to perform inservice inspection, due to difficulties of access, and the information gained would not justify this high personnel exposure. All primary system pipe welds are included in the quality assurance program outlined in Table 4.4-1 and received preservice volumetric inspection to verify weld integrity, except no volumetric inspection of pressure containing welds in piping 2 in. and smaller were performed. A pipe break 2 in. or smaller in size is well within the capability of the safety injection system and will not cause core damage. All pressure containing welds in piping greater than 2 in. in size were included in the base line volumetric inspection. The integrally welded external support attachments to auxiliary piping are inspected. The geometry of the restraints precludes meaningful volumetric inspection.

The location of the reactor vessel biological shield makes several areas of the Reactor Coolant System pressure boundary inaccessible to inspection. Although the areas are inaccessible for inservice inspection, they have all received preservice volumetric inspection to insure weld integrity.

Examination of the primary pump flywheels may be conducted at approximately 20-year intervals. A qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels shall be performed. (Reference SER 2005-0008 dated June 6, 2005, and WCAP-15666)

The reactor vessel external supports have limited accessibility for inservice inspection. The bottom portion of the legs are visible from the keyway area, and the top of the support is visible when the sandbox covers around the RPV flange are opened and the plugs are removed.

Technical Specifications require that a program be established and implemented to ensure that steam generator tube integrity is maintained. The Steam Generator Program establishes performance criteria for structural integrity, accident induced leakage, and operational leakage. Meeting these performance criteria provides reasonable assurance of maintaining tube integrity during normal and accident conditions.

Table 4.4-1 REACTOR COOLANT SYSTEM NONDESTRUCTIVE EXAMINATION

(Sheet 1 of 3)

Component	Type of Examination*
1. Steam Generator	
1.1 Tubesheet	
1.1.1 Forging	UT(1), MT
1.1.2 Cladding	UT(1), PT(2)
1.2 Channel Head	
1.2.1 Casting	RT, MT
1.2.2 Cladding	PT
1.3 Secondary Shell and Head Plates	UT
1.4 Tubes	UT, ET
1.5 Nozzles (Forgings)	UT, MT
1.6 Weldments	
1.6.1 Shell, longitudinal	RT, MT
1.6.2 Shell, circumferential	RT, MT
1.6.3 Cladding (Channel Head Tubesheet joint cladding restoration	PT
1.6.4 Steam and Feedwater Nozzle to Shell	RT, MT
1.6.5 Support Brackets	MT
1.6.6 Tube to Tubesheet	PT
1.6.7 Instrument connections (primary and secondary)	MT
1.6.8 Temporary attachments after removal	MT
1.6.9 After hydrostatic test (all welds and complete channel head where accessible)	MT
1.6.10 Nozzle Safe Ends (if forgings)	RT, PT
1.6.11 Nozzle Safe Ends (if weld deposit)	PT
2. Pressurizer	
2.1 Heads	
2.1.1 Casting	RT, MT
2.1.2 Cladding	PT
2.2 Shell	
2.2.1 Plates	UT, MT
2.2.2 Cladding	PT
2.3 Heaters	
2.3.1 Tubing(3)	UT, PT
2.3.2 Centering of element	RT
2.4 Nozzle	UT, PT

Table 4.4-1 REACTOR COOLANT SYSTEM NONDESTRUCTIVE EXAMINATION

(Sheet 2 of 3)

Component	Type of Examination*
2. Pressurizer (continued)	
2.5 Weldments	
2.5.1 Shell, longitudinal	RT, MT
2.5.2 Shell, circumferential	RT, MT
2.5.3 Cladding	PT
2.5.4 Nozzle Safe End (if forging)	RT, PT
2.5.5 Nozzle Safe End (if weld deposit)	PT
2.5.6 Instrument Connections	PT
2.5.7 Support Skirt	PT
2.5.8 Temporary Attachments after removal	MT
2.5.9 All welds and cast heads after hydrostatic test	MT
2.6 Final Assembly	
2.6.1 All accessible surfaces after hydrostatic test	MT
3. Piping	
3.1 Fittings (Castings)	RT, PT
3.2 Fittings (Forgings)	UT, PT
3.3 Pipe	UT, PT
3.4 Weldments	
3.4.1 Circumferential	RT, PT
3.4.2 Nozzle to run pipe (No RT for nozzles less than 3 in.)	RT, PT
3.4.3 Instrument connections	PT
4. Pumps	
4.1 Castings	RT, PT
4.2 Forgings	PT
4.2.1 Main Shaft	UT, PT
4.2.2 Main Studs	UT, PT
4.2.3 Flywheel (Rolled Plate)	UT
4.3 Weldments	
4.3.1 Circumferential	RT, PT
4.3.2 Instrument Connections	PT
5. Reactor Vessel	
5.1 Forgings	
5.1.1 Flanges	UT, MT

Table 4.4-1 REACTOR COOLANT SYSTEM NONDESTRUCTIVE EXAMINATION

(Sheet 3 of 3)

Component	Type of Examination*
5. Reactor Vessel (continued)	
5.1 Forgings (continued)	
5.1.2 Studs	UT, MT
5.1.3 Head Adapters	UT, PT
5.1.4 Head Adapter Tube	UT, PT
5.1.5 Instrumentation Tube	UT, PT
5.1.6 Main Nozzles	UT, MT
5.1.7 Nozzle Safe Ends (If forging is employed)	UT, PT
5.2 Plates	UT, MT
5.3 Weldments	
5.3.1 Main Steam	RT, MT
5.3.2 CRD Head Adapter Connection	PT
5.3.3 Instrumentation Tube Connection	PT
5.3.4 Main Nozzles	RT, MT
5.3.5 Cladding	UT(4), PT
5.3.6 Nozzle Safe Ends (If forging)	RT, PT
5.3.7 Nozzle Safe Ends (If weld deposit)	RT, PT
5.3.8 Head adapter forging to head adapter tube	RT, PT
5.3.9 All welds after hydrotest	PT
6. Valves	
6.1 Castings	RT, PT
6.2 Forgings (No UT for valves 2 in. and smaller)	UT, PT

Notes:

(1) Flat surfaces only
(2) Weld deposit areas only
(3) Or a UT and ET
(4) UT of Clad bond to base metal

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