CHAPTER 4

REACTOR COOLANT SYSTEM

4.1 DESIGN BASIS

The Reactor Coolant System, shown in Plant Drawings 9321-F-27338 and -27473 [Formerly Figure 4.2-2A & -2B], consists of four similar 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.

4.1.1 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. Demineralized light water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance presented in Chapter 3. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

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

By appropriate selection of the inertia of the reactor coolant pumps, the thermal-hydraulic effects are reduced to a safe level during the pump coast-down which would result from a loss of flow situation. The layout of the system assures natural circulation capability following a loss of flow to permit decay heat removal without overheating the core. Part of the system's piping is used by the Safety Injection System to deliver cooling water to the core during a Loss-of-Coolant Accident.

4.1.2 General Design Criteria

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

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1976, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980,

and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

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 of 7/11/67)

The Reactor Coolant System is of primary importance in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication and inspection conformed to the applicable provisions of recognized codes and good nuclear practice (Section 4.1.7). Details of the quality assurance programs, test procedures and inspection acceptance levels are given in Section 4.3.1 and Section 4.5. Particular emphasis was placed on the quality assurance of the reactor vessel to obtain material whose properties were uniformly within tolerances appropriate to the application of the design methods of the code.

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 of 7/11/67)

All piping, components and supporting structures of the Reactor Coolant System were designed to seismic Class I requirements (SEE Chapter 16). They are capable of withstanding:

a) The Operational Basis Earthquake accelerations within code allowable working stresses.

b) The Design Basis Earthquake accelerations acting in the horizontal and vertical direction simultaneously with no loss of function.

The Reactor Coolant System is located in the Containment, whose design, in addition to being a seismic Class I structure, also considered accidents or other applicable natural phenomena with sufficient margin to compensate for the limits of accuracy on measurements, the quantity of data and the period of time in which historical data have been accumulated on the natural phenomena. Details of the containment design are given in Chapter 5.

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 of 7/11/67)

Records of the design of the major Reactor Coolant System components and the related engineered safety features components are maintained by the Authority and will be retained throughout the life of the plant as dictated by plant procedures.

Records of fabrication were maintained in the manufacturers' plants as required by the appropriate code. They will be available to the Authority throughout the life of the plant. Construction records are available at the site and/or in the office of the Authority 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 of 7/11/67)

The dynamic effects during blowdown following a Loss-of-Coolant Accident were evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Support structures were designed with consideration given to fluid and mechanical thrust loadings.

The steam generators are supported, guided and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines 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 line 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.

4.1.3 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 of 7/11/67)

The Reactor Coolant System in conjunction with its control and protective provisions was 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 retaining boundary of the Reactor Coolant System was carried out in strict accordance with the applicable codes. In addition, there were areas where equipment specifications for Reactor Coolant System components go beyond the applicable codes. Details are given in Section 4.5.1.

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

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

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

Isolatable 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 of 7/11/67)

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, and of runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units. This equipment provided 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 was the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, and condensate runoff. In addition, assuming no operator action, the liquid inventory in the process systems and containment sump can be used for gross indication of leakage.

However, sensitivity of the processing systems and containment sump system can be improved with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every 4 hours).

Further details are supplied in Section 4.2.7.

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 of 7/11/67)

The Reactor Coolant Pressure Boundary is capable of accommodating, without further rupture, the static and dynamic loads which would be imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Chapter 14.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since core depletion is primarily followed with boron dilution, only the rod cluster control assemblies in the controlled 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 assure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value which precludes any resultant damage to the primary system pressure boundary, i.e., gross fuel dispersion in the coolant and possible excessive pressure surges.

The failure of a rod mechanism housing causing a rod cluster to be rapidly ejected from the core was evaluated as a theoretical, thought not a credible accident. While limited fuel damage could result from this hypothetical event, the 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 adequately protected. (See Chapter 14)

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 failure. Consideration is given: (a) to the provision 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. (GC 34 of 7/11/67)

The Reactor Coolant Pressure Boundary was designed to reduce to an acceptable level the probability of a rapidly propagating type failure.

Assurance of adequate fracture toughness in the reactor vessel material was provided by compliance, insofar as possible, with the requirements for fracture toughness testing included in the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. In cases where it as not possible to perform all tests in accordance with these requirements, conservative estimates of material fracture toughness were made using information available at the time of design.

Assurance that the fracture toughness properties remain adequate throughout the service life of the plant is provided by a radiation surveillance program.

Safe operating heatup and cooldown limits are established according to Section III, ASME Boiler and Pressure Vessel Code, Appendix G 2000, Protection Against Nonductile Failure, issued in the Summer 1972 Addenda.

Changes in fracture toughness of the core region plates, weldments, and associated weld heat affected zone due to radiation damage are monitored by a surveillance program based on ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels(1). The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing by Charpy V-notch, dropweight test, and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading specimens carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals. Further details are given in Section 4.1.6.

All pressure containing components of the Reactor Coolant System were designed, fabricated, inspected and tested in conformance with the applicable codes. Further details are given in Section 4.1.7.

<u>Reactor Coolant Pressure Boundary Surveillance</u>

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 leak-tight 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 of 7/11/67)

The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces of the vessel and certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. 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. Further details are given in Section 4.5.

4.1.4 Design Characteristics

Design Pressure

The Reactor Coolant System design and operating pressures together with the safety, power 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 system relief valve characteristics. The design pressures and data for the respective system components are listed in Tables 4.1-2 through 4.1-6. Table 4.1-7 gives the design pressure drop of the system components.

Design Temperature

The design temperature for each component was selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. The design and operating temperatures of the respective system components are listed in Tables 4.1-2 through 4.1-6.

Seismic Loads

The seismic loading conditions were established by the “Operational Basis Earthquake” and “Design Basis Earthquake”. The former was selected to be typical of the largest probable ground motion based on the site seismic history. The latter was selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties.

For the “Operational Basis Earthquake” (OBE) loading condition, the Nuclear Steam Supply System was designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment needed for this purpose were required to operate within normal design limits. The seismic design for the “Design Basis Earthquake” (DBE) was intended to provide a margin in design that assured capability to shut down and maintain the nuclear facility in a safe condition. In this case, it was only necessary to ensure that the reactor coolant system components did not lose their capability to perform their safety function. This is referred to as the “no-loss-of-function” criteria and the loading condition as the “no-loss-of-function earthquake” loading condition.

The criteria adopted for allowable stresses and stress intensities in vessels and piping subjected to normal loads plus seismic loads are defined in Chapter 16. These criteria assure the integrity of the Reactor Coolant System under seismic loading.

For the combination of normal and Operational Basis Earthquake loadings, the stresses in the support structures were kept within the limits of the applicable codes.

For the combination of normal and Design Basis Earthquake loadings the stresses in the support structures were limited to values as necessary to assure their integrity and to maintain the stresses in the Reactor Coolant System components within the allowable limits as previously established.

4.1.5 Cyclic Loads

All components in the Reactor Coolant System were designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes and are not intended to be an accurate representation of actual transients or actual operating experience.

To provide the necessary high degree of integrity for the equipment in the Reactor Coolant System, the transient conditions selected for equipment fatigue evaluation were based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients. To a large extent, the specific transient operating conditions considered for equipment fatigue analyses were based upon engineering judgement and experience. Those transients were 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 operating life. For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients.

1. Heatup and Cooldown

For design evaluation, the heatup and cooldown cases were represented by continuous heatup or cooldown at a rate of 100°F per hour which corresponds to a heatup or cooldown rate under abnormal or emergency conditions. 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 usually attained because of the Over Pressure Protection System (OPS) discussed in Section 4.3 and other limitations such as:

a) Criteria for protection against non-ductible failure which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature.

b) Slower initial heatup rates when using pumping energy only.

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

The heatup and cooldown rates, administratively imposed by plant operating procedures, are limited to 50°F per hour for normal operation. Ideally, heatup and cooldown would occur only before and after refueling. In practice, additional scheduled and unscheduled plant cooldowns may be necessary for plant maintenance.

2. Unit Loading and Unloading

The unit loading and unloading cases were conservatively represented by a continuous and uniform ramp power change of 5% per minute between no load and full load. The reactor coolant temperature varies with load as prescribed by the temperature control system. The number of each operation was specified at 14,500 times. In practice, the plant is operated at base load.

3. 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 was designed to restore plant equilibrium without reactor trip following a ±10% step change in demand. The turbine load power range for automatic reactor control initiated from nuclear plant equilibrium conditions, is in the range between 15% and 100% of full load. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs so that peak reactor coolant temperature is minimized. Concurrently, the reactor coolant temperature is restored to its programmed set point at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step load decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the 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 is ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature set point change is made as a function of turbine-generator load as determined by first stage (inlet) turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant's decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash, which reduces the rate of pressure decrease.

Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

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

raised to a value above its initial equilibrium value at the beginning if the transient.

The number of each operation was specified at 2000 times.

4. Large Step Decrease in Load

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature automatically initiates a secondary side steam dump that prevents a reactor shutdown or the lifting of steam generator safety valves.

This transient capability definition brackets the transient design bases used for the Regulating Systems as discussed in Section 7.3.

The number of occurrences of this transient was specified at 200 times.

5. 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 removes the core residual heat and prevents 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 was specified at 400 times.

6. Hydrostatic Test Conditions

The pressure tests outlined below apply to field pressure 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.

a. Primary Side Hydrostatic Test before Initial Startup at 3110 psig

This hydrostatic test was performed at a minimum water temperature of 100 F, imposed by a reactor vessel material NDTT value 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 designed on the basis of 5 cycles of this hydro test.

b. Reactor Coolant System Leakage Test

This test is performed at normal operating pressure following each refueling outage prior to startup in accordance with ASME Section XI. Additional tests are performed following repairs, replacements or modifications of the RCS in accordance with ASME Section XI.

7. Loss of Load Without Immediate Turbine or Reactor Trip

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

8. Loss of Flow

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

All components in the Reactor Coolant System were 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 were determined for each of these transients through systematic analytical procedures. These stress intensity values S_{ij} (i, j = 1,2,3) were 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} were determined. From these values, the largest S_{alt} (alternating stress intensity) was found.

For this largest value of S_{alt}, an allowable number of cycles (N) was determined through design fatigue curves established for different materials. The ratio of design cycles (n) to allowable cycles (N) gave the usage factor ^{u}i (i = 1,2,3…, n). Usage factor was determined in this manner for all transients. The cumulative usage factor was determined by summing the individual usage factors. The cumulative usage factor (U = $^{u}1^{+}$ $^{u}2^{+u}3^{+}\ldots^{+u}n$) was never allowed to exceed a value of 1.0.

Although loss of flow and loss of load transients were not included in the tabulation, since the tabulation was only intended to represent normal design transients, the effects of these transients were analytically evaluated and were included in the fatigue analysis for primary system components.

Over the range from 15% of full power up to and including, but not exceeding, 100% of full power, the Reactor Coolant System and its components were designed to accommodate 10% of full power step changes in plant load and 5% of full power per minute ramp changes without reactor trip. The Reactor Coolant System can accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump systems make it possible to accept a 10% to 50% change in load, at a maximum turbine unloading rate of 200% per minute from approximately 100% load with steam dump without reactor trip (load rejection capability depends on full power Tavg; see Section 7.3.2). However, for component stress analysis purposes, this was analyzed as a step change in load from 100% to 50% load.

4.1.6 Service Life

The service life of the 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 material radiation damage effects.

The NDTT shift of the vessel material and welds, due to radiation damage effects, is monitored by a radiation damage surveillance program which conforms with ASTM-185 standards.

Reactor vessel design was based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material, as a result of operations such as leak testing and plant heatup and cooldown.

To establish the service life of the Reactor Coolant System components as required by the ASME (Section III) Boiler and Pressure Vessel Code for Class “A” vessels, the unit operating conditions were established. These operating conditions included the cyclic application of pressure loadings and thermal transients.

The number of thermal and loading cycles used for design purposes are listed in Table 4.1-8.

4.1.7 Codes and Classifications

All pressure containing components of the Reactor Coolant System were of U.S. manufacture and were designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1-9.

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 combine with the primary steady state stresses.

References

1) WCAP-8475, "Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program", Westinghouse Class 3, January 1975.

TABLE 4.1-1

REACTOR COOLANT SYSTEM PRESSURE SETTINGS

	Pressure, psig
Design Pressure	2485
Operating Pressure (at pressurizer)	2235
Safety Valves	2485
Power Relief Valves	2335
Pressurizer Spray Valve (Begin to open)	2260
(Fully open)	2310
High Pressure Trip	2385
Low Pressure Trip	1800
Hydrostatic Test Pressure	3110

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, ft-in (Bottom Head O.D. to top of Control Rod Mechanism Housing)	43-9 $\frac{11}{16}$
Water Volume, (with core and internals in place, ft^3)	4647
Thickness of Insulation, min, in	3
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head Studs, in	7
ID of Flange, in	167 $\frac{1}{16}$
OD of Flange, in	205
ID at Shell, in	173
Inlet Nozzle ID, in	27½
Outlet nozzle ID, in	29
Clad Thickness, min, in	$\frac{5}{32}$
Lower Head Thickness, min, in	5 $\frac{5}{16}$
Vessel Belt-Line Thickness, min, in	8 $\frac{5}{8}$
Closure Head Thickness, in	7
Reactor Vessel Inlet Temperature, F	517.3
Reactor Vessel Outlet Temperature, F	611.7
Reactor Coolant Flow, lb/hr	1.388×10^8

TABLE 4.1-3

PRESSURIZER & PRESURIZER RELIEF TANK DESIGN DATA

Pressurizer	
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design/Operating Temperature, F	680/653
Water Volume, Full Power, ft^{3*}	1080
Steam Volume, Full Power, ft^3	720
Surge Line nozzle Diameter, in/Pipe Schedule	14/Sch 140
Shell ID, in/Calculated Minimum Shell Thickness, in	84/4.1
Minimum Clad Thickness, in	0.188
Electric Heaters Capacity, kW	1730
Heatup rate of Pressurizer using Heaters only, F/hr	55 (approximately)
Power Relief Valves	
Number	2
Set Pressure (open), psig	2335
Capacity, lb/hr Saturated steam/valve	179,000
Safety Valves	
Number	3
Set Pressure, psig	2485
Capacity, lb/hr Saturated steam/valve	420,000
Pressurizer Relief Tank	
Design pressure psig	100
Rupture Disc Release Pressure psig	100
Design temperature, F	340
Normal water temperature, F	Containment Ambient
Total volume, ft^3	1800
Rupture Disc Relief Capacity, lb/hr	1.224×10^6

* 60% of net internal volume

TABLE 4.1-4

STEAM GENERATOR DESIGN* DATA

Number of Steam Generators	4
Design Pressure, Reactor Coolant/Steam, psig	2485/1085
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig	3107
Design Temperature, Reactor Coolant/Stem, °F	650/600
Reactor Coolant Temperature to Steam Generator, °F	602.5
Reactor Coolant Temperature from Steam Generator, °F	540.7
Reactor Coolant Flow, (gpm)	88,600
Total Heat Transfer Surface Area, ft^2	43,467
Heat Transferred at Design (Licensed) Power Level, (807.5 Mwt) Btu/hr	2755×10^6
Steam Conditions at Full Load, Outlet Nozzle:	
Steam Flow, 10^6 lb/hr	3.2914 / 3.4906
Steam Temperature, °F	510.9 / 510.8
Steam Pressure, psia	750.4 / 749.6
Feedwater Temperature, °F	390.0 / 433.6
Overall Height, ft-in	63 – 1.62
Shell OD, upper/lower, in	166/127
Shell Thickness, upper/lower, (minimum), in	3.5/2.62
Number of U-tubes	3214
U-tube Diameter, in	0.875
Tube Will Thickness, (Average) in	0.050
Number of manways/ID, in	4/16
Number of handholes/ID, in	6/6
Number of inspection ports/ID, in	1/3

*The values on this table apply at the design full-load power level of 3230 MWt (four loops) unless noted otherwise. The values shown are for a single steam generator.

TABLE 4.1-4
(Cont.)

STEAM GENERATOR DESIGN* DATA

	3230 MWt	Zero Power
Reactor Coolant Water Volume ft^3 (cold)	924.1	924.1
Primary Side Fluid Heat Content, Btu	24.1×10^6	23.86×10^6
Secondary Side Water Volume, ft^3**	1626.9	2666
Secondary Side Steam Volume, ft^3**	3100	2061
Secondary Side Fluid Heat Content, Btu**	45.05×10^6	72.8×10^6

* The values on this table apply to the design full load power level of 3230 MWt (four loops) unless noted otherwise. The values shown are for a single steam generator.

** These values correspond to a normal operating water level of 52% narrow range span, and may vary with changes in water level.

*** Values provided for 3230 MWt correspond to a feed temperature of 433.6°F.

TABLE 4.1-5

REACTOR COOLANT PUMPS DESIGN DATA

Number of Pumps	4
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	555
Net Positive Suction Head, Ft	170
Developed Head, ft	272
Capacity, gpm	89,700
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump discharge Nozzle ID, in	27½
Pump Suction Nozzle, ID, in	31
Overall Unit Height, ft	28.38
Water Volume, ft^3	192
Pump-Motor Moment of Inertia, $lb\text{-}ft^2$	82,000
Motor Data:	
Type	AC Induction, Single Speed, Air Cooled
Voltage	6600
Insulation Class	B Thermalastic Epoxy
Phase	3
Frequency, cps	60

TABLE 4.1-5
(Cont.)

REACTOR COOLANT PUMPS DESIGN DATA

Starting Current, amp	2950
Input (hot reactor coolant), kW	4250
Input (cold reactor coolant) kW	5600
Power, hp (nameplate)	6000

TABLE 4.1-6

REACTOR COOLANT PIPING DESIGN DATA

Reactor Inlet Piping, ID, in	27½
Reactor Inlet Piping, nominal thickness, in	2.375
Reactor Outlet Piping, ID, in	29
Reactor Outlet Piping, nominal thickness,	2.50
Coolant Pump Suction Piping, ID, in	31
Coolant Pump Suction Piping, nominal thickness, in	2.625
Pressurizer Surge Line Piping, ID, in	11.5
Pressurizer Surge Line Piping, nominal thickness,	1.25
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design Temperature, F	650
Design Temperature (pressurizer surge line) F	680
Water Volume (all 4 loops including surge line), ft^3	1156

TABLE 4.1-7

REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP

	Pressure Drop, psi
Across Pump Discharge Leg	1.1
Across Vessel, including nozzles	46.7
Across Hot Leg	1.3
Across Steam Generator	32.3
Across Pump Suction Leg	3.0
Total Pressure Drop	84.4

TABLE 4.1-8

THERMAL AND LOADING CYCLES

Transient Condition	Design Cycles+**	Loading Conditions
1. Plant heatup at 100°F per hour	200*	Normal
2. Plant cooldown at 100°F per hour	200	Normal
3. Plant loading at 5% of full power per minute	14,500	Normal
4. Plant unloading at 5% of full power per minute	14,500	Normal
5. Step load increase of 10% of full power (but not to exceed full power)	2,000	Normal
6. Step load decrease of 10% of full power	2,000	Normal
7. Step load decrease of 50% of full power	200	Normal
8. Reactor trip	400	Upset
9. Hydrostatic test at 3110 psig pressure, 100 F temperature	5 (pre-operational)	Test
10. Hydrostatic test at 2485 psig pressure and 400 F temperature	200 (post-operational)	Test
11. Steady state fluctuations – the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6 F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig.		
12. Loss of load, without immediate turbine trip or reactor trip	80	Upset
13. Partial loss of flow, one pump only	80	Upset

TABLE 4.1-8
(Cont.)

THERMAL AND LOADING CYCLES

Transient Condition	Design Cycles+**	Loading Conditions
14. Operating Basis Earthquake (OBE)	5++	Upset
15. Design Basis Earthquake (DBE)	1++	Faulted

\+ Estimated for equipment design purposes and not intended to be an accurate representation of actual transients or to reflect actual operating experience. See Section 4.1.5.

* This transient includes pressurizing to 2235 psig.

** Piping and Valves included in the Reactor Coolant System boundary are designed, analyzed and fabricated in accordance with their applicable codes.

++ The upset conditions include the effect of the specified earthquake for which the system must remain operational or must regain its operational status. The faulted conditions include the earthquake for which safe shutdown is required. For fatigue studies, Class I components were analyzed for five OBE's and one DBE in addition to other fatigue producing events in the above listed four loading conditions. Each earthquake is considered to produce ten peak stress magnitudes.

TABLE 4.1-9

REACTOR COOLANT SYSTEM – CODE REQUIREMENTS

The edition of the ASME Code, Section III and addenda to which the major components in the Reactor Coolant System were designed and fabricated are:

Component	Code Edition	Class	Applicable Addenda
Reactor Vessel	1965	A	Winter 1965 and Code Cases 1332, 1335, 1339, 1359
Rod Drive Mechanism	1965	A	Summer 1966
Replacement Steam Generators			
- Tube side	1983	1	Summer 1984
- Shell side	1983	1	Summer 1984
Pressurizer	1965	A	Summer 1966
Pressurizer Relief Tank	1965	C	Summer 1966
Pressurizer Safety Valves	1965		Summer 1966
Reactor Coolant Pump Volute	- Westinghouse design per ASME III Article 4.		

In addition the reactor coolant pipe was designed to ANSI B31.1-1955.

The loop 32 hot leg elbow replaced during the cycle 6/7 refueling outage in conjunction with the steam generator replacement was designed and fabricated to ASME Code Section III, 1983 edition, class 1 requirements, including addenda through summer 1984, although the elbow was required to meet only B31.1-1955 criteria.

4.2 SYSTEM DESIGN AND OPERATION

4.2.1 General Description

The Reactor Coolant System consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains steam generator, a pump, loop piping, and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system and auxiliary system connections is shown in Plant Drawing 9321-F-27383 and -27473 [Formerly Figures 4.2.2A and 4.2-2B]. A schematic flow diagram denoting principal parameters under normal steady state full power operating conditions is shown in Figure 4.2-2.

The containment boundary shown on the flow diagram in Plant Drawing 9321-F-27473 [Formerly Figure 4.2-2B] indicates those major components which are to be located inside the Containment. The intersection of a process line with this boundary indicates a functional penetration.

Reactor Coolant System design data are listed in Tables 4.1-2 through 4.1-6. A power level of 100% rated output for 80% of the time is considered an estimate of ideal operation over the service life of the system.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by a pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressure control system is described in Chapter 7. Spring-loaded 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.

4.2.2 Components

Reactor Vessel

The reactor vessel is cylindrical in shape with a hemispherical bottom and a flanged and gasketed removable upper head. The vessel was designed in accordance with Section III (Nuclear Vessels) of ASME Boiler and Pressure Vessel Code. Figure 4.2-3 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 ninety-five percent of the total coolant flow is effective for heat removal from the core. The remainder of the flow includes the flow through the RCC guide thimbles, the leakage across the outlet nozzles, and the flow deflected into the head of the vessel for cooling the upper flange. 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. It is attached to the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the 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 vessel which result from heat generated by the absorption of gamma energy. This protection is further described in Section 3.2.3.

Forty-eight core instrumentation nozzles are located on the lower head.

The reactor closure head and the reactor vessel flange are joined by fifty-four 7 in 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 leakoff connection is also provided beyond the outer O-ring seal.

The vessel is insulated with metallic reflective-type insulation supported from the nozzles. Insulation panels are provided for the reactor closure head which are supported on the refueling seal ledge and vent shroud support rings.

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

Surveillance specimens made from reactor vessel steel are located between the reactor vessel wall and the thermal shield. These specimens are examined at selected intervals to evaluate reactor vessel material NDTT changes as described in Section 4.5.2.

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 3.2-23.

Reactor vessel design data are listed in Table 4.1-2.

Pressurizer

The general arrangement of the pressurizer is shown in Figure 4.2-4, and the design characteristics 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 basis for establishing the pressure relieving capacity for the Reactor Coolant Pressure Boundary is the loss of 100 percent of the heat sink i.e. steam flow to the turbine, when the thermal output of the reactor is at 100 percent of its rated power, with appropriate credit taken for operation of the secondary system safety valves and the Reactor Protection System.

Overpressure protection is described in Section 4.3.4.

The pressurizer 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 section of the vessel regulate the Reactor Coolant System pressure by keeping the water and steam in the pressurizer at saturation temperature. The heaters are capable of raising the temperature of the pressurizer and contents at approximately 55 F/hr during startup of the reactor.

The pressurizer was 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 a decrease in plant load, the pressurizer spray system, which is fed from the cold leg of a coolant loop, condenses steam in the vessel to prevent the pressurizer pressure from reaching the set point of the power operated relief valves. The spray valves on the pressurizer are power operated. In addition, the spray valves can be operated manually by a switch in the Control Room. Spray valve position is monitored by indicating lights in the Control Room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping. Isolation valves and bypasses are provided at each pressurizer spray valve. The outer bypass, which was provided to allow a small continuous spray flow to the pressurizer during on line maintenance of the valves, was removed by a subsequent modification.

During a negative pressure surge, caused by an increase in plant load, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer is constructed of low alloy steel with internal surfaces clad with austenitic stainless steel. The heaters are sheathed in austenitic stainless steel.

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

Steam Generators

Each loop contains a Westinghouse Model 44F vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 4.2-5. Principal design parameters are listed in Table 4.1-4. The steam generators were designed and manufactured in accordance with Section III (Nuclear Vessels) of the ASME Boiler and Pressure Vessel Code and were installed during the cycle 6/7 refueling outage as replacements for the original Model 44 steam generators.

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 into the channel head and to the U-tubes.

Feedwater to the steam generator enters just above the top of the U-tubes through an elevated feedwater ring. The water exits the feedwater ring through inverted J-tubes installed on the top of the ring. This minimizes the potential for damaging water hammer events to occur. The water flows downward through an annulus between the tube wrapper

and the shell and then upward through the tube bundle where it is converted to a steam-water mixture.

The steam-water mixture from the tube bundle passes through a low pressure drop, swirl vane modular primary separator assembly which imparts a centrifugal motion to the mixture and reduces the water content in the mixture. The separated water passes through a perforated section of riser tube, impinges on a small external downcomer sleeve, and combines with the feedwater for another pass through the tube bundle.

The remaining higher quality steam-water mixture rises through additional secondary separators which limit the moisture content of the steam to one tenth of one percent or less under all design load conditions. The steam outlet nozzle is equipped with an integral steam flow limiting device. Manways, handholes, and an inspection port are provided for maintenance and inspection of the secondary side.

The steam generator pressure boundary is constructed of low alloy steel. The heat transfer tubes are thermally treated Inconel 690. The interior surface of the channel head and nozzles are clad with austenitic stainless steel, and the side of the tube sheet in contact with the reactor coolant is clad with Inconel. The tube-to-tubesheet joint is welded.

<u>Reactor Coolant Pumps</u>

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. Auxiliary system piping connections to the reactor coolant pumps are shown in Plant Drawing 9321-F-27383 [Formerly Figure 4.2-2A]. The reactor coolant pump estimated performance and net positive suction head characteristics 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 speed.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser and out through passages in the diffuser and out through a discharge nozzle in the side of the casing. 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, a second 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.

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 in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount which leaks through the second seal, is also collected and removed from the pump. Pump seal injection flow is indicated in the Control Room as described in Section 9.2

Component cooling water is supplied to the motor bearing oil coolers and the thermal barrier cooling coil.

Each of the reactor coolant pumps is provided with an anti-reverse rotation device in the motor. This anti-reverse mechanism consists of eleven pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and two shock absorbers.

After the motor has come to a stop, one pawl engages the ratchet plate and, as the motor tends to rotate in the opposite direction, the ratchet plate also rotates until stopped by the shock absorbers. The rotor remains in this position until the motor is energized again. After the motor has come up to speed, the ratchet plate is returned to its original position by the spring return.

When the motor is started, the pawls drag over the ratchet plate until the motor reaches approximately 80 revolutions per minute. At this time, centrifugal force holds the pawls in the elevated position until the speed falls below the above value.

Normal loop flow, with all pumps running is approximately 9474 pounds/second (per loop). The following table indicates the loop flow distribution for less than four pumps running:

	Running Pump Loop Flow	Stopped Pump Loop Flow	Core Flow
3 Pumps Running*	10,244 (lb/sec)	-3,048 (lb/sec)	27,686 (lb/sec)
2 Pumps Running*	10,771 (lb/sec)	-1,944 (lb/sec)	17,655 (lb/sec)
1 Pump Running*	11,035 (lb/sec)	- 913 (lb/sec)	8,295 (lb/sec)

The above table assumes that the anti-reverse flow mechanism functions properly.

In the highly unlikely event that the anti-reverse flow device does not function properly and allows the pump to rotate freely in reverse, the following flow distributions would be realized:

	Running Pump Loop Flow	Stopped Pump Loop Flow	Core Flow
3 Pumps Running*	10,360 (lb/sec)	-4,837 (lb/sec)	26,244 (lb/sec)
2 Pumps Running*	10,856 (lb/sec)	-2,924 (lb/sec)	15,863 (lb/sec)
1 Pump Running*	11,067 (lb/sec)	-1,313 (lb/sec)	7,127 (lb/sec)

Each loop is provided with flow and temperature instruments. The loop temperature information is valid for all loops, active or inactive. The loop flow information is valid only for active loops. Inactive loop flow can be estimated by calculating total flow from a heat balance and then subtracting the active loop flow.

It is important to note that the Reactor Control and Protection Systems were designed to prevent abnormally induced transients or abnormal operating conditions. Plant operating instructions describe the various combinations of plant power level and reactor coolant flows which will result in plant trips.

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.

*NOTE: Reactor power operation is not permitted with less than four loops operating.

An extensive test program was conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long term tests were conducted on less than full scale prototype seals as well as on full size seals. Operating experience with other large size, controlled leakage shaft seal pumps was also available to confirm the seal design.

The primary coolant pump flywheel is shown in Figure 4.2-8. The flywheel was fabricated from two rolled, vacuum-degassed, ASTM A-533 steel plates. The plates are bolted together with bolts aligned perpendicular to the plane of the plates. Thus the bolts carry no stress during operation.

The material specification is ASTM A-533 Grade B Class I, plus supplementary material testing requirements and Charpy tests, as detailed in Section 4.3.

Pressurizer Relief Tank

Principal design parameters of the pressurizer relief tank are given in Table 4.1-3. It is shown on Figure 4.2-9.

Steam and water discharge from the power relief and safety vales pass 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 a drain to the Waste Disposal System 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 percent of the full power pressurizer steam volume.

The tank is protected against a discharge exceeding the design value by two rupture discs which discharge into the Reactor Containment. The rupture discs on the relief tank have a combined relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure (and the rupture discs setting) is twice the calculated pressure resulting from the maximums 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 percent of the set point pressure at full flow.

The pressurizer relief tank, by means of its connection to the Waste Disposal System, provides a means for removing any non-condensable gases from the Reactor coolant System which might collect in the pressurizer vessel. The tank is constructed of carbon steel with a corrosion resistant coating on the internal surface.

Piping

The general arrangement of the Reactor Coolant System piping is shown on the plant layout drawing in Chapter 1. Piping design data are presented in Table 4.1-6.

The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 inch ID in the hot legs, 27.5 inches ID in the cold legs and 31 inches ID between each loop's steam generator outlet and its reactor coolant pump suction. The pressurizer relief line, which connects the pressurizer safety and relief valves' outlet to the inlet nozzle flange on the pressurizer relief tank was constructed of carbon steel.

Smaller piping, including the pressurizer surge and spray lines, drains and connections to other systems are austenitic stainless steel. All piping connections are welded except for flanged connections at the pressurizer relief tank and at the relief and 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 lines from the residual heat removal loop (safety injection lines)
2) Both ends of the pressurizer surge line,
3) Pressurizer spray line connection to the pressurizer, and
4) Charging lines and auxiliary charging line connections.

Valves

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 either two sets of packing and an intermediate leakoff connection or have been designed with live-loaded packing, which will either control or migrate the potential for valve stem leakage due to modulating service.

Component Supports

The support structures for the reactor coolant system components are described in Appendix 4B.

4.2.3 Pressure-Relieving Devices

The Reactor Coolant System is protected against overpressure by control and protective circuits such as the high pressure trip and by code relief valves connected to the top head of the pressurizer. The 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 Plant Drawing 9321-F-27473 [Formerly Figure 4.2-11] and the valve design parameters are given in Table 4.1-3.

A reactor head vent system is also provided to exhaust non-condensible gases or steam from the primary system that could inhibit natural circulation core cooling.

Two power operated relief valves (PORVs) and three code safety valves (SVs) are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray. The PORVs also operate from the Overpressure Protection System described in Section 4.3 to prevent RCS pressure from exceeding the limits of Appendix G

of Section III of the ASME Pressure Vessel and Boiler Code 1998 Edition through the 2000 Addenda.

The instrument N_2 system for the PORVs is tapped from the N_2 supply line to the safeguards accumulators. The actual take-off point for this N_2 system is downstream of the pressure regulator valve NNE-863. Each PORV is supplied with its own accumulator. The PORV accumulators individually hold 6 cu ft of N_2 at a minimum pressure of 550 psig. During low temperature shutdown operations, the Overpressure Protection System requires an N_2 supply of sufficient capacity which, in case of loss of the main N_2 supply, can support the number of PORV cycles resulting from an overpressure event of 10 minute duration. This N_2 supply is provided by one Safety Injection Accumulator having its associated N_2 fill valve blocked open.

A platform within the pressurizer missile shield provides access to the pressurizer safety valves for testing purposes. This access platform does not interfere with maintenance which may require removal of the safety and/or power operated relief valves.

The pressurizer relief tank is protected against a steam discharge exceeding the design pressure valve by two rupture discs which discharge into the Reactor Containment. The rupture disc relief conditions are given in Table 4.1-3.

The design criteria require that transient flow loads from valve discharge be combined with deadweight loads, seismic loads, and loads due to internal pressure. Thrust, bending, and torsion (as applicable) from each of these loads was determined. Primary stress in the supports under this load combination was limited to 1.33 times allowable stress or rated load in accordance with Manufacturer's Standardization Society Standard MSS-SP-58 for standard supports or AISC-69 requirements for non-standard supports. In those instances where the integrity of supports is also dependent on reinforced concrete anchorage, the concrete behavior limits and anchor bolt behavior were in accordance with I.E. Bulletin 79-02. (8)

In response to NUREG-0737, Item II.D.1, the Electric Power Research Institute (EPRI), under contract with the Westinghouse Owners Group, performed full scale tests on typical safety and relief valves and associated piping configurations under simulated transient conditions.

EPRI issued the results of the test program to participating utilities along with evaluation guidelines for applying the test results to plant specific pressurizer safety and relief valves and associated piping. These guidelines have been used to evaluate the adequacy and integrity of the pressurizer safety and relief valve piping during plant transients which challenge the safety and relief valves.

The evaluation of the pressurizer safety and relief valve piping considered the following two transient cases:

Case 1: Sequential actuation of the power operated relief valves (PORV's) and safety valves (SV's) at their respective set pressures with pressurizer pressure increasing at a rate of 130 psi/sec and loop seal temperature at the inlet to the SV's of 260°F.

Case 2: Block valves upstream of the PORV's closed and only SV's open at their set pressures with pressurizer pressure increasing at a rate of 144 psi/sec and loop seal temperature at the inlets to the SV's of 260°F.

The peak transient pressures, temperatures and flow rates in the discharge piping were computed using the RELAP-5 MOD 1, Cycle 14 computer code which has been verified during the EPRI test program to yield results which closely match actual pressures, temperatures and flow rates witnessed during the testing. A post processor code (FORCE) was used to translate the thermal hydraulic results of the RELAP code into transient forces on the piping and a structural code (STARDYNE) was used to determine the resulting pipe stresses and support transferred loads. In addition, the structural code NUPIPE-II was used to determine the pipe stresses and support transferred loads due to deadweight loads, normal pressure loads, thermal loads and seismic loads.

These evaluations indicated the need to upgrade the original design of the pressurizer safety and relief valve discharge system for purposes of complying with NUREG-0737 Item II.D.1. This upgrade was accomplished by a series of modifications which included: discharge piping support upgrade, discharge piping branch connection reinforcement, replacement of pressurizer safety valve internals, and re-routing of pressurizer safety valve loop seal drain lines to provide for continuous drainage back to the pressurizer. Continuous drainage is provided to preclude the occurrence of water hammer in the discharge piping in the event the valves open.

These modifications served to reduce the stresses in the safety and relief valve discharge piping to acceptable levels which are below the stress limits in the EPRI guidelines for the various load combinations considered. In addition, the stresses in the piping due to thermal loads are below the ANSI B31.1 1967 Power Piping Code stress limits. The constant supports were designed and fabricated in accordance with ANSI B31.1 whereas the hydraulic restraints were designed and fabricated in accordance with the ANSI B31.7 Nuclear Power Piping Code. Finally, the weld reinforcing pads were manufactured and installed in accordance with the requirements of ANSI B31.1.

Subsequent to the analyses and modifications described above, various pipe supports were further upgraded based on reanalyses of the pressurizer PORV inlet and outlet piping and supports. These reanalyses were performed to assess the effect to upgrading the PROV block valve operators with larger and heavier operators to enhance their design capabilities. Upgrade of the supports was deemed necessary to ensure that the integrity of the piping they support and the safety equipment they service will remain intact and to demonstrate that the design criteria used for Seismic Class I pipe supports is not violated with the added loading.

4.2.4 Protection Against Proliferation of Dynamic Effects

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 Chapter 6.

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.

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

Lateral bracing is provided near the steam generator upper tube sheet elevation to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing was provided at a lower elevation to resist pipe rupture loads.

Missile protection afforded by the arrangement of the Reactor Coolant System is illustrated in the containment structure drawings which are presented in Chapter 5.

4.2.5 Materials of Construction

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

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

The water in the secondary side of the steam generators is held within the appropriate chemistry specifications to control deposits and corrosion inside the steam generators. Secondary side chemistry is monitored as described in the Technical Specifications.

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

It is characteristic of stress corrosion that combinations of alloy 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. In the presence of this environment, stress-corrosion cracking can occur in stainless steels having high residual stresses resulting from manufacturing procedures.

The use of lead in the materials of the secondary side of Indian Point 3 was minimized to the practical limit of that occurring as trace elements in alloys and as such is insignificant.

All external insulation employed in the Reactor Coolant System is compatible with the structural materials of the component. The cylindrical shell exterior and closure flanges to the reactor vessel are insulated with metallic reflective insulation. The closure head is insulated with low halide-content insulating material.

Assurance of adequate fracture toughness of the reactor vessel is provided by compliance with the requirements for fracture toughness included in the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. In cases where it is not practicable to perform all tests in accordance with this Code, conservative estimates of material fracture toughness were made to demonstrate compliance with the Code requirements.

The techniques used to measure and predict the integrated fast neutron (E greater than 1 MeV) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum fast neutron exposure of the reactor vessel is identical to that described for the irradiation samples. Since the neutron spectra at the samples and vessel inner surface are identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel is obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The maximum integrated fast neutron exposure of the vessel inner surface was computed to be 0.922 x 10^{19} n/cm^2 at 27.1 effective full power years, at a power of 3216 MWt. Similarly, the maximum integrated fast neutron exposure at the ¼T location was computed to be 5.5 x 10^{18} n/cm^2 (11). With this exposure the end-of-life ΔRT_{NDT} was estimated to be 170.6°F, and RT_{PTS} temperature was not expected to be over 268°F, which is below the NRC screening criteria of 270°F.

The maximum integrated fast neutron exposure of the vessel inner surface was computed to be 1.560 x 10^{19} n/cm^2 for the period of extended operation, or 48 effective full power years, at a power of 3216 MWt. Similarly, the maximum integrated fast neutron exposure at the 1/4T location was computed to be 9.298 x 10^{18} n/cm^2 at EOL. With this exposure, the end-of-life 1/4T RT_{NDT} was estimated to be 255.8°F (plate B2803-3), and RT_{PTS} temperature was estimated to be 279.9°F (plate B2803-3), which exceeds the NRC screening criterion of 270°F. For all other locations, the RT_{PTS} temperature is projected to be less than the NRC screening criterion. As required by 10 CFR 50.61(b)(4), a plant-specific safety analysis for plate B2803-3 will be submitted to the NRC three years prior to reaching the RT_{PTS} screening criterion. Each core redesign is evaluated to assure that leakage is less than assumed in analyses to predict the effect of neutron embrittlement.

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by Westinghouse Corporation and the results have been evaluated and reported. (1) Based on the Westinghouse evaluation, heatup and cooldown curves were developed for up to 9.26 EFPYs of reactor operation.

Generic Letter 88-11 requested that licensees use the methodology of Regulatory Guide 1.99, revision 2. “Radiation Embrittlement of Reactor Vessel Material”, to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A of 10 CFR part 50, Appendix G. Capsules X and Z were analyzed (9)(10) and new pressure-temperature curves were developed using this methodology.

The maximum shift in RT_{NDT} after 20 EFPYs of operation is projected to be 230.1°F at the 1/4T and 188.8°F at the 3/4T vessel wall locations for Plate B2803-3 the controlling plate. The maximum adjusted RT_{NDT} after 48 EFPYs of operation is projected to be 255.8°F at the 1/4T and 207.9°F at the 3/4T vessel wall locations for Plate B2803-3 the controlling plate.

Plate B2803-3 was also the controlling plate for the operating periods of 2 EFPYs, 9 EFPYs, 11.00 EFPYs, 13.3 EFPYs and 16.2 EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value or RT_{NDT} at the end of 20 EFPYs. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table 4.4.1.

Changes in fracture toughness of the core region plates or forgings, weldments, and associated heat treated zones due to radiation damage will continue to be monitored by a surveillance program which conforms with ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels. (3) Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals. To this extent, the second, third and forth reactor vessel material surveillance capsules were removed in 1982, 1987 and 2003 respectively. These capsules have been tested and the results have been evaluated and reported. (4)(5)(6)(9)(10) No change to the technical specification heatup and cooldown limit curves was required as a result of the Capsule Y and Capsule Z analyses. However, the new analytical methodology introduced in Reg. Guide 1.99, Rev. 2, resulted in the generation of new heatup-cooldown curves [Deleted].

The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing by Charpy V-notch, drop weight and tensile specimens and post-irradiation testing by Charpy V-notch, tensile, and wedge opening loading specimens carried out during the lifetime of the reactor vessel.

Information from the radiation surveillance program described above is also used for evaluations required by the NRC's Pressurized Thermal Shock (PTS) Rule (10 CFR 50.61). Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the reactor vessel as a function of fluence. All of the projected RT_{PTS} values are within the NRC screening values for PTS at 48 EFPY with the exception of plate B2803-3, which has an EOL RT_{PTS} of 279.9°F and is 9.9°F above the screening criterion. As required by 10 CFR 50.61(b)(4), a plant-specific safety analysis for plate B2803-3 will be submitted to the NRC three years prior to reaching the RT_{PTS} screening criterion. At present, it is estimated that plate B2803-3 will reach the screening criterion at approximately 37 EFPY. (7)

4.2.6 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 4.1.5. The normal system heat-up rate is $\leq$60°F/hr, and cooling rate is $\leq$50°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 pressurized liquid homogeneous with the coolant.

The fastest cooldown rates which result form the hypothetical case of a break of a main stream line are discussed in Chapter 14. Refer to Section 4.4 for further information.

4.2.7 Leakage

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

1) Two radiation sensitive instruments provide the capability for detection of leakage from the Reactor Coolant System. The containment air particulate monitor is quite sensitive to low leak rates. The containment radiogas monitor is much less sensitive but can be used as a backup to the air particulate monitor.

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

3) A leakage detection system is included which determines leakage losses from all water and steam systems within the Containment including that from the Reactor Coolant System. This system collects and measures moisture condensed from the containment atmosphere by the cooling coils of the fan cooling units. It relies on the principle that all leakages up to sizes permissible with continued plant operation will be evaporated into the containment atmosphere. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary.

4) An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer. Assuming no operator action, the increase in containment sump level provides a less sensitive means of detecting leakage. However, sensitivity of the processing systems and containment sump system can be improved with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every 4 hours).

Leakage detection methods are described in detail and evaluated in Chapter 6.

Leakage Prevention

Reactor Coolant System components were manufactured to exacting specifications which exceeded normal code requirements (as outlined in Section 4.1.3). In addition, because of the welded construction of the Reactor Coolant System and the extensive non-destructive testing to which it was subjected (as outlined in Section 4.5), it is considered that leakage through metal surfaces or welded joints is very unlikely.

However, some leakage from the Reactor Coolant System is permitted by the reactor coolant pump seals. Also, all sealed joints are potential sources of leakage even though the most appropriate sealing device was selected in each case. Thus, because of the large number of joints and the difficulty of assuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable. Leakage from the reactor through its head flange will leak off between the double o-ring seal and actuate an alarm in the Control Room.

Whereas leakage prevention is important from the standpoint of RCS inventory loss, it can also be of concern with regard to corrosion of carbon steel components within the reactor

coolant pressure boundary. Leakage past the canopy seals of the reactor vessel closure head penetrations has been reported at several plants including Indian Point 3. While this type of leakage is not a safety issue due to the minimal amounts possible, it does represent a boric acid corrosion concern. For these reasons, CRDM and CET Seal Clamp Assemblies designed by the reactor manufacturer have been installed at the twelve spare reactor vessel head penetrations and at the five core exit thermocouple penetrations to prevent canopy seal weld leaks in these locations. The CRDM and CET Seal Clamp Assemblies meet all of the ASME Boiler and Pressure Vessel Code requirements applicable to the Indian Point 3 reactor vessel.

Locating Leaks

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

Methods of leak location which can be used during plant shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the Reactor Coolant System in the leaking fluid and deposited by the evaporation process.

4.2.8 Water Chemistry

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

Following a significant boration or dilution, the operator normally will sample both the Reactor Coolant System hot legs and the pressurizer liquid for record purposes and to check that homogenization of the pressurizer liquid with the recirculating reactor coolant has been completed. For a cold shutdown, the operator borates the system prior to the start of cooldown.

All 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 qualify listed in Table 4.2-2. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and the Sampling System which are described in Chapter 9.

4.2.9 Reactor Coolant Flow and Temperature 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 flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation $\Delta P/\Delta P_o = (w/w_o)^{2.0}$ where ΔP_o is the referenced pressure differential with the corresponding referenced flow rate w_o and ΔP is the pressure differential with the corresponding flow rate w. The full flow reference point was established during initial plant startup. The low flow trip point was then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse 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 14.1.6 provides for an allowance of 3% measurement inaccuracy.

As a result of the calibration techniques used, the absolute accuracy of the coolant flow measurement is not relevant. As indicated in Chapter 14, the limiting trip point assumed for analysis was 87% loop flow. As indicated above, this represents a 3% of flow allowance below the allowable value (low flow trip point) of 90% or higher which is dictated by the Technical Specifications. Since the trip point is calibrated as a function of full flow output of the instrument and since the flow rate of the reactor is verified to be equal or greater than the design flow rate during startup testing, the actual trip point would be 89% or greater based on the 1% repeatability.

Startup tests provide a means for verifying that reactor coolant flow is equal to or greater than the design flow rate. The core flow rate cap be verified with an accuracy better than 10% by correlating a secondary system heat balance and the inlet and outlet core temperatures. In addition measurements of pump input power and loop ΔP can be made at hot shutdown condition for various configurations of running pumps, and the absolute flow rate of each pump is verified to be greater than the design flow.

Hot leg temperature measurement of each loop is accomplished with three fast response, narrow range single element RTD's mounted in the three scoops of each hot leg. Cold leg temperature measurement of each loop is accomplished with one fast response, narrow range duel element RTD mounted in the manifold of each cold leg. In addition a single wide range RTD is provided in each cold leg at the discharge of the RCP and near the entrance to each steam generator.

4.2.10 Reactor Coolant System Vent Collection System

A vent collection system consisting of stainless steel pipes is provided as a means to collect water and contain gases during venting of the Reactor Coolant System during cold shutdown.

The system is located inside the Containment and consists of a vent collecting header which is connected to the Reactor Coolant System venting points. The vent header discharges gases inside the Containment in the area of the purge exhaust system. Liquids are continuously drained to the Reactor Coolant Drain Tank or to the Containment Sump.

The Vent Collection System is used only when the reactor is in cold shutdown; vent connections are disconnected and the Reactor Coolant System venting points flanged off before the reactor is put into hot shutdown mode.

4.2.11 Reactor Head Vent System

The basic function of the Reactor Vessel Head Vent System is to remove non-condensible gases or steam from the Reactor Vessel Head. This system is designed to mitigate a possible condition of inadequate core cooling due to a loss of natural circulation resulting from the accumulation of non-condensible gases in the Reactor Coolant System. The Reactor Head Vent System connects to the reactor head via an existing part length control rod drive mechanism conoseal port. The vent piping downstream of the concoseal port connects to two parallel paths. Each path contains two series normally closed valves. The two flow paths join together and discharge into the pressurizer safety and relief valve

discharge header. The common vent line downstream of the valves contains a flanged spool piece which is removable for refueling. These flanged connections are therefore outside the reactor coolant pressure boundary.

The Reactor Head Vent System is operated manually from the main control room. The solenoid operated isolation valves are full open/close type valves which are controlled from individual control switches in the main control room.

4.2.12 Reactor Vessel Level Indication System (RVLIS)

The basic function of the RVLIS is to monitor the water level in the reactor vessel or relative voids in the RCS during accident conditions. However, the system is designed to function under all plant normal, abnormal, and accident conditions, including LOCA, post-LOCA, and during and after a seismic event. Refer to Section 7.5.2 for more detailed information about the operation of this system.

References

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(2) Hazeton, W.S., S. L. Anderson and S. E. Yanichko. “Basis for Heatup and Cooldown Limit Curves,” WCAP-7924, July, 1972.

(3) Yankchko, S. E. and J. A. Davidson, “Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program,” WCAP-8475, January, 1975.

(4) Yanichko, S. E. and Anderson. “Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 1,” WCAP-10300-1. March, 1983.

(5) Kaiser, W. T. “Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 2,” WCAP-10300-2, March, 1983: Heatup and Cooldown Curves for Normal Operation.

(6) Yanichko, S. E. “Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 3,” WCAP-10300-3, March, 1983: J-Integral Testing of IX-WOL Fracture Mechanics Specimens.

(7) Perone, V. A. et al. “Indian Point Unit 3 Reactor Vessel Fluence and RT_{PTS} Evaluations,” WCAP-11045, January, 1986 and Revision 1 thereto dated June,1989.

(8) NRC IE Bulletin 79-02, dated March 8, 1979, “Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts.”

(9) Yanichko, S. E. “Analysis of Capsule Z from the New York Power Authority Indian Point 3 Reactor Vessel Radiation Surveillance Program,” WCAP-11815, March, 1988.

(10) Entergy Report to NRC “Capsule X Material Surveillance Report, NL-04-092,” 07/29/04.

(11) Calculation CN-RCDA-03-88, Rev. 0, “Indian Point Unit 3 Stretch Power Uprate Evaluation for Reactor Vessel Intergrity,” Westinghouse Electric Co., December 2003.

TABLE 4.2-1

MATERIALS OF CONSTRUCTION OF THE
REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Materials
Reactor Vessel	Pressure Plate	SA-302, Gr. B
	Shell & Nozzle Forgings	A-508 Class 2
	Cladding, Stainless Weld Rod	type 304 equivalent
	Thermal Shield and Internals	A-240, type 304
	Insulation	Stainless steel, Aluminum
Replacement Steam Generator	Pressure Plate	SA-533, Gr. B Class 1
	Cladding, Stainless Weld Rod	type 304 equivalent
	Cladding for Tube Sheets	Inconel Weld Deposit
	Tubes	Inconel (SB-163)
	Channel Head Forgings	SA-508 Class 3
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
Pressurizer Relief Tank	Shell	A-285 Gr. G
	Heads	A-285 Gr. C
	Internal surface coating	Amercoat 55
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

TABLE 4.2-2

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Chemistry Parameter	MODE	Requirement
Electrical Conductivity	1, 2, 3, 4, 5	Determined by the concentration of boric acid and alkali present.
Solution pH (at 25 C)	1, 2, 3, 4, 5	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration).
Oxygen, ppm, max. (>250 F)	1, 2, 3, 4	0.10
Chloride, ppm, max. (>250 F)	1, 2, 3, 4	0.15
Flouride, ppm, max. (>250 F)	1, 2, 3, 4	0.15
Hydrogen, cc (STP)/kg H_20	1, 2	25-50
Total Suspended Solids, ppm, max.	1, 2, 3, 4, 5	1.0
pH Control Agent (Li^7OH)	1, 2	Varying to maintain minimum RCS $pH_{(t)}$ of 6.9, not to exceed 3.5 ppm Lithium
Boric Acid as ppm Boron	1, 2, 3, 4, 5, 6	Variable from 0 to 4000

4.3 SYSTEM DESIGN EVALUATION

4.3.1. 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 the design and stress analyses, the material selection and fabrication, and the quality control and operations control.

Reactor Vessel

A stress evaluation of the reactor vessel has been carried out in accordance with the rules of Section III of the ASME Boiler and Pressure Vessel 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.

The State of New York has adopted ASME Code Section III and imposes no additional design requirements beyond those listed in this code.

A summary of fatigue usage factors for components of the reactor vessel is given in Table 4.3-2.

For the Indian Point 3 reactor vessel, the maximum thermal stress due to gamma ray heating occurs in the cylindrical portion of the vessel adjacent to the core and its value is about 2500 psi. This additional thermal stress does not augment the stress intensity values considerably. The maximum stress intensity values under steady state and transient operating conditions are still far below the allowable limits of N-414 of ASME Boiler and Pressure Vessel Code Section III. The effect of gamma ray heating on the cumulative usage factor is negligible.

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 presently in service. 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. A complete list of thermal and loading cycles and their frequencies used for design purposes are given in Table 4.1-8.

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

Assurance of adequate fracture toughness of the Reactor Coolant System is provided by compliance, insofar as possible, with the requirements for fracture toughness included in Appendix G to Section III of the ASME Boiler and Pressure Vessel Code.

The heatup and cooldown curves for the plant are based on the actual measured fracture toughness properties of the vessel materials (see the Technical Specifications) determined in accordance with the above mentioned fracture toughness requirements. Where sufficient tests to comply with these requirements for fracture toughness testing were not performed, conservative estimates of fracture toughness properties are used. Maximum allowable pressures as a function of the rate of temperature change and the actual temperature

relative to the vessel RT_{NDT} are established according to the methods given in Appendix G, Protection Against Nonductile Failure, published in Section III of the ASME Boiler and Pressure Vessel Code. Curves incorporating allowances for instrument error in measurement of temperature and pressure are given in the Technical Specifications.

These curves are based on the predicted RT_{NDT} of the vessel and include appropriate estimates of ΔRT_{NDT} caused by radiation. Estimated ΔRT_{NDT} values are derived by using Figure 4.4-1 and the fluence at 1/4T corresponding to the maximum for the service period applicable. (See Figure 4.2-10)

The results of the radiation surveillance program are used to verify the ΔRT_{NDT} predicted from Figure 4.4-1 as discussed in Section 4.4.

The use of an RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material automatically provides additional conservatism for the non-irradiated regions. Therefore, the flanges, nozzles, and other regions not affected by radiation will be favored by additional conservatism approximately equal to the assumed ΔRT_{NDT}.

Change in fracture toughness of the core region plate or forging, weld metal and of the associated heat affected zone due to radiation damage are monitored by a surveillance program based on ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels. For additional details of the irradiation program, see Section 4.5. In addition, assurance of adequate fracture toughness is further provided by the evaluations required by the NRC's Pressurized Thermal Shock (PTS) Rule (10CDF50.61). Information from the radiation surveillance program referenced above is used to demonstrate that the RT_{PTS} values generated in accordance with PTS rule methodology remain below the NRC screening values for PTS.

The vessel closure contains fifty-four 7-inch studs. The stud material has a minimum yield strength of 104,400 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is approximately 54,210 psi. This means that twenty eight of the fifty four studs have the capability of withstanding the hydrostatic end load on the vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

As part of the plant operator training program, supervisory and operating personnel are instructed in reactor vessel design, fabrication and testing, as well as present and future precautions necessary for pressure testing and operating modes. The need for record keeping is stressed as such records are helpful for future summation of time at power levels and temperatures which tend to influence the irradiated properties of the material in the core region.

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

<u>Control Rod Housing</u>

An interaction analysis was performed on the CRDM housings. The flange was assumed to be a ring and the tube a long cylinder. The different values of Young's Modulus and coefficients of thermal expansion of the tubes were taken into account in the analysis. The

local flexibility was considered at appropriate locations. The closure head was treated as a perforated spherical shell with modified elastic constants. The effects of redundants on the closure head were assumed to be local only. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation was made for the “J” weld.

Closure Head Flange and Shell

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

Main Closure Studs

A similar analysis to the one described for the closure head flange and shell was performed for the vessel flange to vessel shell juncture and for the main closure studs.

Inlet Nozzle and Vessel Support

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

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

Outlet Nozzle and Vessel Support

The method of analysis for the outlet nozzle and vessel support was the same as for the inlet nozzle.

Vessel Wall Transition

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

Core Barrel Support Pads

Thermal, mechanical and pressure stresses were calculated at various locations on the pad and at the vessel wall. Mechanical stresses were calculated by the flexure formula for bending stress in a beam, pressure stresses were taken from the analysis of the vessel to bottom head juncture and thermal stresses were determined by the conservative method of

skin stresses. The stresses due to the cyclic loads were multiplied by a stress concentration factor where applicable and used in the fatigue evaluation.

Bottom Head to Shell Juncture

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

Bottom Head to Instrument Penetrations

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

The location and geometry of the areas of discontinuity and/or stress concentration are shown in Figures 4.3.-1, 4.3.-2 and 4.3-3.

In addition to the analyses of reactor vessel components, the following transients were analyzed since they cause temperature and pressure excursions influencing the cumulative fatigue of the reactor vessel:

Transient Analyses

Loss of Load Transient. This is the most severe anticipated transient on the Reactor Coolant System. This transient applies to a step decrease in turbine load from full power occasioned by the loss of turbine load without immediately initiating a reactor trip. The reactor and turbine eventually are assumed to trip as a consequence of a high pressurizer level trip initiated by the Reactor Protection System.

Figure 4.3-4 gives the pressure-temperature transient assumed in the analysis for usage factor. This design basis transient is more severe than that reported in Section 14.1.8.

Loss of Flow Transient. This transient applies to a partial loss of flow accident from full power in which a reactor coolant pump is tripped out of service as a result of a loss of power to that pump. The consequences of such an accident 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 sizeable reduction in the hot leg coolant temperature of the affected loop. Figure 4.3-5 gives the temperature transient assumed in the analysis for usage factor. This design basis transient is more severe than that reported in Section 14.1.6.

The number of occurrences of the loss of load and the loss of flow transients was generally specified at 80.

Reactor Coolant System Stress Analyses

All components in the Reactor Coolant System were designed to withstand the effects of these and other transients that would 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 were determined for each of these transients through systematic analytical procedures. These stress intensity values S_{ij}(i, j = 1, 2, 3) were plotted against a time interval for each cycle. This plot would represent one or more stress cycles. For each cycle, extreme values of S_{max} and S_{min} were determined. From these values, the largest S_{alt} (alternating stress intensity) was found.

For this value of S_{alt} , an allowable number of cycles (N) was determined through design fatigue curves established for different materials. The ratio of design cycles (n) to allowable cycles (n) gave the usage factor (u). The usage factor was determined in this manner for all transients. The cumulative usage factor was determined by summing the individual usage factors. The cumulative usage factor ($U=u_1 + u_2 + u_3 \ldots$) was never allowed to exceed a value of 1.0.

Subsection N415.2 of the 1965 Edition of the ASME Code was used for calculating the usage factors.

Details of thermal and seismic analyses are summarized below(15). These analyses are directly applicable to Indian Point 3.

Thermal Stresses

Maximum thermal stresses in the core barrel would occur if cold water were injected from the accumulators due to the occurrence of a Loss-of-Coolant Accident. The barrel is exposed to cold water in the downcomer annulus and to somewhat hotter water in the compartments between barrel and baffle, producing a thermal gradient across the barrel wall.

The lower support structure is cooled more uniformly because of the large and numerous flow holes, and consequently, thermal stresses are lower.

The method used to obtain the maximum barrel stresses was as follows:

1) Temperature distribution across the barrel wall was computed as a function of time taking into consideration water temperatures and film coefficients

2) Assuming that the obtained thermal gradients were axisymetrically distributed, which is conservative for stresses, maximum thermal stresses were computed in the barrel which was considered as an infinite cylinder

3) Thermal stresses were added to primary stresses, including seismic, in order to obtain the maximum stress state of the barrel.

Results of the analyses showed that the maximum thermal stresses in the barrel wall were well below the allowable criteria given for design by Section III of the ASME Code.

Seismic Analysis of Reactor Internals

The maximum stresses were obtained by combining the contributions from the horizontal and vertical earthquakes in the most conservative manner. The following paragraphs describe the horizontal and vertical contributions.

Horizontal Earthquake Model and Procedure

The Containment Building along with the reactor vessel support, the reactor vessel, and the reactor internals were included in this analysis. The mathematical model of the building, attached to ground, was identical to that used to evaluate the building structure. The reactor internals were mathematically modeled as beams, concentrated masses, and linear springs.

All masses, water, and metal were included in the mathematical model. All beam elements had the component weight or mass distributed uniformly, e.g., the fuel assembly mass and barrel mass. Additionally, wherever components were attached somewhat uniformly, their mass was included as an additional uniform mass, e.g., baffles and formers acting on the core barrel. The water near and about the beam elements was included as a distributed mass.

Horizontal components were considered as concentrated masses acting on the barrel. These concentrated masses also included components attached to the horizontal members since these were the media through which the reaction was transmitted. The water near and about these separated components was considered as being an additive at these concentrated mass points.

The concentrated masses attached to the barrel represented the following: a) the upper core support structure, including the upper vessel head and one-half the upper internals; b) the upper core plate, including one-half the thermal shield and the other half of the upper internals: c) the lower core plate, including one-half of the lower core support columns; d) the lower one-half of the thermal shield; and e) the lower core support, including the lower instrumentation and the remaining half of the lower core support columns.

The modulus of elasticity was chosen at its hot value for the three major materials found in the vessel, internals, and fuel assemblies. In considering shear deformation, the appropriate cross-sectional areas were selected along with a value for Poisson's ratio. The fuel assembly moment of inertia was derived from experimental results by static and dynamic tests performed on fuel assembly models. These tests provided stiffness values for use in this analysis.

The fuel assemblies were assumed to act together and were represented by a single beam. The following assumptions were made with regard to connection restraints. The vessel is pinned to the vessel support which is the surrounding concrete structure and part of the Containment Building. The barrel is clamped to the vessel at the barrel flange and spring connected to the vessel at the lower core barrel radial support. This spring corresponds to the radial support stiffness for two opposite supports acting together. The beam representing the fuel assemblies is pinned to the barrel at the locations of the upper and lower core plates.

Model analysis plus the response spectrum method[16] were used in this analysis. The modal analysis was studied by the use of a transfer matrix method.

The maximum deflection, acceleration, etc., were determined at each particular point by summing the absolute values obtained for modes. With the shear forces and bending moments determined, the earthquake stresses were then calculated.

Figure 4.3-6 shows the mathematical model studied.

Analytical Model for Vertical Earthquake Model and Procedure

The reactor internals were modeled as a single degree of freedom system for vertical earthquake analysis. The maximum acceleration at the vessel support was increased by the amplification due to the building-soil interaction.

There were no interfaces in the analysis (e.g., dynamic to static, elastic to plastic).

Reactor Coolant Pumps Evaluation

Pump Motor Overspeed

During normal operation, the reactor coolant pumps are supplied from the unit auxiliary bus and therefore are tied to the turbine generator frequency (speed). On occurrence of unit (turbine) trip, the pump electrical buses are transferred from the auxiliary transformer without any intentional delay.

On most electrical and mechanical events which cause the turbine to be tripped, the reactor coolant pump buses and the unit are tripped simultaneously and the pumps will therefore not exceed their normal or pretrip running speed. If for some unlikely reason the only plant trip is a turbine overspeed trip (mechanical – hydraulic trip), then the pump trip will be initiated by the turbine hydraulic system and the trip point will be between 106 and 110 percent of the turbine generator synchronous speed. The turbine overspeed trip point is set at a value not to exceed 106 percent of synchronous speed (1908 rpm).

Missile Prevention

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

It may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft would fail in torsion just below the coupling to the motor. This would constitutes a loss of coolant flow in the one loop, the effect of which is analyzed in Chapter 14.

Following the seizure, the motor would continue to run without any overspeed, and the flywheel would maintain its integrity, as it would still be supported on a shaft with two bearings.

There are no other credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing would be precluded by shearing of the graphitar into the bearing. Any seizure in the seals would result in a shearing of the anti-rotation pin in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Indication of pump malfunction under these conditions would be initially by high temperature signals from the bearing water temperature detector and later by excessive No. 1 seal leakoff indications. Following these signals, pump vibration levels would be checked. These would show excessive levels, indicating some mechanical trouble. The pump would then be shut down for investigation.

The design specifications for the reactor coolant pumps included as a design condition the stresses generated by a design basis earthquake ground acceleration of 0.15g. Besides examining the externally produced loads from the nozzles and support lugs, an analysis was made of the effect of gyroscopic reaction on the flywheel and bearings and in the shaft, due to rotational movements of the pump about a horizontal axis, during the maximum seismic disturbance.

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

Precautionary measures, taken to preclude missile formation from primary coolant pump components, assure that the pumps will not produce missiles under any anticipated accident condition. Each component of the primary pump motors has been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing.

<u>Reactor Coolant Pump (RCP) Flywheel</u>

a) Methods of Fabrication

The pump flywheels were fabricated from vacuum degassed steel accordance with ASTM A-533 CL-1, Grade B.

The flywheel blanks were flame-cut from 8 inch and 5 inch plates with allowance for exclusion of flame-affected metal. They were then machined to the specific dimensions, and the bolt holes were drilled. These plates were subjected to 100 percent volumetric ultrasonic examination.

The two plates, machined to 7.25 inches and 4.45 inches in thickness, were then bolted together, the finished flywheel was attached to the motor shaft, and the whole unit was balanced to yield vibration levels at operating speed less than 0.001 inches double amplitude.

The supplier certification data, presented in Table 4.3-5, shows the Charpy V-notch test results for the Indian Point 3 flywheels at +10° F.

Acceptability of the above flywheel material for Indian Point 3, in comparison to Safety Guide No. 14 toughness criteria, was determined by the following two steps:

1) A reference curve describing the lower bound fracture toughness behavior for the material in question was established

2) Charpy (C_V) impact energy values obtained in certification tests (Table 4.3-5) at 10° F were used to fix the position of the heat in question on the reference curve.

A lower bound K_{Id} reference curve (see Figure 4.3-8) was constructed from dynamic fracture toughness data generated by Westinghouse on A-533 Grade B, Class I steel. [13][14] All data points were plotted on the temperature scale relative to the NDT temperature. The construction of the lower bound, below which no single test point falls, combined with the use of dynamic data when flywheel loading is essentially static represents a large degree of conservatism.

The applicability of a 30 ft-lb Charpy energy reference value was derived from sections on Special Mechanical Property Requirements and Tests in Article 3, Section III of the ASME Boiler and Pressure Vessel Code. The implication was that the test temperature lies a safe margin above NDTT. Indian Point 3 flywheel plates exhibited an average value greater than 30 ft-lbs in the weak direction and, therefore, met the specific requirement “a” stated in Safety Guide No. 14 that NDTT must be no higher than 10° F. Making the conservative assumption that all materials in compliance with the Code requirements were characterized by an NDT temperature of 10° F, one was able to reassign the “zero” reference temperature position in Figure 4.3-8 a value of 10° F.

Flywheel operating temperature at the surface is 120° F. The lower bound toughness curve indicated a value of 116 ksi-in^{½} at the (NDTT + 110) position corresponding to operating temperature. Safety Guide No. 14 requirement “c” was fulfilled with considerable margin for safety. The flywheel analysis was reviewed by Westinghouse (Reference 20) and it was concluded that a 10°F increase in flywheel surface temperature to 130° F has no adverse effect on the flywheel integrity or the conclusions of the analysis.

By assuming a minimum toughness at operating temperature in excess of 100 ksi-in^{½}, it was seen by examination of the correlation in Figure 4.3-9 that the C_V upper shelf energy must be in excess of 50 fl-lb; therefore, the requirement “b” that the upper shelf energy must be at least 50 ft-lb was satisfied.

Based on the above discussion, the flywheel materials met the Safety Guide No. 14 toughness criteria on the basis of supplier certification data.

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

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

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

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

So, in each of these cases, where the functional requirements of the component controlled its dimensions, it was seen that if the requirements discussed previously were met, the stress-related failure cases were more than adequately satisfied.

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

b) Methods of Quality Assurance

An NDTT less than +10 F was specified. A minimum of three Charpy tests were made from each plate, parallel and normal to the rolling direction, to determine that each blank satisfied design requirements.

The finished flywheels were subjected to 100% volumetric ultrasonic inspection.

The finished machined bores were also subjected to magnetic particle, or liquid penetrant examination.

The design-fabrication techniques yielded flywheels with primary stress at operating speed (as shown in Figure 4.3-7) which was less than 50% of the minimum specified material yield strength at room temperature (100 to 150 F).

The flywheel calculated stresses at operating speed were based on stresses due to centrifugal forces (Figure 4.3-7). The stress resulting from the interference fit

of the flywheel on the shaft was less than 2000 psi at zero speed, but this stress became zero at approximately 600 rpm because of radial expansion of the hub.

A fracture mechanics evaluation (WCAP-15666-A) was made on the reactor coolant pump flywheel. This evaluation justifies operation of the RCPs with flywheel inspections at least every 20 years.

An ultrasonic inspection capable of detecting at least 1/2" deep cracks from the ends of the flywheel and a dye penetrant or magnetic particle test of the bore will be performed when the RCP motor is sent out for refurbishment in accordance with the Preventive Maintenance Program.

c) Normal Operating Speed

The primary coolant pumps run at 1189 rpm and may operate briefly at overspeeds up to 109% (1295 rpm) during loss of outside load.

For conservatism, however, 125% of operating speed was selected as the design speed for the primary coolant pumps. For the overspeed condition, which would not persist for more than 30 seconds, pump operating temperatures would remain at about the design value.

d) Bursting Speed

Bursting speed of the flywheels has been calculated on the basis of Robinson's results[1] to be 3900 rpm, more than three times the operating speed. This is confirmed using Griffith-Irwin theory.[2]

IP3 License Amendment 221, issued July 2, 2004 revised the plant technical specifications to extend RCP flywheel inspection interval from 10 to 20 years. The amendment was based on NRC approval (May 5, 2003) of WCAP-15666 developed and submitted through the Westinghouse Owners Group. The technical justification provided in WCAP-15666 is based on a combination of deterministic fracture mechanics methods and risk based considerations of credible maximum flywheel speeds under specified operational or accident conditions.

Steam Generators

The pressure boundary integrity of the steam generator, including the primary to secondary pressure boundary is assured by compliance of the design, fabrication, analysis, inspection, and testing activities with the criteria and requirements of the ASME Boiler and Pressure Vessel Code. The stress report for the Model 44F steam generators currently installed in Indian Point Unit 3 included an evaluation for faulted conditions including large break LOCA and steam line break (loss of secondary pressure). The stress intensities calculated for these conditions are less than the applicable limits from the ASME Code. The criteria and requirements of Section III of the ASME Code (1965 edition, Winter 1965 Addenda) were used for the evaluation.

The evaluation of the stress intensity levels in the tubesheet and channel head for these faulted conditions is based on an evaluation of interactions of the complex structure of the channel head, tubesheet and lower shell. A finite element computer program is used for the

evaluation. The structure is modeled in terms of discrete elements with loading and boundary conditions applied to these elements. The system of simultaneous linear equations resulting from the modeling is solved to determine the stress conditions. This method of stress analysis is well established for reactor coolant system components. For the tubesheet calculations, the guidelines of Article I-9, ASME Code, Section III were used to calculate the ligament efficiency based on nominal pitch dimension and maximum hold dimensions.

The postulated rupture of a primary pipe is assumed to impose a maximum pressure differential of 1100 psi across the tubes and tubesheet. The maximum local primary membrane plus primary bending stress in the tubesheet under these conditions is 21,620 psi. This is well below the ASME Code stress intensity allowable of 84,000 for this condition. The stress intensities in the channel head, channel head to tubesheet weld, and the tubesheet to lower shell weld have lower values and are well below applicable limits for this condition.

The postulated rupture of a secondary pipe is assumed to impose a maximum pressure differential of 2485 psi across the tubes and tubesheet. The maximum local primary membrane plus primary bending stress in the tubesheet under these conditions is 54,650 psi. This is well below the ASME Code stress intensity allowable of 84,000 for this condition (Tables 4.3-3 and 4.3-4). The stress intensities in the channel head, channel head to tubesheet weld, and the tubesheet to lower shell weld have lower values and are well below applicable limits for this condition.

The stress intensity in the tubes has also been considered for the postulated faulted conditions. In addition to the pressure differential stresses, LOCA blowdown forces result in a bending load on the tubes. The fatigue due to vibration of the tubes during a steam line break does not need to be considered in the evaluation. The requirements of the ASME Code are met for these faulted conditions.

In addition to analytical results, results of destructive pressure testing or representative tubes demonstrates a factor of safety against tube collapse due to external pressure. The results of the pressure testing have been used to calculate a collapse pressure for tubes of the size and material in the Model 44F steam generators. The lower bound collapse pressure for the tubes in the Indian Point steam generators is 2369 psi considering tube ovality, tube wear, and tube corrosion.

The structural evaluation of the tubes uses an allowance of 2 mils of uniform wall thinning. This value is based on published values and operating experience for corrosion and erosion-corrosion. The plugging limit for indication of tube degradation is based on the requirements of IWB-3521 of the ASME Boiler and Pressure Vessel Code Section XI or an analysis meeting the requirements of Regulatory Guide 1.121 and not on the allowance for uniform thinning in the structural evaluation.

The loading conditions considered include the maximum potential earthquake loading conditions superimposed on the loss of secondary pressure effects. The dynamic effects of the fluid and the acceleration of the steam generator result in a small increase in the equivalent pressure loading compared to the base pressure differential. The stress intensity for the combined loading condition is well below limits.

The fluid dynamic load on the tube support plate for the steam line break conditions has been considered. The analysis has determined that the tube supports will be restrained without deformation of the tubes.

In addition, the secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70 ^{o}F.

4.3.2 Reliance on Interconnected Systems

The principal heat removal systems which are interconnected with the Reactor Coolant System are the Steam and Power Conversion, the Safety Injection and the Residual Heat Removal Systems. The Reactor Coolant System is dependent upon the steam generators and upon the steam, feedwater, and condensate system for decay heat removal under normal operating conditions until RHR can be placed in service (reactor coolant temperature between 250° F and 350° F). The layout of the system ensures the natural circulation capability to permit adequate core cooling following a loss of all main reactor coolant pumps.

The flow diagram of the Steam and Power Conversion System is shown on Figure 10.2-1. 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 Axiliary Feedwater System will supply water to the steam generators in the event that the main feedwater pumps are inoperative.

The Safety Injection System is described in Section 6.2. The Residual Heat Removal System is described in Section 9.3.

4.3.3 System Integrity

A complete stress analysis, which reflected consideration of fall design loadings detailed in the design specification, was prepared by the manufacturer. The analysis showed that the reactor vessel, steam generators, reactor coolant pump casings and pressurizer complied with the stress limits of Section III of the ASME Code. A similar analysis of the piping showed that it complied with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V-notch toughness tests were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generators and pressurizer to provide assurance that hydrotesting and operation will be in the ductile region at all times. In addition, dropweight tests were performed on the reactor vessel plate material.

As an assurance of system integrity, all components in the system were hydrotested at 3110 psig prior to initial operation. Replacement steam generators were shop tested on the primary side at 3107 psig, and hydrotested after installation in accordance with ASME Section XI requirements.

A summary of Charpy V-notch and dropweight test results for the reactor vessel plates and forgings is given in Section 4.4.

Furnace Sensitized Components

The following pressure or strength-bearing stainless steel component parts in the Reactor Coolant System have become furnace-sensitized during the fabrication sequence:

Reactor Vessel
Eight primary nozzle safe ends (forgings) which were overlaid in the field with stainless steel weld metal

Steam Generators
Two primary nozzle safe ends per generator – weld metal buttering

Pressurizer
All nozzle safe ends (forgings) in top and bottom heads.

Vibration and Cycle Loads

Vibration loads were considered in the design for the reactor internals, the steam generator tube bundles, and the reactor coolant pipe. Reactor coolant pump vibration is insignificant. Instrumentation is provided to check the vibration level of these pumps if an abnormal condition is suspected.

Reactor Internals

Model tests of the Indian Point 3 reactor internals were performed for normal operating and transient conditions. Results of the combined analytic and experimental work were factored into the design.

Predicted stresses and deflections were in agreement with tests on reactors having similar internals design. The results of the vibration tests performed on the Ginna rector (reported in WCAP-7408-L, Westinghouse proprietary report) [18] confirmed that the tests agree very closely with the predicted performance and margins. A more extensive testing program was performed during pre-operational testing for Indian Point 2.

Allowable stress amplitude for flow induced vibration was established on the basis of the material fatigue properties (endurance limit of 20,000 psi for 10^{10} cycles). Since infinite cycle fatigue was a criterion, no limits were then necessary for frequency. Displacement amplitudes for reactor internals vibration were not governing; stress limits were more restrictive.

An analysis of the dynamic response of the Indian Point 2 internals under seismic and blowdown loads was made. Allowable criteria were established and stresses and deflections were determined to assure that seismic and blowdown loads will not prevent core shutdown or will not interfere with the effectiveness of the emergency core cooling system (reported in detail in WCAP-7822, Westinghouse non-proprietary report).[15] This analysis applies directly to Indian Point 3.

Steam Generators

a) Tube Vibration Analysis

In the design of Westinghouse Model 44F steam generators used in Indian Point 3, the possibility of tube degradation due to either mechanical or flow-induced excitation was considered. This evaluation included detailed analysis of the tube support systems as well as an extensive research program with tube vibration model tests.

Consideration was given to potential sources of tube excitation including primary fluid flow within the U-tubes, mechanically-induced vibration, and secondary fluid flow on the outside of the tubes. The effects of primary fluid flow and mechanically-induced vibration are considered to be negligible during normal operation. The primary source of potential tube degradation due to vibration is the hydrodynamic excitation by the secondary fluid on the outside of the tubes, and this area has been emphasized in both analyses and tests, including evaluation of steam generator operating experience.

Three potential tube vibration mechanisms due to hydrodynamic excitation by the secondary fluid on the outside of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulence, and fluidelastic vibration mechanisms.

Vortex shedding is possible, at most, only for the outer few rows in the wrapper inlet region of steam generators such as the Model 44F for which non-uniform, two-phase turbulent flow exists throughout most of the tube bundle. Moderate tube response caused by vortex shedding is observed in some carefully controlled laboratory tests on idealized tube arrays. However, no evidence of tube response caused by vortex shedding is observed in steam generator scale model tests simulating the wrapper inlet region. Bounding calculations consistent with laboratory tests parameters confirmed that vibration amplitudes would be less than .001 inches, even if the carefully controlled laboratory conditions were unexpectedly reproduced in the steam generator.

Flow-induced vibrations due to flow turbulence are also small: root mean square amplitudes are less than allowances used in tube sizing, and these vibrations cause stresses which are two orders of magnitude below fatigue limits for the tubing material. Therefore, neither unacceptable tube wear nor fatigue degradation is anticipated due to secondary flow turbulence in the Model 44F design configuration.

Fluidelastic tube vibration is potentially more severe than either vortex shedding or turbulence because it is a self-excited mechanism: relatively large tube amplitudes can feedback proportional driving forces if an instability threshold is exceeded. Tube support spacing incorporated into design of both the tube support plates and the anti-vibration bars in the U-bend region provides tube response frequencies such that the instability threshold is not exceeded for secondary fluid flow conditions. This approach provides large margins against

initiation of fluidelastic vibration for tubes which are effectively supported by the Model 44F tube support configuration.

Small clearances between the tubes and supporting structure are required for steam generator fabrication. These clearances introduce the potential that any given tube support location may not be totally effective in restraining tube motion if there is a finite gap around the tube at the location. Fluidelastic tube response within available support clearance is therefore theoretically possible if secondary flow conditions exceed the instability threshold assuming no support at the location with a gap around the tube.

This potential has been investigated both with tests and analyses for the U-bend region where secondary flow conditions have the potential to exceed the instability threshold if a tube does not contact two or more sequential supports as a result of fabrication tolerances. Tube vibration response is shown to have wear potential within available design margins even for limiting tube fit-up conditions which are not expected. Corresponding tube bending stresses remain more than two orders of magnitude below fatigue limits as a consequence of vibration amplitudes constrained by available clearances. These analyses and tests for limiting postulated fit-up conditions include simultaneous contributions from flow turbulence.

Analyses and tests therefore demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the Model 44F steam generators at Indian Point 3. Operating experience with similar steam generators supports this conclusion.

b) <u>Tubesheet Analysis</u>

The evaluation of the Indian Point Unit 3 steam generator tubesheets was performed according to the rules of the ASME Boiler and Pressure Vessel Code. The design criteria encompassed consideration of both steady state, transient and emergency operations specified in the steam generator Design Specification for the plant.

The evaluation of the tubesheet involved the heat conduction and stress analysis of the tubesheet, channel head, and secondary shell structure for particular steady design conditions for which the Code stress limitations were to be satisfied, and for discrete points during transient operation for which the temperature/pressure conditions were to be known in order to evaluate stress minima and maxima for fatigue life usage.

The stress analysis of the tubesheet complex consisted of performing an interaction analysis between the tubesheet and the channel head and attached lower shell to determine the interaction forces and moments between parts of the structure. A finite element computer program is used to calculate the stress intensities. The tubesheet calculations were made with a ligament efficiency determined using guidelines from Article I-9 from the ASME Code, Section III (1965) Edition, Winter 1965 Addenda). The stress analysis considered stress due to symmetric temperature and pressure differential as well as asymmetric

temperature distribution due to temperature drop across the tubesheet divider lane.

The fatigue analysis of the tubesheet was performed at potentially critical regions, such as the junction between the tubesheet and the channel head and secondary shell, as well as representative locations in the perforated region of the tubesheet. The nominal stress results from the primary chamber components interaction analysis are used as the basis for the fatigue evaluation. These stresses are modified by stress concentration factors which are a function of position around the circumference of the hole. The fatigue evaluation is performed at the hole evaluated for the nominal location. The Model 44F steam generators for Indian Point Unit 3 did not have any misdrilled holes to be evaluated. Fatigue usage is evaluated for several locations around the circumference of the hole.

In all cases evaluated, the Indian Point Unit 3 steam generator tubesheet complex met the stress limitations and fatigue criteria.

c) <u>Tube-Tubesheet Analysis</u>

The tubes in the Indian Point Unit 3 Model 44F steam generators are hydraulically expanded the full depth of the tubesheet. This expansion of the tubes into the tubesheet adds to the strength of the tubesheet for some loading conditions. However, the effect of the tubes is not accounted for in the analysis.

Due to the expansion of the tube into the tubesheet, the effects of fluid induced vibration of the tube terminate at or near the secondary surface of the tubesheet. The fatigue usage is calculated for the tube at the tube to tubesheet juncture at the secondary surface of the tubesheet. An appropriate fatigue strength reduction factor is used at that location. The calculated fatigue usage for the tube to tubesheet juncture on the secondary side is less than the Code limit.

d) <u>Tube to Tubesheet Weld</u>

At the primary surface of the tubesheet the tube is joined to the tubesheet with a weld of the tube to the tubesheet cladding. The weld is not subject to any effects of tube vibration on the secondary side. The stress analysis of the weld considers thermal stresses due to increasing and decreasing transient temperature conditions. A finite element model was used to determine the states of stress in the tube to tubesheet weld. The model included the effect of adjacent tube holes o n the stiffness of the tubesheet. The stress state of the tubesheet surrounding the tube was based on the tubesheet analysis. The weld was evaluated for fatigue usage using an appropriate fatigue strength reduction factor. The calculated fatigue usage for the weld is less than the Code limit.

<u>Reactor Coolant Pumps</u>

The RCP motors are equipped with a Bently-Nevada vibration monitoring system that provides continuous monitoring of both shaft and motor frame vibration for trending and troubleshooting RCP vibration. The system includes a touchscreen VGA display monitor and digital paperless recorder located in the Control Room and a remote data acquisition

(DAQ) computer located in the MUX room outside the Control Room. Additionally, vibration alarms are provided for the RCPs. A Control Room alarm will be annunicated if an RCP vibration exceeds the predetermined level. A noise detector is also located adjacent to each motor and may be placed in service via a selector switch in the Control Room.

Each of the reactor coolant pumps is also equipped with two International Research and Development vibration pickups mounted at the top of the motor support stand. They are mounted in a horizontal plane and pick up radial vibrations of the pump. One is aligned parallel to the pump discharge; the other is aligned perpendicular to the pump discharge. Their signals are taken to a multi-point selector switch mounted outside the reactor containment. Both signals form each reactor coolant pump are taken to this selector switch. Also supplied is a vibration meter. This is a portable device that may be plugged into the selector switch, so the signal from any one pickup may be monitored at one time. The vibration levels of the reactor coolant pumps can thus be checked and the pumps were checked during full flow tests for flow induced vibration (Reference 19).

No analysis for vibration loading was performed for the reactor coolant pumps.

Primary System Piping

The reactor Coolant System piping was designed for normal and faulted conditions. For faulted conditions, the piping was designed and analyzed for seismic loads and blowdown forces due to a Loss-of-Coolant Accident. By design, the main piping of the reactor coolant loop is not subjected to induced pressure pulse vibration from the reactor coolant pump impeller or the pistons of the charging pump.

The perturbating frequency of the reactor coolant pump is quite high when compared to the piping natural frequency. Frequency separation, therefore, insures a very small probability for self-excited or sympathetic vibration. This is borne out of satisfactory operation of several representative coolant loops.

4.3.4 Overpressure Protection

An Overpressure Protection System (OPS) is required to prevent the reactor vessel pressure from exceeding the Technical Specifications (Appendix G) [Deleted]. The problem arises inasmuch as the reactor vessel steel has less ductility at low temperatures. As the reactor vessel is irradiated during its lifetime, the limitations become even more stringent.

The OPS is a three-channel analog curve tracking arrangement which can initiate an appropriate chain of coincidence logic for the purpose of automatically preventing a violation of the technical specification temperature/pressure limit curve for the reactor vessel.

Wide range RCS temperature signals will be used to perform two primary functions in this system:

1) Provide the arming and disarming function, and

2) Serve as the independent variable in computing the reference curve which is the system pressure limit that must be adhered to. The basis of the OPS curve is the Appendix G isothermal cooldown curve increased by 10% [Deleted] and then adjusted to allow for effects of the design basis heat input and mass input events.

The arming function of the Overpressure Protection System is activated when the RCS temperature is below a predetermined value as specified in the Technical Specifications. Below this temperature, one half of a two-out-of-two (temperature-pressure) coincidence logic is satisfied to allow the air operated valves (PCV-455C, 456) to open automatically in the event of an impending overpressure condition. This automatic opening of the valves is initiated by the other half of the two-out-of-two coincidence logic which is described below.

These same temperature signals are also fed into three respective function generators whose task is to output values of pressure as a function of the input temperature which are the maximum RCS pressures allowed at those temperatures. The difference between the Appendix G [Deleted] curve pressures and the actual RCS pressure transmitted by three 0 to 1500 psig transmitters is computed in each of the three channels, and if any two-out-of-three of these differences is smaller than a preset minimum, a trip open condition will be initiated for each pressurizer power operated relief valve. The power operated relief valves are normally operated using the N2 system with accumulators at each valve.

However, the Overpressure Protection System requires an N2 supply of sufficient capacity which, in case of loss of the main N2 supply, can support the number of PORV cycles resulting from an overpressure event of 10 minute duration. This N2 supply is provided from one Safety Injection Accumulator having its associated N2 fill valve blocked open.

The system also includes appropriate accessory equipment to provide adequate testability, calibration facilities and operator surveillance instrumentation.

The main function of the power operated relief valves (PORVs) is to open automatically at 2335 psig to protect against pressure surges. Under the OPS, the PORVs will also open automatically when the reactor coolant temperature is within the temperature range specified in the Technical Specifications, to relieve the RCS from exceeding the Appendix G [Deleted] curves. Under this mode of operation, the PORVs will relieve solid water rather than steam-water mixture at 2335 psi.

Westinghouse conducted a generic reactor vessel overpressure study and the analytical results showed that only one PORV is necessary to turn around the design mass input overpressure incident and the design thermal input overpressure incident. The motor operated valves and power operated relief valves feed to the pressurizer relief tank. The tank has sufficient capacity to accept the expected short term flow from the OPS.

The electrical activation uses two-out-of-three logic for valve activation. The electrical supply is from the instrument buses with backup by the station emergency batteries. Thus, the system is single failure proof in addition to having a secured power supply.

The system allows for accurate control of system heatup and cooldown. Spurious opening and/or closing of the PORVs is essentially eliminated by the new two-out-of-three logic (if one channel were to fail the valve would not be able to malfunction).

4.3.5 System Incident Potential

The potential of the Reactor Coolant System as a cause of accidents was evaluated by investigating the consequences of certain credible types of components and control failures as discussed in Section 14.1 and 14.2. Reactor coolant pipe rupture is evaluated in Section 14.3.

4.3.6 Loose Parts Monitoring

The metal impact monitor was designed to enable early detection of loose metallic parts which may be in the steam generator or the reactor vessel. Upon the occurrence of an impact of loose metallic parts, a pressure wave is generated in the reactor system component causing minute displacements in the component material. The step excitation of the impact produces a broadband frequency response with peak amplitude response at resonant frequencies. Many of these resonant frequencies lie in the audible frequency range and are called “Bell” frequencies. Certain of these “Bell” frequencies are especially sensitive to impact excitation due to differing modes of preferred vibration response.

The displacements attendant with an impact wave are insignificant. However, the accelerations caused by the moderately high audio frequencies are very significant; for this reason, acceleration is the parameter chosen to indicate impact.

Acceleration Measurement

Acceleration is measured by the use of special transducers that convert accelerations to electrical signals. These transducers are mounted at specially selected monitoring points on the exterior of the Reactor Coolant System. Monitoring points normally in use during plant operation are at the top and bottom of the reactor vessel and above and below each steam generator tube sheet with transducers mounted on the generator shell. Additional monitoring points are available above and below each steam generator transition cone and above each feedwater nozzle.

Each transducer contains a piezoceramic material fabricated in a manner that provides for changes in compression of the material in response to accelerations. A small electric charge proportional to acceleration is generated by piezoelectric behavior. The charge is converted to a voltage signal by a charge preamplifier that operates in a feedback manner to continuously compensate for charge input changes to maintain a near-zero differential input voltage. A small current flows in the cable connecting the transducer to the charge pre-amplifier resulting in virtual elimination of cable effects on the acceleration signal. The charge preamplifier output voltage can then be treated as a normal instrument signal requiring only normal shielding and cable considerations.

The sensitivity of this measurement to impact is determined by the impact energy (determined by mass and impact velocity) and the distance from the point of impact to the measurement point (damping and geometry changes attenuate the traveling wave).

Impact Monitoring

The voltage signal from the charge preamplifier is further amplified and then fed to a monitoring device (Digital Metal Impact Monitor). This device excludes all frequency information except for the "Bell" frequencies from the impact. The selected signals are further processed to provide an output DC signal proportional to the impact energy (as seen by the transducer) and a DC signal indicating the rate of occurrence of impact repetition.

Records and Displays

Every 24 hours and whenever the alarm setpoint is exceeded the impact energy level and rate of occurrence can be displayed on a printer. These records serve to establish a history for establishing when and where impacts were observed. The rate at which impacts occur gives an indication of the amount of debris present in the monitored area while the impact energy is a measure of the weight of the debris.

A test circuit is provided for determining the continuity of the monitoring system. The test signal inputs a representative wave form similar to a metallic impact.

Cabling

The signal cables are an integral part of the accelerometer. They are constructed of high temperature and radiation resistant materials and transmit the accelerometer signal to the charge preamplifier.

4.3.7 Cold Shutdown RCS Level Indication

The original system used to monitor RCS level indication during cold shutdown conditions was replaced and upgraded with a new system in accordance with NRC Generic Letters 87-12 – "Loss of RHR While RCS Is Partially Filled" and 88-17 – "Loss of Decay heat Removal." The new system provides two independent RCS water level indicators and the Westinghouse Ultrasonic Level Measuring System for use during reduced inventory conditions. Filling the RCS while under vacuum was a process introduced in RO11. In order to maintain compliance with Generic Letter 88-17, another RCS Level Monitoring system (Mansell Level Monitoring System, i.e. MLMS) was introduced during RO11 which was capable of providing RCS level indication while in reduced inventory and mid-loop conditions while the RCS is under vacuum as well as non-vacuum conditions. Hand held UT is required to be used with MLMS in mid-loop condition to increase the range of level indication in the hot leg. The four system includes:

1) Two redesigned, independent water level columns, with the supply and vent lines for the level gauges being constructed of stainless steel piping for strength and compatibility with the REC. The level gauges are constructed of 2 inch diameter transparent rigid polyvinyl chloride (PVC) tube and span a range of the RCS from the 78 foot elevation (=10% in the pressurizer) down to the 60.8 foot elevation (inside bottom of hot leg piping), well below the RCS "midloop" elevation of 62' – 0". The columns are independently supplied from the low point drain lines of the RCS intermediate loop piping, one level column from loop #32 and the other from loop #34. The level gauges can be viewed in the Control Room by remotely operated cameras.

2) A Westinghouse Ultrasonic Level measuring System (ULMS) that provides control room indications of water level in the reactor coolant piping during non-power operations. The ULMS consists of the following three basic components.

   a) The sensor, which transmits and receives the ultrasonic waves,

   b) The preamp/pulsar module, which drives the sensor and preamplifies the received echo signal, and

   c) The signal processing module, which provides an analog output signal proportional to the ultrasonic wave travel time in the water.

3) A MLMS that is capable of providing the Control Room indications of water level in the reactor coolant piping during non-power operation. The MLMS utilizes sensing locations above and below where the RCS water levels are expected to remain during outages. The MLMS consists of two independent indication loops, a wide range and a narrow range loop. The wide range loop has an indication range between 121 foot elevation down to the 62 foot elevation. The wide range high point connects to a vent on the pressurizer relief line. The wide range loop low point connects to a drain off of PT-413. The narrow range loop has an indication range between 77 foot elevation down to 62 foot elevation. The narrow range high point connects to a drain off of PT-433. The pressure transducer assembly (consisting of two redundant transducers per assembly) is connected to each of the four sensing locations. The pressure transducers transmit signal to two computers located in the Control Room, i.e. one narrow range and one wide range computer. The pressure difference between the high and low points is calculated by the computers and is displayed as water level on computer screens, as well as LED displays in the Control Room. The computers have an alarm feature.

4) Two hand held UT devices capable of providing indication of water level inside the reactor coolant piping during non-power operations. The UT devices are manually positioned at the 6:00 position of the RCS hot legs and level in the pipe is read locally and communicated back to the Control Room. This method of level indication is only used while the RCS is in mid-loop and may be used to comply with Generic Letter 88-17, but must be used in conjunction with the MLMS system in mid-loop operation.

In order to comply with the requirements of Generic Letter 88-17, two independent continuous RCS water level indications are necessary when the RCS is in a reduced inventory condition. Water level indication should be periodically checked and recorded by an operator or automatically and continuously monitored and alarmed. The ULMS and MLMS system provide an alarm function. The water level columns and UT devices do not. Any combination of two independent indicators may be used for level indication provided they are capable of performing within the RCS pressure (vacuum) ranges. Examples include: one level column and ULMS; one MLMS and one ULMS; two MLMS and two UT devices; two UT devices, etc. Operations procedures delineate specific actions required for the particular combination of level indicators in service.

4.3.8 Saturation Alarm and Recorder (Analog System)

The saturation recorder provides subcooling trend data to the operator which will note if saturation conditions in the Reactor Coolant System occur.

The system involves a calculating device to develop a saturation curve, an alarm duplex bistable (alarm in CCR disabled), an isolating amplifier, a high selector, a low selector and a summing unit to form a difference signal between reactor coolant temperature and saturation temperature.

The RCS temperature is determined by the use of four RCS temperature loops taken from all four hot leg instrument loops. The four signals from these loops are channeled into a device which selects the highest temperature signal and feeds it to a duplex bistable and a summing unit.

The programmed RCS saturation temperature is determined by the use of two RCS pressure loops. The two signals from these loops are channeled into a device which selects the lowest pressure signal and feeds it to a characterizer where a corresponding saturation temperature signal is derived. This temperature signal is fed into the duplex bistable and summing unit. The duplex bistable trips when the difference in temperature (RCS vs. saturation) approaches saturation conditions and when the difference in temperature is at saturation. These alarm functions are disabled from alarming in the CCR.

This system is operational during the natural circulation cooldown mode because the temperature readings in this mode are derived from the RCS loops.

References

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2) Winne, D. H. and B. M. Wundt, “Application of the Griffith-Irwin Theory of Crack Propagation to the Bursting Behavior of Discs, Including Analytical and Experimental Studies,” Transactions of ASME, Paper No. 57-A-249, December 1, 1957.

3) DELETED 6/2000

4) DELETED 6/2000

5) DELETED 6/2000

6) DELETED 6/2000

7) DELETED 6/2000

8) DELETED 6/2000

9) DELETED 6/2000

10) DELETED 6/2000

11) DELETED 6/2000

12) DELETED 6/2000

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14) Corten, H. T. and R. H. Sailors, “Relationship Between Material Fracture Toughness Using Fracture Mechanics and Transition Temperature Tests,” UILU-ENG 71-60010, August 1, 1971.

15) WCAP-7822, “Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation” (Non-Proprietary).

16) Housner, George W., “Vibration of Structures Induced by Seismic Waves,” Shock and Vibration Handbook, Volume 3, Chapter 50, ed. Harris and Crede.

17) DELETED 6/2000

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19) Westinghouse NSD Technical Bulletin TB 75-3, “Reactor Coolant Pump (RCP) Vibration Limits for Type 93 & 93A Pumps”

20) Westinghouse Letter No. INT-98-227, “RCP Flywheel Integrity,” dated October 5, 1998.

21) Dominion Engineering Inc. Report R-4147-00-1, “Reactor Vessel Tensioning Optimization Stress Report – Indian Point Units 2 & 3.”

TABLE 4.3-1

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

Area	Stress Intensity (psi)	Allowable Stress 3Sm (psi) (Operating Temperature)
Control Rod Housing	55,300	69,900
Head Flange	51,820	80,100
Vessel Flange	52,890	80,100
Closure Studs	109,400	110,300
Primary Nozzles – Inlet	45,500	80,100
Outlet	49,390	80,100
Core Support Pad	55,740	69,900
Bottom Head to Shell	37,700	80,100
Bottom Instrumentation	56,800	69,900
Nozzle Belt to Shell	42,630	80,100
CRDM Adapter Plug	27,630	48,600

TABLE 4.3-2

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

Item	Usage Factor*
Control rod housing	0.124
Head Flange	0.024
Vessel Flange	0.023
Stud Bolts	0.944
Primary Nozzles – Inlet	0.049
Outlet	0.259
Core Support Pad (Lateral)	0.052
Bottom Head to Shell	0.020
Bottom Instrumentation	0.206
Nozzle Belt to Shell	0.002
CRDM Adapter Plug	0.0036

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

TABLE 4.3-3

STRESSES DUE TO MAXIMUM STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL (2485 PSI)

Stress	(668 F) Computed Value	Allowable Value
Primary Membrane Stress	6,960 psi	56,000 psi (0.70 Su)
Primary Membrane plus Primary Bending Stress	54,650 psi	84,000 psi (1.05 Su)

TABLE 4.3-4

RATIO OF ALLOWABLE STRESSES TO COMPUTED STRESSES
FOR A STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL OF 2485 PSI

Component Part	Stress Ratio
Channel Head	1.90
Channel Head-Tube Sheet Joint	1.67
Tubes	2.08
Tube Sheet (Average Ligament)	1.54

TABLE 4.3-5

CHARPY V-NOTCH TEST RESULTS IP-3 FLYWHEELS AT +10°F
(CERTIFICATION DATA)

1. Flywheel No. 1

A. 5 In Thick Plate — Heat No. C4344 Slab No. 3

	1	2	3
"Weak" direction (ft-lbs)	30	32	33
90° to "weak" direction (ft-lbs)	60	75	80

B. 8 In Thick Plate — Heat No. C4908 Slab No. 1A

	1	2	3
"Weak" direction	57	57	53
90° to "weak" direction	38	44	49

2. Flywheel No. 2

A. 5 In Thick Plate — Heat No. C4679 Slab No. 3

	1	2	3
"Weak" direction	96	101	104
90° to "weak" direction	50	47	53

B. 8 In Thick Plate — Heat No. C4909 Slab No. 1

	1	2	3
"Weak" direction	58	52	57
90° to "weak" direction	95	78	77

TABLE 4.3-5
(Cont.)

CHARPY V-NOTCH TEST RESULTS IP-3 FLYWHEELS AT +10°F
(CERTIFICATION DATA)

3. Flywheel No. 3

A. 5 In Thick Plate — Heat No. C4679 Slab No. 3

Results as for Flywheel No. 2 (Case 2A)

B. 8 In Thick Plate — Heat No. C4909 Slab No. 1

Results as for Flywheel No. 2 (Case 2B)

4. Flywheel No. 4

A. 5 In Thick Plate — Heat No. C4679 Slab No. 3

Results as for Flywheel No. 2 (Case 2A)

B. 8 In Thick Plate — Heat No. C4909 Slab No. 1

Results as for Flywheel No. 2 (Case 2B)

4.4 SAFETY LIMITS AND CONDITIONS

4.4.1 System Heatup and Cooldown Rates

The operating limits for the Reactor Coolant System heatup and cooldown rates are defined in the Technical Specifications. These limits are recalculated periodically using methods derived from Appendix G, Protection Against Non-Ductile Failure, of Section III of the ASME Boiler and Pressure Vessel Code [Deleted]. The ASME approach utilizes fracture mechanics concepts and is based on the reference nil-ductility transition temperature, RT_{NDT}. The method calculation for the Indian Point 3 operating limits for heat up and cooldown rates is described in detail in Reference 10.

RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature or the temperature 60°F less than the 50 ft-lb (and 35-mil lateral expansion) temperature as determined from Charpy specimens oriented normal (transverse) to the major working 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_{IC} curve) which appears in Appendix G of the ASME Code. The K_{IC} curve is a lower bound of [Deleted] the static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IC} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

RT_{NDT} and, in turn, the operating limits of the nuclear power plant can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of the pressure vessel steel are monitored by the “Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program” (WCAP-8475, Reference 2). Surveillance capsules are periodically removed from the reactor, one at a time, and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} as an adjustment for radiation embrittlement. The radiation induced ΔRT_{NDT} is estimated from Regulatory Guide 1.99, Revision 2. The adjusted RT_{NDT} is used to index the material to the K_{IR} curve and, in turn, to reset the operating limits of the plant to take into account the effects of irradiation on the reactor vessel materials.

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. The analysis of this capsule (T) (Reference 3) forms the basis of the original technical specification operating limits for RCS heatup and cooldown rates for up to 9.26 Effective Full Power Years (EFPY’s). The second and third reactor vessel material surveillance capsules were removed in 1982 and 1987, respectively. These capsules have been tested and the results have been evaluated and reported (References 4,5,6,7). No change to the technical specification heatup and cooldown limit were required as a result of the Capsule Y and Capsule Z analyses.

In May 1991, 10 CFR 50.61 (Reference 9) was amended to include the provisions of Reg. Guide 1.99, Rev. 2 (Reference 8), which provided a slightly different methodology for calculation of reactor vessel embrittlement.

The heatup-cooldown curves were accordingly regenerated using this new rule for plant operating periods [Deleted]. Future Technical Specifications Amendment proposals for greater operating periods will be submitted, using this new rule, as appropriate (References

10 and 13). The fourth surveillance capsule (Capsule X) was removed in 2003. No changes to the Technical Specifications were required as a result of Capsule X analysis (Ref. 12).

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure will continue to be obtained form ongoing and future capsule analyses in accordance with the Indian Point 3 reactor vessel radiation surveillance program.

The use of an RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material, automatically provides additional conservatism for the non-irradiated regions. Therefore, the flanges, nozzles, and other regions not affected by radiation will be favored by additional conservatism approximately equal to the assumed ΔRT_{NDT}.

Table 4.4-1 provides the material toughness test requirements and data that were specified and reported for plates, forgings, piping, and weld material prior to plant operation. Specifically, the following data were provided:

1) The maximum NDT temperature as obtained from DWT results.
2) The maximum temperature corresponding to the 50 ft-lb value of the Cv fracture energy.
3) The minimum upper shelf Cv energy value for the weak direction (WR direction in plates) of the material.

For the limiting reactor vessel beltline materials, the original end-of-life RT_{PTS} was estimated to be within the screening criteria of 10 CFR 50.61 for plate metal and welds (Reference 11). (Reg. Guide 1.99, Rev. 2 defines the property RT_{PTS} as an indicator of vessel embrittlement.) Refer to Section 4.2.5 for a discussion of projected RT_{PTS} values for the period of extended operation.

The analysis of the reactor vessel material contained in surveillance Capsule T showed that the irradiated properties of the limiting reactor vessel beltline materials exceeded those predicted. However, due to the effects of low-leakage core loading strategy, measured beltline properties were within predicted values by the time Capsule Z was analyzed.

4.4.2 Reactor Coolant Activity Limits

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the coolant could constitute a hazard only if the reactor coolant system barrier is breached, and then only if the coolant contains excessive amounts of activity which could be released to the environment. The plant systems were designed for operation with activity in the Reactor Coolant System corresponding to 1 percent fuel defect. The waste gas system is designed such that rupture of a gas decay tank, following a refueling shutdown wherein the gaseous activity is removed from the reactor coolant to the waste gas tanks for decay, will not result in an offset whole body exposure in excess of 0.5 rem. In the event of a steam generator tube rupture a high activity level signal at the condenser air ejector exhaust will divert the radioactive discharge back into Containment.

4.4.3 Maximum Pressure

The Reactor Coolant System serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the uncontrolled release of fission

products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. Thus, the safety limit of 2735 psig (110% of design pressure) has been established. This represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section III. Reactor Coolant System pressure settings are given in Table 4.1-1.

4.4.4 System Minimum Operating Conditions

Minimum operating conditions for the Reactor Coolant System for all phases of operation are given in the Technical Specifications and Technical Requirements Manual.

References

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4) Yanichko, S. E. and Anderson, "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program, Volume 1," WCAP-10300-1, March 1983.

5) Kaiser, W. T., "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 2," WCAP-10300-2, March 1983: Heatup and Cooldown Curves for Normal Operation.

6) Yanichko, S. E., "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 3," WCAP-10300-3, March 1983: J-Integral Testing of IX-WOL Fracture Mechanics Specimens.

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8) Regulatory Guide 1.99, Rev. 2

9) 10 CFR 50.61 "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Vol. 56, No. 94, Effective May 15, 1991.

10) "Final Report on Pressure Temperature Limits for Indian Point 3 Nuclear Power Plant," ABB Combustion Engineering, July 1990.

11) Memorandum REC:92-003, dated February 10, 1992, F. Gumble to P. Kokolakis

12) Laubham, T., “Analysis of Capsule X from Entergy’s Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program,” WCAP-16251-NP, Rev. 0, July 2004.

13) Westinghouse WCAP-17954-P dated December 2014 “Indian Point Unit 3 Heatup and Cooldown Limit Curves for Normal Operation"

TABLE 4.4-1

MATERIAL TOUGHNESS TEST REQUIREMENTS AND DATA

Component	Material Grade	Cu (%)	(Drop Wt.) NDTT (F)	50 Ft-lb/35 Mil Temp. Long. (F)	50 Ft-lb/35 Mil Temp. Trans. (F)	RT_{NDT} (F)	Minimum Upper Shelf Long. (Ft-lb)	Minimum Upper Shelf Trans. (Ft-1b)
I. Reactor Vessel								
a. Closure Hd. Dome	A533,B,C1 1	0.13	+10	93	148[a]	88	85	55[b]
b. Clos. Hd. Peel Seg.	A533,B,C1 1	0.14	0	83	112[a]	52	108	70[b]
c. Clos. Hd Peel Seg.	A533,B,C1 1	0.14	+10	51	80[a]	20	128	83[b]
d. Clos. Hd. Peel Seg.	A533,B,C1 1	0.13	+10	78	99[a]	39	117	76[b]
e. Head Flange	A508, C1 2	NA	+ 3*	-22	- 8[a]	3	117	76[b]
f. Vessel Flange	A508, C1 2	NA	+38*	-5	16[a]	38	141	92[b]
g. Inlet Nozzle	A508, C1 2	NA	+20*	-28	-2[a]	20	154	100[b]
h. Inlet Nozzle	A508, C1 2	NA	+45*	-8	40[a]	45	120	78[b]
i. Inlet Nozzle	A508, C1 2	NA	+40*	-7	20[a]	40	158	103[b]
j. Inlet Nozzle	A508, C1 2	NA	+12*	-14	0[a]	12	155	101[b]
k. Outlet Nozzle	A508, C1 2	NA	+60*	60	150[a]	90	72.5	47[b]
l. Outlet nozzle	A508, C1 2	NA	+60*	4	44[a]	60	105.5	69[b]
m. Outlet Nozzle	A508, C1 2	NA	+60*	4	44[a]	60	96	62[b]
n. Outlet Nozzle	A508, C1 2	NA	+60*	-2	43[a]	60	123.5	80[b]
o. Upper Shell	A533, B,C1 1	NA	-50	74	128[a]	68	90 (95% shear)	58[b]
p. Upper Shell	A533, B,C1 1	0.20	-40	84	130[a]	70	100	65[b]
q. Upper Shell	A533, B,C1 1	NA	-40	43	82[a]	22	127	82.5[b]
r. Inter. Shell	A533, B,C1 1	0.20	-50	24	65**	5	134	97**

TABLE 4.4-1
(Cont.)

MATERIAL TOUGHNESS TEST REQUIREMENTS AND DATA

Component		Material Grade	Cu (%)	(Drop Wt.) NDTT (F)	50 Ft-lb/35 Mil Temp. Long. (F)	50 Ft-lb/35 Mil Temp. Trans. (F)	RT_{NDT} (F)	Minimum Upper Shelf Long. (Ft-lb)	Minimum Upper Shelf Trans. (Ft-1b)
I.	Reactor Vessel								
s.	Inter. Shell	A533, B,C1 1	0.22	-50	23	56**	-4	113	86**
t.	Inter. Shell	A533, B,C1 1	0.20	-40	40	77**	17	113	85**
u.	Lower Shell	A533, B,C1 1	0.19	0	74	109**	49	90	65**
v.	Lower Shell	A533, B,C1 1	0.22	-20	6	55**	-5	134	89**
w.	Lower Shell	A533, B,C1 1	0.24	-10	68**	134**	74	105**	62**
x.	Bot. Hd.								
	Peel Seg.	A533, B,C1 1	0.13	-40	23	62[a]	2	103	67[b]
y.	Bot. Hd.								
	Peel Seg.	A533, B,C1 1	0.16	-40	33	56[a]	-4	108	70[b]
z.	Bot. Hd.								
	Peel Seg.	A533, B,C1 1	0.13	-40	38	69[a]	9	106	69[b]
aa.	Bottom Hd. Dome	A533, B,C1 1	0.13	-30	60	107[a]	47	80	52[b]
ab.	WELD	-	0.15**	0*	-	5**	0	-	112**
ac.	HAZ	-	NA	NA	-	10**	-	-	111**

NA – Not Available

* Estimated (60^{O} F or 100 Ft-lb temp., whichever is less for forgings; O^{O} F or 30 Ft-1b temp, whichever is higher for welds).

a) Estimated when no transverse data are available. (77 Ft-lb/54 mil longitudinal temp.)

b) Estimated when no transverse data are available. (65% of longitudinal shelf)

** Westinghouse data (all other data provided by the vessel fabricator).

4.5 INSPECTIONS AND TESTS

4.5.1 Inspection of Materials and Components Prior to Operation

Table 4.5-1 summarizes the quality assurance program for all Reactor Coolant System components. In this table, all of the non-destructive tests and inspections which were required by Westinghouse specifications on Reactor Coolant System components and materials are specified for each component. All tests required 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. The fabrication and quality control techniques used in the fabrication of the Reactor Coolant System were equivalent to those used for the reactor vessel.

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

1) Ultrasonic Testing – Westinghouse required that a 100% volumetric ultrasonic test (both shear wave and longitudinal wave) of reactor vessel plate be performed. Section III Class A vessel plates were required by code to receive only a longitudinal wave ultrasonic test. The 100% volumetric ultrasonic test by both techniques was a severe requirement, but it assured that the plates were of the highest quality.

2) Radiation Surveillance Program – In the surveillance program, the evaluation of the radiation damage was based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile and wedge opening loading (WOL) fracture mechanics test specimens.

4.5.2 Reactor Vessel Surveillance

The reactor vessel surveillance program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature and the fracture mechanics approach. The reactor vessel surveillance program includes specimens from the most limiting plate used in the core region of the reactor vessel. Three capsules meet the requirements of ASTM-E-185-70 except that the limiting plate specimen orientation is transverse (weak direction) rather than longitudinal (strong direction). Five additional capsules do not meet E-185-70 since they do not include HAZ specimens. The program is essentially in accordance with ASTM-E-185-70, and ASTM-E-185-79 “Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,” except that three capsules contain, in addition to the specified Charpy specimens taken from the weld metal, the heat affected zone and the ASTM reference plate, core region base metal specimens oriented normal (transverse) to the principal rolling direction of the plate rather than parallel to the principal rolling direction.

Five additional capsules, not required by ASTM-E-185 but included in the program, contain both longitudinal and transverse Charpy specimens taken from the limiting core region material, and longitudinal tensile and Charpy specimens of one of the other core region plates; however, they do not contain weld heat affected zone specimens of ASTM reference correlation monitor specimens. The surveillance program does not include thermal control specimens. These specimens were not required since the surveillance specimens are exposed to the combined neutron irradiation and temperature effects and the test results

provide the maximum transition temperature shift. Thermal control specimens as considered in ASTM-E-185 would not provide any additional information on which the operational limits for the reactor are set.

Test stock from four plates (three intermediate shell plates and the most limiting lower shell plate) used in the core region of the reactor vessel, and a weldment containing representative (as deposited) weld metal (but no HAZ representing the limiting plate) were retained as per Section 3.1.2 of ASTM E-185-70.

Chemical analyses (excluding nitrogen and iron) as per Section 3.1.3 of ASTM #-185-70 were obtained for the limiting core region plate and weld metal.

The reactor vessel surveillance program uses eight specimen capsules. The capsules are located about 3 inches from the vessel wall directly opposite the center of the core and are retained in guide baskets welded to the outside of the thermal shield. Sketches of an elevation and plan view showing the location and dimensional spacing of the capsules with relation to the core, thermal shield and vessel are shown in Figures 4.5.1 and 4.5.2, respectively. The capsules can be removed when the vessel head is removed and can be replaced when the internals are removed. The capsules contain reactor vessel steel specimens from shell plates located in the core region of the reactor and from associated weld metal and heat affected zone metal. In addition, three capsules contain correlation monitors made from fully documented specimens of SA-533 Grade B, Class 1 material obtained through Subcommittee II of ASTM Committee E10, “Radioisotopes and Radiation Effects”. The capsules contain tensile specimens, Charpy V-notch specimens (which include weld metal and heat affected zone material) and WOL specimens.

Sixty-four Charpy V-notch specimens (oriented with respect to the weak direction) for one of the lower shell plates were included in the reactor vessel surveillance program. This lower shell plate is the limiting material in the core region as defined by E-185.

Dosimeters, including Ni, Cu, Co-Al, Ed shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238, for capsules V, Y & S were placed in filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys were included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion and to ensure good thermal conductivity.

The complete capsule was helium leak tested. Vessel material sufficient for at least 2 capsules will be kept in storage. This material represents four plates (three intermediate shell plates and the most limiting lower shell course plate) used in the core region of the reactor vessel and a representative weldment. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01% is made for surveillance material and as deposited weld metal.

Each of three capsules contain the following specimens:

CAPSULES V. Y***** and S

Material	No. of Charpys	No. of Tensiles	No. of WOL's
Plate B2803-3*	8	2	-
Weld Metal	8	2	4
Heat affected zone metal	8	-	-
ASTM Reference	8	-	-

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

DOSIMETERS

Copper

Nickel

Cobalt – Aluminum (0.15% Co)

Cobalt – Aluminum (Cadmium shielded)

U-238 (Cadmium shielded)

Np-237 (Cadmium shielded)

THERMAL MONITORS

97.5% Pb, 2.5% Ag (579 F Melting Point)

97.5% Pb, 1.75% Ag, 0.75% Sn (590^{o} F Melting Point)

Five additional capsules contain the following specimens:

Capsules	Material	No. of Charpys	No. of Tensiles	No. of WOL's
W and T ****	Plate B2803-3*	8	-	-
	Plate B2803-3**	8	2	-
	Plate B2802-1***	8	1	6
	Weld Metal	8	-	-
X and U	Plate B2803-3*	8	-	-
	Plate B2803-3**	8	2	-
	Plate B2802-2***	8	2	6
	Weld Metal	8	-	-
Z*****	Plate B2803-3*	8	-	-
	Plate B2803-3**	8	2	-
	Plate B2802-3***	8	2	6
	Weld Metal	8	-	-

* Lower shell plate specimens oriented normal (transverse) to the principal rolling direction of the plate.

** Lower shell plate specimens oriented parallel (longitudinal) to the principal rolling direction of the plate.

*** Intermediate shell plate specimens oriented parallel (longitudinal) to the principal rolling direction of the plate.

**** Capsule T has been removed and analyzed.

***** Capsule Z has been removed and analyzed.

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

Dosimeters

Copper
Nickel
Cobalt-Aluminum (0.15% Co)
Cobalt-Aluminum (Cadmium shielded)

Iron

Thermal Monitors

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

The fast neutron exposure of the specimens occurs at a rate equal to or faster than the maximum exposure experienced by the vessel wall with the specimens being located between the core and the vessel. Since some of these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the RT_{NDT} determinations of these specimens are representative of the vessel at a later time in service.

Data from the WOL fracture toughness specimens provide additional information for use in determining allowable stresses for irradiated material.
The calculated average fast neutron exposure at the vessel clad-base metal interface was 5.86 x 10^8 n/cm^2 (E greater than 1 MeV at the end of Cycle 12). The reactor vessel surveillance capsules are located at 4°, 40°, and 220° as shown in Figure 4.5-2. The design basis lead factor and the plant specific lead factor are listed below.

Capsules at	Design Basis Lead Factor	Plant Specific Lead Factor
4°	1.07	1.30
40°	3.74	3.74
220°	3.46	3.44

Correlations between the calculations and the measurements on the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Appendix 4A. The analysis of the reactor vessel material contained in Surveillance Capsule T for Indian Point 3 was reported in WCAP-9491, April 1979. The analysis of the reactor vessel material contained in Surveillance Capsule Y for Indian Point 3 was reported in WCAP-10300, March 1983. The analysis of the reactor vessel material contained in Surveillance Capsule Z for Indian Point 3 was reported in WCAP-11815, March 1988. The analysis of reactor vessel material contained in Surveillance Capsule X for Indian Point Unit 3 was reported in WCAP – 16251-NP, July 2004. The Capsule T report indicated that the damage rate of the plate and weld metal due to irradiation is in excess of that predicted by the Westinghouse trend curves and that the calculated lead factors were slightly higher than originally estimated. However, due to the effects of low-leakage core loading strategy, measured belting plate and weld metal properties were found to be slightly better than design by the time Capsule Z was analyzed.

The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and readjustment of the calculated wall exposure are made by use of data on all the capsules withdrawn as was done for Capsule T, Capsule Y, and Capsule Z.

The tentative schedule for removal of the capsules is as follows:

Capsule	Location	Lead Factor	Withdrawal Date	Withdrawal EFPY	Capsule Fluence (n/cm2)
T	40^o	3.43	Removed 1978 Refueling Outage	1.4	2.63E+18
Y	40^o	3.49	Removed 1982 Refueling Outage	3.2	6.92E+18
Z	40^o	3.48	Removed 1987 Refueling Outage	5.5	1.04E+19
S	40^o	3.46	Retired in Place	N/A	N/A
X	4^o	1.49	Removed 2003 Refueling Outage	15.5	8.74E+18
U	4^o	1.52	RFO23	Approx. 37 EFPY	Approx. 1.86E+19
V	4^o	1.52	Spare	N/A	N/A
W	4^o	1.52	Spare	N/A	N/A

The times for removal will be adjusted based on the Reactor Vessel Surveillance Program.

4.5.3 Primary System Quality Assurance Program

Table 4.5-1 summarizes the quality assurance program with regard to inspections performed on primary system components, including the replacement steam generators installed during the cycle 6/7 refueling outage. In addition to the inspections shown in Table 4.5-1, there were those performed by the equipment supplier to confirm the adequacy of material received and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME Code requirements. The inspection procedures and acceptance standards required on original pipe materials and piping fabrication were governed by USAS B31.1 and Westinghouse requirements, and were equivalent to those performed on ASME coded vessels. The loop 32 RCS hot leg elbow replaced in conjunction with the steam generator was fabricated and inspected to ASME Code Section III requirements.

Procedures for performing the examinations were consistent with those established in the ASME Code Section III and were reviewed by qualified Westinghouse engineers. These procedures were developed to provide the highest assurance of quality material and fabrication. They considered not only the size of possible flaws, but equally as important, how the material was fabricated, the orientation and type of possible flaws, and the areas of most severe conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming and fabrication processes received a 100% surface inspection by magnetic particle or liquid penetrant testing after all these operations were completed.

All reactor coolant plate material was subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. All forgings received the same inspection. In addition, 100% of the material volume was covered in these tests as an added assurance over the grid basis required in the code.

Westinghouse Quality Control engineers monitored the supplier's work and witnessed 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 test and qualification of supplier personnel. An independent surveillance of the conformance to the fabrication and installation specifications and the quality control requirements of, amongst other things, the original Reactor Coolant System components was carried out by the United States Testing Company for Consolidated Edison. Comparable independent surveillance was also carried out during fabrication and installation of the replacement steam generators.

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.

Consolidated Edison engineers witnessed the hydrostatic test of the reactor vessel.

Field erection and field welding of the Reactor Coolant System during original plant construction were performed so as to permit exact fitting of the 31" 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" 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. During replacement steam generator fabrication and installation, customized steam generator primary nozzle coordinates, temporary pipe restraints, mechanical and optical templating methods, and precision machining were all employed in order to ensure restoration of the RCS to its original configuration.

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.

Section III of the ASME B&PV Code required that nozzles carrying significant external loads be attached to the shell by full penetration welds. This requirement was satisfied for 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 required the use of both preheat and postheat. Preheat requirements, non-mandatory 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. Both preheat and postheat of weldments served a common purpose: the production of tough,

ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones whereas post-heating achieves this by tempering any hard zones which may have formed due to rapid cooling.

Quality control techniques used in the fabrication of the Reactor Coolant System were equivalent to those used in the manufacture of the reactor vessel which conformed to Section III of the ASME Boiler and Pressure Vessel Code.

The piping was designed to the USAS B31.1 (1955) Code for Power Piping using the allowable stresses found in Nuclear Code Cases N-7 and N-10 for pipe and fittings, respectively. Results of piping reanalysis for seismic performance are presented in Section 16.3.5.

While the governing code for design, fabrication, inspection and testing of original RCS piping was USAS B31.1 (1955), the quality assurance requirements imposed by Westinghouse in the purchase and examination of the reactor coolant piping assured that the quality level of the plant is comparable to that delineated by USAS B31.7, Class I, Code for Nuclear Piping. This is demonstrated by the following comparison of original RCS quality assurance measures to selected provisions of USAS B31.7. The RCS reconnection activities associated with the replacement of steam generators during the cycle 6/7 refueling outage were governed by ANSI B31.1-1986 requirements but also met or exceeded the original RCS quality assurance measures, including the measures described below where applicable.

a) All materials conformed to ASTM specifications listed for B31.7 Class I, Nuclear Piping. In addition, all materials were certified, identified, and marked to facilitate traceability thus complying with the requirements of USAS B31.7, Class I, Code for Nuclear Piping

b) Piping base materials were examined by quality assurance methods having acceptance criteria which met the requirements set forth in USAS B31.7, Class I, Code for Nuclear Piping

c) All welding procedures, welding, and welding operators were qualified to the requirements of ASME Section IX, Welding Qualifications, which was in compliance with the requirements of USAS B31.7, Class I, Code for Nuclear Piping

d) All welds were examined by NDT methods and to the extent prescribed in USAS B31.7 for Class I, Nuclear Piping

e) All branch connection nozzle welds of nominal sizes of 3” and larger were 100% radiographed. This exceeded the requirements of USAS B31.7, Class I piping since it included nominal sizes of 6” and larger for 100% radiography

f) All finished welds were liquid penetrant examined on both the outside and inside (if accessible) surfaces as required by USAS B31.7, Class I. In addition, nozzle welds in nominal sizes 2” and smaller were progressively examined after each ¼ inch increment of weld deposit in lieu of radiography

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

Hence, the Westinghouse quality assurance requirements implemented in the procurement of Indian Point 3 piping and fittings were equal to and in some instances exceeded the requirements of USAS B31.7.

The design and stress criteria specified in USAS B31.7 are not directly comparable to that of USAS B31.1 (1955 for piping design and 1967 for piping stress qualification). The following described how USAS B31.1 (1955 and 1967) were used in the primary coolant piping and the ASME B&PV Code Section III, Subsection NB, 1986 Edition for the pressurizer surge line including the effects of thermal stratification in the Indian Point Unit 3 design. A thermal expansion flexibility analysis was performed on the main primary coolant piping and pressurizer surge line (including the effects of thermal stratification) in accordance with the criteria set forth in USAS B31.1 (1955 and 1967) for the reactor coolant piping and the ASME B&PV Code Section III 1986 Edition for the pressurizer surge line including the effects of thermal stratification. For the reactor coolant piping the analysis was performed to ensure that the stress range and number of thermal cycles (usage factor) are safely within the limits prescribed in B31.1. As per the requirements of USAS B31.1, no fatigue analysis is required and hence, no fatigue analysis of the reactor coolant piping is performed. For the pressurizer surge line including the effects of thermal stratification, the analysis was performed to ensure that the stress range and number of thermal cycles (usage factor) are safely within the limits prescribed in ASME B&PV Code Section III, Subsection NB,1986 Edition. In addition, seismic analyses were performed on the composite piping, which included the combined effects of all the sustained (pressure and weight) loading plus seismic vertical/horizontal loading components. The resultant reactions of the piping due to separate and combined effects of thermal, sustained and seismic loading were factored into the checking of the final design of the equipment nozzles with which the piping is interconnected. In turn, the equipment supporting structures were checked for adequate design including the added effects of these same loadings. Thus the total design analysis including pipe, equipment and structures considered the effects of thermal expansion, sustained and seismic loadings with a normal usage factor.

For considering and protecting against the dynamic effects of postulated ruptures, the Reactor Coolant Loop (RCL) LOCA analysis is performed for postulated breaks in the following branch lines:

- The Surge and the Residual Heat Removal (RHR) lines on the hot leg,
- and the Accumulator line in the cold leg.

The RCL is also evaluated for the secondary side breaks at the main steam line and feedwater line terminal end nozzle locations at the steam generator.

Thermally induced stresses arising from temperature gradients were limited to a safe and low order of magnitude in assigning a maximum permissible time rate of temperature change on plant heat up, cool down, and incremental loadings in the plant operation procedure.

An added margin of conservatism was obtained through the use of thermal sleeves in 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.

The use of thermal sleeves was not a specific requirement in B31.7. The seismic reanalysis effort for the Reactor Coolant System piping is described in Section 16.3.5.

Shop and field fabrication requirements, documentation, and quality assurance examinations all complied with those found in USAS B31.7 for Class I Nuclear Piping. Effective with the implementation of the EN Welding Program, in lieu of the above, in-process quality assurance examinations for safety related piping welds and the acceptance criteria will be in accordance with ASME Section III 1992 Edition.

Electroslag Welding

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

1) The electroslag welding procedure employing "one-wire" technique was qualified in accordance with the requirements of ASME B&PV Code, Section IX, and Code Case 1355 plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from a 5 inch thick weldment and successfully tested:
   a) 6 Transverse Tensile Bars – as welded
   b) 6 transverse Tensile Bars – 2050 F, H_20 Quench
   c) 6 Transverse Tensile Bars – 2050 F, H_20 Quench + 750 F stress relief heat treatment
   d) 6 Transverse Tensile Bars – 2050 F, H_20 Quench, tested at 650 F
   e) 12 Guided Side Bend Test Bars

2) The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The acceptance standards were ASTM E-186, Severity level 2 (except no category D or E defectiveness was permitted) and ASME Section III, Paragraph N-627, respectively.

3) The edges of the electroslag weld preparation were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME Section III, Paragraph N-627.

4) The completed electroslag weld surfaces were ground flush with the casting surface. The, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Section III, Paragraph N-627.

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

6) Heat treatment furnace charts were recorded and certified.

Two of the Indian Point 3 reactor coolant pump casings were electroslag welded. The efforts discussed below were performed for quality assurance of the components:

1) The electroslag welding procedure employing "two- and three-wire" techniques was qualified in accordance with the requirements of the ASME B&PV Code Section IX and Code Case 1355 plus supplementary evaluations as required by Westinghouse. The following test specimens were removed from an 8 inch

thick and from a 12 inch thick weldment and successfully tested for both the “2-wire” and “3-wire” techniques, respectively:

a) Two wire electroslag process – 8” thick weldment
   1. 6 Transverse Tensile Bars – 750 F postweld stress relief
   2. 12 Guide Side Bend Test Bars

b) Three wire electroslag process – 12” thick weldment
   1. 6 Transverse Tensile Bars – 750 F postweld stress relief
   2. 17 Guided Side Bend Test Bars
   3. 21 Charpy Vee Notch Specimens
   4. Full section macroexamination of weld and heat affected zone
   5. Numerous microscopic examinations of specimens removed from the weld and heat affected zone regions
   6. Hardness survey across weld and heat affected zone.

2) A separate weld test was made using the “2-wire” electroslag technique to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and techniques as such an occurrence was anticipated during production applications due to equipment malfunction, power outages, etc. The following test specimens were removed from an 8 inch thick weldment in the stop-restart-repaired region and successfully tested:
   a) 2 Transverse Tensile Bars – as welded
   b) 4 Guided Side Bend Test Bars
   c) Full section macroexamination of weld and heat affected zone.

3) All of the weld test blocks in 1) and 2) above were radiographed using a 24 MeV Betatron. The radiographic quality level obtained was between one-half of 1% to 1%. There were no discontinuities evident in any of the electroslag welds.

   a) The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2 except no category D or E defectiveness was permitted for section thickness up to 4½ inches and ASTM E-280 severity level 2 for section thicknesses greater than 4½ inches. The penetrant acceptance standards were ASME B&PV Code Section III, Paragraph N-627.

   b) The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME B&PV Code Section III, Paragraph N-627.

   c) The completed electroslag weld surfaces were ground flush with the casing surface. Then the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME B&PV Code Section III, Paragraph N-627.

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

e) Heat Treatment furnace charts were recorded and certified.

The two remaining Indian Point 3 reactor coolant pump casings were submerged arc welded. Quality Assurance procedures and Quality Assurance inspections equivalent to the above were also exercised on these casings.

4.5.4 Non-Destructive Testing

Section XI of the ASME Boiler and Pressure Vessel Code sets the requirements for both the pre-operational and operational non-destructive testing of nuclear reactor coolant system.

The plant was examined to the fullest extent practical in accordance with Section XI, IS-141 and IS-142, even though this plant was ordered and designed before the code was effective. Examinations nonetheless followed coed requirements wherever the design of the plant allowed.

Non-destructive testing was performed by one of several methods, as specified in Section XI and its applicable reference:

1) Visual Examination
   a) Direct Visual
   b) Remote Visual
   c) Indirect Visual

2) Surface Examination
   a) Magnetic Particle
   b) Liquid Penetrant

3) Volumetric Examination
   a) Radiographic
   b) Ultrasonic

Test personnel were qualified in accordance with all code requirements.

Pre-Service Inspection

Section XI, IS-232 required pre-operational examination of essentially 100% of the pressure containing welds within the reactor coolant system boundary.

The plant components were examined in accordance with the requirements wherever it was possible and practical to do so in order to provide base line date for subsequent inservice inspections.

The pre-service examination for the original plant components was performed at the plant site after the components had been installed. With the exception of the reactor coolant pipe to channel head weld which received a pre-service examination after the replacement steam generators were installed, pre-service examination of replacement steam generator pressure boundary welds was performed at the manufacturer's shop. Primary and

secondary side hydrostatic tests were performed after installation. Personnel qualifications, equipment and records met the requirements of applicable codes. Onsite examinations were necessarily limited by the design and accessibility restrictions of the plant.

In-Service Inspection

Operational examinations as set forth in ASME Section XI are performed to the fullest extent practical at the required intervals.

The structural integrity of the Reactor Coolant System is maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence resulting from the inspections required by the ISI Program and indicating that potential defect implications have initiated or enlarged are investigated, including evaluation of comparable areas of the Reactor Coolant System.

Non-destructive test methods, personnel, equipment and records conform to the requirements of ASME Section XI.

The use of conventional non-destructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several features were incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques as they become available:

1) Shop ultrasonic examinations were performed on all thermally clad surfaces to an acceptance and repair standard which assures an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed was ¼" x ¾"

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

3) To establish baselines for Post-Operational Ultrasonic Testing of the Reactor Vessel, during the manufacturing stage selected areas of the reactor vessel were ultrasonic tested and mapped to facilitate the in-service inspection program.

   The areas 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) Exterior surface of the closure head form the flange knuckle to the cooling shroud
   e) Nozzle to upper shell weld
   f) Middle shell to lower shell weld
   g) Upper shell to middle shell weld.

The pre-operational ultrasonic testing of these areas was performed after hydrostatic testing of the reactor vessel.

A qualified inspector employed by an insurance company authorized to write boiler and pressure vessel insurance certified all examinations.

Means of access to the Reactor Coolant Pressure Boundary were provided as necessary for the surveillance programs as detailed in the ISI Program. This inspection program is in compliance with Section XI of the ASME Code for in-service inspection of nuclear reactor coolant systems.

During the design phase, careful consideration was given to provide access for both visual and non-destructive in-service inspections of the reactor coolant primary and associated auxiliary systems and components within the boundaries established in accordance with the Section XI Code.

Specific provisions made for inspection access in the design of the reactor vessel, system layout and other major primary coolant components were:

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

2) The reactor vessel shell in the core areas was designed with a clean, uncluttered cylindrical inside surface to permit future positioning of test equipment without obstruction

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

4) The cladding to base metal interface was ultrasonically examined to assure satisfactory bonding to allow the volumetric inspection of the vessel welds and base metal from the vessel inside surface

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

6) [Deleted]

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

8) Removable plugs were provided in the primary shield, providing limited access for inspection of the primary nozzle safe-end welds

9) Manways were provided in the steam generator channel head to provide access for internal inspection

10) A manway was provided in the pressurizer top head to allow access for internal inspection

11) The insulation covering all component and piping welds 6 inches in diameter and larger and covering the adjacent base metal was designed for ease of removal and replacement in areas where external inspection is planned

12) Removable plugs were provided in the primary shield concrete above the main coolant pumps to permit removal of the pump motor and to provide internal inspection access to the pumps.

The Indian Point 3 reactor vessel was built to the 1965 edition of the ASME Code Section III and all addenda through the Summer 1965 issue. ASME Section XI in-service inspection was not a requirement at that time. However, accessibility and techniques are available for inspecting all welds requiring inspection by ASME Section XI on the vessel, except for the closure head dome and bottom head dome circumferential welds and the control rod mechanism housing and bottom instrumentation tube attachment welds which were completed prior to the issuance of Section XI. Particular design improvements applied to the reactor vessel to facilitate in-service inspection include Items 1) through 4) and 7) above.

The data and results of the pre-operational examination serve as baseline data for the in-service inspection program.

In-service inspection of seismic Class I pressure retaining components, such as vessels, piping, etc. within the Reactor Coolant Pressure Boundary is performed in accordance with Section XI of the ASME Code, as described in the IP3 Inservice Inspection Program Plan for the applicable interval, with certain exceptions whenever specific relief is granted by the NRC.

The engineered safety features, the reactor shutdown systems, the cooling water systems, and the radioactive waste treatment systems which are necessary for plant operation are provided as redundant systems. This redundancy provides the capability for system and/or component outage (per Technical Specification requirements) to perform operability tests/checks or repair/maintenance. Periodic testing is in accordance with Technical Specification requirements and the IP3 Pump and Valve Testing Plan. The Pump and Valve Testing Plan is in accordance with ASME Section XI with certain exceptions whenever specific relief is granted by the NRC.

Pre-operational Vibration Test Program

During hot functional testing, the piping was observed and any vibration problems were eliminated. Also, any other piping vibrations that were deemed excessive were eliminated. Observation for piping vibration was made by persons experienced in piping design when systems were operated in normal modes during hot functional testing. When piping vibrations were observed, and evaluation was made to determine corrective action.

Class I (seismic) systems were checked out and run prior to hot functional testing in accordance with Section 13.1.

During the normal course of the pre-operational test program, specific attention was directed to evaluating possible vibration problems during the performance of the following transients:

PRE-OPERATIONAL TEST	SPECIFIC TRANSIENTS
1. Reactor Coolant System Heatup	Operational Test of Charging Pumps (Step Changes) Reactor Coolant Pump Start Operation of Pressurizer Power-Operated Relief Valves Operation of Pressurizer Spray Valves Operation of Letdown Isolation Valves
2. Reactor Coolant System at Temperature	Operation of Pressurizer Power-Operated Relief Valve Reactor Coolant Pumps (Stopping and Starting)
3. Reactor Coolant System Cooldown	Initiation of Residual Heat Removal
4. Emergency Core Cooling Full Flow Test	Initiation and Termination of the Following: A. Safety Injection Pumps B. Residual Heat Removal Pumps
5. Chemical and Volume Control System Test	Operational Test of Positive Displacement Charging Pumps (Stop and Start)

Amplitudes of vibration will theoretically cause the pipe to reach its elastic limit. Charts or monographs were provided during pre-operational testing to define these amplitudes as a function of pipe size, span and schedule as an aide for the operator and cognizant engineer to determine acceptability.

The acceptance of an observed vibration was based on operator and cognizant engineer experience. In addition, systems and components were physically examined (visually) for the following types of deficiencies which are indicative of a possible vibration problem:

1) Cracks in the grouting of equipment foundations
2) Leaking gaskets in piping systems and pump seals
3) Leaks from flanged connections in piping systems
4) Metal to metal contact indications on piping systems restraints.

If the above types of indications were observed, further investigation was performed to establish and correct any adverse conditions.

TABLE 4.5-1

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*	LT*
1. Steam Generator						
1.1 Tube Sheet						
1.1.1 Forging		yes		yes		
1.1.2 Cladding		yes(1)	yes			
1.2 Channel Head						
1.2.1 Forging		yes		yes		
1.2.2 Cladding		yes(1)	yes			
1.3 Secondary Sheet & Head						
1.3.1 Plates		yes				
1.3.2 Shell Transition Cone (forging)		yes		yes		
1.4 Tubes		yes			yes	
1.5 Nozzles (forgings)		yes		yes		
1.6 Weldments						
1.6.1 Shell, longitudinal	yes			yes		
1.6.2 Shell, circumferential	yes			yes		
1.6.3 Cladding (Channel Head-Tube Sheet joint cladding restoration)		yes(1)	yes			
1.6.4 Steam and Feedwater Nozzle to shell	yes			yes		
1.6.5 Support brackets				yes		
1.6.6 Tube to tube sheet			yes			yes
1.6.7 Instrument connections (primary and secondary)				yes		
1.6.8 Temporary attachments after removal				yes		
1.6.9 After hydrostatic test (all shell welds and Tube-sheet to channel head)				yes		
1.6.10 Nozzle safe ends (weld deposit)	yes		yes			
2. Pressurizer						
2.1 Heads						
2.1.1 Casting	yes			yes		
2.1.2 Cladding			yes			
2.2 Shell						
2.2.1 Plates		yes		yes		
2.2.2 Cladding			yes			

TABLE 4.5-1
(Cont.)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*	LT*
2.3 Heaters						
2.3.1 Tubing(++++)		yes	yes			
2.3.2 Centering of element	yes					
2.4 Nozzle		yes	yes			
2.5 Weldments						
2.5.1 Shell, longitudinal	yes			yes		
2.5.2 Shell, circumferential	yes			yes		
2.5.3 Cladding			yes			
2.5.4 Nozzle Safe End (if forging)	yes		yes			
2.5.5 Nozzle Safe End (if weld deposit)			yes			
2.5.6 Instrument Connections			yes			
2.5.7 Support Skirt				yes		
2.5.8 Temporary Attachments after removal				yes		
2.5.9 All welds and cast heads after hydrostatic test				yes		
2.6 Final Assembly						
2.6.1 All accessible surfaces after hydrostatic test				yes		
3. Piping						
3.1 Fittings (Castings)	yes		yes			
3.2 Fitting (Forgings)		yes	yes			
3.3 Pipe		yes	yes			
3.4 Weldments						
3.4.1 Circumferential	yes		yes			
3.4.2 Nozzle to run pipe (no RT for nozzles less than 3 inches)	yes		yes			
3.4.3 Instrument connections			yes			
4. Pumps						
4.1 Casting	yes		yes			
4.2 Forgings		yes	yes			
4.2.1 Main Shaft		yes	yes			
4.2.2 Main Studs		yes	yes			
4.2.3 Flywheel (Rolled Plate)		yes				
4.3 Weldments						
4.3.1 Circumferential	yes		yes			
4.3.2 Instrument connections			yes			

TABLE 4.5-1
(Cont.)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*	LT*
5. Reactor Vessel						
5.1 Forgings						
5.1.1 Flanges		yes		yes		
5.1.2 Studs		yes		yes		
5.1.3 Head Adapters		yes	yes			
5.1.4 Head Adapter Tube	yes	yes				
5.1.5 Instrumentation Tube		yes	yes			
5.1.6 Main Nozzles		yes		yes		
5.1.7 Nozzle Safe-Ends (If forging is employed)		yes	yes			
5.2 Plates		yes		yes		
5.3 Weldments						
5.3.1 Main Seam	yes			yes		
5.3.2 CRD Head Adapter Connection			yes			
5.3.3 Instrumentation tube connection			yes			
5.3.4 Main nozzles	yes			yes		
5.3.5 Cladding		yes(+++)	yes			
5.3.6 Nozzle Safe-Ends (If forging)		yes				
5.3.7 Nozzle Safe-Ends (If weld deposit)	yes		yes			
5.3.8 Head adapter forging to head adapter tube	yes		yes			
5.3.9 All welds after hydrotest				yes		
6. Valves						
6.1 Castings	yes		yes			
6.2 Forgings (No UT for valves two inch and smaller)		yes	yes			

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

(+) Flat Surfaces Only
(++) Weld Deposit Areas Only
(+++) UT of Clad Bond-to-Base Metal
(++++) Or a UT and ET
(1) For clad defects and for bond to base metal

APPENDIX 4A

NEUTRON DOSIMETRY

1. MEASUREMENT OF INTEGRATED FAST NEUTRON FLUX

In order to obtain a correlation between fast neutron (E greater than 1.0 MeV) exposure and the changes observed in radiation induced properties in the test specimens, a number of fast neutron monitors are included as an integral part of the Reactor Vessel Surveillance Program. In particular, the surveillance capsules contain detectors employing the following reactions:

Fe^{54} (n,P) Mn 54 (Fe from test specimens)
Ni^{58} (n,P) Co^{58}
Cu^{63} (n, ~) Co^{60}
Np^{237} (n,f) CS^{137}
U^{238} (n,f) Cs^{137}

In addition, thermal neutron flux monitors, in the form of bare and cadmium shielded Co-Al wire, are included within the capsules to enable an assessment of the effects of isotopic burnup on the response of the fast neutron detectors.

The use of activation detectors such as those listed above does not yield a direct measure of the energy dependent neutron flux level at the point of interest. Rather, the activation process is a measure of the integrated effect that the time and energy dependent neutron flux on the target material. An accurate estimate of the average neutron flux level incident on the various detectors may be derived from the activation measurements only if the parameters of the irradiation are well known. In particular, the following variables are of interest:

1. The operating history of the reactor
2. The energy response of the given detector
3. The neutron energy spectrum at the detector location
4. The physical characteristics of the detector

The procedure for the derivation of the fast neutron flux from the results of the Fe^{54}(n,P)Mn^{54} reaction is described below. The measurement technique for the other dosimeters, which are sensitive to different portions of the neutron energy spectrum, is similar.

The Mn^{54} product of the Fe^{54}(n,P)Mn^{54} reaction has a half-life of 314 days and emits gamma rays of 0.84 MeV energy which are easily detected using gamma spectrometry. In irradiated steel samples, chemical separation of the Mn^{54} may be performed to ensure freedom from interfering activities. This separation is simple and very effective, yielding sources of very pure Mn^{54} activity. In some samples, all of the interferences may be corrected for by the gamma spectrometric methods without any chemical separation.

The analysis of the sample requires that two procedures be completed. First, the Mn^{54} disintegration rate per unit mass of sample and the iron content of the sample must be measured as described above. Second, the neutron energy spectrum at the detector location must be calculated.

For this analysis, the DOT[1] two-dimensional multigroup discrete ordinates transport code is employed to calculate the spectral data at the location of interest. Briefly, the DOT calculations utilize a 21 group energy scheme, an S_8 order of angular quadrature, and P_3 (Reference 2) expansion of the scattering matrix to compute neutron radiation levels within the geometry of interest. The reactor geometry employed here includes a description of the radial regions internal to the primary concrete (core barrel, neutron pad, pressure vessel and water annuli) as well as the surveillance capsule and an appropriate reactor and core baffle description. Thus, distortions in the fission spectrum due to the attenuation of the reactor internals are accounted for in the analytical approach.

Having the measured activity, sample weight, and neutron energy spectrum at the location of interest, the calculation of the threshold flux is as follows:

The induced Mn^{54} activity in the iron flux monitors may be expressed as

$$D = (No/A) f_i y \int_E \sigma(E)\phi(E) \, dE \sum_{J=1}^{N} F_J \left(1 - e^{-\lambda T_J}\right) e^{-\lambda(T - T_J)}$$

where:	D	=	Induced Mn^{54} activity	(dps/ gmFe)
	No	=	Avogadro's number	(atoms/ gm-atom)
	A	=	Atomic weight of iron	(gm/ gm-atom)
	f_i	=	Weight fraction of Fe^{54} in the detector	
	y	=	Number of product atoms produced per reaction	
	$\sigma(E)$	=	Energy dependent activation cross-section for the Fe^{54}(n,P)Mn^{54} reaction	(barns)
	$\phi(E)$	=	Energy dependent neutron flux at the detector at full reactor power	(n/ cm^2-sec)
	λ |	=	Decay constant of Mn^{54}	(sec-1)
	F_J	=	Fraction of full reactor power during the Jth time interval, T_J	
	t_J	=	Length of the J^{th} irradiation period	(sec)
	T	=	Elapsed time between initial reactor startup and sample counting	(sec)

The parameters F_J, t_J, and T depend on the operating history of the reactor and the delay between capsule removal and sample counting.

The integral term in the above equation may be replaced by the following relation:

$$\int_E \sigma(E)\phi(E)dE = \overline{\sigma}\,\overline{\phi}E_{TH} = \overline{\phi\quad} E_{TH} \sum_{E_{TH}}^{\infty} \sigma_S(E)\phi_S(E) \Big/ \sum_{E_{TH}}^{\infty} \phi_S(E)$$

where: $\overline{\sigma}$ = Effective spectrum average reaction cross-section for neutrons above energy, E_{TH}

$\overline{\phi}\ E_{TH}$ = Average neutron flux above energy, E_{TH}

$\sigma_S(E)$ = Multigroup Fe^{54}(n,P)Mn^{54} reaction cross-section

compatible with the DOT energy group structure

$\phi_s(E)$ = Multigroup energy spectra at the detector location

obtained

from the DOT analysis

Thus,

$$D = (No/A) f_i\, y\, \overline{\sigma}\,\overline{\phi}E_{TH} \sum_{J=1}^{n} F_J \quad \left(1 - e^{-\lambda T_J}\right) e^{-\lambda(T - T_J)}$$

or, solving for the threshold flux

$$\overline{\phi}E_{TH} = D/(No/A) f_i\, y\, \overline{\sigma} \quad \sum_{J=1}^{n} F_J \left(1 - e^{-\lambda T_J}\right) e^{-\lambda(T - T_J)}$$

The total fluence above energy E_{TH} is then given by

$$\phi_{E_{TH}} = \overline{\phi}E_{TH} \sum_{J=1}^{n} F_J T_J$$

where $\sum_{J=1}^{n} F_J T_J$ represents the total effective full power seconds of reactor

operation up to the time of capsule removal.

Because of the relatively long half-life of Mn^{54}, the fluence may be accurately calculated in this manner for irradiation periods up to about two years. Beyond this time, the calculated average flux begins to be weighted toward the later stages of irradiation and some inaccuracies may be introduced. At these longer irradiation times, therefore, more reliance must be placed on the Np^{237} and U^{238} fission detectors with their 30 year half-life product (Cs^{137}).

No burnup correction was made to the measured activities, since burnout of the Mn^{54} product is not significant until the thermal flux level is about 10^{14} n/cm^2-sec.

The error involved in the measurement of the specific activity of the detector after irradiation is estimated to be 6 5 percent.

2. CALCULATION OF INTEGRATED FAST FLUX

The energy and spatial distribution of neutron flux within the reactor geometry is obtained from the DOT(1) two dimensional Sn transport code. The radial and azimuthal distributions are obtained from an R, u computation wherein the reactor core as well as the water and steel annuli surrounding the core are modeled explicity. The axial variations are then obtained from an R, Z DOT calculation using the equivalent cylindrical core concept. The neutron flux at any point in the geometry is then given by

$$\phi(E,R,\theta,Z)=\phi(E,R,\theta) \quad F(Z)$$

Where $\phi(E,R,\theta)$ is obtained directly from the R,θ calculation and $F(Z)$ is a normalized function obtained directly from the R,Z analysis. The core power distributions used in both the R,θ and R,Z computations represent the expected average over the life of the station.

Having the calculated neutron flux distributions within the reactor geometry, the exposure of the capsule as well as the lead factor between the capsule and the vessel may be determined as follows:

The neutron flux at the surveillance capsule is given by

$$\phi c \quad = \quad \phi(E, R_c, \theta_c, Z_c)$$

and the flux at the location of peak exposure on the pressure vessel inner diameter is

$$\phi_{v-\max} \quad = \quad \phi(E, R_v, \theta_{v-\max}, Z_{v-\max})$$

The lead factor then becomes

$$LF = \quad \phi_c / \phi_{v-\max}$$

Similar expressions may be developed for points within the pressure vessel wall; and, thus, together with the surveillance program dosimetry, serve to correlate the radiation induced damage to test specimens with that of reactor vessel.

The specific activity of each of the activation monitors is determined by using established ASTM procedures.

References

1. R. G. Soltesz, et al., “Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Volume 5 – Two-Dimensional Discrete Ordinates Technique, “ WANL-PR-(LL)-034, Aug. 1970

2. WCAP-11057, “Indian Point Unit 3 Reactor Vessel Fluence and RT-PTS Evaluations for Consideration of Life Extension,” Westinghouse Electric Corp, June 1989 Rev. 1

APPENDIX 4B

EVALUATION OF REACTOR COOLANT SYSTEM AND SUPPORTS UNDER COMBINED SEISMIC AND BLOWDOWN LOADS

1. Description of reactor coolant system component support structures
2. Analysis of reactor coolant system and supports under combined loads

APPENDIX 4C

PROCEDURE FOR PLUGGING A TUBE IN A STEAM GENERATOR (DELETED)

APPENDIX 4D

SENSITIZED STAINLESS STEEL

Introduction

Westinghouse has evaluated the use of sensitized stainless steel for reactor components in pressurized water reactors. The results of this evaluation are summarized in WCAP-7477-L (Westinghouse proprietary) which covers the nature of sensitization conditions leading to stress corrosion and associated problems with both sensitized and non-sensitized stainless steel. The results of extensive testing and service experience that justify the use of stainless steel in the sensitized condition for components in Westinghouse systems is presented in the report.

Sensitized stainless steel is subject to stress corrosion, and must not be exposed to certain environments which will cause cracking. Chlorides and flourides are the most important contaminants, although oxygen, low pH, elevated temperature and high stress generally must also be present to cause cracking. When subjected to environments that cause cracking, the cracks are usually intergranular in sensitized stainless steel.

The stainless steel safe ends on the reactor vessel, pressurizer, and steam generator nozzles may become somewhat sensitized during stress relief of the vessel. The Post Weld Heat Treatment (PWHT) temperatures and minimum time are consistent with ASME Section III requirements. The degree of sensitization of the safe ends varies from plant to plant, depending on the materials used and the detailed processing performed by the various vendors. For Indian Point 3, the specific design and construction practices are discussed in the following sections. The outer diameter and inner diameter safe ends of the reactor vessel were overlayed with type 312L and Inconel weld metal to eliminate the exposure of sensitized stainless steel in areas where there is limited accessibility for inservice inspection and plant maintenance. There is complete accessibility to the remaining RCS components. The pre-operational inspection of the RCS components provided assurance that there was no stress corrosion cracking of sensitized stainless steel.

All core structural load bearing members were made from annealed type 304 stainless steel, so there is no possibility of sensitization, with the exception of the core barrel itself, which required stress relief during manufacture at temperatures over 750 F. The stress relieving operation was conducted in a manner to minimize the possibility of severe sensitization, while maintaining the necessary conditions for relieving residual fabrication stresses. This consisted of heating to 1650 F to 1750 F, holding at this temperature for several hours, then cooling very slowly in the furnace. This treatment results in massive carbide precipitation at the grain boundaries, and agglomeration of the carbides, instead of the formation of detrimental continuous carbide films. Further, the long times at high temperatures cause diffusion of chromium into the grain boundary areas that were depleted in chromium by the precipitation of chromium carbides. This combination of formation of massive carbides, plus diffusion of chromium back into the depleted zone is referred to as "desensitization", and is commonly used to prevent severe sensitization of parts requiring heat treatments that otherwise would cause severe sensitization of the material. Stress tests run according to ASTM A393 were performed on core barrel material given this heat treatment, and results verified that severe sensitization is prevented. Material that does not or would not be expected to pass ASTM A393 is considered to be severely sensitized.

Severe sensitization of component parts of the Reactor Coolant Pressure Boundary was avoided by the following methods:

- Reactor Vessel - The type 316 safe ends were overlaid with Inconel and type 308L weld metal on the ID and OD after post welding heat treatment.
- Steam Generators - The safe ends were made of weld metal, type 309 and 308L, containing enough ferrite to preclude severe sensitization.
- Pressurizer - The nozzles were made of type 316 stainless steel, but the very short post welding heat treatment time used (less than 10 hours) is not expected to cause severe sensitization of type 316 which is more resistant to sensitization than type 304.

The welding processes employed for field use were shielded metal arc welding (SMAW) and gas tungsten arc welding (GTAW). Both welding processes were used individually or in combination to qualify procedures per the requirements of ASME Code Section IX. Quality controls employed included the use of qualified weld and inspection procedures as well as verification of the maximum interpass temperatures by use of Tempil-Stiks or contact pyrometers.

The effect of nitrogen addition on the corrosion resistance of stainless steel in the PWR environment is discussed in WCAP-7735, "Topical Report - Sensitized Stainless Steel in W PWR NSSS", August 1971.

Further justification that nitrogen addition does not adversely affect the corrosion resistance of sensitized austenitic stainless steel can be obtained from the following literature:

1) C. J. Smithells, Metal Reference Book, Vol. II, p. 621, Plenum Press (1967).

2) L. R. Scharistein, "Effects of Residual Elements on the General Corrosion Resistance of Austenitic Stainless Steels", Effects of Residual Elements on Properties of Austenitic Stainless Steels, ASTM, STP 418 (1967).

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

4) Jones and Laughlin data, sheet-type 304-N Stainless Steel, Jones and Laughlin Steel Corp., Stainless and Strip Division, Warren, Michigan.

All piping in Indian Point 3 was fabricated in a manner to assure that it will not be sensitized. All pipes and fittings were purchased in the sensitized (carbide solution treated) condition. Heat treatment after bending was also a carbide solution treatment at 1900 F or above. No heat treatment was permitted after welding. Welding was done with closely controlled interpass temperature, both in the shop and in the field, to assure freedom from sensitization.

Reactor Coolant System Nozzle Safe Ends

1. Reactor Vessel Primary Nozzle Safe Ends

A. Method of Fabrication (See Figure 4D-1)

1) Wrought Stainless Steel - A-508, class 2 nozzle forging clad with 308 stainless steel welded to type 316 forging with Inconel weld metal. Attached prior to final post weld heat treatment.

2) Forging was overlayed on ID and OD with type 312L stainless and Inconel weld metal. This was performed in the shop after the final post weld heat treatment.

B. Inspection

1) Forging Safe Ends were examined by UT and PT at Combustion Engineering using Section III acceptance standards.

2) Weld overlay of the ID and OD surfaces was examined by UT and PT. The acceptance standards are shown below:

Ultrasonic Acceptance Standards

Rejectable Defect Indications:

a) Those exceeding 90% screen height and exceeding 1/2" length
b) Those exceeding 90% screen height and 1/2" or less in length if not separated by 2" from a similar indication
c) Those with range of 50% to 90% screen height and exceeding 1-1/2" in length
d) Those with range of 50% to 90% screen height and 1" to1-1/2" in length if not separated by 2" from a similar indication.

Penetrant Inspection Acceptance Standards

The following relevant indications were unacceptable:
a) Any cracks and linear indications
b) Rounded indications with dimensions greater than 3/16"
c) Four or more rounded indications in a line separated by 1/16 in or less edge-to-edge
d) Ten or more rounded indications in any six square inches of surface with the major dimension of this area not to exceed six inches with the area taken in the most unfavorable location relative to the indications being evaluated.

2. Steam Generator Primary Nozzle Safe Ends (See Figure 4D-2)

   A. Method of Fabrication

   308 stainless steel weld metal buttering applied to low alloy steel (SA508 Class 3 forging) nozzles prior to final post weld heat treatment. Stainless weld metal for the first layer is type 309L (modified) and for the balance is type 308L.

   B. Inspection

   Buttered safe ends were examined by PT and RT using ASME B&PV Code Section III acceptance standards.

3. Pressurizer (See 4D-3)

   A. Method of Fabrication

   Wrought stainless steel pipe or Type 316 forgings welded to carbon steel (A-216 Grade WCC Casting with 308 stainless steel cladding) nozzles with type 309 (modified) and 308L weld metal before PWHT. The surge nozzle safe end is fabricated from SA- 312 pipe, type 316 and the spray, relief, and safety nozzle safe ends from SA-182 forgings, type 316.

   B. Inspection

   Wrought material was examined by UT and PT using Section III acceptance standards.
   Reactor Coolant System Construction

All primary piping and fittings were given a solution annealing treatment consisting of heating to 1900 - 1950 F, holding 1 hour per inch of thickness and water quenching. This assured that the material would not be sensitized.

Main coolant pipe welds are of type 308L or 316 stainless steel. Welding was performed during original plant construction by the manual metal arc process after the root pass was completed using an insert followed by three layers using the manual gas shielded tungsten arc process. The maximum energy input possible with the manual metal arc process is on the order of 20,000 joules per linear inch of weld. With the large heat sink available in this thick walled pipe (2.375 to 3.00"), and the interpass temperature control of 350 F maximum, there was no sensitization of the solution treated pipe during welding.

Comparable welding controls to avoid primary piping sensitization were also employed during steam generator replacement, however, automatic gas metal arc welding processes were used after the root and hot passes were manually completed. The use of inserts was not required during the reconnection of primary piping to the replacement steam generators.

Venting provisions were made at high points throughout the Reactor Coolant System to relieve entrapped air when the system is filled and pressurized. Principally, vents were installed on the reactor vessel head, the pressurizer, and the reactor coolant pumps. Additional vents are available on the control rod drive mechanisms, on instruments, and on a number of connecting

pipes. For normal venting of the Reactor Coolant System, only the principal venting points are utilized. The amount of oxygen which could be trapped in the remaining small volumes becomes negligible as the system is pressurized and the oxygen is scavenged by the hydrazine, specifically added for this purpose prior to operation. During operation, the oxygen levels are kept low consistent with water chemistry requirements as described in the Technical Requirements Manual.

Reactor Coolant System Operational Stresses

To avoid unusual stresses in areas where nozzle safe ends are joined to the piping, precautions were taken to eliminate unnecessary stresses due to erection of the various components of the Reactor Coolant System. The primary coolant system piping closure pieces are two pipe fitting subassemblies located between the steam generator and the primary coolant pump. The 40 degree elbow of the loop piping was first installed on the steam generator outlet nozzles. Then the gap to be closed by the closure pieces was physically measured between the 40 degree elbow outlet and the inlet nozzle of the pump. These measured dimensions for each individual loop were compensated and adjusted for the expected field weld shrinkage. The resulting net true dimensions were then transmitted to the pipe shop fabricator who prepared the final closure pipe subassemblies for each primary coolant loop. Upon welding these specially dimensioned pipe subassemblies in place, the primary coolant system closure was accomplished for each loop in a condition which was free from cold spring. During steam generator replacement, customized steam generator primary nozzle coordinates, temporary pipe restraints, mechanical and optical templating methods, and precision machining were all employed to ensure restoration of the RCS to its original configuration.

As a precaution that the behavior of the Reactor Coolant System during operating conditions was as predicted, measurements were made during incremental temperature increases during the hot functional test. The measurements were made to check the movement of the components at temperature and pressure to insure interferences were not present. The data taken during the test were compared with the flexibility analysis predictions and evaluated.

Inservice Inspection Capability

As a final check on the adequacy of the precautions taken to avoid any Reactor Coolant System failure as a result of severely sensitized stainless steel, a post-operational inspection plan was developed for the nozzle safe ends within the Reactor Coolant System Boundary. The pressurizer and steam generator stainless steel safe ends which were subjected to the furnace atmosphere during final stress relief are accessible for visual, surface and volumetric inspection upon removal of the insulation at each safe end. The reactor vessel safe ends which were subjected to the furnace atmosphere are accessible for limited inspection by removal of the special access plugs provided in the primary concrete just above each nozzle. Upon removal of these plugs and the insulation of the safe end, approximately 120 degrees of the top segment of the safe ends are accessible for direct visual and surface examination.

A specially designed in-vessel, remote, ultrasonic, inservice inspection tool was developed which can be affixed to the upper vessel flange after removal of the head. This tool is intended for ultrasonic examination of the vessel circumferential and longitudinal welds, nozzle-to-vessel welds and nozzle-to-safe-end welds. Some of these examinations utilizing this invessel tool were performed in the 1979 refueling outage. No indications were revealed by this testing.