CHAPTER 4
REACTOR COOLANT SYSTEM

4.0 GENERAL DESCRIPTION

The reactor coolant system, shown in Plant Drawing 9321-2738 [Formerly UFSAR Figure 4.2-1], consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant circulation pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control.

4.1 DESIGN BASES

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 Section 3.2. 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 any uncontrolled release to the secondary system and to other parts of the plant to acceptable values under conditions of either normal or abnormal reactor behavior. During transient operation, the system heat capacity attenuates thermal transients generated by the core or extracted by the steam generators. 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 coastdown that would result from a loss-of-offsite power situation. The layout of the system ensures the natural circulation capability following a loss of offsite power, to permit decay heat removal without overheating the core. Portions of the system piping are 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 (GDC) that apply to the reactor coolant system are given below.

4.1.2.1 Quality Standards

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

The reactor coolant system is of primary importance with respect to its safety function in protecting the health and safety of the public.

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

4.1.2.2 Performance Standards

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

All piping, components, and supporting structures of the reactor coolant system are designed to Class I requirements, as discussed in Sections 1.11 and 4.1.4.3.

The reactor coolant system is located in the containment whose design, in addition to being a Class I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Chapter 5.

4.1.2.3 Records Requirements

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

Records of the design of the major reactor coolant system components and the related engineered safety features components are maintained for the life of the plant.

Records of fabrication are maintained in the manufacturers' plants as required by the appropriate code or other requirements pending submittal to Westinghouse or ENIP2. They are available at any time throughout the life of the plant. Records of changes made to the plant as described in the FSAR are maintained for the life of the plant.

4.1.2.4 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)

4.1.2.4.1 Original Design Basis

The dynamic effects during blowdown following a loss-of-coolant accident are evaluated in the detailed layout and design of the high-pressure equipment and barriers that afford missile protection. Support structures are designed with consideration given to fluid and mechanical thrust loadings.

Original plant design basis required that the steam generators be supported, guided, and restrained in a manner that 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.

Original plant design basis also required that the mechanical consequences of a pipe rupture as a result of forces created by a reactor coolant system pipe rupture be restricted by design such that the functional capability of the engineered safety features would not be impaired.

4.1.2.4.2 Revised Design Basis

In 1989, the NRC approved elimination of the necessity for considering and protecting against dynamic effects of postulated primary loop pipe ruptures from the design basis of Indian Point Unit 2. "Leak before break" technology was applied as permitted by revised General Design Criterion 4 of 10CFR50, Appendix A. References 1, 2, 3 and 4 contain further information.

With the elimination of the necessity for considering and protecting against the dynamic effects of postulated primary loop ruptures from the design basis of Indian Point 2, breaks have been postulated in the following branch lines: the Accumulator branch line in the cold leg, the Pressurizer Surge line and the Residual Heat Removal (RHR) line in the hot leg. This is discussed in Reference 5. These breaks are the new design basis breaks with respect to considering and protecting against the dynamic effects of postulated ruptures.

In general, these new breaks are significantly less severe than the original design basis breaks. The following is a description of how the dynamic effects of these new breaks have been considered and protected against. The dynamic effects have been grouped into three categories as follows:

1. Containment subcompartment pressurization
2. Break reaction forces (i.e. forcing function analysis, asymmetric blowdown loading) used in structural support design and for confirming the structural integrity of systems and components
3. Missile Protection (pipe whip, jet impingement and missiles)

4.1.2.4.2.1 Containment Subcompartment Pressurization

The dynamic effects of these new breaks with respect to containment subcompartment pressurization is discussed in Section 14.3.5.4.3.2.

4.1.2.4.2.2 Break Reaction Forces

As discussed in Reference 5, the new break locations were used to develop hydraulic forcing functions for revised structural analyses. These analyses demonstrate that for the new break locations the structural integrity of the reactor vessel internals, core components including fuel assemblies, and the reactor coolant loop will be maintained and will preserve the ability to maintain a coolable geometry for the rated stretch power conditions. The dynamic effects of postulated breaks with respect to break reaction forces is discussed further in Sections 1.3.7, 1.11.3, 4.2.4, 4.3.1.3, Appendix 4B.7, 5.1.1.1.5, 5.1.5, 6.1.1.5, and 6.2.2.5.

4.1.2.4.2.3 Missile Protection

Since the new break locations remain inside the missile barrier, engineered safety features and associated systems remain protected from loss of function due to dynamic effects and missiles, which might result from these breaks. This protection is discussed further in Sections 1.3.7, 4.2.4, 5.1.2.5, 6.1.1.3, 6.2.2.5, and 8.2.2.6.

4.1.3 <u>Principal Design Criteria</u>

The criteria that apply solely to the reactor coolant system are given below.

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

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

The reactor coolant system in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions and to maintain the stresses within applicable code stress limits.

Fabrication of the components that constitute the pressure retaining boundary of the reactor coolant system is carried out in strict accordance with the applicable codes. In addition, there are 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 that might otherwise reduce the system's structural integrity during its service lifetime.

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

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

4.1.3.2 Monitoring Reactor Coolant Leakage

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

Positive indications in the control room of leakage of coolant from the reactor coolant system to the containment are provided by equipment that 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 provides indication of normal environmental conditions within the containment. Any increase in the observed parameters could be an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate runoff, and in the case of significant leakage, the liquid inventory in the process systems and containment sump.

Further details are supplied in Sections 4.2.7 and 6.7.

4.1.3.3 Reactor Coolant Pressure Boundary Capability

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

The reactor coolant boundary is shown to be capable of accommodating without further rupture the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Section 14.2.6.10.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, only the rod cluster control assemblies in the controlling groups are inserted in the core at power, and at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to ensure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and positions as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value that 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 is evaluated as a theoretical, though 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 are shown to be adequately protected. Refer to Section 14.2.6.

4.1.3.4 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 provisions for control over service temperature and irradiation effects, which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those, which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes. (GDC 34)

The reactor coolant pressure boundary is designed to reduce to an acceptable level the probability of a rapidly propagating type failure. In the core region of the reactor vessel it is expected that the notch toughness of the material will change as a result of fast neutron exposure. This change is evidenced as a shift in the nil-ductility transition temperature (NDTT), which is factored into the operating procedures in such a manner that full operating pressure is not obtained until the affected vessel material is above the design transition temperature (DTT) in the ductile material region. The pressure during startup and shutdown at the temperature below NDTT is maintained below the threshold of concern for safe operation.

The DTT is a minimum of NDTT plus 60°F and dictates the procedures to be followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of the DTT is increased during the life of the plant as required by the expected shift in NDTT and as confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the plant lifetime. Further details are given in Section 4.1.6.

All pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested in conformance with the applicable codes. Further details are given in Section 4.1.7.

4.1.3.5 Reactor Coolant Pressure Boundary Surveillance

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

The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces 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. Monitoring of the nil-ductility transition temperature properties of the core region plates, forgings, weldments, and associated heat-treated zones is performed in accordance with ASTM E185

(Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors). Samples of reactor vessel plate materials are retained and cataloged in case further engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The fracture mechanics specimens are the wedge opening loading type specimens. The observed shifts in nil ductility transition temperature of the core region materials with irradiation are used to confirm the calculated limits to startup and shutdown transients.

To define permissible operating conditions below the design transition temperature, a pressure range is established, which is bounded by a lower limit for pump operation and an upper limit, which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected design transition temperature, brittle fracture during normal operation is not considered to be a credible mode of failure.

4.1.4 Design Characteristics

4.1.4.1 Design Pressure

The reactor coolant system design and operating pressure together with the safety, power relief, and pressurizer spray valves setpoints and the protection system setpoint pressures are listed in Table 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.

4.1.4.2 Design Temperature

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

4.1.4.3 Seismic Loads

The seismic loading conditions are established by the "design earthquake" and "maximum potential earthquake." The former is selected to be typical of the largest probable ground motion based on the site seismic history. The latter is selected to be the largest potential ground motion at the site according to seismic and geological factors and their uncertainties.

For the design earthquake loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. Therefore, for this loading condition critical structures and equipment needed for this purpose are required to operate within normal design limits. The seismic design for the maximum potential earthquake is intended to provide a margin in design that ensures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the reactor coolant system components do not lose their capability to perform their safety function. This has come to be referred to as the "no-loss-of function" criteria and the loading condition as the "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 Section 1.11. These criteria ensure the integrity of the reactor coolant system under seismic loading.

For the combination of normal and design earthquake loadings, the stresses in the support structures are kept within the limits of the applicable codes.

For the combination of normal and no-loss-of-function earthquake loadings, the deflections and stresses in the support structures are limited to values as necessary to ensure their integrity and to maintain supported equipment within their stress limits as stated in Table 1.11-2.

4.1.5 Cyclic Loads

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operations. The number of thermal and loading cycles used for design purposes and the bases thereof are given in Table 4.1-8. There is a station program, which tracks these thermal and loading cycles. 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. For example, the number of cycles for plant heatup and cooldown at 100°F per hr was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number could be increased significantly; however, it is the intent to represent a conservative realistic number rather than the maximum allowed by the design.

Although loss-of-flow and loss-of-load transients are not included in the tabulation because the tabulation is only intended to represent normal design transients, the effects of these transients have been analytically evaluated and are included in the fatigue analysis for primary system components.

Over the range from 15-percent full power up to and including but not exceeding 100-percent of full power, for the purpose of cyclic load definition, the reactor coolant system and its components are designed to accommodate 10-percent of full power step changes in plant load and 5-percent of full power per minute ramp changes without reactor trip. The reactor coolant system will accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump system makes it possible to accept a step load decrease of 50-percent of full power without reactor trip. These transient capability definitions bracket the transient design bases used for the Regulating Systems as discussed in Section 7.3.

4.1.6 Service Life

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

The reactor vessel is the only component of the reactor coolant system that is exposed to a significant level of neutron irradiation and it is therefore the only component that is subject to material radiation damage effects.

The nil-ductility transition temperature shift of the vessel material and welds, due to radiation damage effects, is monitored by a radiation damage surveillance program, which conforms with ASTM E185 standards.

Reactor vessel design is 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 have been established for the licensed life. These operating conditions include 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

The quality assurance criteria specified below apply to all nuclear Class I piping and fittings. Effective with the implementation of the Entergy Nuclear (EN) Welding Program, in lieu of what is specified below, in-process quality assurance examinations for safety related piping welds and the acceptance criteria will be in accordance with ASME Section III 1992 Edition.

All pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested in conformance with the applicable codes listed in Table 4.1-9.

Shop and field fabrication requirements, documentation, and quality assurance examinations all comply with those found in USAS B31.7 for Class I nuclear piping.

Quality control techniques used in the fabrication of the reactor coolant system are equivalent to those used in the manufacture of the reactor vessel, which conforms to Section III of the ASME Boiler and Pressure Vessel Code.

The piping is designed to the USAS B31.1 (1955 and Summer 1973) Code for Power Piping using the allowable stresses found in the Nuclear Code Cases N-7 and N-10 for pipe and fittings, respectively.

The quality assurance requirements required by Westinghouse in the purchase and examination of the reactor coolant piping ensures that the quality level of a Westinghouse plant is comparable to that delineated by USAS B31.7, Class I, Code for Nuclear Piping. Effective with the implementation of the EN Welding Program, in lieu of the above, the in-process quality assurance examinations for safety related piping welds and the acceptance criteria will be in accordance with ASME Section III 1992 Edition.

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

2. Piping base materials are examined by quality assurance methods having acceptance criteria that meet the requirements set forth in USAS B31.7, Class I, Code for Nuclear Piping.

3. All welding procedures, welders, and welding operators are qualified to the requirements of ASME Section IX, Welding Qualifications, which is in compliance with the requirements of USAS B31.7, Class I, Code for Nuclear Piping.

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

5. All branch connection nozzle welds of nominal sizes of 3-in. and larger are 100-percent radiographed. This exceeds the requirements of USAS B31.7 for Class I piping, since it includes nominal sizes of 6-in. and larger for 100-percent radiography.

6. All finished welds are liquid penetrant examined on both the outside and inside (if accessible) surfaces as required by USAS B31.7, Class I. In addition, nozzle welds in nominal sizes 2-in. and smaller are progressively examined after each 0.25-in. increment of weld deposit in lieu of radiography.

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

Hence, the Westinghouse quality assurance requirements implemented in the procurement of Indian Point Unit 2 piping and fittings are equal to and in some instances exceed the requirements of USAS B31.7.

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

The design and stress criteria specified in USAS B31.7 are not directly comparable to that of USAS B31.1 (1955 and Summer 1973). The following describes how USAS B31.1 (1955 and Summer 1973) was used in the design of the primary coolant piping and the ASME B&PV Code Section III, Subsection NB, 1986 Edition for the presurizer surge line including the effects of Thermal Stratification on Indian Point Unit 2. A thermal expansion flexibility stress 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 Summer 1973) 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 is 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 loop 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 analysis were performed on the composite piping, which included the combined stress effects of all the sustained (pressure and weight) loadings plus seismic vertical / horizontal loading components. The resultant reactions of the piping due to the separate and combined effects of thermal, sustained, and seismic loadings were factored into the piping as interconnected. In turn, the equipment supporting structures were checked for adequate design including the added effects of these same loadings. Thus the total design analyses including pipe, equipment, and structures considered the effects of thermal expansion, sustained, and seismic loadings.

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

An added margin of conservatism is 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 line, and residual heat return nozzle connections to the primary coolant loop piping. The thermal sleeve is no longer in place on the 10" SI line to the 23 Cold Leg. A detailed analysis demonstrated that the fatigue usage factor and stresses for the nozzle in line 353 still meet the requirements of ASME Section III of the Boiler and Pressure Vessel Code for continued operation through the life of the plant with the thermal sleeve not in place.

REFERENCES FOR SECTION 4.1

1. Letter from Stephen B. Bram, Con Edison, to Document Control Desk, NRC, Subject: Leak-Before-Break, dated May 23, 1988.

2. Letter from Stephen B. Bram, Con Edison, to Document Control Desk, NRC, Subject: Leak-Before-Break (LBB) Submittal (TAC 68318), dated November 18, 1988.

3. Letter from Stephen B. Bram, Con Edison, to Document Control Desk, Subject: Leak-Before-Break (LBB) Submittal (TAC 68318), dated January 12, 1989.

4. Letter from Donald S. Brinkman, NRC, to Stephen B. Bram, Con Edison, Subject: Safety Evaluation Report on Elimination of Dynamic Effect of Postulated Primary Loop Pipe Ruptures from Design Basis for Indian Point Unit 2 (TAC No. 68318), dated February 23, 1989.

5. WCAP-12187 – Consolidated Edison Company of New York, Inc., Indian Point Unit 2, NSSS Stretch Rating – 3083.4 MWT, Engineering Report, Volume II. - Westinghouse Proprietary Class 2

TABLE 4.1-1
Reactor Coolant System Pressure Settings

Category	Pressure (psig)
Design pressure	2485
Operating pressure (at pressurizer)	2235_{1}
Safety valves	2485_{1}
Power relief valves	2335_{1}
Pressurizer spray valves (open)	2260_{1}
High pressure trip	$\leq 2363_{1}$
High pressure alarm	$2300/2335_{1,2}$
Low pressure trip	$\geq 1928_{1}$
Low pressure alarm	2185_{1}
Hydrostatic test pressure	3110

Notes:
1. Nominal values
2. The fixed high alarm PC-456F is a redundant alarm, and will not annunciate unless there is a failure of the 2300 psig alarm/control circuit.

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 OD to top of control rod mechanism housing)	43-9 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 1/16
OD of flange, in.	205
ID at shell, in.	173
Inlet nozzle ID, in.	27 1/2
Outlet nozzle ID, in.	29
Clad thickness, min., in.	5/32
Lower head thickness, min., in.	5 5/16
Vessel belt-line thickness, min., in.	8 5/8
Closure head thickness, in.	7
Reactor coolant inlet temperature, °F	514.3[1]
Reactor coolant outlet temperature, °F	605.8[1]
Reactor coolant flow, lb/hr	1.268×10^8

Notes:
1. Reactor Coolant inlet temperature is for the low T_{avg} case and Reactor Coolant outlet temperature is for the high T_{avg} case.

TABLE 4.1-3
Pressurizer and Pressurizer 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_1
Steam volume, full power, ft^3	720_1
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	1800
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 per valve	179,000
Safety valves	
Number	3
Set pressure, $psig_2$	2485
Capacity, lb/hr saturated steam per valve	408,000

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

Notes:
1. Present operation is at a T_{avg} of 562°F. In the safety analysis discussed in section 14.1, a reduced flow is assumed to account for a postulated 25% steam generator tube plugging. Actual values will depend on T_{avg} and actual percentage of tube plugging.
2. Allowance for error is specified in the Technical Specifications.

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/steam, °F	650/556	
Reactor coolant flow, Thermal Design, gpm/loop	80,700	
Total heat transfer surface area, ft^2	43,467	
Heat transferred, Btu/hour	2755×10^6	
Steam conditions at full load, outlet nozzle:	Low T_{avg} *	High T_{avg} *
Steam flow, lb/hr	3.50×10^6	3.51×10^6
Steam temperature, °F	488.0	513.3
Steam pressure, psia	610.1	766.3
Feedwater temperature, °F	436.2	436.2
Overall height, ft-in.	63-1.625	
Shell OD, upper/lower, in.	166/127.0	
Shell thickness, upper/lower, in.	3.5/2.63	
Number of U-tubes	3214	
U-tube diameter, in.	0.875	
Tube wall thickness, (average), in.	0.050	
Number of manways/ID, in.	4/16	
Number of handholes/ID, in.	6/6	
Number of Inspection Openings/ID, in.	1/3	

	3230 MWt Low T_{avg}/High T_{avg} *	Zero Power
Reactor coolant water volume (unplugged), ft^3	924	924
Primary side fluid heat content, Btu	23.67×10^6 / 24.11×10^6	23.630×10^6
Secondary side water volume, ft^3	1493/1599	2778.5
Secondary side steam volume, ft^3	32434/3128	1949
Secondary side fluid heat content, Btu	39.96×10^6 / 75.31×10^6	

Note:
*Refers to low (548.9 deg F) and high (571.9 deg F) T_{avg} and 0% tube plugging cases for design.

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 [1]
Net positive suction head, ft	170 [1]
Developed head, ft	272 [1]
Capacity, gpm	89,700 [1]
Seal water injection, gpm	8
Seal water return, gpm	3
Pump discharge nozzle ID, in.	27 1/2
Pump suction nozzle ID, in.	31
Overall unit height, ft	28.38
Water volume, ft^3	192
Pump-motor moment of inertia, lb/ft^2	82,000
Motor Data:	
Type	AC induction, single speed, air cooled
Voltage	6600
Insulation class	B thermoplastic epoxy
Phase	3
Frequency, cps	60
Starting Current, amp	2950
Input (hot reactor coolant), kW	4221 [1]
Input (cold reactor coolant), kW	5673 [1]
Power, HP (nameplate)	6000

Note:

1. These values represent the pump hydraulic design point. Actual heads, flows, temperatures, currents, and powers are dependant upon system parameters such as reactor internals changes, percentage of steam generator tube plugging, and plant operating T_{avg}. For use in analyses or evaluations, values reflecting the current conditions should be obtained.

TABLE 4.1-6
Reactor Coolant Piping Design Data

Reactor inlet piping ID, in.	27 1/2
Reactor inlet piping nominal thickness, in.	2.375
Reactor outlet piping ID, in.	29
Reactor outlet piping nominal thickness, in.	2.50
Coolant pump suction piping ID, in.	31
Coolant pump suction piping nominal thickness, in.	2.656
Pressurizer surge line piping ID, in.	11.5
Pressurizer surge line piping nominal thickness, in.	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	1.2
Across vessel, including nozzles	51.5
Across hot leg	1.1
Across steam generator	31.8
Across pump suction leg	2.8
Total pressure drop	88.4

Notes:

1) DP’s based on best estimate flow conditions.

TABLE 4.1-8
Thermal and Loading Cycles

	Transient Condition	Design Cycles[1]
1.	Plant heatup at 100°F per hr	200[2] [Deleted]
2.	Plant cooldown at 100°F per hr	200 [Deleted]
3.	Plant loading at 5-percent of full power per min	14,500 [Deleted]
4.	Plant unloading at 5-percent of full power per min	14,500 [Deleted]
5.	Step load increase of 10-percent of full power (but not to exceed full power)	2000 [Deleted]
6.	Step load decrease of 10-percent of full power	2000 [Deleted]
7.	Step load decrease of 50-percent of full power	200 [Deleted]
8.	Reactor trip	400 [Deleted]
9.	Hydrostatic test at 3110 psig pressure	5 (preoperational)
10.	Hydrostatic test at 2485 psig pressure and 400°F temperature	5 (postoperational)
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. [Deleted]	

Notes:
1. Estimated for equipment design purposes [Deleted] and not intended to be an accurate representation of actual transients, or to reflect actual operating experience.
2. This transient includes pressurizing to 2235 psig.

TABLE 4.1-9
Reactor Coolant System - Design Code Requirements

COMPONENT	CODE	CODE EDITION	APPLICABLE ADDENDA
Reactor vessel	ASME III[1] Class A	1965	Summer 1965 and Code Cases 1332, 1335, 1339, 1359
Control rod drive mechanism	ASME III[1] Class A	1965	Summer 1966
Steam generators			
Tube side	ASME III[1] Class A	1965	Summer 1966
Shell side[4]	ASME III[1] Class C	1965	Summer 1966
Reactor coolant pump volute[5]	ASME III[1] Class A	1965	Winter 1965[2]
Pressurizer	ASME III[1] Class A	1965	Summer 1966
Pressurizer relief tank	ASME III[1] Class C	1964	Winter 1965
Pressurizer safety valves:			
Old Buy	ASME III[1]	1971	Winter 1972
New Buy		1974	Summer 1975
Reactor coolant piping	USAS B31.1[3]	1955	
System valves, fittings, piping	USAS B31.1[3]	1955	

Notes:

1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.
2. Not stamped, but built in accordance with this edition and addenda.
3. USAS B31.1 Code for pressure piping.
4. The shell side of the generator conforms to requirements for Class A vessels and is so stamped as permitted under the rules of Section III.
5. The reactor pump, though not a coded vessel, was designed to Section III of the ASME Boiler and Pressure Vessel Code.

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 a steam generator, a pump, loop piping, and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system is shown in Plant Drawing 9321-2738 [Formerly UFSAR Figure 4.2-1], and a schematic flow diagram in Figure 4.2-2. The total design volume of the reactor coolant system, at rated operating conditions, is approximately 12,250-ft^3. The nominal liquid volume of the reactor coolant system, at rated operating conditions and with 0% Steam Generator tube plugging, is 11,350 cubic feet.

The containment boundary shown on the flow diagram indicates those major components, which are to be located inside the containment. The intersection of a process line with this boundary indicates a functional penetration.

Reactor coolant system design data are listed in Tables 4.1-2 through 4.1-6.

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

4.2.2.1 Reactor Vessel

The reactor vessel is cylindrical with a hemispherical bottom and a flanged and gasketed removable upper head. The vessel is 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 95-percent of the total coolant flow is effective for heat removal from the core. The remainder of the flow includes the flow through the rod cluster control 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 that result from heat generated by the absorption of gamma energy. This protection is further described in Section 3.2.3.

Fifty-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 0-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two 0-rings to monitor leakage across the inner 0-ring. A leakoff connection is also provided beyond the outer 0-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 are 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 will be examined at selected intervals to evaluate reactor vessel material nil-ductility transition temperature changes as described in Section 4.5.2. The factor by which the maximum specimen exposure exceeds that at the vessel wall (at the location of maximum vessel wall exposure) has a maximum value of 3.5. Four of the eight irradiation specimens will lead the vessel wall maximum exposure by this factor.

Ring forgings have been used for closure flanges; no other forgings have been used in the reactor vessel shell sections. The eight primary inlet and outlet nozzles have been provided with nozzle safe ends (forgings). These safe ends have been overlaid in the field with stainless steel weld metal. The Charpy V-notch and drop weight tests for the reactor vessel plates and forgings are discussed in Section 4.2.5.

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 3.2-47.

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

4.2.2.2 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 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 is designed to accommodate positive and negative surges caused by load transients. The surge line, which is attached to the bottom of the pressurizer, connects the pressurizer to a hot leg of a reactor coolant loop. During a positive surge caused by a decrease in plant load, the 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 setpoint 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 valve controller in the control room. A small continuous spray flow is provided to ensure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping.

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.

4.2.2.3 Steam Generators

Each loop contains a vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 4.2-5. Principal design parameters are listed in Table 4.1-4. The steam generators are designed and manufactured in accordance with Section III (Nuclear Vessels) of the ASME Boiler and Pressure Vessel Code. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

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. Manways are provided to permit access to the U-tubes and the moisture-separating equipment.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam. Certain plant operating conditions affecting the steam generator can result in the steam generator water level dropping below the feedwater sparging ring. As a result of these conditions and the waterhammer that can occur in the feedwater system, "J" tubes are installed on the feedwater sparging rings inside the steam generators. These "J" tubes preclude the rapid draining of the feedwater sparging rings and prevent steam from entering these rings even if they are uncovered. In the very remote event that the feedwater system would experience another large pressure wave, additional pipe restraints were installed in 1974 along the feedwater pipe. This modification is intended to preclude the rebound-type failure of the feedwater line at the containment penetration supports. All modifications that were made and the test program that was performed as a result of these conditions were accomplished in accordance with the Quality Assurance Program for operating nuclear plants that was currently in effect.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly, which imparts a centrifugal motion to the mixture and separates the water particles from the steam. The water spills over the edge of the swirl vane housing and combines with the feedwater for another pass through the tube bundle. The steam rises through additional separators, which further reduce the moisture content of the steam.

A steam-generator blowdown system exists to perform several functions. Primarily it is used in maintaining the secondary side water chemistry of the steam generators within specifications. It also provides water samples from the secondary side of the steam generator as well as a means of draining the shell sides for inspection and/or maintenance.

The steam generator is constructed primarily of low alloy steel. The heat transfer tubes are Inconel. The tubes undergo thermal treatment following tube-forming operations. The interior surfaces of the channel heads 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-tube sheet joint is welded. The primary nozzles are provided with safe ends with weld metal overlay.

Tubes are examined and defective tubes are repaired, plugged or sleeved as required by the technical specifications. The upper limit for tube plugging is 10-percent. *[**Note** – Fuel pellet thermal conductivity degradation evaluation resulted in a reduction of the maximum steam generator tube plugging from 10% to 5%.]*

Nozzle dam retention rings are permanently welded to the channel head cladding and provide a means of attachment for the temporary installation of nozzle dams during refueling and/or maintenance outages.

The inspection ports and handholes are on the secondary side of the steam generators and any possible leakage (the only possible failure) would be inside containment. The possibility of secondary loss of water has been evaluated in Section 14.1.9, Loss of Normal Feedwater, and Section 14.2.5, Rupture of a Steam Pipe. These sections show that the types of failure possible due to secondary leakage or loss of water have already been analyzed.

4.2.2.4 Reactor Coolant Pumps

Each reactor coolant loop contains a vertical, single-stage centrifugal pump that employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 4.2-6 and the principal design parameters for the pumps are listed in Table 4.1-5. The reactor coolant pump estimated performance and 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-percent design flow introduces no operational restrictions, because the pumps operate at full speed.

During normal operation, the reactor coolant pumps are supplied from the unit auxiliary bus and are therefore tied to the turbine-generator frequency (speed). On occurrence of unit turbine trip, the pump electrical buses are transferred from the unit auxiliary transformer to the station auxiliary transformer with an intentional delay of 30 sec. Further details are given in Section 14.1.8.3.

On most electrical events, which cause the turbine to be tripped, the reactor coolant pump buses are transferred to offsite power and the unit is tripped simultaneously and the pumps will therefore not exceed their normal running speed. If for some unlikely reason the only plant trip is a turbine over-speed trip an over-frequency trip relay circuit is provided that will trip the turbine-generator. This trip circuit first locks out the 6.9 kV dead bus transfer at 62.2 ±0.1 Hz (1866 ± 3 rpm) and then trips the main generator at 62.5 ±0.1 Hz (1875 ± 3 rpm). Termination of power to the in-house 6.9 kV buses 1-4 limits the reactor coolant pumps overspeed to maintain the RCS flow condition below the design limit per section 4.2.2.5.4.

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 exits through a discharge nozzle in the side of the casing. The motor-impeller can be removed from the casing of 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 that directs the controlled leakage out of the pump, and a third seal that 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 that leaks through the second seal is also collected and removed from the pump.

Component cooling water is supplied to the motor bearing oil coolers of the reactor coolant pumps. The component cooling water system also provides cooling flow to the thermal barrier heat exchanger of the reactor coolant pumps to minimize heat transfer from the high-temperature primary coolant to the seal area environment, to cool primary system water that could leak through the thermal barrier labyrinth seals, and to provide adequate seal cooling in the event that seal injection flow was lost.

In the event of loss of offsite power, the reactor coolant pump motor is deenergized and both cooling water supplies (seal injection and component cooling flow) are terminated; however, the plant diesel generators are immediately started and the component cooling water pumps are automatically loaded (in sequence) onto the emergency buses and started (no operator action required). Once the automatic loading of the emergency buses has been completed, the operator has the option of manually loading a charging pump onto one of the diesels and reestablishing normal seal injection flow. 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.

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 has also been available.

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. Both high and low oil levels would signal an alarm in the control room to alert the operator to possible pending bearing problems. Each motor bearing contains embedded temperature detectors, and so initiation of failure, separate from or in combination with a loss of oil, would be indicated and alarmed in the control room as a high bearing temperature. This would require a reactor trip followed by 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 constitute a loss of coolant flow in the one loop, the effect of which is analyzed in Section 14.1.6.

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 in 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. Indications of pump malfunction in these conditions would be initially by high temperature signals from the bearing water temperature detector, and excessive No. 1 seal leakoff indications, respectively. Following these signals, pump vibration levels would be checked. These would show excessive levels, indicating some mechanical trouble. Again, the pump would be shut down for investigation.

The design specifications for the reactor coolant pumps include as a design condition that the pumps are designed to withstand seismic load equivalent to 0.28 g in the vertical direction and 0.40 g in the horizontal direction and the seismic loads shall be considered acting simultaneously. 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 does any bearing stress in the pump or motor exceed or even approach a value, which the bearing could not carry.

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

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

As is generally the case with machines of this size, the shaft dimensions are predicated on avoidance of shaft critical speed conditions, rather than actual levels of stress. There are many machines as large as, and larger than these, that are designed to run at speeds in excess of first shaft critical. However, it is considered desirable in a superior product to operate below first critical speed; the reactor coolant pumps are designed in accordance with this philosophy. This results in a shaft design, which even under the severest postulated transient, gives 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 gives assurance of the conservative stress levels experienced during these transients.

So in each of these cases, where it is the functional requirements of the component that control its dimensions, it can be seen that if these are met, the stress-related failure cases are more than adequately satisfied.

It is 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.

4.2.2.5 Reactor Coolant Pump Flywheel Integrity

The reactor coolant pump flywheels were fabricated from two rolled, vacuum-degassed, ASTM A-533 Grade B Class 1 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 flywheel blanks were flame-cut from the plates, with allowance for exclusion of flame-affected metal. They were then machined to the specified dimensions and the bolt holes were drilled.

Two plates were then bolted together, the finished flywheel attached to the motor shaft, and the whole unit balanced to yield vibration levels at operating speed less than 0.001-in. double amplitude. The reactor coolant pump flywheel is shown in Figure 4.2-8.

A nil-ductility transition temperature less than +10^{o}F was specified. A minimum of three Charpy tests, parallel and normal to the rolling direction, were made from each plate to determine that each blank satisfied design requirements.

The finished flywheels were subjected to 100-percent volumetric ultrasonic inspection.

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

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 4.2-9) less than 50-percent of the minimum specified material yield strength at room temperature (100 to 150^{o}F).

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.

4.2.2.6 Pressurizer Relief Tank

Principal design parameters of the pressurizer relief tank are given in Table 4.1-3. The tank is shown on Figure 4.2-10.

Steam and water discharge from the power relief and safety valves pass to the pressurizer relief tank, which is partially filled with water. 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 full power pressurizer steam volume.

The tank is protected against a discharge exceeding the design value by two rupture disks that discharge into the reactor containment. The rupture disks 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 disk's bursting pressure) is twice the calculated pressure resulting from the maximum safety valve discharge described above. This margin is to prevent deformation of the disk. The tank and rupture disk 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 setpoint pressure at full flow.

The pressurizer relief tank, by means of its connection to the waste disposal system, provides a means for removing any noncondensable gasses 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.

4.2.2.7 Piping

A schematic of the reactor coolant piping is shown on Figure 4.2-2. The general arrangement of the loop piping is shown on Plant Drawings 9321-2502, 9321-2506, 9321-2508 [Formerly UFSAR Figures 5.1-3, 5.1-5, and 5.1-7]. Piping design data are presented in Table 4.1-6.

The austenitic stainless steel reactor coolant piping and fittings that make up the loops are 29-in. ID in the hot legs, 27.5-in. ID in the cold legs, and 31-in. ID between the steam generator outlet and reactor coolant pump suction. The pressurizer relief line, which connects the outlets of the pressurizer safety and relief valves to the inlet nozzle flange on the pressurizer relief tank, is 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 safety valves, and the vacuum fill connection closure.

In response to NRC Bulletin 88-11, thermal stratification effects on the pressurizer surge line have been evaluated for the design life of the plant[18]. The stress and fatigue analyses results are within the ASME Code allowables for the surge line.

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) See Section 4.1.7.

2. Both ends of the pressurizer surge line.

3. Pressurizer spray line connection to the pressurizer.

4. Charging lines and auxiliary charging line connections.

4.2.2.8 <u>Valves</u>

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 mitigate the potential for valve stem leakage due to modulating service.

4.2.2.9 <u>Component Supports</u>

The support structures for the reactor coolant components are described in Appendix 4B and Chapter 5.

4.2.3 <u>Pressure-Relieving Devices</u>

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 Figure 4.2-1; the valve design parameters are given in Table 4.1-3. Valve sizes are determined as indicated in Section 4.3.4.

Power-operated relief valves and code safety valves are provided to protect against pressure that is beyond the pressure limiting capacity of the pressurizer spray. Acoustic sensors installed on the code safety valve discharge lines provide indication in the control room of the "flow" or "no flow" condition of the safety valves. Direct valve position indication is also provided for the power-operated relief valves.

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

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 a concrete shield wall. This wall provides shielding to permit access into the containment annular region during full-power operation for inspection and maintenance of miscellaneous equipment. This shielding wall also provides 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 is 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 given in Chapter 5.

[Historical Information Only] This paragraph is retained for historical purposes only.
The integrity of the reactor coolant system as may be affected by asymmetric loss-of-coolant accident loads due to postulated pipe breaks in the primary loop coolant piping was considered in WCAP-9117 (Reference 3) and in a subsequent submittal to the NRC on June 15, 1978 (Reference 4). The combination of LOCA and safe shutdown earthquake loads applied to the results of WCAP-9117 was described in the September 3, 1980 follow-up submittal (Reference 5), and is applicable to both Indian Point Units 2 and 3, based on the similarity between the two units. The safe shutdown earthquake results for Unit 3 would be similar to those for Unit 2, and the total strains also apply to Unit 2. The NRC Safety Evaluation Report (Reference 6) concluded that the assessment of asymmetric loss-of-coolant accident and safe shutdown earthquake loads for the Indian Point Unit 2 was acceptable. This conclusion was based upon the installation of pipe motion limiters in the primary shield wall and was contingent upon the verification of shield plug assumptions, which included the determination of the effects of the plugs as missiles, and the determination that structural components do not inhibit plug displacement. This verification was provided in a subsequent submittal to the NRC on June 10, 1986 (Reference 22).

In 1989, the NRC approved changes to the design bases with respect to dynamic affects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4.

4.2.5 Materials Of Construction

Each of the materials used in the reactor coolant system is selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1.

All reactor coolant system materials that are exposed to the coolant are corrosion resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. Reactor coolant chemistry is further discussed in Section 4.2.8.

It is characteristic of stress corrosion that combinations of alloy and environment, which result in cracking are usually quite specific. Environments that have been shown to cause stress-corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism, and the presence of chlorides and free oxygen. With regard to the former, experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator. In the presence of this environment, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. However, the steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel alloy has excellent resistance to general and pitting-type corrosion.

Considerable experience with Inconel in steam generator and heat exchanger applications has been accumulated in the industry. Since 1962, widespread adoption of Inconel for steam generator tubes in nuclear stations is evident: as for example, Connecticut-Yankee; San Onofre; PM-1, Sundance; PM-3A, McMurdo Sound; CVTR; NPD, and Hanford N-Reactor. Materials with lead traces in the overall composition were present in the secondary side of the referenced plants. The use of lead in the materials of the secondary side of the Indian Point plant has been minimized to the practical limit of that occurring as trace elements in metallurgical alloys and as such is insignificant.

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

The reactor vessel was fabricated by Combustion Engineering, Inc. A sketch of the reactor vessel showing all materials in the beltline region is shown in Figure 4.2-11. Information on each of the welds and plates in the beltline region is shown in Tables 4.2-2 through 4.2-5, and Tables 4.2-5 through 4.2-8, respectively. Information relative to weld and plate material included in the material surveillance program is shown in Tables 4.2-2 and 4.2-6 through 4.2-8.

The reactor vessel plate or forging material opposite the core is purchased to a specified Charpy V-notch test result of 30-ft-lb or greater at a corresponding nil-ductility transition temperature (NDTT) of 40°F or less, and the material is tested to verify conformity to specified requirements and to determine the actual NDTT value (see Table 4.2-9). In addition, this plate is 100-percent volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods.

The remaining material in the reactor vessel and other reactor coolant system components meets the appropriate design code requirements and specific component function.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility, which ensures a low NDTT, and thereby gives assurance that the finished vessel can be initially hydrostatically tested and operated near room temperature within the restrictions of NDTT + 60°F. The stress limits established for the reactor vessel are dependent upon the temperature at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the NDTT. An initial maximum value of NDTT of 40°F has been established during fabrication in this region.

The techniques used to measure and predict the integrated fast neutron (E > 1 MeV) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum neutron (E > 1 MeV) 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 will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The evaluation of the second surveillance capsule is discussed in detail in Reference 8. The analysis for the third surveillance capsule, removed during the 1984 refueling outage, is documented in Reference 9.

The analysis of the fourth surveillance Capsule V, removed during the 1987-88 refueling outage (end of Cycle 8), is documented in Reference 11. The Capsule V received a fast neutron (E > 1 MeV) fluence of 5.3 x 10^{18} n/cm^2 in 8.6EFPY at the end of Cycle 8. See Reference 11.

The maximum integrated fast neutron (E > 1 MeV) exposure of the vessel for 32 EFPYs is calculated to be 1.39 x 10^{19} n/cm^2 based on the measurements from the fourth surveillance Capsule V. Fast neutron fluences corresponding to 32 EFPYs at various reactor vessel thicknesses are given in Table 4.2-10.

For the extended period of operation (60 years), the maximum integrated fast neutron (E > 1 MeV) exposure of the vessel for 48 EFPYs is calculated to be 1.906 x 10^{19} n/cm^2 based on calculations using Regulatory Guide 1.99 methodology. Fast neutron fluences corresponding to 48 EFPYs at various reactor vessel thicknesses are given in Table 4.2-10.

The calculated neutron exposure exceeds the value of 0.85 x 10^{19} n/cm^2 (E > 1MeV) reported in the First Supplement to the Preliminary Facility Description Safety Analysis Report. The reasons for the increase are:

1. Anticipated increase in reactor power from 2758 MWt to 3071.4 MWt in Cycle 10 and then to 3114.4 MWt in Cycle 16 and subsequently to 3216 MWt in Cycle 17.
2. Revision of analysis methodology including upgrading of neutron cross sections and codes.
3. Core design considerations involving changes in loading patterns.
4. Extended period of operation.

The above projected exposure to reactor vessel from Capsule V measurements is based upon using the standard loading pattern (only fresh fuel assemblies at core periphery) in Cycles 1 thru 5 and the low leakage loading pattern (mixture of fresh and spent fuel assemblies) or L^3P starting from Cycle 6. Furthermore, it assumes Indian Point Unit 2 operation at stretch power operation at 3071.4 MWt core power level starting from Cycle 10, then at the Appendix K uprated power level of 3114.4 MWt starting in Cycle 16 and subsequently at 3216 MWt starting in Cycle 17.

The maximum reference temperature, RT_{NDT} for the Indian Point Unit 2 vessel core beltline materials at the 1/4 thickness and the 3/4 thickness after 32 effective full power years of operation are projected to be 240°F and 194°F, respectively, based on calculations performed per Regulatory Guide 1.99, Revision 2, using data obtained from evaluation of Surveillance Capsule V. (Ref.11). This data provides the basis for subsequent calculation of Adjusted

Reference Temperature values for determination of allowable pressure/temperature limits for operation to 25 EFPY, as described in Reference 19.

For the extended period of operation, the maximum reference temperature RT_{NDT} for the Indian Point Unit 2 vessel core beltline materials at the ¼ thickness and the ¾ thickness after 48 effective full power years of operation is projected to be 238.3°F and 174.8°F, respectively (at circumferential weld 9-042), based on calculations performed per Regulatory Guide 1.99, Revision 2.

To evaluate the NDTT shift of welds, heat affected zones, and base material for the vessel, test coupons of these material types have been included in the reactor vessel surveillance program described in Section 4.5.2. The methods used to measure the initial NDTT of the reactor vessel baseplate material are given in Appendix 4A.

The reference nil ductility transition temperatures for pressurized thermal shock evaluation (RT_{PTS}) have been estimated[11,12,13] in accordance with 10 CFR 50.61(b)(2). The values at 15 EFPY and also at the end of the license term are well below the screening criteria of 270°F (for plates and axial weld materials) and 300°F (for circumferential weld materials), based on a low-leakage core design. The NRC has accepted this analysis.[14] Additional information in response to Generic Letter 92-01, Revision 1, is given in reference 21.

The projected RT_{PTS} values at 48 EFPY are within the established screening criteria of 270°F (for plates and axial weld materials) and 300°F (for circumferential weld materials), based on calculations performed per Regulatory Guide 1.99, Revision 2.

With regard to electroslag welding of Class I components, the Indian Point Unit 2 90-degree elbows were electroslag welded. The following efforts 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 Boiler and Pressure Vessel Code, Section IX and Code Case 1355 plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from a 5-in. thick weldment and successfully tested:

   a. Six transverse tensile bars - as welded.
   b. Six transverse tensile bars - 2050°F, H_2O quench.
   c. Six transverse tensile bars - 2050°F, H_2O quench + 750°F stress relief heat treatment.
   d. Six transverse tensile bars - 2050°F, H_2O quench, tested at 650°F.
   e. Twelve guided side bend test bars.

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

3. The edges of the electroslag weld preparations 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. Then the electroslag weld and adjacent base material were 100-percent radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code Section III, Paragraph N-627.

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

6. Heat treatment furnace charts were recorded and certified.

Two of the Indian Point Unit 2 reactor coolant pump casings were electroslag welded. The following efforts were performed for quality assurance of these two components.

1. The electroslag welding procedure employing two- and three-wire technique was qualified in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section IX and Code Case 1355 plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from an 8-in.-thick and from a 12-in.-thick weldment and successfully tested for both the two-wire and the three-wire techniques, respectfully.

    a. Two-wire electroslag process - 8-in.-thick weldment.

        (1) 6 transverse tensile bars - 750^{o}F postweld stress relief.
        (2) 12 guided side bend test bars.

    b. Three-wire electroslag process - 12-in.-thick weldment.

        (1) 6 transverse tensile bars - 750^{o}F postweld stress relief.

        (2) 17 guided side bend test bars.

        (3) 21 Charpy V-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.

    c. A separate weld test was made using the two-wire electroslag technique to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and techniques as such an occurrence was anticipated during production applications due to equipment malfunction, power outages, etc. The following test specimens were removed from an 8-in.-thick weldment in the stop-restart-repaired region and successfully tested.

        (1) 2 transverse tensile bars - as welded.

(2) 4 guided side bend test bars.

(3) Full section macroexamination of weld and heat affected zone.

d. All of the weld test blocks in (a), (b), and (c) above were radiographed using a 24-MeV Betatron. The radiographic quality level obtained was between 0.05 to 1-percent. There were no discontinuities evident in any of the electroslag welds.

(1) The casting segments were surface conditioned for 100-percent radiographic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2 (except no category D or E defectiveness was permitted) for section thickness up to 4.5-in. and ASTM E-280 severity level 2 for section thicknesses greater than 4.5-in. The penetrant acceptance standards were ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-627.

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

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

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

(5) Heat treatment furnace charts were recorded and certified.

The two remaining Indian Point Unit 2 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.

4.2.6 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 reactor coolant system heatup, cooldown, and leak test limitations curves are included in the Technical Specifications. Starting with a minimum water level, sufficient electrical heaters are installed in the pressurizer to permit a heatup rate of 55°F/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant. The fastest cooldown rates, which result from the hypothetical case of a break of a main steam line are discussed in Section 14.2.5.

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. Radiation sensitive instruments provide the capability for detection of leakage from the reactor coolant system. The containment air particulate monitors are quite sensitive to low leak rates. The containment radiogas monitors are less sensitive but are used in addition to the air particulate monitor.

2. A third mechanism used in leak detection is the humidity detectors. These provide a means of measuring overall leakage from all water and steam systems within the containment, which can affect containment humidity. The humidity monitoring method is considered supplemental to the radiation monitoring methods.

3. A leakage detection system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units. The condenser moisture includes, of course, any leaks from the cooling coils themselves. This system provides a dependable and accurate means of measuring the total leakage from these sources. Condensate flows of approximately 1.0 gpm to 15 gpm per detector can be measured by this system. Condensate flows can be determined using weir calibration curves in conjunction with the weir water head displayed by the weir water meter, or by direct reading of the weir integrated condensate flow on the weir meter.

4. An increase in the amount of coolant makeup water, which is required to maintain normal level in the pressurizer or an increase in containment sump level provide additional means of detecting leakage.

The Technical Specifications provide the requirements and bases for leakage detection.

In considering potential leakage from the reactor coolant system containing primary coolant at high pressure, four categories are described and evaluated in Section 6.7.1. These include leakage paths to the reactor coolant drain tank, leakage paths to the pressurizer relief tank, leakage paths to the containment environment, and leakage paths to the interconnecting systems.

4.2.7.1 Maximum Leak Rates

The maximum leak rate from an unidentified source that will be permitted during normal operation is specified in the Technical Specifications. Leakage from the reactor coolant system is collected in the containment or by the other closed systems. These closed systems are: the steam and feedwater system, the waste disposal system, and the component cooling system. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of unidentified leakage is a conservative limit on what is allowable before the guidelines of 10 CFR 20 would be exceeded.

With the limiting reactor coolant activity and assuming initiation of a leak from the reactor coolant system to the component cooling system, the radiation monitor that samples the component cooling pump discharge downstream of the component cooling heat exchangers would annunciate in the control room and initiate closure of the surge tank vent line in the component cooling system. In the case of failure of the closure of the vent line and resulting continuous discharge in the atmosphere via the component cooling surge tank vent, the resultant dose at the site boundary would be within the limit allowed by 10 CFR 20.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation specified by the Technical Specifications for a source of leakage not identified is sufficiently above the minimum detectable leakage rate to provide a reliable indication of leakage. The leakage limit is well within the capacity of one coolant charging pump.

The conservative approach that is used in the design and fabrication of the components that constitute the primary system pressure boundary together with the operating restrictions, which are imposed for system heatup and cooldown give adequate assurance that the integrity of the primary system pressure boundary is maintained throughout plant life. The periodic examination of the primary pressure boundary via the inservice inspection program (specified in the Technical Specifications) will physically demonstrate that the operating environment will have no deleterious effect on the primary pressure boundary integrity.

The maximum unidentified leak rate that is permitted during normal operation is well within the sensitivity of the leak detection systems incorporated within the containment, and it reflects good operating practice based on operating experience gained at other PWR plants. Detection of leakage from the primary system directs the operator's attention to potential sources of leakage, such as valves, and permits timely evaluation to ensure that any associated activity release does not constitute a public hazard, that the reactor coolant inventory is not significantly affected, and that the leakage is well within the capability of the containment drainage system. See also Section 6.7 for a further discussion of leakage detection.

4.2.7.2 Leakage Prevention

Reactor coolant system components are manufactured to exacting specifications, which exceed normal code requirements (as outlined in Section 4.1.7). In addition, because of the welded construction of the reactor coolant system and the extensive nondestructive testing to which it is 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 design from the reactor coolant pump seals. Also, all sealed joints are potential sources of leakage even though the most appropriate sealing device is selected in each case. Thus, because of the large number of joints and the difficulty of ensuring 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.

4.2.7.3 Locating Leaks

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

The Reactor Coolant System shall be tested for leakage at normal operating pressure prior to plant startup following each refueling outage, in accordance with the requirements of the applicable edition and addenda of the ASME Section XI Code. Leak test of the Reactor Coolant System is required by the ASME Boiler and Pressure Vessel Code, Section XI, to ensure leak tightness of the system during operation. The test frequency and conditions are specified in the Code.

Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of the applicable edition and addenda of the ASME Section XI Code. For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak-tightness during normal operation.

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. 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 required reactor coolant water quality. Maintenance of the water quality to minimize corrosion is accomplished using the chemical and volume system and sampling system that is described in Chapter 9.

4.2.9 Reactor Coolant Flow Measurement

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:

$$\frac{\Delta P}{\Delta P_o} = \left(\frac{\omega}{\omega_o}\right)^2$$

where ΔP_o is the reference pressure differential with the corresponding referenced flow rate ω_o, and ΔP is the pressure differential with the corresponding referenced flow rate. 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 PWR plants. The expected absolute accuracy of the channel is within 10-percent and field results have shown the repeatability of the trip point to be within 1-percent. The analysis of the loss-of-flow transient is presented in Section 14.1.6.

4.2.10 Reactor Coolant Vent System

4.2.10.1 Design Basis

The remote reactor coolant vent system has been designed and installed in accordance with NUREG-0737 Item II.B.1 to allow for remote manual venting of gases from the reactor vessel head should they accumulate there. The power-operated relief valve system acts as the remote operated vent system for the pressurizer (see Section 4.2.3) and as a redundant backup to the vessel head vent system.

4.2.10.2 System Description

4.2.10.2.1 Power-operated Relief Valve System

The power-operated relief valve system is discussed in Sections 4.2.3 and 4.3.4.

4.2.10.2.2 Remote Reactor Head Vent System

The original manual reactor vessel head vent line has been extended and two motor operated valves have been installed in series to facilitate venting of the reactor vessel head from the control room. The release point is located above the operating floor at an elevation of approximately 105-ft and is situated so that the discharge of the system will not impinge on any structures, systems, or components essential to the reactor safe shutdown or mitigation of a design basis accident.

The power-operated relief valve system relieves to the pressurizer relief tank. The remote reactor head vent, and the power-operated relief valves and their associated block valves are in three separate lines and are supplied with three independent emergency power sources so that at least one vent path will remain functional after the single failure of an emergency power train.

Potential seat leakage through both valves is vented directly to the containment atmosphere and is detected and monitored as part of the reactor coolant system leakage requirements specified in the Technical Specifications.

The two series motor operated head vent valves are closed and deenergized during normal plant operation. The circuit breakers will be locked open to prevent inadvertent operation. If the need should arise for venting the reactor, the two breakers of the remote head vent valves will be reenergized and the valves opened as necessary from the central control room accident assessment panel.

4.2.10.3 Design Criteria

The reactor coolant vent system piping, valves, components, and supports are classified seismic Class I and Class A. They have been designed and installed in accordance with the original requirements for reactor coolant pressure boundary installations, and ASME and ANSI codes applicable to Indian Point Unit 2. The piping, valves, and fittings were fabricated from stainless steel and are compatible with reactor coolant chemistry.

To alleviate the potential hazard of missiles, the remote reactor head vent system was installed such that it does not come close to and have the ability to damage safety-related systems required for safe reactor shutdown or mitigation of a design basis accident.

4.2.10.4 Design Evaluation

Consistent with NUREG-0737 Item II.B.1 Clarification A(4), the new remote reactor head vent system was designed with sufficient flow restriction that in the event of inadvertent opening or line breaks, normal makeup charging flow from the chemical and volume control system is capable of precluding actuation of the safety injection system. The original reactor vessel head vent consisted of a 3/4-in. line with a manual (locally operated) shutoff valve and bolted blind flange and was used only for routine operations when the reactor was shut down. When the remotely operated head vent system was installed, the blind flange was removed and additional nominally 3/4-in. tubing (9/16-in. ID) was run from the existing 3/4-in. NPS line to the new motor-operated head vent valves. Those portions of the system, which were revised were designed and constructed to the same criteria as the original Indian Point Unit 2 pressure boundary components.

A specific calculation has been performed for the worst case break location for the revised vent system (i.e., the interface between the 9/16-in. tubing and the original 3/4-in. head vent piping). This calculation determined that even at this worst case location, the break flow would be well within the capacity of two chemical and volume control system charging pumps without actuating safeguards equipment. Thus, failure of the vent system would not result in a break size corresponding to the definition of a loss-of-coolant accident.

4.2.11 Reactor Vessel Level Indication System

The reactor vessel level indication system (RVLIS) has been installed in accordance with the requirements of NUREG-0737. The system is mainly part of the "Inadequate Core Cooling (ICC) Instrumentation" in improving the reliability of the plant operator to diagnose the approach of inadequate core cooling and to assess the adequacy of responses taken to restore cooling. The system also provides assistance to the operator in determining the presence of voids in the vessel. Additional information is given in Section 7.5.2.

4.2.11.1 Design Basis

The system has been designed to provide continuous indication of coolant level inside the reactor and to assist the operator in determining the presence of voids in the vessel. The system was designed by Westinghouse as a Class A-Class 1E system.

4.2.11.2 System Description

The RVLIS, shown in Plant Drawing 208798 [Formerly UFSAR Figure 4.2-12], is based on the differential pressure principle as sensed by taps located at the top and bottom of the reactor vessel. The top tap is installed on an unused control rod penetration and the bottom tap uses an unused incore instrument thimble to the seal table. The differential pressure is transmitted through filled capillary systems to transmitters outside containment. The temperature sensors are mounted on each capillary inside the containment. The signals from the temperature sensors and transmitters are routed to a Class 1E panel in the cable spreading room. The temperature compensated signals of level indication are indicated on the accident assessment panel in the Unit 1/Unit 2 central control room.

The reactor vessel level indication system, which was installed in response to NUREG-0737, Inadequate Core Cooling Instrumentation, has been approved by the NRC.

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13. Letter from Murray Selman, Con Edison, to Document Control Desk, NRC, transmitting supplemental information, dated January 12, 1987.

14. Letter from Marylee M. Slosson, NRC, to Murray Selman, Con Edison, Subject: Projected Values of Material Properties for Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, dated February 27, 1987.

15. Deleted

16. Deleted

17. Deleted

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19. Letter from Jefferey F. Harold, NRC, to Paul H. Kinkel, Con Edison, Subject: Issuance of Amendment for Indian Point Nuclear Generating Unit No. 2 (TAC No. M96944), dated February 27, 1998.

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23. Letter from R. Capra, NRC, to S. Bram,, Con Edison, dated December 24, 1987, Subject: Indian Point Nuclear Generating Unit No. 2 Steam Generator Girth Weld Repair (TAC 66684).

24. Letter from R. Capra, NRC, to S. Bram,, Con Edison, dated October 28, 1988, Subject: Steam Generator Girth Weld Repair Safety Evaluation for Indian Point Nuclear Generating Unit No. 2 (TAC 66684).

25. Letter from R. Capra, NRC, to S. Bram,, Con Edison, dated January 4, 1991, Subject: Review of Mid-Cycle Steam Generator Girth Weld and Feedwater Inlet Nozzle Work (TAC 72961).

26. WCAP-15629, Rev. 1, “Indian Point Unit 2 Heatup and Cooldown Curves for Normal Operation and PTLR Support Documentation,” Westinghouse Electric Co., December 2001.

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 and nozzle forgings	SA-302, Gr. B / SA-336
	Cladding, stainless weld rod	Type 304 equivalent
	Thermal shield and internals	A-240 Type 304
	Insulation	Stainless steel, Aluminum
Steam generator	Pressure plate	SA-533, Grade A Class 2
	Cladding, stainless weld rod	Type 304 equivalent
	Cladding for tube sheets	Inconel
	Tubes	SB-163, Thermally Treated (Code Case N-20)
	Channel head castings	SA-216 WCC
Pressurizer	Shell	SA-302 Gr. B
	Heads	SA-216 WCC
	External plate (support skirt)	SA-516, Gr. 70
	Cladding, stainless	Type 304 equivalent
	Internal plate	SA-240 Type 304
	Spray Nozzle	SA-376 Type 316
Pressurizer relief tank	Shell	A-285 Gr. C
	Heads	A-285 Gr. C
	Internal surface coating	Amercoat 55 system
Piping	Pipes	A-376 Types 304 and 316
	Fittings	A-351 CF8M
	Nozzles	A-182 Type F316
Pump	Shaft	Type 304
	Impeller	A-351 CF8
	Casing	A-351 CF8M
Valves	Pressure containing parts	A-351 CF8 and CF8M; A-182 Type F316, and ASME SA182 Type F316 ASTM A479 Type 316

TABLE 4.2-2
Identification of Indian Point Unit 2 Reactor Vessel Beltline Region Weld Metal

Weld Location	Welding Process	Weld Control No.	Weld Wire Type	Weld Wire Heat No.	Flux Type	Lot No.	Post-Weld Heat Treatment
Nozzle shell vertical seam 1-042 A, B, and C	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1125 ± 25°F 25 hr-FC
Inter shell vertical seam circle seam 8-042	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1125 ± 25°F 25 hr-FC
Nozzle shell to inter seam 2-042 A, B, and C	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1125 ± 25°F 25 hr-FC
Inter shell to lower shell circle seam 9-042	Submerged Arc	M1.03	RACO 3 +Ni 200	34B009 N9867A	Linde 1092	3708	1150 ± 25°F 40 hr-FC
Lower shell vertical seams 3-042 A and B	Submerged Arc	-	RACO 3 +Ni 200	W5214 -	Linde 1092	3576	1150 ± 25°F 40 hr-FC
Surveillance weld	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1150 ± 25°F 19 3/4 hr-FC

TABLE 4.2-3
Chemical Composition of Reactor Vessel Beltline Region Weld Metal

Weld Wire Type	Weld Wire Heat No.	Flux Type	Lot No.	C	Mn	P	S	Si	Mo	Cr	Ni	Cu
				Weight Percent								
RACO 3	W5214	Linde 1092	3600	.11	1.20	.021	.012	.19	.52	--	--	--
RACO 3	34B009	Linde 1092	3708	.14	2.01	.010	.017	.04	.51	--	--	--$_1$
RACO 3	W5214	Linde 1092	3576	.12	1.15	.021	.012	.21	.56	--	--	--
Surveillance Weld - Not Performed												

Notes:

1. Chemical analysis of bare wire - No as-deposited analysis available.

TABLE 4.2-4
Mechanical Properties of Reactor Vessel Beltline Region Weld Metal

Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	^{T}NDT$_1$ (°F)	Energy at 10°F (ft-lbs)	^{RT}NDT$_1$ (°F)	Shelf Energy (ft-lbs)	YS (ksi)	UTS (ksi)	Elong Percent	RA Percent
RACO 3	W5214	Linde 1092	3600	0	103,93,95	0	--	65.5	80.0	31.0	71.5
RACO 3	34B009	Linde 1092	3708	0	84,71,90	0	--	67.9	84.2	31.0	69.8
RACO 3	W5214	Linde 1092	3576	0	57,51,69	0	--	68.5	85.0	27.5	68.5
Surveillance Weld				0	78,74,81	0	121	64.75	80.85	27.7	72.7

NOTES:

1. Estimated per NRC Standard Review Plan Section 5.3.2.

TABLE 4.2-5
Maximum 32 EFPY Fluence at Vessel Inner Wall Locations

Plate or Weld Location	Seam or Plate No.	Fluence (n/cm^2)
Nozzle Shell Vertical Seam	1-042A	6.6×10^{17}
Nozzle Shell Vertical Seam	1-042B	4.4×10^{17}
Nozzle Shell Vertical Seam	1-042C	1.1×10^{18}
Nozzle Shell to Inter. Shell Circle Seam	8-042	1.3×10^{18}
Intermediate Shell Vertical Seam	2-042A	8.8×10^{18}
Intermediate Shell Vertical Seam	2-042B	8.8×10^{18}
Intermediate Shell Vertical Seam	2-042C	5.0×10^{18}
Intermediate Shell to Lower Shell Circle Seam	9-042	1.6×10^{19}
Lower Shell Vertical Seam	3-042A	7.0×10^{18}
Lower Shell Vertical Seam	3-042B	7.0×10^{18}
Nozzle Shell Plate	B2001-1	1.3×10^{18}
Nozzle Shell Plate	B2001-2	1.3×10^{18}
Nozzle Shell Plate	B2001-3	1.3×10^{18}
Intermediate Shell Plate	B2002-1	1.6×10^{19}
Intermediate Shell Plate	B2002-2	1.6×10^{19}
Intermediate Shell Plate	B2002-3	1.6×10^{19}
Lower Shell Plate	B2003-1	1.6×10^{19}
Lower Shell Plate	B2003-2	1.6×10^{19}

TABLE 4.2-6
Identification of Reactor Vessel Beltline Region Plate Material

Component	Plate No.	Heat No.	Mat'l Spec No.	Supplier	Heat Treatment Austenitize	Heat Treatment Temper	Heat Treatment Stress Relief
Nozzle shell	B2001-1	B4679	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 60hr-FC
Nozzle shell	B2001-2	B4701	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 60hr-FC
Nozzle shell	B2001-3	A9870	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 50hr-FC
Inter shell	B2002-1[1]	B4688	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 50hr-FC
Inter shell	B2002-2[1]	B4701	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 50hr-FC
Inter shell	B2002-3[1]	B4922	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 40hr-FC
Lower shell	B2003-1	B4791	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 40hr-FC
Lower shell	B2003-2	B4782	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 40hr-FC

Notes:

1. Surveillance Material.

TABLE 4.2-7
Chemical Composition of Reactor Vessel Beltline Region
Plate Material, Weight Percent

Plate No.	C	Mn	P	S	Si	Ni	Mo	Cu
B2001-1[1]	0.22	1.35	0.010	0.022	0.24	0.50	0.46	0.20
B2001-2[1]	0.23	1.27	0.011	0.021	0.23	0.43	0.47	0.14
B2001-3[1]	0.23	1.35	0.012	0.025	0.26	0.50	0.48	0.19
B2002-1[2]	0.20	1.28	0.010	0.019	0.25	0.65	0.46	0.19
B2002-2[2]	0.22	1.30	0.014	0.020	0.22	0.46	0.50	0.17
B2002-3[2]	0.22	1.29	0.011	0.018	0.25	0.60	0.46	0.25
B2003-1[1]	0.23	1.33	0.011	0.025	0.23	0.66	0.48	0.20
B2003-2[1]	0.21	1.30	0.010	0.021	0.23	0.48	0.45	0.19

Notes:

1. Surveillance Material - No analysis performed other than reported by supplier.
2. Best estimate Cu and Ni weight percent [26].

TABLE 4.2-8
Mechanical Properties of Reactor Vessel Beltline Region Plate Material

Plate No.	^{T}NDT (°F)	^{RT}NDT$_1$ (°F)	Shelf Energy$_1$ (ft-lb)	YS (ksi)	UTS (ksi)	Elongation (percent)	RA (percent)	
B2001-1	-10	24	69	67.25	87.75	26.00	64.45	
B2001-2	-10	18	63.5	63.25	85.25	27.25	65.75	
B2001-3	-10	25	69	65.25	86.75	25.00	63.75	
B2002-1	-20	34	70	70.75	91.50	25.00	64.75	
B2002-2	-30	21	73	65.00	85.25	26.50	67.00	
B2002-3	-10	21	73.5	68.95	90.50	26.75	67.75	
B2003-1	-20	20	71	65.75	87.25	27.75	65.50	
B2003-2	-20	-20	88	61.25	81.60	30.75	70.50	
B2002-1	-	34	76	67.17	88.40	25.20	67.6	
B2002-2	-	34	75	64.55	87.15	27.65	69.8	Surveillance
B2002-3	-	39	72.5	65.32	87.32	26.30	67.0	Test Data

Notes:

1. Estimated from longitudinal data per NRC Standard Review Plan Section 5.3.2.

TABLE 4.2-9
Summary of Charpy V-notch and Drop Weight Tests

Component	Grade	30-ft-lb Fix (°F)	Drop Weight NDT (°F)
Head dome	A533B CL1	-2	10
Head peel segment	A533B CL1	-10	10
Head peel segment	A533B CL1	12	0
Upper shell plate	A533B CL1	33	-10
Upper shell plate	A533B CL1	31	-10
Upper shell plate	A533B CL1	9	-10
Intermediate shell plate	A533B CL1	14	-20
Intermediate shell plate	A533B CL1	-11	-30
Intermediate shell plate	A533B CL1	18	-10
Lower shell plate	A533B CL1	-5	-20
Lower shell plate	A533B CL1	-32	-20
Bottom peel segment	A533B CL1	-12	-20
Bottom peel segment	A533B CL1	-9	-10
Bottom dome	A533B CL1	8	-30
Head flange	A508 CL2	10	-
Vessel flange	A508 CL2	-18	-
Inlet Nozzle	A508 CL2	-102	-
Inlet Nozzle	A508 CL2	-84	-
Inlet Nozzle	A508 CL2	-95	-
Inlet Nozzle	A508 CL2	-51	-
Outlet Nozzle	A508 CL2	-32	-
Outlet Nozzle	A508 CL2	<10	-
Outlet Nozzle	A508 CL2	-45	-
Outlet Nozzle	A508 CL2	<10	-

TABLE 4.2-10
Reactor Vessel Beltline Fluence

Fast Neutron Fluence (>1 MeV)
32 Effective Full Power Years
(n/cm^2)[1]

Reactor vessel Interior surface	1.39×10^{19}
¼ vessel thickness (1/4 T)	9.04×10^{18}
¾ vessel thickness (3/4 T)	3.48×10^{18}

Fast Neutron Fluence (>1 MeV)
48 Effective Full Power Years
(n/cm^2)

	Vessel Plates and Circumferential Welds (45^o azimuthal position)[2]	Axial Welds (30^o azimuthal position)[2]
Interior surface	1.906×10^{19}	1.295×10^{19}
¼ vessel thickness (1/4 T)	1.136×10^{19}	7.72×10^{18}
¾ vessel thickness (3/4 T)	4.04×10^{18}	2.74×10^{18}

Notes:
1. These values are calculated based upon experimental results from the measurements on the fourth surveillance capsule V. See Reference 11.
2. The 30^o fluences are used to calculate embrittlement parameters for the beltline axial welds because these welds are at azimuthal locations of 0, 15, and 30 degrees. The 45^o fluences are used for all other reactor vessel beltline components.

4.2 FIGURES

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 design and stress analysis, material selection and fabrication, quality control, and operations control.

4.3.1.1 Reactor Vessel

A stress evaluation of the reactor vessel has been carried out in accordance with the rules of Section III of the ASME Nuclear 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 most significant transients with regard to cumulative fatigue of the reactor vessel are loss of load transient and loss-of-flow transients. A summary of fatigue usage factors for components of the reactor vessel is given in Table 4.3-2. 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 now in service.
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.

To preclude the possibility of brittle failure, a reactor vessel material surveillance program that meets the requirements of 10CFR50 App. H, is implemented to monitor the change in reactor vessel materials due to neutron radiation.

The radiation induced shift in Reference Temperature nil-ductility transition (RT_{NDT}) is periodically assessed during the life of the plant by testing of vessel material samples that are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. Regulatory Guide 1.99, Rev.2, “Radiation Embrittlement of Reactor Vessel Materials”, is utilized to predict the radiation induced change in the (RT_{NDT}) and calculate a new Adjusted Reference Temperature (ART). To compensate for any increase in the (RT_{NDT}) caused by irradiation, the heatup and cooldown pressure temperature limits given in the Technical Specifications are periodically changed to comply with 10CFR50, Appendix G, “Fracture Toughness Requirements”.

The vessel closure contains fifty-four 7-in. studs. The stud material has a minimum yield strength of 104,100 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is approximately 40,000 psi. This means that 21 of the 54 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.

The normal operating temperature always exceeds even the highest anticipated DTT during the life of the plant. Thus the emphasis of conservative operation is placed on heatup and cooldown because long term irradiation of the vessel raises the DTT and thereby limits the heatup or cooldown rates. The conservatism in setting up the temperature-pressure relationship limits stated above are:

1. Use of a stress concentration factor of 4 on assumed flaws in calculating the stresses.
2. Use of nominal yield of material instead of actual yield.
3. Neglecting the increase in yield strength resulting from radiation effects.

4.3.1.2 Steam Generators

Calculations confirm that the steam generator tube sheet will withstand the loading (which is a quasi-static rather than a shock loading) by loss of reactor coolant.

The rupture of primary or secondary piping has been assumed to impose a maximum pressure differential of 2485 psi across the tubes and tube sheet from the primary side or a maximum pressure differential of 1035 psi across the tubes and tube sheet from the secondary side, respectively. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet). This criterion prevents any violation of the containment boundary.

An examination of stresses under these conditions shows that for the case of a 2485 psi maximum tube sheet pressure differential, the stresses are within acceptable limits.

The tubes were designed to the requirements (including stress limitations) of Section III for normal operation, assuming 1700 psi as the normal operating pressure differential. Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

An evaluation determined the extent of tube wall thinning that could be tolerated under accident conditions. The worst-case loading conditions are assumed to be imposed upon uniformly thinned tubes at the most critical location in the steam generator. Under such a postulated design basis accident, vibration is short enough duration that there is no endurance issue to be considered.

The steam generator tubes, existing originally at their minimum wall thickness and reduced by a conservative general corrosion and erosion loss, provide an adequate safety margin (sufficient wall thickness) in addition to the minimum required for a maximum stress less than the allowable stress limit, as defined by the ASME Code.

Studies have been made on tubing of the size in the replacement steam generators under accident loadings. The results show that the maximum Level D Service condition stress due to combined pipe rupture and safe shutdown earthquake loads is less than the allowable limit. The tube thickness required to achieve the acceptable stress is less than the minimum steam generator tube wall thickness, which is reduced to account for assumed general corrosion and erosion rate. Thus, an adequate safety margin is exhibited. The general corrosion rate is based on a conservative weight-loss rate for Alloy 600 tubing in flowing, 650°F primary-side reactor coolant fluid. The estimated weight loss, based on testing when equated to a thinning rate and projected over a 60-year design objective, is much less than the assumed corrosion allowance of 3 mils. This leaves the remainder of the general corrosion allowance for thinning on the secondary side.

Potential sources of tube excitation are considered, including primary fluid flow within the U-tubes, mechanically induced vibration, and secondary fluid flow on the outside of the U-tubes. The effects of primary fluid flow and mechanically induced vibration, including those developed by the canned-motor pump, are acceptable during normal operation. The primary source of potential tube degradation due to vibration is the hydrodynamic excitation of the tubes by the secondary fluid. This area has been emphasized in both analyses and tests, including evaluation of steam generator operating experience.

Three potential tube vibration mechanisms related to hydrodynamic excitation of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulence, and fluid-elastic vibration mechanisms.

Non-uniform, two-phase turbulent flow exists throughout most of the tube bundle. Therefore, vortex shedding is possible only for the outer few rows of the inlet region. 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 inlet region. Bounding calculations consistent with laboratory test parameters confirmed that vibration amplitudes would be acceptably small, 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. These vibrations cause stresses that are two orders of magnitude below fatigue limits for the tubing material. Therefore, neither unacceptable tube wear nor fatigue degradation due to secondary flow turbulence is anticipated.

Fluid elastic tube vibration is potentially more severe than either vortex shedding or turbulence because it is a self-excited mechanism. Relatively large tube amplitudes can feed back proportionally large tube driving forces if an instability threshold is exceeded. Tube support spacing in 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 for tubes effectively supported. This approach provides large margins against initiation of fluid elastic vibration for tubes effectively supported by the tube support system.

Small clearances between the tubes and the 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 that location. Fluid-elastic tube response within available support clearances is therefore theoretically possible if secondary flow conditions exceed the instability threshold when no support is assumed 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 provided 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, based on previous experience in fabricating steam generators with fit-up control typical of the replacement steam generator. The replacement steam generator includes a number of features that minimize the potential for tube wear at tube supports. Provisions to minimize the potential for wear include the spacing between the tube supports, the configuration of the broached hole through the support plate, the surface finish of the broached hole in the tube support plate, the clearance between the tube and the hole in the tube support plate, and the tube support plate material selection.

Tube bending stresses corresponding to tube vibration response remain more than two orders of magnitude below fatigue limits as a consequence of vibration amplitudes constrained by available clearances. The analyses and tests for limiting postulated fit-up conditions include simultaneous contributions from flow turbulence.

As outlined, analyses and tests demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the replacement steam generators. Operating experience with steam generators having the same size tubes and similar flow conditions supports this conclusion.

The U-bend fatigue (discussed in NRC Bulletin 88-02) is not a consideration in the replacement steam generators. The mechanism considered in Bulletin 88-02 requires denting of the top tube support plate. But this is not expected with the stainless steel tube support plates in the replacement steam generator.

The stress limits for Service Level D that allow inelastic deformation are supplemented with the requirements of "Rules for Evaluation of Service Loadings with Level D Service Limits," Appendix F of ASME Code, Section III. The limits and rules of Appendix F confirm that pressure boundary integrity and core support structural integrity are maintained but do not confirm operability. The limits and rules of Appendix F do not apply to the portion of the component or support in which the failure has been postulated.

The structural stress analyses performed on the replacement steam generators consider the loadings specified. These loads result from thermal expansion, pressure, weight, earthquake, pipe rupture, and plant operational thermal and pressure transients. Dynamic effects of pipe rupture, including the loss of coolant accident, are not included in loading combinations when the leak-before-break criteria are satisfied.

The combination of safe shutdown earthquake plus pipe rupture loads by square-root-sum-of-the-squares is considered. The dynamic effects of pipe rupture that are combined with safe shutdown earthquake in loading combinations are combined using the square-root-sum-of-the-squares method.

The integrity of the pressure boundary of safety-related components is provided by the use of the ASME Code. The replacement steam generators, including the transition cone, lower shell, tubesheet, and channel head are constructed to the requirements of the ASME Code, Section III, 1980 Edition, plus winter 1981 addenda, which is reconciled to the design code of record, the 1965 ASME Boiler and Pressure Vessel Code, Section III, plus Addenda thru summer 1966. Using the methods and equations in the ASME Code, stress levels in the components and supports are calculated for various load combinations. These load combinations may include the effects of internal pressure, dead weight of the component and insulation, and fluid, thermal expansion, dynamic loads due to seismic motion, and other loads. The evaluation of the stress levels and fatigue usage for the steam generator pressure boundary is calculated for the specified loading conditions and demonstrates that the values are less than the allowable limits. These calculations are documented in a Stress Report as required by the ASME Code. Evaluation of the secondary shell in contact with secondary water assumes a 0.050 inch corrosion allowance. The analysis of the support plates assumes no corrosion allowance.

The ASME Code, Section III requires that a design specification be prepared for ASME components. The specification conforms to and is certified to the requirements of ASME Code, Section III. The Code also requires a design report for safety-related components, to demonstrate that the as-built component meets the requirements of the relevant ASME Design Specification and the applicable ASME Code. The design specifications and design reports will be completed by the Combined License applicant or his agent. Design specifications for ASME components and piping are prepared utilizing procedures that meet the ASME Code. The design report includes as-built reconciliation.

4.3.1.3 Piping

The reactor coolant system piping has been designed for normal and emergency conditions. For the emergency condition, the piping has been 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 vibrations from the reactor coolant pump impeller or from the pistons of the charging pump.

In 1989, the NRC approved changes to the design bases with respect to dynamic affects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4.

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 residual heat removal systems. The reactor coolant system is dependent upon the steam generators, and the steam, feedwater, and condensate systems for decay heat removal from normal operating conditions to a reactor coolant temperature of approximately 350°F. The layout of the system ensures the natural circulation capability to permit adequate core cooling following a loss of power to all main reactor coolant pumps. Further details are given in Section 14.1.6.1.

The NRC reviewed the Indian Point 2 response to issues concerning natural circulation cooldown in their safety evaluation report dated August 1, 1983 (reference 20), and determined that Con Edison met the requirements of Generic Letter 81-21.

The flow diagram of the steam and power conversion system is shown on Plant Drawings 227780, 9321-2017, 235308 [Formerly UFSAR Figure 10.2-1 sheets 1 to 3], 9321-2025

[Formerly UFSAR Figure 10.2-4], 9321-2018 and 235307 [Formerly UFSAR Figure 10.2-5 sheets 1 and 2], and 9321-2019 [Formerly UFSAR Figure 10.2-7]. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the feedwater system may be pumped into the steam generators and the resultant steam vented to the atmosphere. The auxiliary feedwater system will supply water to the steam generators in the event that the main feedwater system is unavailable.

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 that reflects consideration of all design loadings detailed in the design specification has been prepared by the manufacturer. The analysis shows that the reactor vessel, steam generator, reactor coolant pump casing, and pressurizer comply with the stress limits of Section III of the ASME Code. A similar analysis of the piping shows that it complies with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V-notch toughness test curves were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator, and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times. In addition, drop weight 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.

4.3.4 Overpressure Protection

The reactor coolant system is protected against overpressure by safety valves located on the top of the pressurizer. The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10-percent, in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The capacity of the pressurizer safety valves is determined from considerations of: (1) the reactor protective system, and (2) accident or transient conditions, which may potentially cause overpressure.

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

Details of the analysis are reported in Section 14.1.8. Experience has shown that the safety valve capacity so determined is adequate for all the other transients as the results of Section 14.1 show.

The report "Summary Report of Safety and Relief Valve Installation and Re-Analysis for ASME Class 1 and Class 2 Systems in Indian Point Unit No. 2" (Reference 5) describes the general scope, design and installation criteria, significant assumptions, methods of analysis, and maximum combined stresses for those applicable safety and relief valves in the reactor coolant system, main steam system, chemical and volume control system, safety injection system, component cooling water system, and service water system.

In response to NUREG-0737 Section II.D.1, a test program for the pressurizer safety and relief valves was formulated by the Electric Power Research Institute to provide full-scale test data confirming the functional ability of the reactor coolant system power-operated relief valves and safety valves for expected operating and accident conditions, and to obtain sufficient piping thermal-hydraulic load data to permit confirmation of models that may be used for plant-unique analysis of safety and relief discharge piping systems. The Indian Point 2 plant-specific evaluations regarding this generic issue are contained in Con Edison submittals to NRC dated July 1, 1982, September 15, 1982, June 15, 1984, June 14, 1985 and October 18, 1985. This program satisfied the requirements of NUREG-0737, as documented in NRC's Safety Evaluation Report (SER) dated August 5, 1987 (Reference 16).

Item II.K.3.2 of NUREG-0737 required licensees of pressurized water reactors to submit a report to the NRC staff documenting the various actions taken to decrease the probability of a small break LOCA caused by a stuck-open PORV and show how these actions constitute sufficient improvements to safety. Based upon the results of the report submitted in response to item II.K.3.2, licensees were to assess whether an automatic PORV isolation system was required. If required, licensees were to submit a system design that uses the PORV block valve to automatically protect against a small break LOCA caused by a stuck open PORV.

The Westinghouse Owners Group submitted a generic report to the NRC staff in response to Item II.K.3.2 (Reference 17). Con Edison's response to the NRC on this matter (Reference 18) adopted the conclusions reached in the aforementioned report as applicable to IP2, namely that the concept of an automatic PORV block valve closure system cannot be warranted on the basis of providing additional protection against a PORV LOCA. On this basis, Con Edison proposed no modifications to provide automatic isolation of the PORVs.

The NRC reviewed Con Edison's submittal and found that the requirements of NUREG-0737, Item II.K.3.2 were met with the existing PORV, safety valve and reactor high pressure trip setpoints and that an automatic PORV isolation system was not required for IP2 (Reference 19).

4.3.4.1 Reactor Coolant System Overpressure Protection System

An overpressure protection system to prevent reactor coolant system pressure exceeding the 10 CFR 50 Appendix G curves has been installed. It is a three-channel, analog, curve-tracking arrangement, which would initiate an appropriate chain of coincidence logic for the purpose of automatically preventing a violation of the operating Technical Specifications temperature/pressure curves for the reactor vessel.

In order to develop the overpressure protection system setpoint limit curve for the Technical Specifications, heatup and cooldown limit curves are calculated (Ref.21), using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} (ART) of the limiting material in the core region of the reactor vessel is determined by using unirradiated reactor vessel material fracture toughness properties, estimating the radiation induced change in RT_{NDT}, and adding margin for uncertainty. The unirradiated RT_{NDT} is designed as the higher of either the drop weight nil ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35 mils lateral expansion (transverse to the primary working direction), less 60F degrees. The method used to calculate the ART values at 1/4T and 3/4T locations, (where T is the thickness of the reactor vessel at the beltline region not including the cladding), complies with Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials”.

The heatup and cooldown curves are generated using the most limiting ART values and the methodology documented in Westinghouse Report WCAP-14040-NP-A, Rev.2, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown limit curves”, with the following exceptions:

1) The fluence values used are calculated fluence values, rather than best estimate fluence values.
2) The Kic critical stress intensities are used in place of Kia critical stress intensities, in compliance with ASME Code Case N-640,
3) The 1996 version of Appendix G to ASME Section XI is used instead of the 1989 version, and
4) Pressure-temperature limit curves were generated with the most limiting circumferential weld ART in conjunction with Code Case N-588. These curves are bounded by curves using the standard axial flaw methodology of the ASME Code 1996, App. G with the ART from the limiting plate material.

The heatup and cooldown pressure-temperature limit curves (“10CFR50 App. G limits”) so obtained (Ref.21) are valid for 48 EFPY.

Thermal-hydraulic analysis (Ref.22) accounting for the effects of pressure bias and pressure overshoot during mass and heat input transients, is utilized to develop setpoint curves to actuate the Power Operated Relief Valves (PORVs) and prevent the reactor coolant system from exceeding the 10 CFR50 App. G limits. The analysis demonstrates that a single power operated relief valve is capable of mitigating the worst possible mass or heat input transient, thereby ensuring that the peak reactor coolant system pressure for the Indian Point Unit No. 2 remains below the 10 CFR50 Appendix G limits were such a transient to occur. Overpressure Protection System setpoint curves are further adjusted for instrument error, and these latter curves are utilized as heatup and cooldown limits in plant operating procedures. Also, additional administrative controls are utilized to protect the Residual Heat Removal System from reactor coolant overpressurization events.

The overpressure protection system does not change the primary system operation or relief system operation during normal plant operation. The system allows for more close control of system heatup and cooldown through more accurate instrumentation and monitoring. Spurious opening and/or closing of the power operated relief valves is essentially eliminated by the new two-out-of-three logic (if one channel were to fail the valve would not malfunction). Thus, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased.

The overpressure protection system maintains the existing operational function of all existing plant components. There is an expansion of certain component functions to enhance the controllability of primary system pressure during heatup and cooldown. Inasmuch as there is better control of existing plant components and no change to their operation, the possibility for an accident or malfunction of a different type than any evaluated previously is not increased.

The overpressure protection system allows heatup and cooldown guidelines to be strictly followed both by automatic and manual means, thereby reducing the possibility of violating significant parameters and maintain an orderly heatup and cooldown. Thus the margin of safety as defined in the bases for the facility Technical Specifications is not reduced.

NRC acceptance of the Indian Point 2 low temperature overpressure protection system and the relevant Consolidated Edison submittals are contained in References 12, 13, 14 and 15.

4.3.4.2 Nitrogen System

A nitrogen system actuates the power operated relief valves (PORVs) PCV-455C and PCV-456. The PORV nitrogen system is tapped from the existing nitrogen supply header to Safety Injection (SI) system accumulators at a location downstream of pressure regulator valve PCV-942 and relief valve RV-1816, which is set at 1100 psig. The nitrogen pressure to the PORVs is reduced to 100 psig by pressure regulator valves PRV-3100 and PRV-3101. Containment isolation of the PORV nitrogen system is provided by valves 4312 and 863.

The instrument nitrogen system includes two accumulators, each holding approximately 13-ft^3 of nitrogen. In case the nitrogen supply is lost, these accumulators, with a minimum initial pressure of 600 psig, can support cycling (full open/close) the power operated relief valves for a minimum of 10 minutes.

The nitrogen system is provided with pressure indicating alarms located on the SKF panel in the control room to provide information to the operator in case of low pressure in the nitrogen accumulators. Pressure alarms also on the SKF panel provide indication for nitrogen supply regulator malfunction.

The PORV nitrogen system is designated Class A, Seismic Class I. The accumulators are designed to ASME Section VIII, Div.1 and piping is designed to ANSI B31.1. The PORV nitrogen system piping can withstand 1100 psi, the relief setting of valve RV-1816. The design of the PORV nitrogen system further considered the potential to generate missiles. To ensure that none of the components of the nitrogen system would become a source of missiles, the valves are forged, have a bolted clamseal bonnet and have stems which back seat. This rules out the possibility of ejecting valve stems as, even if it were assumed that the stem threads fail, the back seat or the upset end cannot penetrate the bonnet and thereby become a missile.

Also, the valves have been designed against bonnet-body connection failure and subsequent bonnet ejection by means of (1) using the design practice of ASME Section VIII, which limits the allowable stress of bolting material to less than 20-percent of its yield strength, (2) using the design practice of ASME Section VIII for flange design, and (3) by controlling the load during the bonnet-body connection stud tightening process. The pressure-containing parts except the flange and studs are designed per criteria established by USAS B16.5. Flanges and studs, where used, are designed in accordance with ASME Section VIII.

4.3.4.3 Evaluation of the Overpressure Protection System

With the overpressure protection system enabled, the power-operated relief valves will open automatically to prevent the reactor pressure vessel pressure from exceeding the Appendix G limits during a temperature range and for the effective full power years as defined in the Indian Point Unit 2 Technical Specifications, and there is a pressure excursion over the setpoint. With the Overpressure Protection System enabled, the power-operated relief valve isolation motor-operated valves are in the open position. Existing wide-range cold leg reactor coolant system temperature signals (TE-413, 433, and 443) are designed 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 Appendix G limit to which the system pressure limit must be adhered.

The arming function is initiated when the reactor coolant system temperature falls below a temperature defined by the Technical Specification. At the OPS enable temperature, the motor-operated valves (MOV-535 and 536) on the pressurizer will either be manually or automatically opened and the overpressure protection system logic system will be armed to prevent a possible overpressurization condition. Also, one half of a two-out-of-two coincidence logic will be satisfied to allow the relief valves to open in the event of an impending overpressure condition.

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 reactor coolant system pressures (Appendix G limit pressures) allowed at those temperatures. The difference between these maximum permissible reactor coolant system pressures and the actual reactor coolant system pressure, transmitted by 0 to 1500-psig transmitters (PT-413, 433, 443), 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 designated as train "A" (MOV-536 and PCV-456) and train "B" (MOV-535 and PCV-455C).

Various alarms and lights related to reactor coolant system overpressurization arming, actuation or non-availability of train "A" or "B" are located in the SGF and FB panels.

The alarms to indicate arming of the reactor coolant system overpressurization trains and actuation of the reactor coolant system overpressurization train "A" and train "B" are located on the SG panel. The motor-operated valves can be closed in the armed region by putting the motor-operated valves selector switch into the full locked position. White lights (one for each train), which indicate that the reactor coolant system overpressurization train is not available are located on the FB panel above the control switches.

As a protection against a common air supply failure causing inoperability of both power operated relief valves, the air system has been replaced by a nitrogen system with accumulators to supply each valve. (See description of nitrogen supply system, Section 4.3.4.2.) The electrical supply is from the 125VDC power panels, which are supplied by 480VAC through 125VDC battery chargers with backup by the station emergency batteries. The electrical activation uses two-out-of-three logic for valve actuation.

Manual disconnect switches provide a means to interrupt the power to a SOV, which will then result in the closure of the associated PORV. Operation with the switches closed permits the PORVs to open or close automatically or be manually operated to perform their pressure relief function. In the event of a fire in most of the fire zones in Fire Area A, these switches can be manually opened to prevent or mitigate the spurious opening of the PORVs due to a hot short in the control circuitry. To ensure that the PORV’s can perform their pressure relief function, the block valves are interlocked to open automatically when the pressurizer pressure reaches a preset limit below the pressure at which the PORVs open. In addition, the PORV actuation and reclosure setpoint calibration is checked each 24 months. Operation with the PORVs and block valves closed will prevent the spurious opening of both a PORV and its associated block valve in the event of a fire in certain fire zones.

4.3.5 Incident Potential

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

4.3.6 Redundancy

Each loop of the reactor coolant system contains a steam generator and a reactor coolant pump. Operation at reduced reactor power is possible with one loop out of service as limited by the facility Technical Specifications. For added reliability, power to the reactor coolant pumps is normally supplied by electrically separated buses as shown in Plant Drawing 231592 [Formerly UFSAR Figure 8.2-5]. The remote reactor head vent valves and the power operated relief valves and block valves are supplied with diverse and independent emergency power sources as described in Section 4.2.10.

REFERENCES FOR SECTION 4.3

1. H. Kihara and K. Masubuchi, "Effect of Residual Stress on Brittle Fracture," Welding Journal, Volume 38, April 1959.

2. Robertson, "Propagation From Brittle Fracture in Steel," Journal of the Iron and Steel Institute, 1953.

3. L. Porse, "Reactor Vessel Design Considering Radiation Effects," ASME Paper Number 63-WA-100.

4. Deleted

5. Letter from W. J. Cahill, Con Edison, to J. F. O'Leary, NRC, Subject: "Summary Report of Safety and Relief Valve Installation and Re-Analysis of ASME Class 1 and Class 2 Systems in Indian Point Unit 2," dated July 13, 1972.

6. Deleted

7. Deleted

8. Deleted

9. Westinghouse Electric Corporation, "Consolidated Edison Company of New York, Inc., Indian Point Unit 2 NSSS Stretch Rating - 3083.4 MWT Engineering Report," WCAP 12187 (Proprietary).

10. Deleted

11. Deleted

12. Letter (with attachments) from A. Schwencer, NRC to W. Butler, NRC, Subject: Safety Evaluation of the Electrical, Instrumentation and Control (EI&C) Aspects of the Low Temperature Overpressure Protection System, dated June 14, 1978.

13. Letter (with attachments) S. Varga, NRC to J. O'Toole, Con Edison, Subject: Indian Point 2- Low Temperature Overpressure Protection System, dated April 24, 1984.

14. Letter (with attachments) S. Varga, NRC to J. O'Toole, Con Edison, Subject: Low Temperature Overpressure Protection System at Indian Point Nuclear Generating Plant, Unit 2, dated June 28, 1984.

15. Letter (with attachments) J. O'Toole, Con Edison to S. Varga, NRC, Subject: Response to NRC Safety Evaluation Report, dated July 17, 1984.

16. Letter (with attachments) from M. Slosson, NRC to M. Selman, Con Edison, Subject: Safety Evaluation Report, NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves for Indian Point Nuclear Generating Unit No. 2, dated August 5, 1987.

17. WCAP-9804, "Probabilistic Analysis and Operational Data is Response to NUREG-0737, Item II.K.3.2, for Westinghouse NSSS Plants", Westinghouse Electric Corporation, February 1981

18. Letter (with attachments) from J. D. O'Toole, Con Edison, to D. G. Eisenhut, NRC, Response to NUREG-0737, Clarification of TMI Action Plan Requirements, Dated February 26, 1981

19. Letter (with attachments) from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: NUREG-0737 Items II.K.3.1 - Automatic PORV Isolation and II.K.3.2 - Report on PORVs the Indian Point Nuclear Generating Plant, Dated September 13, 1983

20. Letter (with attachment) from S.A. Varga, NRC, to J.D. O'Toole, Con Edison, Subject: Natural Circulation Cooldown For The Indian Point Nuclear Generating Plant, Unit No. 2 (IP-2), Dated August 1, 1983.

21. Westinghouse Electric Company Report: WCAP16752-NP, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation, January 2008.

22. [Deleted]Calculation No. FMX-00270 Rev. 2, "Indian Point Unit 2 Overpressure Protection System (OPS) Thermal Hydraulic Analysis, Setpoint Development and Technical Specification Revision," November 28, 2012

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	77,700 (1)	69,900
Head flange	45,370	80,100
Vessel flange	52,140	80,100
Closure studs	109,400	110,400
Primary nozzles – inlet outlet	45,500 49,390	80,100 80,100
Core support pad	55,280	69,900
Bottom head to shell	34,100	80,100
Bottom instrumentation	55,500	69,900
Nozzle belt to shell	37,900	80,100
Head Adapter Plugs	27,630	48,600

Note:
1. A simplified elastic plastic analysis was performed to justify exceeding the $3S_m$ limit.

TABLE 4.3-2
Summary of Cumulative Fatigue Usage Factors for
Components of the Reactor Vessel

Item	Usage Factor[1]
Control rod housing	0.01
Head flange	0.0107
Vessel flange	0.0229
Stud bolts	0.944
Primary nozzles - inlet	0.050
outlet	0.281
Core support pad (lateral)	0.904
Bottom head to shell	0.004
Bottom instrumentation	0.201
Nozzle belt to shell	0.0029
Head Adapter Plugs	0.0036

Notes:
1. As defined in Section III of the 1965 ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

TABLE 4.3-3
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TABLE 4.3-4
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4.4 SAFETY LIMITS AND CONDITIONS

4.4.1 System Heatup And Cooldown Rates

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

The stress level of material in the reactor vessel, or in other reactor coolant system components, is a combination of stresses caused by internal pressures and by thermal gradients. The latter are significant as they may result from a rate of change of reactor coolant temperature. Operating restrictions are imposed to limit the combined stresses to 20-percent of minimum yield stress when at the design transition temperature (DTT). The DTT is defined as the initial nil-ductility transition temperature (NDTT) plus the increase in NDTT due to irradiation experienced, plus 60^{o}F. This stress limit (20-percent of yield strength) is reduced linearly to a value of 10-percent of yield at a temperature 200^{o}F below DTT. Curves are incorporated in the plant operating procedures, which define the operating limits for initial operation and for end of life operation. To establish the latter, an adjustment is made for the maximum expected NDTT shift (240^{o}F), which the reactor vessel material will experience because of the fast neutron dose it will receive. The predicted shift will be verified by the surveillance program testing. The limits for initial operation are used to define operational limitations, and these curves are periodically updated to reflect irradiation exposure of the vessel and the results of the surveillance program.

4.4.2 Reactor Coolant Activity Limits

The plant systems are designed for operation with activity in the reactor coolant systems corresponding to 1-percent fuel defects. The accident analyses presented in Chapter 14 include the calculation of doses resulting from the release of activity initially contained in the primary system. The reactor coolant system operational activity limit is defined in the Technical Specifications.

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-percent 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.

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 nondestructive 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 are 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 are equivalent to those used for the reactor vessel.

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

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

2. Radiation surveillance program - In the surveillance programs, the evaluation of the radiation damage is based upon pre-irradiation and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading fracture mechanism type.

4.5.2 Reactor Vessel Surveillance Program

This program is directed toward evaluation of the effects of radiation on the fracture toughness of reactor vessel steels based on the transition temperature and fracture mechanics approaches, and is in accordance with ASTM E-185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors."

The reactor vessel surveillance program uses eight specimen capsules, which are located about 3-in. from the vessel wall directly opposite the center portion of the core. The capsules can be removed when the vessel head is removed. The capsules contain reactor vessel steel specimens from the shell plates and forgings located in the core region of the reactor, associated weld metal, and heat affected zone metal. In addition, correlation monitors made from fully documented specimens of SA302 Grade B material obtained through Subcommittee II of ASTM Committee E10, Radioisotopes and Radiation Effects, are inserted in the capsules. The 8 capsules contain at least 27 tensile specimens, 256 Charpy V-notch specimens (which will include weld metal and heat-affected zone material), and 42 wedge opening loading specimens. Dosimeters including pure Ni, Al-Co (0.15-percent Co), Cd shielded Al-Co, Cd0 shielded Np-237, and Cd0 shielded U-238 are placed in the impact specimens, tensile specimens, or filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion.

Irradiation of the specimens will be higher than the irradiation of the vessel because the specimens are located in the vicinity of the core corners and are closer to the core than the vessel itself. Since these specimens will experience higher irradiation and are actual samples from the materials used in the vessel, the nil-ductility transition temperature (NDTT) measurements will be representative of the vessel at a later time in life. Data from fracture toughness samples (wedge opening loading specimens) are expected to provide additional information for use in determining allowable stresses for irradiated material.

The Indian Point Unit 2 reactor vessel surveillance program was developed on the requirements provided in ASTM E-185 in effect at the time of construction. The details of the program are provided in WCAP-7323, “Consolidated Edison Co., Indian Point Unit No. 2 Reactor Vessel Surveillance Program”, Dated May 1969. The requirements of this program, currently form the basis for the reactor vessel surveillance program, as modified by the requirements of 10CFR50, Appendix H which state that the “… test procedures and reporting requirements must meet the requirements of ASTM E-185-82 to the extent practicable for the configuration of the specimens in the capsule.”

The following is a list of the surveillance program capsules along with the actual (past) and anticipated (future) withdrawal schedule based on the latest fluence and embrittlement calculations performed in accordance with the requirements of Regulatory Guide 1.99, Revision 2 (WCAP-15629).

Capsule	Location	Lead Factor	Withdrawal Date	Withdrawal EFPY (vessel)	Capsule Fluence (n/cm^2)
T	320°	3.42	End of Cycle 1	1.42	2.53×10^{18}
Y	220°	3.48	End of Cycle 2	2.34	4.55×10^{18}
Z	40°	3.53	End of Cycle 5	5.17	1.02×10^{19}
V	4°	1.18	End of Cycle 8	8.6	4.92×10^{18}
S	140°	3.5	Retired in Place	N/A	N/A
U*	176°	1.2	Spare	Spare	N/A
W*	184°	1.2	End of Cycle 28	Approx. 42 EFPY	2.0×10^{19}
X*	356°	1.2	Spare	Spare	N/A

*The withdrawal schedule for these three capsules is interchangeable due to the common lead factor and the common materials in the capsules

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Results of Surveillance Capsule analyses are discussed in Section 4.2.5.

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. In addition to the inspections shown in Table 4.5-1, there are those that the equipment supplier performed to confirm the adequacy of material he received and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator are governed by ASME Code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication are governed by USAS B31.1 and Westinghouse requirements, and are equivalent to those performed on ASME coded vessels. Procedures for performing the examinations are consistent with those established in the ASME Code Section III and are 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 the flaws, but equally as important, how the material was fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming, and fabricating processes received a 100-percent surface inspection by magnetic particle or liquid penetrant testing after all these operations were completed, although flaws in plates are inherently laminations in the center. 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-percent 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, witnessing key inspections not only in the supplier 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, required tests, and qualification of supplier personnel. An independent surveillance of the conformance to the fabrication and installation specifications and the quality control requirements of, among other things, the reactor coolant system components, was carried out by the United States Testing Company for Con Edison.

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 ensure that installation welds were of equal quality.

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

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

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

4.5.4 Inservice Inspection Considerations

The inservice inspection and testing program is discussed in Chapter 1.

4.5.5 Reactor Coolant System Surveillance

A preoperational and inservice structural surveillance program for the reactor vessel and reactor coolant system boundary was originally established as part of the Indian Point Unit 2 initial plant conditions. This program was designed to ensure the continued integrity of the reactor coolant system boundary and included specifications, as follows:

1. Prior to initial plant operation, an ultrasonic survey was made of reactor vessel shell welds, vessel nozzles, vessel flange welds, piping system butt welds, and major welds on the pressurizer, steam generator, coolant piping and components to establish preoperational system integrity, and establish baseline data.

2. An inspection interval of 10 years was established.

3. Postoperational nondestructive inspections were provided for. The results obtained from compliance with this specification were to be evaluated after 5 years, and the conclusions of this evaluation reviewed with the NRC.

4. The structural integrity of the reactor coolant system boundary was to be maintained throughout the life of the plant at the level required by the original acceptance standards. Any evidence as a result of the inspections that defects have initiated or grown, were to be investigated, including evaluation of comparable areas of the reactor coolant system.

5. The following definitions apply to the nondestructive inspection methods.

    a. UT - Volumetric examination using ultrasonic techniques.

    b. RT - Volumetric examination using radiography.

    c. PT - Surface examination using liquid penetrant methods.

    d. V - Visual examination by direct vision or by means of remote viewing devices.

    e. IV - Indirect visual examination performed during periods when the reactor coolant system is subjected to hydrostatic test pressure.

6. Detailed records of each inspection shall be maintained to allow comparison and evaluation of future inspections.

Current requirements for the primary system surveillance program are discussed in Section 5.5.6 of the facility Technical Specifications and in the Inservice Inspection and Testing Program, Chapter 1.

During the first ten year inspection of the reactor vessel, an indication was discovered in a longitudinal weld in the lower shell course. While the NRC in their October 16, 1984 safety evaluation concurred that the size of the indication was acceptable for plant operation, they required an augmented inspection program for the reactor vessel, which was incorporated into the Technical Specifications. By safety evaluation dated July 12, 1988, the NRC concluded that the required augmented inspection could be discontinued.

In addition, inservice surveillance of the steam generator tubes that are part of the primary coolant pressure boundary is detailed in Section 5.5.7 of the Technical Specifications. This surveillance program is to ensure their continued integrity and includes inspection requirements, corrective measures, reports, and NRC approval as a condition for plant operability. This program for inservice inspection of steam generator tubes exceeds the requirements of Regulatory Guide 1.83, Revision 1, July 1975.

4.5.6 Reactor Coolant Vent System Testing

The testing of the remote reactor head vent and power operated relief valves system valves is performed in accordance with ASME Code Section XI requirements for Category B valves.

TABLE 4.5-1 (Sheet 1 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT_1	UT_2	PT_3	MT_4	ET_5
1. Steam generator					
1.1 Tube sheet					
1.1.1 Forging		Yes		Yes	
1.1.2 Cladding		Yes_6	Yes_7		
1.2 Channel head					
1.2.1 Casting	Yes			Yes	
1.2.2 Cladding			Yes		
1.3 Secondary shell and head					
1.3.1 plates		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		
1.6.4 Steam and feedwater nozzles to shell	Yes			Yes	
1.6.5 Support brackets				Yes	
1.6.6 Tube to tube sheet			Yes		

TABLE 4.5-1 (Sheet 2 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT_1	UT_2	PT_3	MT_4	ET_5
1.6.7 Instrument connections					
(primary and secondary)				Yes	
1.6.8 Temporary attachments					
after removal				Yes	
1.6.9 After hydrostatic test					
(all welds and complete					
channel head)				Yes	
1.6.10 Nozzle safe ends	Yes		Yes		
(if forgings)					
1.6.11 Nozzle safe ends			Yes		
(if weld deposit)					
2. Pressurizer					
2.1 Heads					
2.1.1 Casting	Yes			Yes	
2.2.2 Cladding			Yes		
2.2 Shell					
2.2.1 Plates		Yes		Yes	
2.2.2 Cladding			Yes		
2.3 Heaters					
2.3.1 $Tubing_8$		Yes	Yes		
2.3.2 Centering of element	Yes				

TABLE 4.5-1 (Sheet 3 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT_1	UT_2	PT_3	MT_4	ET_5
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 weld surfaces after hydrostatic test				Yes	
3. Primary Coolant Piping					
3.1 Fittings (castings)	Yes		Yes		
3.2 Fittings (forgings)		Yes	Yes		

TABLE 4.5-1 (Sheet 4 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT[1]	UT[2]	PT[3]	MT[4]	ET[5]
3.3 Pipe[9]	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-in.)	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		
5. Reactor Vessel					
5.1 Forgings					
5.1.1 Flanges		Yes		Yes	
5.1.2 Studs		Yes		Yes	

TABLE 4.5-1 (Sheet 5 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT_1	UT_2	PT_3	MT_4	ET_5
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_{10}		Yes	
5.3.6 Nozzle-safe ends (if forging)	Yes		Yes		
5.3.7 Nozzle safe ends (if weld deposits)	Yes		Yes		
5.3.8 Head adaptor forging to head adaptor tube	Yes		Yes		
5.3.9 All welds after hydrotest				Yes	

TABLE 4.5-1 (Sheet 6 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT_1	UT_2	PT_3	MT_4	ET_5
6. Valves					
6.1 Castings	Yes		Yes		
6.2 Forgings (No UT for valves two inches and smaller)	Yes	Yes			

Notes:
1. RT - Radiographic.
2. UT - Ultrasonic.
3. PT - Dye Penetrant.
4. MT - Magnetic Particle.
5. ET - Eddy Current.
6. Flat Surfaces Only.
7. Weld Deposit Areas Only.
8. Or a UT and ET.
9. Except pressurizer surge line - UT only.
10. UT of Clad Bond-to-Base Metal.

4.6 METAL IMPACT MONITORING SYSTEM

4.6.1 General

The metal impact monitoring system is designed to enable early detection of any debris, detached internal structural items, and hardware present in the reactor coolant system.

A metal impact monitoring system for Indian Point Unit 2 was installed during the 1976 refueling outage and was operational when the plant returned to service in September 1976. At that time, component "signature acquisition" of the nuclear steam supply system components (baseline data) was obtained at selected plant operating conditions for future reference. The metal impact monitoring system was modified during the 1982 refueling outage.

4.6.2 Description

This system involves the use of a metal impact monitoring system capable of detecting changes in reactor coolant system vibrations and converting that input into an electronic signal thereby providing an indication to operating personnel that an undesireable level of foreign material may be present in the reactor coolant. While the installed system has no control capability, it is nevertheless quite valuable as an advisory system.

Metal impact monitoring is accomplished by the installation of specially developed transducers (accelerometers) mounted on the exterior of the reactor coolant system and steam generators. When the interior of the reactor coolant system is struck by bouncing debris, the structure is shock excited producing local wall accelerations that are detected by the transducers, amplified, conditioned, and fed to the metal impact monitoring system. The metal impact monitoring system further conditions the signals for recording and display in the control room.

The transducers are located on the following equipment:

1. Reactor vessel head.
2. Incore instrumentation penetration (below reactor vessel).
3. Steam generators.

APPENDIX 4A
DETERMINATION OF REACTOR PRESSURE
VESSEL NIL-DUCTILITY TRANSITION TEMPERATURE (NDTT)

4A.1 MEASUREMENT OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

The energy dependent neutron fluxes at the irradiation samples are obtained from the DOT[1], a two-dimensional discrete ordinates transport theory code. Dosimeters in the surveillance program include CdO shielded U-238, Np-237, Co-Al, Cu, Ni, Cd shielding Co-Al, and Fe from specimens, which will be contained in the capsule assemblies.

The specific activities of the dosimeters are to be determined by the multichannel analyzer and NAI scintillation detector. The equipment calibration shall be accomplished with ^{54}Mn and ^{60}Co radioactivity standards obtained from the U.S National Bureau of Standards or the equivalent. All activities will be corrected to the time-of-removal (TOR) at reactor shutdown.

Infinite dilute saturated activities (A_{SAT}) will be calculated for each of the dosimeters because A_{SAT} is directly related to the product of the energy dependent microscopic activation cross-section and the neutron flux density. The relationship between A_{TOR} and A_{SAT} is given by:

$$\frac{A_{TOR}}{A_{SAT}} = \sum_{m=1}^{m=n} P_m \left(1 - e^{-\lambda T} m\right) \left(e^{-\lambda t} m\right)$$

Where: λ = decay constant for the activation product, 1/day
t_m = decay time after operating period m, days
T_m = operating days P_m = average fraction of full power during operating period
P_m = average fraction of full power during operating period

The primary result desired from the dosimeter analysis in the total neutron fluence (E > 1 MeV) that the surveillance specimens and pressure vessel have received. The average flux density at full power is given by:

$$\phi = A_{SAT}/N_o\bar{\sigma}$$

Where: ϕ = energy dependent neutron flux density, n/cm^2-sec
$\bar{\sigma}$ = spectrum averaged activation cross-section, cm^2
N_o = number of target atoms per mg

The total neutron flux fluence is then equal to the product of the averaged neutron flux and the equivalent reactor operating time at full power.

4A.2 CALCULATION OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

In the analysis of the neutron environment within a pressurized water reactor geometry, predictions of the spatial neutron flux magnitude, and energy spectra are made with the DOT (two-dimensional discrete ordinates transport theory code). First, the radial and azimuthal distributions are obtained from an R, θ computation normalized to the reactor core power density representative of the axial midplane. A second calculation in R, Z geometry is used to provide relative axial variations of neutron flux in the pertinent regions of the pressure vessel. A three-dimensional description of the neutron environment is then constructed by assuming separability and using the relation:

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

Where ϕ (R, θ, E) represents the absolute neutron flux magnitude at the core midplane as determined from the R, θ computation and F(Z, E) is the relative axial distribution obtained from the R, Z analysis and normalized to unity at the core midplane.

From a neutronic standpoint, the inclusion of the surveillance capsule structures in the R, θ analytical model is significant. Neutron dosimetry from these capsules provides a means for evaluating the analytical model by direct comparison with measurement. Since the presence of the capsules has a marked impact on both the neutron flux magnitude and energy spectrum, a meaningful comparison of measurement and calculation can be made only if these perturbation effects are properly accounted for in the analysis.

Two distinct sets of transport calculations are carried out. The first, a single computation in the conventional forward mode, is used primarily to obtain relative neutron energy distributions throughout the reactor geometry as well as to establish relative radial distributions of exposure parameters (ϕ (E > 1.0 MeV), ϕ (E > 0.1 MeV), and dpa) through the vessel wall. The neutron spectral information is required for the interpretation of neutron dosimetry withdrawn from surveillance capsules as well as for the determination of exposure parameter ratios: i.e., dpa/ϕ(E > 1.0 MeV), within the pressure vessel geometry. The relative radial gradient information is required to permit the projection of measured exposure parameters to locations interior to the pressure vessel wall; i.e., the 1/4T, 1/2T, and 3/4T locations.

The second set of calculations consists of a series of adjoint analyses relating the fast neutron flux (E > 1.0 MeV) at surveillance capsule positions, and several azimuthal locations on the pressure vessel inner radius to neutron source distributions within the reactor core. The importance functions generated from these adjoint analyses provide the basis for all absolute exposure projections and comparison with measurement. These importance functions, when combined with cycle specific neutron source distributions, yield absolute predictions of neutron exposure at the locations of interest for each of the operating fuel cycles; and establish the means to perform similar predictions and dosimetry evaluations for all subsequent fuel cycles. It is important to note that the cycle specific neutron source distributions utilized in these analyses include not only spatial variations of fission rates within the reactor core; but, also account for the effects of varying neutron yield per fission and fission spectrum introduced by the build-in of plutonium as the burnup of individual fuel assemblies increased.

The absolute cycle specific data from the adjoint evaluations together with relative neutron energy spectra and radial distribution information from the forward calculation provide the means to:

1. Evaluate neutron dosimetry obtained from the surveillance capsule program.

2. Extrapolate dosimetry results to key locations at the inner radius and through the thickness of the pressure vessel wall.

3. Enable a direct comparison of analytical prediction with measurement.

4. Establish a mechanism for projection of pressure vessel exposure as the design of each new fuel cycle evolves.

The forward transport calculation is carried out in R, θ geometry using the DOT two-dimensional discrete ordinates code[1] and the SAILOR cross-section library[2]. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In these analyses anisotropic scattering was treated with a P_3 expansion of the cross-sections and the angular discretization was modeled with an S_8 order of angular quadrature. The reference forward calculation is normalized to a core power.

All adjoint analyses are also carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation from the SAILOR library. Adjoint source locations are chosen at several azimuthal locations at the pressure vessel inner radius and at the geometric center of surveillance capsules positioned at 4° and 40° relative to the core cardinal axes. Again these calculations are run in R, θ geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case, ϕ (E > 1.0 MeV). Having the importance functions and appropriate core source distributions, the response of interest could be calculated as:

$$R(r,\theta) = \int_r \int_\theta \int_E I(r,\theta,E)S(r,\theta,E)r\,dr\,d\theta\,dE$$

where:	$R(r, \theta)$	=	ϕ (E > 1.0 MeV) at radius r and azimuthal angle θ
	$I(r, \theta, E)$	=	Adjoint importance function at radius r, azimuthal angle θ, and neutron source energy E.
	$S(r, \theta, E)$	=	Neutron source strength at core location r, θ. and energy E.

Forward transport as well as the adjoint analyses for Indian Point Unit 2 were carried out and summarized in Reference 3.

In the R, θ analysis, the discretization of the angular flux is represented by a symmetric S_8 quadrature. However, in the R, Z case the use of this relatively low order quadrature set can often prove to be inadequate. At large depths within the pressure vessel, the axial distribution of neutron flux is dominated by neutron streaming in the annulus between the pressure vessel wall and the primary biological shield. To account for this effect a high resolution angular quadrature is required. Therefore, in this analysis a 124 angle asymmetric quadrature is employed. For regions of the reactor, which are above the core midplane, this quadrature is constructed with 109 angles biased in the upward directions, i.e., the direction of prime interest, and 15 angles biased downward. For analysis below the core midplane, the quadrature is reversed with 109 angles biased in the downward direction. Complete descriptions of both the symmetric S_8 and the asymmetric 124 angle quadratures are given in Reference 1.

The calculated fast neutron flux distributions may be used in conjunction with damage trend curves to predict the degree of embrittlement of the reactor vessel steel over its service life. The accuracy of these neutron flux profiles depends on the analyst's ability to define an appropriate core power distribution, the adequacy of the cross-sections used in the transport analysis, and the applicability of the geometric modeling of the reactor. Taken as a whole, these factors combine to yield an overall uncertainty of 20-percent in the prediction of neutron flux and fluence within the pressure vessel wall.

4A.3 MEASUREMENT OF THE INITIAL NIL-DUCTILITY TRANSITION TEMPERATURE OF THE REACTOR PRESSURE VESSEL BASE PLATE AND FORGINGS MATERIAL

The unirradiated or initial NDTT of pressure vessel reactor materials was measured by two methods. These methods were the drop weight test per ASTM E208 and the Charpy V-notch impact test (Type A) per ASTM E23.

The NDTT is defined in ASTM E208 as the temperature at which a drop weight test specimen is broken in a series of tests in which duplicate no-break performance occurs at a temperature of 10^{o}F higher.

The NDTT temperature, as determined by drop weight tests is the RT_{NDT} if, at 60^{o}F above the NDTT, at least 50-ft-lbs of energy and 35 mils lateral expansion are obtained in Charpy V tests on specimens oriented in the weak direction (traverse to the direction of maximum working).

The NDTT has been correlated with Charpy V-notch impact tests results.

For SA 302B and A508 Class 2 steels the Charpy V-notch "fix" temperature, which corresponds to NDTT is the temperature at 30-ft-lbs in accordance with Section III Table N-421 of the ASME Code for Nuclear Vessels. The curve of the temperature versus energy observed in breaking the specimen was plotted.

To obtain this curve 15 tests were performed, which include three tests at five different temperatures. The intersection of the energy versus temperature curve with the 30-ft-lbs ordinate is designated as NDTT.

As part of the Westinghouse surveillance program referred to above, Charpy V impact tests, tensile tests, and fracture mechanics specimens are taken from the plate of forging material. To assess any possible uncertainties in the consideration of NDTT shift for welds, heat affected zone and base metal, test specimens of these three "material types" have been also included in the reactor vessel surveillance program.

Encapsulated specimens are located on the outside diameter surface of the thermal shield where the fast neutron flux density is about three times that at the adjacent vessel wall surface. The capsules also contain several dosimeter materials for experimentally determining the average neutron flux density at each capsule location during the exposure period.

REFERENCES FOR APPENDIX 4A

1. R. G. Soltesz, et al, "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation - Volume 5, Two-Dimensional Discrete Ordinates Transport Techinque," WANL-PR-(LL)-304, August, 1970.

2. "ORNL RSIC Data Library Collection DLC-76, SAILOR Coupled Self Shielded, 47 Neutron, 20 Gamma-Ray, P_3, Cross Section Library for Light Water Reactors".

3. S.L. Anderson "Plant Specific Fast Neutron Exposure Evaluation of the Indian Point Unit 2 Reactor Pressure Vessel and Surveillance Capsules Fuel Cycles 1 through 9". Westinghouse report PSE-REA-88/127, July 1988.

APPENDIX 4B
SUPPORT STRUCTURES FOR REACTOR COOLANT SYSTEM COMPONENTS

The reactor coolant system components and their supports are designed as seismic Class I components as discussed in Section 1.11. In 2003, the reactor coolant loop and its component supports were re-analyzed due to a power uprate. This latest analysis does not consider the coincident combination of blowdown and seismic loads.

4B.1 REACTOR VESSEL

The reactor vessel support structure consists of a circular box section ring girder fabricated of carbon steel plates. The bottom flange of the girder is in continuous contact (except for openings for neutron detectors) with a non-yielding concrete foundation.

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

4B.2 STEAM GENERATORS

The steam generators are supported within a caged structural system consisting of four connected columns welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction with a system of "Lubrite" plates, hydraulic snubbers, guides, and stops to accommodate piping expansion. The "Lubrite" plates, hydraulic snubbers, guides, and stops were originally designed as a rigid support to resist the action of seismic and pipe break loads.

In 2000, the number of hydraulic snubbers supporting the steam generator frame in the direction of the hot leg, has been reduced from the original six down to two per steam generator. The two remaining snubbers are located at the upper support point of the frame at Elevation 92'-0". The analysis of the reactor coolant loop and of the steam generator support structure accounts for the replacement steam generator and for the reduced number of hydraulic snubbers.

4B.3 REACTOR COOLANT PUMP

Each reactor coolant pump is supported on a three-legged structural system consisting of three connected columns fabricated of carbon steel members, structural sections, and pipe. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement, and a system of tie rods and anchor bolts, which restrain the structure from movement beyond the calculated limits.

4B.4 PRESSURIZER

The pressurizer is supported on a free-standing structural system consisting of six connected columns fabricated of carbon steel members, all welded together and secured at the base by anchor bolts.

4B.5 PIPING

The reactor coolant piping layout is designed on the basis of providing "floating" supports for the steam generator and reactor coolant pump in order to absorb the thermal expansion from the fixed or anchored reactor vessel. A comprehensive thermal analysis has been performed to ensure that stresses induced by linear thermal expansion are within code limits.

4B.6 APPLICABILITY OF UNIT 3 PIPE BREAK ANALYSES TO UNIT 2

A report (Reference 1) entitled, "Analysis of Reactor Coolant System for Postulated Loss-of-Coolant Accident: Indian Point Unit 3 Nuclear Power Plant," has been submitted to the NRC. This report postulates pipe breaks at the locations in the primary loop, which induce the most severe asymmetric loads on the reactor vessel. The analyses performed included the effects of the addition of pipe motion limiters and demonstrate the adequacy of the entire system.

Reference 4 of Section 4.2 addresses the applicability of this report to Unit 2. Because of the similarity of the plants the nature of the system response, and the installation in Unit 2 of the modifications discussed in that report, the conclusions stated for Unit 3 in that report are found to be applicable to Unit 2.

4B.7 LEAK BEFORE BREAK

In 1989, the NRC approved elimination of the necessity for considering and protecting against dynamic effects of postulated primary loop pipe ruptures from the design basis of Indian Point Unit 2 as discussed in Section 4.1.2.4. "Leak before break" technology was applied as permitted by revised General Design Criterion 4 of 10CFR50, Appendix A. References 2, 3, 4 and 5 contain further information.

REFERENCES FOR APPENDIX 4B

1. "Analysis of Reactor Coolant System for Postulated Loss-Of-Coolant Accident: Indian Point Unit 3 Nuclear Power Plant." WCAP-9117 (Proprietary) and WCAP-9130 (Non-Proprietary), Westinghouse Electric Corporation.

2. May 23, 1988 letter, Bram to Document Control Desk, subject: Leak-Before-Break (LBB).

3. November 18, 1988 letter, Bram to Document Control Desk, subject: Leak-Before-Break (LBB) Submittal (TAC 68318).

4. January 12, 1989 letter, Bram to Document Control Desk, subject: Leak-Before-Break (LBB) Submittal (TAC 68318).

5. Letter from Donald Brinkman, NRC, to Stephen B. Bram, Con Edison, Subject: Safety Evaluation Report on Elimination of Dynamic Effect of Postulated Primary Loop Pipe Ruptures from Design Basis for Indian Point Unit 2 (TAC No. 68318), dated February 23, 1989.

APPENDIX 4C
SENSITIZED STAINLESS STEEL

4C.1 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 (Reference 1), which cover 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 that will cause cracking. Chlorides and fluorides 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 Unit 2, 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 overlaid with type 308L and Inconel weld metal to eliminate any question of intergranular attack in areas where there is limited accessibility for inservice inspection and plant maintenance. There is complete accessibility to the remaining reactor coolant system components. The pre-operational inspection of the reactor coolant system components provides assurance that there is no stress corrosion cracking of sensitized stainless steel.

4C.2 REACTOR COOLANT SYSTEM NOZZLE SAFE-ENDS

4C.2.1 Reactor Vessel Primary Nozzle Safe-Ends

1. <u>Method of Fabrication</u> (See Figure 4C-1)

   a. Wrought stainless steel - Type 316 Forging welded to SA-336 nozzle with Inconel weld metal. Attached prior to final post weld heat treatment.

   b. Forging was overlaid on ID and OD with type 308L stainless and Inconel weld metal. This was performed in the field after the primary coolant piping was attached to the nozzles.

2. Inspection

a. Forging safe-ends were examined by ultrasonic testing and penetrant testing at Combustion Engineering using Section III acceptance standards.

b. Weld overlay of the ID and OD surfaces was examined by ultrasonic testing and penetrant testing. The acceptance standards are shown below:

(1) Ultrasonic Acceptance Standards

Each discontinuity that produced a response equal to or exceeding the calibration reference line and was 0.5-in. or greater in length was considered rejectable and removed.

Discontinuities that produced a response equal to or greater than the calibration reference line and exceed 0.25-in., but were less than 0.5-in. in length were considered acceptable if separated by a minimum distance of 2-in. from similar discontinuities.

Each discontinuity that produced a response between 50 and 100-percent of the calibration reference line and exceeded one inch but was not more than 1.5-in. in length, were acceptable if separated by a minimum distance of 2-in. from similar indications.

(2) Penetrant Inspection Acceptance Standards

(a) Examination of welds by liquid penetrant methods were made over an area including the welds and base metal extending for at least 0.5-in. on each side of weld.

(b) Surfaces examined by fluid penetrant methods were free of laps, fissures, cracks, other linear indications.

(c) Weld area and adjacent wrought type base metal(s) - In any 6-in. length of weld and adjacent base metal examined, there were no indications greater than 0.62-in. in maximum dimension, nor were there more than six indications with sum of maximum dimensions specified herein. Any 6-in. length of weld was interpreted to denote the 6-in. length selected in the least favorable location with respect to the discontinuities disclosed by the inspection test. All surfaces examined were free of linearly disposed indications of four or more indications in a line and each separated by 1/16-in. or less, edge to edge.

(d) Weld area and adjacent cast type base metal(s) - In any 6-in. length of weld examined, there were no indications greater than those defined in 3, above. The adjacent cast base metal was free of random indications in excess of those shown in the following table for a distance of not less than 0.5-in. from toe(s) of weld:

Size of Indications, In.	Number per Square In.
> 1/8	None
>1/16 < 2/8	2
< 1/16	10

(e) All surfaces examined were free of linearly disposed indications of four or more indications in a line and each separated by 1/16-in. or less, edge to edge. Rounded indications were those which were circular or elliptical with the length less than twice the width.

4C.2.2 STEAM GENERATOR PRIMARY NOZZLE SAFE-ENDS (See Figure 4C-2)

1. Method of Fabrication

   Weld metal buttering applied to carbon steel (A-216 Casting) nozzles prior to final post weld heat treatment. Stainless weld metal for the first layer was type 309L, and for the balance was type 308L.

2. Inspection

   Buttered safe-ends were examined by penetrant testing and radiography testing using ASME Boiler and Pressure Vessel Code Section III acceptance standards.

4C.2.3 PRESSURIZER (See Figure 4C-3)

1. Method of Fabrication

   Wrought stainless steel pipe or forgings welded to carbon steel (A-216 Casting) nozzles with type 309 weld metal before post weld heat treatment. The surge nozzle safe-end was fabricated from SA-312 pipe, type 316, and the spray, relief, and safety nozzle safe-ends from SA-182 forgings, type 316.

2. Inspection

   Wrought material was examined by ultrasonic testing and penetrant testing using Section III acceptance standards.

4C.3 REACTOR COOLANT SYSTEM CONSTRUCTION

All primary piping and fittings were given a solution annealing treatment consisting of heating to 1900 - 1950°F, holding 1 hr/in. of thickness, and water quenching. This ensured that the material would not be sensitized.

Main coolant pipe welds are of type 308 or type 316 stainless steels. Welding was performed 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-in.) and the interpass temperature control of 350°F maximum, there will be no sensitization of the solution-treated pipe during welding.

Venting provisions have been made at high points throughout the reactor coolant system to relieve entrapped air when the system is filled and pressurized. Principally, vents are installed on the reactor coolant pumps with additional vents 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 used. 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 Specifications. In addition to the high point vents, a connection is installed downstream of the Power-operated Relief Valves to permit pulling the air out of the system under vacuum during system refilling.

4C.4 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 were 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 is free from cold spring.

As a precaution that the behavior of the reactor coolant system during operating conditions would be as predicted, measurements were made at incremental temperature increases during the hot functional test. The measurements were made to check the movement of the components at temperature and pressure to ensure interferences were not present. Data taken during the test were compared with the flexibility analysis predictions and evaluated.

4C.5 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 postoperational inspection plan was developed for the nozzle safe-ends within the reactor coolant system boundary. The pressurizer and steam generator stainless steel safe-ends that 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 on the safe-end, approximately 120-degrees of the top segment of the safe-ends are accessible for direct visual, surface, and remote volumetric inspection.

As specially designed devices for remote ultrasonic inspection and applicable procedures become available, and when metallurgical considerations indicate that this type of inspection is appropriate and necessary, such inspections will be accomplished utilizing the internal access to the reactor vessel safe-ends. Requirements for inspection of the reactor coolant system are detailed in the facility Technical Specifications.

REFERENCES FOR APPENDIX 4C

1. WCAP-7477L (Proprietary), Westinghouse Electric Corporation.

APPENDIX 4C FIGURES

Figure No.	**Title**
Figure 4C-1	Primary Nozzle Combustion Engineering Reactor Vessel
Figure 4C-2	Primary Nozzle Tampa Steam Generators
Figure 4C-3	Spray or Surge Nozzle Tampa Pressurizer