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 <u>Performance Objectives</u>

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 <u>General Design Criteria</u>

General design criteria (GDC) that apply to the reactor coolant system are given below.

4.1.2.1 <u>Quality Standards</u>

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 [Deleted] 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^{o}F per hr was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number [Deleted] 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. [Deleted]

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 Valves

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion-resistant materials. Connections to stainless steel piping are welded. Valves that perform a modulating function are equipped with either two sets of packing and an intermediate leakoff connection or have been designed with live-loaded packing which will either control or mitigate the potential for valve stem leakage due to modulating service.

4.2.2.9 Component Supports

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

4.2.3 Pressure-Relieving Devices

The reactor coolant system is protected against overpressure by control and protective circuits such as the high-pressure trip and by code relief valves connected to the top head of the pressurizer. The relief valves discharge into the pressurizer relief tank, which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in 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. [Deleted]

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 5.3×10^{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 1.39×10^{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 1.906×10^{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 0.85×10^{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^{3}P 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 <u>Leakage Prevention</u>

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 <u>Locating Leaks</u>

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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5. Letter from P. Zarakas, Con Edison, to T. Novak, NRC, Subject: Effects of Combining Seismic (SSE) and LOCA Responses on WCAP-9117 Analysis, dated September 3, 1980.

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12. Letter from J.D. O'Toole, Con Edison, to S.A. Varga, NRC, Subject: Reference Transition Temperature For Pressurized Thermal Shock Evaluations, dated January 22, 1986.

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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22. Letter from J. O'Toole, Con Edison, to M Slosson, NRC, Dated June 10, 1986.

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
		Stainless steel,
	Insulation	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	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		Flux		Weight Percent								
Type	Heat No.	Type	Lot No.	C	Mn	P	S	Si	Mo	Cr	Ni	Cu
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		Flux		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
Type	Heat No.	Type	Lot No.								
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	Austenitize	Heat Treatment Temper	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. [Deleted]

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 (if forgings)	Yes		Yes		
1.6.11 Nozzle safe ends (if weld deposit)			Yes		
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	RT1	UT2	PT3	MT4	ET5
3.3 Pipe9	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^o and 40^o 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, θ)	=	ϕ (E > 1.0 MeV) at radius r and azimuthal angle θ
	I (r, θ, E)	=	Adjoint importance function at radius r, azimuthal angle θ, and neutron source energy E.
	S (r, θ, 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°F higher.

The NDTT temperature, as determined by drop weight tests is the RT_{NDT} if, at 60°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. Method of Fabrication (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^{o}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^{o}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

CHAPTER 14
SAFETY ANALYSIS

14.0 INTRODUCTION

This chapter evaluates the safety aspects of the plant and demonstrates that the plant can be operated safely and that exposures from credible accidents do not exceed the applicable limits.

14.0.1 Accident Classification

This chapter is divided into four sections, each dealing with a different behavior category:

1. Core and Coolant Boundary Protection Analysis, Section 14.1 -The incidents presented in Section 14.1 generally have no offsite radiation consequences.

2. Standby Safeguards Analysis, Section 14.2 –The accidents presented in Section 14.2 are more severe and may cause the release of radioactive material to the environment.

3. Rupture of a Reactor Coolant Pipe, Section 14.3 - The accident presented in Section 14.3, the rupture of a reactor coolant pipe, is the worst-case accident and is the primary basis for the design of engineered safety features. It is shown that even this accident meets the applicable limits.

4. Anticipated Transients Without Scram, Section 14.4 -The accidents presented in Section 14.4 were assumed to occur without the benefit of tripping the reactor. While the failure to trip is unlikely, several accidents were evaluated for which credit was not taken for a reactor trip. The results showed that gross fuel clad damage would not occur if the reactor failed to trip.

14.0.2 General Assumptions

Parameters and assumptions that are common to various accident analyses are described below to avoid repetition in subsequent sections. Reactor characteristics are reviewed at the start of each operating cycle to assure that they are within the bounds assumed in the accident analyses.

14.0.2.1 Steady-State Errors

For most accidents which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit DNBR, as described in Reference 1. This procedure is known as the "Revised Thermal Design Procedure" (RTDP) and these accidents utilized the WRB-1 DNB correlation for both the 15x15 VANTAGE+ fuel design and the upgraded fuel design. The initial conditions for other key parameters are selected in such a manner to maximize the impact to DNBR. Minimum measured flow is used in all RTDP transients. This flow allows for up to 7.9-percent allowance for calorimetric uncertainty.

For accidents which are not DNB limited, or for which the Revised Thermal Design Procedure is not employed, the initial conditions are obtained by applying the maximum steady state errors to

rated values in such a manner to maximize the impact on the limiting parameters and conditions.

The following conservative steady state errors are considered:

1.	Core Power	± 2-percent allowance for calorimetric error
2.	Average Reactor Coolant System Temperature	± 7.5 °F allowance for controller deadband and measurement error
3.	Pressurizer Pressure	+28/-37 psi allowance for steady state fluctuations and measurement error
4.	Reactor Coolant Flow	Thermal Design Flow of 80,700 gpm/loop is assumed and no steady state errors are applied.

For all accidents initiated from full power, a nominal full power vessel average temperature ranging between a minimum and maximum of 549°F to 572.0°F was conservatively chosen for the analysis to bound operation at full power within this temperature range.

14.0.2.2 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies, control rods, and operating instructions. Power distribution may be characterized by the radial factor ($F_{\Delta H}$) and the total peaking factor (F_Q). The peaking factor limits are given in the Core Operating Limits Report (COLR).

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 7.2-19. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the design thermal power level defined in the Technical Specifications.

For transients which may be overpower limited, the total peaking factor (F_Q) is of importance. All transients that may be overpower limited are assumed to begin with plant conditions including power distributions which are consistent with reactor operation as defined in the Technical Specifications.

The incore instrumentation system is employed to verify that actual hot channel factors are, in fact, no higher than those specified in the COLR.

14.0.2.3 Reactor Trip

A reactor trip signal acts to open the two series trip breakers feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the control rods, which then fall by gravity into the core. There are various instrumentation delays associated with each tripping function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The

total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall.

The time delay assumed for each tripping function is as follows:

Tripping Function	Time Delay (sec)
Overpower (nuclear)	0.5
Overtemperature ΔT	2.0
Overpower ΔT	2.0
Low pressurizer pressure	2.0
High pressurizer pressure	2.0
High pressurizer level	2.0
Low reactor coolant flow	
- loop flow detectors	1.0
- reactor coolant pump undervoltage	1.5
- reactor coolant pump underfrequency trip	1.0
Turbine trip	2.0
Low-low steam-generator water level	2.0

The trip levels used in the following analyses are maximum values including the trip setpoint and the error allowance. The trip setpoints are established based on Allowable Values set forth in the Technical Specifications.

The maximum nuclear overpower trip point assumed for the analysis is 116-percent. The trips are calibrated at power such that the calibration error is the calorimetric error of 2-percent. The design allowance for nonrepeatable errors is 6-percent. Nonrepeatable errors include both instrument drift and errors due to process changes such as control rod motion since both are observable as an error between the indicated signal and the known power from calorimetric measurements. In summary, the trip setpoints are less than the trip value assumed in the analyses to ensure that trip occurs within the assumed value when including the design error allowance.

The negative reactivity insertion following a reactor trip is a function of the position versus time of the control rods and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85-percent of the control rod travel.

The reactivity insertion versus time assumed in accident analyses is shown in Figure 14.0-1. The control rod insertion time to dashpot entry is taken as 2.4 seconds. This control rod drop time requirement is specified in the plant Technical Specifications.

REFERENCES FOR SECTION 14.0

1. Friedland, A.J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

2. Deleted

14.0 FIGURES

Figure No.	Title
Figure 14.0-1	Reactivity Insertion vs. Time for Reactor Trip

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

For the following plant abnormalities and transients, the reactor control and protection systems are relied upon to protect the core and reactor coolant boundary from damage:

1. Uncontrolled rod cluster control assembly withdrawal from a subcritical condition.
2. Uncontrolled rod cluster control assembly withdrawal at power.
3. Rod cluster control assembly drop.
4. Chemical and volume control system malfunction.
5. Loss of reactor coolant flow.
6. Startup of an inactive reactor coolant loop.
7. Loss of external electrical load.
8. Loss of normal feedwater.
9. Reduction in feedwater enthalpy incident.
10. Excessive load increase incident.
11. Loss of all normal ac power to the station auxiliaries.
12. Likelihood and consequences of turbine-generator overspeed.

All reactor protection criteria are met presupposing the most reactive rod cluster control assembly is in its fully withdrawn position. Trip is defined for analytical purposes as the insertion of all full-length rod cluster control assemblies, except the most reactive assembly, which is assumed to remain in the fully withdrawn position. This is to provide margin in shutdown capability against the remote possibility of a stuck rod cluster control assembly condition existing at a time when shutdown is required.

Instrumentation is provided for continuously monitoring all individual rod cluster control assemblies together with their respective group position. This is in the form of a deviation alarm system. If a rod should deviate from its intended position, the reactor can be shut down in an orderly manner and the condition corrected. [***Note*** *- See Technical Specifications, Section 3.1, for permissible variances.*] Such occurrences are expected to be extremely rare on the basis of operation and test experience to date.

In summary, reactor protection is designed to prevent cladding damage in all transients and abnormalities listed above. The most probable modes of failure in each protection channel result in a signal calling for the protective trip. The coincidence of two-out-of-three (or two-out-of-four) signals is required where single-channel malfunction could cause spurious trips while at power. A single component or channel failure in the protection system itself coincident with one stuck rod cluster control assembly is always permissible as a contingent failure and does not cause a violation of the protection criteria. The reactor protection systems are designed in accordance with the IEEE "Standard for Nuclear Plant Protection Systems."

14.1.1 Uncontrolled Rod Cluster Control Assembly Withdrawal From A Subcritical Or Low Power Startup Condition

A rod cluster control assembly (RCCA) withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting

in a power excursion. This could occur with the reactor subcritical, at hot zero power, or at power. The "at power" case is discussed in Section 14.1.2. The low power startup condition assumed in this section (1×10^{-9} of nominal power) is less than the power level expected for any shutdown condition.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Section 14.1.5).

The RCCA drive mechanisms are wired into preselected bank configurations, which are not altered during reactor life. The drive mechanisms being wired into preselected bank configurations prevent the RCCAs from being manually withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed, which is well within the capability of the protection system to prevent core damage.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the reactor protection system:

1. Source range flux level trip - actuated when either of two independent source range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when either of two intermediate range flux channels indicates a flux level above the source range cutoff power level.

2. Intermediate range flux level trip - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This trip function is manually bypassed when two-out-of-four power range channels are reading above approximately 10-percent power and automatically reinstated when three-out-of-four channels indicate a power level below this value. To prevent unnecessary reactor trips during power reductions prior to shut down, operating procedures allow these trips to be manually bypassed until they have reset to the untripped condition and the reset has been verified.

3. Power range flux level trip (low setting) - actuated when two-out-of-four power range channels indicate a power level above approximately 25-percent. This trip function may be manually bypassed when two-out-of-four power range channels indicate a power level above approximately 10-percent power and is automatically reinstated when three of the four channels indicate a power level below this value.

4. Power range flux level trip (high setting) - actuated when two-out-of-four power range channels indicate a power level above a preset setpoint. This trip function is always active.

In addition, control rod stops on high intermediate range flux level (one out of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

NOTE: Automatic Rod Withdrawal Has Been Physically Disabled At Indian Point Unit 2.

14.1.1.1 Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: (1) an average core nuclear power transient calculation, (2) an average core heat transfer calculation, and (3) the DNBR calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods in TWINKLE[18] to determine the average power generation with time, including the various total core feedback effects (i.e., Doppler reactivity and moderator reactivity). The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN[19]. The average heat flux with appropriate peaking factors is next used in VIPRE[23] for departure from nucleate boiling ratio calculations.

This accident is analyzed using Standard Thermal Design Procedures. Plant characteristics and initial conditions are discussed in Section 14.0.2.1. In order to give conservative results for a startup accident, the following assumptions are made:

1. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler defect, conservatively low values as a function of power are used.

2. Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A highly conservative value is used in the analysis to yield the maximum peak heat flux.

3. The reactor is assumed to be just critical at hot zero power (no load) T_{avg} (547°F). This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect thereby increasing the neutron flux peak. The initial effective multiplication factor is assumed to be 1.0 since this results in the worst nuclear power transient.

4. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod cluster control assembly release, is taken into account. A 10-percent increase is assumed for the power range flux

trip setpoint raising it from the nominal value of 25-percent to 35-percent. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth rod cluster control assembly is stuck in its fully withdrawn position.

5. The maximum positive reactivity insertion rate assumed (75 pcm/sec) is greater than that for the simultaneous withdrawal of the combination of two sequential control banks having the greatest combined worth at maximum speed (45-in./min). Control rod drive mechanism design is discussed in Section 3.2.3.4.

6. The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high worth position, is assumed in the departure from nucleate boiling analysis.

7. The initial power level was assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). This combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

8. Two reactor coolant pumps are assumed to be in operation. This is conservative with respect to departure from nucleate boiling. No single active failure in any system or equipment available to mitigate the effects of the accident will adversely affect the consequences of the accident.

14.1.1.2 <u>Results</u>

Figures 14.1-1 through 14.1-4 show the transient behavior for the uncontrolled RCCA bank withdrawal incident, with the accident terminated by reactor trip at 35-percent of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region. Figure 14.1-1 shows the nuclear power transient.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for departure from nucleate boiling considerations, is shown in Figure 14.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full power nominal value. There is a large margin to departure from nucleate boiling during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figures 14.1-3 and 14.1-4 show the response of the hot-spot fuel average temperature and the hot-spot clad temperature. The average fuel temperature increases to a value lower than the nominal full power value. The minimum departure from nucleate boiling ratio at all times remains above the limit value.

The calculated sequence of events and summary of the results for this accident are shown in Table 14.1-1. With the reactor tripped, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

The operating procedures would call for operator action to control reactor coolant system boron concentration and pressurizer level using the chemical and volume control system, and to maintain steam generator level through control of the main or auxiliary feedwater system.

Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of 10 min following reactor trip.

14.1.1.3 Radiological Consequences

There are no radiological consequences associated with an uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition event since radioactivity is contained within the fuel rods and the reactor coolant system is maintained within design limits. This is demonstrated by showing that the minimum departure from nucleate boiling ratio remains above the limit DNBR.

14.1.1.4 Conclusions

In the event of a RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR greater than the limit value. Thus, no fuel or clad damage is predicted as a result of departure from nucleate boiling.

14.1.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power

An uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power increase and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the safety analysis limit values.

This event is classified as an ANS Condition II incident (an incident of moderate frequency).

The automatic features of the Reactor Protection System which prevent core damage following the postulated accident include the following:

1. Power range neutron flux instrumentation actuates a reactor trip if two-of-four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an Overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an Overpower ΔT setpoint. This setpoint is automatically varied with coolant temperature to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

4. A high pressurizer pressure reactor trip is actuated from any two-out-of-three pressure channels which is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip is actuated from any two-out-of-three level channels when the reactor power is above approximately 10-percent (Permissive P-7).

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

1. High neutron flux (one-out-of-four power range)
2. Overpower ΔT (one-out-of-four)
3. Overtemperature ΔT (one-out-of-four)

The manner in which the combination of the overpower and overtemperature ΔT trips provide protection over the full range of RCS conditions is described in Chapter 7.

14.1.2.1 Method of Analysis

The transient is analyzed by the RETRAN Code.[21A] This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

This accident is analyzed using the Revised Thermal Design Procedure. [22] Initial reactor power, RCS pressure, and temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 22 of Chapter 14.1.

In performing the analysis, the following assumptions are made to assure bounding results are obtained for all possible normal operational conditions:

1. Reactivity Coefficients - Two cases are analyzed:
   a. Minimum Reactivity Feedback. A least-negative moderator density coefficient of reactivity is assumed, corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed.

   b. Maximum Reactivity Feedback. A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

2. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 116-percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values.

3. The trip reactivity is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position.

4. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the two control banks having the maximum combined worth at maximum speed.

5. A range of initial power levels from 10% to 100% power is considered.

The effect of the axial core power distribution is accounted for by causing a decrease in the Overtemperature ΔT trip setpoint proportional to the decrease in margin to DNB.

14.1.2.2 Results

Figures 14.1-5, 14.1-6 and 14.1-7 show the transient response for a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and margin to DNB is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figures 14.1-8, 14.1-9 and 14.1-10. Reactor trip on Overtemperature ΔT occurs after a longer period and the rise in temperature is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the safety analysis limit values.

Figure 14.1-11 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and Overtemperature ΔT channels. The minimum DNBR is never less than the safety analysis limit values.

Figures 14.1-12 and 14.1-13 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10-percent power, respectively, for minimum and maximum reactivity feedback. The results are similar to the 100-percent power case, except as the initial power is decreased, the range over which the Overtemperature ΔT trip is effective is increased. In all cases the DNBR does not fall below the safety analysis limit value.

The shape of the curves of minimum DNB ratio versus reactivity insertion rate in the reference figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

For transients initiated at 60% power it is noted that:

1. For reactivity insertion rates above approximately 10 pcm/sec reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux lags behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient. As reactivity insertion rate decreases, core heat flux can remain more nearly in equilibrium with the neutron flux. Minimum DNBR during the transient thus decreases with decreasing insertion rate.

2. The Overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured Reactor Coolant System average temperature and pressure. It is important to note that the average temperature contribution to the circuit is lead lag compensated to decrease the effect of the thermal capacity of the RCS in response to power increases.

3. For reactivity insertion rates below10 pcm/sec the Overtemperature ΔT trip terminates the transient.

   For reactivity insertion rates from 10 pcm/sec to approximately 2 pcm/sec the effectiveness of the Overtemperature ΔT trip increases (in terms of increased minimum DNBR) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower and the system lags and delays become less significant.

4. For reactivity insertion rates less than 2 pcm/sec, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which acts as a heat sink on the Reactor Coolant System and results in increased heat removal from the Reactor Coolant System, sharply decreases the rate of increase of the Reactor Coolant System average temperature.

The effect described in item 4 above, which results in the sharp peak in minimum DNBR at approximately 2 pcm/sec, does not occur for transients initiated at 100% power since the steam generator safety valves are not actuated prior to trip (Figure 14.1-11).

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 116-percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will still remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the Overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 116-percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will remain below the fuel melting temperature.

Since the DNBR is not violated at any time during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient. The calculated sequence of events for this accident is shown on Table 14.1-2 for large and small reactivity insertion rates. These sequences of events are for the cases initiated from full power assuming minimum reactivity feedback conditions. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

14.1.2.3 Conclusions

The high neutron flux and Overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the safety analysis limit values.

14.1.3 Incorrect Positioning Of Part-Length Rods

Part-length rods were employed in the original design to improve the axial power distributions as well as to control potential axial xenon oscillations. Subsequent to initial plant operations, however, (during the Cycle 2/3 refueling outage), the part-length rod cluster control assemblies were removed from the reactor.

14.1.4 Rod Cluster Control Assembly Drop

The dropping of a rod cluster control assembly could occur from deenergizing a drive mechanism. It would result in a power reduction and a possible increase in the hot-channel factor. If no protective action occurred, the reactor coolant system would attempt to restore the power to the level that existed before the incident occurred. This would lead to a reduced safety margin or possibly departure from nucleate boiling, depending upon the magnitude of the hot-channel factor.

If a rod cluster control assembly should drop into the core during power operation, this would be detected by the rod bottom signal device, which provides an individual position indication signal for each rod cluster control assembly. The initiation of this signal is independent of lattice location, reactivity worth, or power distribution changes inherent with the dropped rod cluster control assembly. Further indication of a rod cluster control assembly drop would be obtained by independent means, using the out-of-core power range channel signals.

A rod drop signal from any rod position indication channel, or from one or more of the four power range channels, initiates protective action by reducing turbine load by a preset adjustable amount. Bypass switches have been installed which are in the DEFEAT position, so as to bypass the runback. The automatic rod control system has been modified and currently utilizes only the automatic rod insertion feature (the automatic rod withdrawal feature has been disabled by this modification). This action prevents core damage. The automatic turbine runback functionality has been administratively deleted. The rod stop is also redundantly actuated. Rod drop protection is discussed in Section 7.2.

14.1.4.1 Method of Analysis

The transient response following a dropped RCCA event is calculated using a detailed digital simulation of the plant. A dropped RCCA or dropped RCCA Bank causes a step decrease in reactivity and the resulting core power generation is determined using the LOFTRAN computer code [21]. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, rod control system, steam generators, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. Since LOFTRAN employs a point neutron kinetics model, a dropped rod event is modeled as a negative reactivity insertion corresponding to the reactivity worth of the dropped RCCA(s), regardless of the actual configuration of the rod(s) that drop.

A dropped rod cluster control assembly results in a negative reactivity insertion. The core is not adversely affected during this period since power is decreasing rapidly. Following a dropped rod cluster control assembly with turbine runback and automatic rod withdrawal disabled, the plant will establish a new equilibrium condition. Depending on the worth of the dropped RCCA(s), power may be reestablished by reactivity feedback.

When reactivity feedback does not offset the worth of the dropped RCCA(s), a cooldown condition exists until a low pressurizer pressure reactor trip signal is reached. When reactivity feedback is large enough to offset the worth of the dropped RCCA(s), reactor power is reestablished at a new equilibrium condition.

To capture the transient response, dropped rod statepoints designed to bound possible operation without a turbine runback were evaluated. The dropped rod/bank statepoints are based on generic dropped rod analyses performed as part of the Westinghouse Owners Group (WOG) dropped rod protection modification program.[27] The WOG dropped rod protection modification program was specifically performed to support elimination of turbine runback on dropped rod (for Westinghouse plants with this system) and deletion of the negative flux rate trip (for Westinghouse plants without turbine runback on dropped RCCA) for conditions with and without automatic rod withdrawal block. The incident is analyzed using the Revised Thermal Design Procedure and assumes nominal initial conditions as described in Section 14.0.2.1

14.1.4.2 Results

Figures 14.1-14 through 14.1-16 illustrate a typical transient response when reactivity feedback does not offset the worth of the dropped RCCA(s). In this case, BOL conditions are shown with a small negative moderator temperature coefficient (MTC) of -5 pcm/ºF for a dropped RCCA worth of 400 pcm. As a result of the negative reactivity insertion of the dropped rod cluster control assembly, a cooldown condition of the RCS exists. The nuclear power reaches a level lower than that which existed before the incident and the RCS temperature and pressure continue to decrease until a low pressurizer pressure reactor trip signal is reached.

Figures 14.1-17, 14.1-18, and 14.1-19 illustrate a typical transient response when reactivity feedback is large enough to offset the worth of the dropped RCCA(s). In these figures EOL conditions are shown with a large negative moderator temperature coefficient (MTC) of -35 pcm/ºF for a dropped rod cluster control assembly worth of 400 pcm. With a large reactivity feedback, a new equilibrium condition is reached without a reactor trip. The nuclear power returns to nearly the initial power level that existed before the incident while the RCS temperature and pressure are reduced to a slightly lower condition.

The evaluation of the generic WOG dropped rod/bank statepoints considered to bound possible operation without turbine runback show the applicable licensing basis acceptance criteria is met. Specifically, the evaluations performed using the WOG dropped rod/bank statepoints verified that the DNBR licensing basis acceptance criterion is met assuming no turbine runback following a dropped RCCA event for single or multiple dropped RCCAs from the same group of a given bank with rod withdrawal block. It should be noted that no evaluation of single dropped RCCA worths with automatic rod control functioning was performed to confirm the acceptability of the dropped RCCA event for a single failure of a rod-on-bottom signal which automatically blocks rod withdrawal. This is because automatic rod withdrawal has been physically disabled at Indian Point Unit 2 which precludes such occurrences.

For all cases analyzed, the DNBR does not fall below the limit value.

14.1.4.3 Conclusions

Based on the DNBR results for all of the cases analyzed, it has been demonstrated that the DNBR criterion is met, and therefore, it is concluded that dropped RCCAs do not lead to

conditions that cause core damage and that all applicable safety criteria is satisfied for this event.

14.1.5 Chemical And Volume Control System Malfunction

14.1.5.1 Introduction

Reactivity can be added to the core with the chemical and volume control system by feeding reactor makeup water into the reactor coolant system via the reactor makeup control system. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time. The chemical and volume control system is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

There is only a single, common source of dilution water to the blender from the primary water makeup system; inadvertent dilution can be readily terminated by isolating this single source. The operation of the primary water makeup pumps that take suction from the primary water storage tank (PWST) provides the non-borated supply of makeup water to the blender. The boric acid from the boric acid storage tank(s) is blended with the reactor makeup water in the blender, and the composition is determined by the preset flow rates of boric acid and reactor makeup water on the reactor makeup control. The operator must switch from the automatic makeup mode to the dilute mode and move the start-stop switch to start, or, alternatively, the boric acid flow controller could be set to zero. Since these are deliberate actions, the possibility of inadvertent dilution is very small. In order for this dilution water to be added to the reactor coolant system, the charging pumps must be running in addition to the primary water makeup pumps. Also, any diluted water introduced into the volume control tank (VCT) must pass through the charging pumps to be added to the reactor coolant system.

Thus, the rate of addition of diluted water to the reactor coolant system from any source is limited to the capacity of the charging pumps. This addition rate is 294 gpm for all three charging pumps. This is the maximum delivery rate based on a pressure drop calculation comparing the pump curve with the system resistance curve. Normally, only one charging pump is operating while the others are on standby.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the chemical and volume control system. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction. Boron dilution during refueling, startup, and power operation are considered in this analysis.

14.1.5.2 Method of Analysis and Results

14.1.5.2.1 Dilution During Refueling

During refueling the following conditions exist:

1. One residual heat removal pump providing a minimum flow rate of 1000 gpm is normally running except during short time periods as allowed by the technical specifications.

2. The chemical and volume control system and/or safety injection system are aligned so that there is at least one flow path to the core for boric acid injection when there is fuel in the reactor, as required by the Technical Specifications.

3. The minimum boron concentration of the refueling water is at least 2050 ppm or higher to maintain a shutdown of at least 5-percent $\Delta k/k$ with all control rods in; periodic sampling ensures that this concentration is maintained.

4. Neutron sources (e.g., primary sources, secondary sources or recently irradiated fuel assemblies) are installed in the core and detectors connected to instrumentation giving audible count rates are installed outside or within the reactor vessel to provide direct monitoring of the core.

A minimum water volume in the reactor coolant system of 3257-ft^3 is considered. This corresponds to the volume necessary to fill the reactor vessel to mid-loop. The maximum dilution flow of 294 gpm and uniform mixing are also considered. The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the main control room. The count-rate increase is proportional to the multiplication factor.

The boron concentration must be reduced from 2050 ppm to approximately 1390 ppm before the reactor will go critical. This would require more than 30 minutes. This is ample time for the operator to recognize the audible high count-rate signal and isolate the reactor makeup source by closing valves and stopping the primary water makeup pumps and/or charging pumps. The Refueling Operation Surveillance Procedure requires values which are potential sources of un-borated water be tagged closed, and the possibility of inadvertent dilution during refueling is very small. In addition, there could be a source of water from Indian Point Unit 1. Procedures call for isolation of that source should there be an unintended dilution.

14.1.5.2.2 Dilution During Startup

In this mode, the plant is being taken from one long-term mode of operation, Hot Standby, to another, Power Operation. Typically, the plant is maintained in the Startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation.

Conditions assumed for the analysis are:

1. Dilution flow is the maximum capacity of the charging pumps, 294 gpm.

2. A minimum RCS water volume of 8567-ft^3. This corresponds to the active RCS volume taking into account 10% uniform steam generator tube plugging minus the pressurizer and the reactor vessel upper head.

3. The initial boron concentration is assumed to be 1800 ppm, which is a conservative maximum value for the critical concentration at the condition of hot zero power, rods to insertion limits, and no Xenon.

4. The critical boron concentration following reactor trip is assumed to be 1550 ppm, corresponding to the hot zero power, all rods inserted (minus the most reactive RCCA), no Xenon condition. The 250 ppm change from the initial condition noted above is a conservative minimum value.

This mode of operation is a transitory operational mode in which the operator intentionally dilutes (borates) and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution (boration) and subsequently manually withdraw the control rods, a process that takes several hours. The Technical Specifications require that the operator assure that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip nominally set at 2.3 E5 CPS after receiving P-6 from the intermediate range. Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

However, in the event of an unplanned approach to criticality or dilution during power escalation while in the Startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power to a reactor trip on the power range neutron flux - high, low setpoint (nominal 25-percent power). From initiation of the event, there are greater than 15 minutes available for operator action prior to return to criticality.

14.1.5.2.3 Dilution at Power

In this mode, the plant may be operated in either automatic or manual rod control. Conditions assumed for the analysis are:

1. Dilution flow is the maximum capacity of the charging pumps, 294 gpm.

2. A minimum RCS water volume of 8567-ft^3. This corresponds to the active RCS volume (with 10% uniform steam generator tube plugging) minus the pressurizer and reactor vessel upper head.

3. The initial boron concentration is assumed to be 1800 ppm, which is a conservative maximum value for the critical concentration at the condition of hot full power, rods to insertion limits, and no Xenon.

4. The critical boron concentration following reactor trip is assumed to be 1450 ppm, corresponding to the hot zero power, all rods inserted (minus the most reactive RCCA), no Xenon condition. The 350 ppm change from the initial condition noted above is a conservative minimum value.

With the reactor in automatic rod control, the power and temperature increase from boron dilution results in insertion of the control rods and a decrease in the available shutdown margin. The rod insertion limit alarms (LOW and LOW-LOW settings) alert the operator more than 15

minutes prior to losing the required minimum shutdown margin. This is sufficient time to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the Overtemperature ΔT trip setpoint resulting in a reactor trip. The boron dilution transient in this case is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution is conservatively estimated to 1.24 pcm/sec, which is within the range of insertion rates analyzed. Thus, the effects of dilution prior to reactor trip are bounded by the uncontrolled RCCA bank withdrawal at power analysis (Section 14.1.2). Following reactor trip there are greater than 15 minutes prior to criticality. This is sufficient time for the operator to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

14.1.5.3 Conclusions

Because of the procedures involved in the dilution process requiring operator action, an erroneous dilution is considered very unlikely. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to changes in dilution are slow enough to allow the operator to determine the cause of the addition and take corrective action before shutdown margin is lost.

14.1.6 Loss Of Reactor Coolant Flow

14.1.6.1 Description

A loss-of-coolant-flow incident may result from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss-of-coolant flow is a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling with subsequent fuel damage if the reactor is not tripped promptly. The following trip circuits provide the necessary protection against a loss-of-coolant-flow incident and are actuated by:

1. Low voltage on pump power supply bus (above P-7 permissive).

2. Pump circuit breaker opening (one-out-of-four above P-8 permissive, two-out-of-four above P-7 permissive).

3. Low reactor coolant flow (one-out-of-four above P-8 permissive, two-out-of-four above P-7 permissive).

Each pump circuit breaker is automatically tripped on an undervoltage of its associated bus or an underfrequency on any two-out-of-four pump buses.

These trip circuits and their redundancy are further described in Section 7.2.

The most severe partial and complete loss of reactor coolant flow accidents are analyzed to ensure that the reactor trip together with flow sustained by the inertia of the coolant and rotating

pump parts will be sufficient to prevent departure from nucleate boiling. Therefore, the fuel will not be damaged as a result of the most severe credible loss-of-coolant-flow accident.

14.1.6.2 Method of Analysis

The following loss of flow cases were analyzed:

1. Loss of four pumps from full power during four-loop operation.
2. Loss of one pump from full power during four-loop operation.

The normal power supplies for the pumps are the four buses connected to the generator, each of which supplies power to one of the four pumps. When a turbine trip occurs, the pumps are automatically transferred to the buses supplied from an external power line, and the pumps will continue to supply coolant flow to the core. The simultaneous loss of power to the four reactor coolant pumps is a highly unlikely event. Since the pumps are on separate buses, a single bus fault would result in the loss of only one pump.

These transients are analyzed with two computer codes. First, the RETRAN[21a] computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE[23] computer code is then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS flow from RETRAN.

The calculation of DNBR during the transient is made using the nucleate boiling correlations as described in Section 3.2.2.1.2. In addition, the following assumptions were made in the calculations.

14.1.6.2.1 Initial Operating Conditions

The initial operating conditions used for the analysis are consistent with the use of the Revised Thermal Design Procedure (RTDP).[22] These assumptions include the following full power initial operating conditions; nominal value of power, nominal steady state pressure, and maximum steady state average programmed temperature.

14.1.6.2.2 Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used. The least negative moderator temperature coefficient (minimum moderator density coefficient) is assumed (0.0 pcm/^{o}F), since this results in the maximum core power during the initial part of the transient, when the minimum DNBR is reached.

14.1.6.2.3 Reactor Trip

For the one-pump loss-of-flow incidents, the reactor trip is assumed to be actuated by the redundant flow monitoring channel (two-out-of-three), since this results in the largest delay to reactor trip. For the four-pumps loss of flow incident, two cases are considered; reactor trip actuated by redundant bus undervoltage or breaker trip (one-out-of-four or one-out-of-three) and reactor trip on bus underfrequency (two-out-of-four). For the analysis of the four-pump loss-of-flow incident actuated by a bus undervoltage or breaker trip, the loss of flow is assumed to occur at the initiation of the event (i.e., t=0). Hence, with respect to the safety analysis, the undervoltage trip setpoint is irrelevant. However, for the analysis of the four-pumps loss-of-flow

incident actuated by a bus underfrequency, the reactor is assumed to trip after an underfrequency reactor coolant pump trip at 57 Hz following a frequency decay of 5 Hz/sec from an initial frequency of 60 Hz. The trip is conservatively modeled to occur at 1.6 seconds, which includes a maximum reactor trip time delay of 1.0 seconds. Following reactor trip, the reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

The low-flow trip setting is 92-percent of full flow; the trip signal is assumed to be initiated at 85.0-percent of full flow, allowing 7.0-percent for margin and instrumentation uncertainty. Upon reactor trip, it is assumed that the most reactive rod cluster control assembly is stuck in its fully withdrawn position, hence resulting in a minimum insertion of negative reactivity. The negative reactivity insertion upon trip is conservatively assumed to be 4% Δk.

A conservative shape of trip reactivity insertion versus time (based on a RCCA drop time of 2.4 seconds to the dashpot) was also used.

14.1.6.2.4 Heat Transfer Coefficient

The overall heat conductance between the fuel and water regions varies considerably during the transient mostly as a result of the change of fuel gap conductance. The larger heat transfer coefficients calculated at several different power levels, using EOL fuel temperatures, are used. This assumption produces a fast fuel thermal response and maximizes the positive reactivity inserted by Doppler feedback as the core is shutdown.

14.1.6.2.5 Flow Coastdown

Reactor coolant flow coastdown curves are shown in Figure 14.1-21 for the one-pump loss of flow and in Figures 14.1-24 and 14.1-27, for the four-pumps loss of flow accident on bus undervoltage and bus underfrequency, respectively. These curves are based on high estimates of loop pressure losses and include the effect of inertia from the pump flywheels.

14.1.6.3 <u>Results</u>

The time sequence of events and summary of the results for the complete (four-pumps) loss of flow and for the partial (one-pump) loss of flow accidents are shown in Tables 14.1-3 and 14.1-4, respectively.

Figure 14.1-20 shows the nuclear power and heat flux transients for the partial loss of flow from full power operation. Figure 14.1-22 shows the DNBR as a function of the time for this case. The minimum DNBR is reached at about 3.4 seconds after the initiation of the accident. For this case, the DNBR also always remains above the safety limit value.

Figure 14.1-23 shows the nuclear power and heat flux transients for the complete loss of flow from full power operation following a reactor trip on bus undervoltage. Figure 14.1-25 shows the DNBR as a function of time for this case. The minimum DNBR is reached at about 3.3 seconds from the start of the accident and the DNBR always remains above the safety limit value.

Figure 14.1-26 shows the nuclear power and hot channel heat flux transients for the complete loss of flow from full power operation following a reactor trip on bus underfrequency. Figure 14.1-28 shows the DNBR as a function of time for this case. The minimum DNBR is reached at

about 3.6 seconds from the start of the accident and the DNBR always remains above the safety limit value.

14.1.6.4 Conclusions

Since the applicable safety analysis DNBR limit is met for the loss of flow cases considered, there is no cladding damage and no release of fission products into the reactor coolant. Therefore, all applicable safety criteria is met for the loss of flow events.

14.1.6.5 Locked Rotor Accident

A transient analysis was performed for the postulated instantaneous seizure of a reactor coolant pump rotor. Flow through the reactor coolant system is rapidly reduced, leading to a reactor trip on a low-flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer to the secondary system, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and eventually opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect is not included in the analysis.

14.1.6.5.1 Method of Analysis

As was the case for the loss of flow accident previously analyzed, the locked rotor analysis was performed assuming a full power initial condition with all four loops in operation and the same two computer codes are used to analyze this transient. The RETRAN[21a] computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE[23] computer code is then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS flow from RETRAN.

The following effects of the locked rotor event were investigated:

1. Primary pressure transient.
2. Fuel clad temperature transient (this is calculated assuming film boiling in order to give the worst possible results).
3. DNB transient (for determining the amount of rods in DNB for the offsite dose release calculations).

14.1.6.5.1.1 Initial Conditions

Except for the DNB evaluation, performed using the Revised Thermal Design Procedure, the locked rotor accident was analyzed assuming that at the beginning of the postulated event (at the time the shaft in one of the reactor coolant pumps is assumed to seize), the plant is in operation under the most adverse steady-state operating conditions; i.e., 102% of the NSSS design thermal power, with maximum steady-state pressurizer pressure and level, and maximum steady-state coolant average temperature.

14.1.6.5.1.2 Evaluation of the Pressure Transient

For the peak pressure evaluation, the initial pressure is conservatively estimated as 28 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure.

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin 1 second after the flow in the affected loop reaches 85.0-percent of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The safety valves start operating at 2485 psig and their combined capacity for steam relief is 42-ft^3/sec.

14.1.6.5.1.3 Evaluation of Fuel Rod Thermal Transient

The evaluation of fuel rod thermal transient is performed at the hot spot. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction.

In the evaluation, the rod power at the hot spot is conservatively assumed to be at least 2.5 times the average rod power (i.e., F_Q = 2.5) at the initial core power level.

14.1.6.5.1.4 Film Boiling Coefficient

The film boiling coefficient is calculated in the VIPRE program [23] using the Bishop-Sandberg-Tong film-boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step, based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and mass flowrate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, film boiling was assumed to start at the beginning of the accident.

14.1.6.5.1.5 Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady-state value consistent with initial fuel temperature to 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value of the gap coefficient is released to the clad at the initiation of the transient.

14.1.6.5.1.6 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{d\left(w^2\right)}{dt} = 33.3x10^6 \exp\left(-\frac{45{,}500}{1.986T}\right)$$

where:

w = amount reacted (mg/cm^2).

t = time (seconds).

T = temperature (°Kelvin).

The reaction heat is 1510 cal/g.

The effect of zirconium-steam reaction is included in the calculation of the hot spot clad temperature transient.

14.1.6.5.1.7 Evaluation of Departure from Nucleate Boiling (DNB) in the Core During the Accident

The evaluation of the number of rods in DNB has been performed using the Revised Thermal Design Procedure.

Nominal values for power, core pressure and core inlet temperature were assumed in the analysis, consistent with the use of the RTDP.

Calculation of the extent of the DNB in the core during the transient has been performed using the VIPRE[23] program.

14.1.6.5.2 Results

Figures 14.1-29 through 14.1-30a show the transient results for one locked rotor with four loops in operation (with loss of offsite power). The results of these calculations and the time sequence of events are also summarized in Table 14.1-5. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits of the ASME code, Section III. Also the clad peak temperature is considerably less than 2700°F. It should be noted that the clad temperature was conservatively calculated assuming that DNB (i.e., film boiling) occurs at the initiation of the transient even if DNB is not expected.

14.1.6.5.3 Fission Product Release

As a result of the accident, fuel clad damage may occur. Due to the potential for leakage between the primary and secondary systems, radioactive reactor coolant is assumed to leak from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or the main steam safety

valves. Iodine and alkali metals group activity is assumed to be contained in the secondary coolant prior to the accident, and some of this activity is also released to the atmosphere as a result of steaming the steam generators following the accident.

There are no rods in DNB as a result of the locked rotor. In determining the offsite doses following the locked rotor accident, it is conservatively assumed that 5% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released. The core activity is provided in Table 14.3-43 and it is assumed that the damaged fuel rods have all been operating at a peaking factor of 1.70. The gap fractions from Table 3 of Regulatory Guide 1.183 (Reference 37) are used. These are 8% for I-131, 10% for Kr-85, 5% for other iodines and noble gases, and 12% for alkali metals. Per the model in Regulatory Guide 1.183, these are the only nuclide groups considered for gap activity.

A pre-existing iodine spike in the reactor coolant is assumed to have increased the primary coolant iodine concentration to 60 μCi/gm of dose equivalent 1-131 prior to the locked rotor accident. The alkali metals and noble gas activity concentrations in the RCS at the time the accident occurs are based on operation with a fuel defect level of one percent. The iodine activity concentration of the secondary coolant at the time the locked rotor ejection accident occurs is assumed to be 0.15 μCi/gm of dose equivalent I-131.

Regulatory Guide 1.183 (Reference 37) specifies that the iodine released from the fuel is 95% particulate (cesium iodide), 4.85% elemental, and 0.15% organic. However, iodine in solution is considered to be all elemental and after it is released to the environment the iodine is modeled as 97% elemental and 3% organic.

The primary to secondary steam generator tube leak used in the analysis is 150 gpd per steam generator (total of 600 gpd).

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere. The residual heat removal system is assumed to be placed in service at 30 hours after the accident and there are no further releases to the environment after this point in time.

An iodine partition factor in the steam generators of 0.01 curies/gm steam per curies/gm water is used. The partition factor for the alkali metal activity in the steam generators is 0.0025 and is based on moisture carryover.

The resultant 2 hour site boundary dose is 0.24 rem TEDE. The 30 day low population zone dose is 0.54 rem TEDE. These doses are calculated using the meteorological dispersion factors discussed in Section 14.3.6.2.1.

The offsite doses resulting from the accident are less than 2.5 rem TEDE, which is 10-percent of the limit value of 10 CFR 50.67 and is the dose acceptance limit from Regulatory Guide 1.183.

The accumulated dose to control room operators following the postulated accident was calculated using the same release, removal and leakage assumptions as the offsite doses, using the control room model discussed in Section 14.3.6.5 and Tables 14.3-50 and 14.3-51. The calculated control room dose is presented in Table 14.3-52 and is less than the 5.0 rem TEDE control room dose limit values of 10 CFR 50.67.

14.1.6.5.4 Conclusions

1. The peak pressure of 2553 psia for the worst case ensures that the integrity of the primary coolant system is not endangered and can be considered as an upper limit, considering the conservative assumptions used in the study.

2. The DNBR always remains above the safety limit value. Hence there are no rods in DNB.

3. The peak clad average temperature of 1813°F, calculated for the hot spot, includes the effect of the zirconium-steam reaction (which is still quite small at that temperature). It can be considered an upper limit since:
   a. The hot spot was assumed to be in departure from nucleate boiling from time zero regardless if DNB occurs.
   b. A high gap coefficient (10000 Btu/hr-ft^2 - °F) was used.
   c. No credit was taken for transition boiling. The heat transfer coefficient for fully developed film boiling was used from time zero.
   d. The nuclear heat released in the fuel at the hot spot was based on a zero moderator coefficient.

4. The radiological consequences of this event are within the limit values.

Based on this, it can be concluded that all the applicable safety criteria for the locked rotor accident are met.

14.1.7 Startup Of an Inactive Reactor Coolant Loop

Technical Specifications require that all 4 reactor coolant pumps be operating for reactor power operation and preclude operation with an inactive loop (except for testing or repair and not to exceed the time specified). This event was originally included in the FSAR licensing basis when operation with a loop out of service was considered. Based on the current Technical Specifications which prohibit at power operation with an inactive loop as indicated above and the changes to the Technical Specifications which deleted all references to three loop operation, this event has been deleted from the updated FSAR.

14.1.8 Loss Of External Electrical Load

14.1.8.1 Description

A major load loss on the plant can result from either a loss of external electrical load or from a turbine trip. For either case, offsite power is normally available for the continued operation of plant components such as the reactor coolant pumps, unless the 6.9 KV fast bus transfer does not take place. The specific case of loss of all ac power to station auxiliaries is discussed in Section 14.1.12. The case of RCP overspeed following a turbine mechanical overspeed trip is addressed in Section 4.2.2.4.

A turbine trip will cause a reactor trip based on a signal derived from the turbine autostop oil pressure unless the reactor is below approximately 20-percent power (P-8). The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and

pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere. Additionally, main feedwater flow would be lost if the turbine condenser were not available. For this situation, steam generator level would be maintained by the auxiliary feedwater system.

The unit was originally designed to accept a step 50% loss of load without actuating a reactor trip. The automatic steam dump system, with 40% steam dump capacity to the condenser, was designed to accommodate this load rejection by reducing the severity of the transient imposed upon the RCS. The reactor power is reduced to the new equilibrium power level at a rate consistent with capability of the Rod Control System. The steam generator relief valves may be actuated, but the pressurizer relief valves and the steam generator safety valves should not lift for the 50% step loss of load with steam dump available.

In the event the steam dump valves fail to open following a large loss of load or in the event of a complete loss of load with steam dump operating, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, the low steam generator level signal, or the overtemperature/overpower ΔT signals. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. However, the pressurizer safety valves and steam generator safety valves are sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the steam dump system. The RCS and main steam supply relieving capacities were designed to ensure safety of the unit without requiring the automatic rod control, pressurizer pressure control and/or steam bypass control systems.

14.1.8.2 Method of Analysis

In this analysis, the behavior of the unit was evaluated for a complete loss of steam load from full power without a direct reactor trip. This was done to show the adequacy of the pressure relieving devices and to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a trip. The turbine was assumed to trip without actuating the turbine trip signal (low auto stop oil pressure). This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient. In addition, for conservatism, no credit was taken for steam dump, main feedwater flow is terminated at the time of turbine trip, and no credit was taken for auxiliary feedwater (except for long-term recovery) to mitigate the consequences of the transient.

In addition to the specific analysis discussed above for a complete loss of steam load from full power, the acceptability of a loss of steam load without direct reactor trip on turbine trip below 35% of 3230.0 MWt NSSS full power was also evaluated.

The total loss of load transients were analyzed with the RETRAN computer program (Reference 21a). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

This accident was analyzed using the Revised Thermal Design Procedure (RTDP) (Reference 22) for DNB concerns (case with pressure control) and for overpressure concerns (case without pressurizer pressure control) using the Standard Thermal Design Procedure (STDP). With the RTDP, the initial conditions assumed for reactor power, RCS pressure and temperature are assumed to be at their nominal values as described in Section 4.0.2.1.

Major assumptions are summarized below:

1. Initial Operating Conditions
   The initial reactor power, RCS pressure, and RCS temperatures are assumed at their nominal values consistent with steady state full power operation for the DNB case analyzed using RTDP. For the peak RCS pressure case, uncertainties are applied in the most limiting directions to the initial core power, reactor coolant pressure and reactor coolant temperature.

2. Moderator and Doppler Coefficients of Reactivity
   The turbine trip is analyzed with minimum reactivity feedback. The minimum feedback cases assume a minimum moderator temperature coefficient and the least negative Doppler coefficient.

3. Reactor Control
   From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

4. Steam Releases
   No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits the secondary steam pressure at the setpoint value.

5. Pressurizer Spray and Power-operated Relief Valves
   Two cases with minimum reactivity feedback conditions were analyzed:
   (a) For the DNB case, full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

   (b) For the overpressure case, no credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

6. Feedwater Flow
   Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur. However, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip.

14.1.8.3 Results

The transient responses for a total loss of load from full power operation are shown on Figures 14.1-31 to 14.1-33 and Figures 14.1-37 through 14.1-39 for two cases; one case with pressure

control, one case without pressure control, both assuming minimum reactivity feedback conditions. Previously, four cases were analyzed; two cases at BOL minimum reactivity feedback conditions and two cases at EOL reactivity feedback conditions. Since the Loss of Load/ Turbine Trip event results in a primary system heatup, the analysis conservatively assumed minimum reactivity feedback conditions for both, with and without pressurizer pressure control which bounds the event with EOL reactivity feedback conditions.

Figures 14.1-31 through 14.1-33 show the transient responses for the total loss of steam load assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the high pressurizer pressure trip channel.

The minimum DNBR is well above the limit value. The pressurizer power operated relief valves are actuated for this case and maintain system pressure below 110-percent of the design value. The steam generator safety valves open and limit the secondary steam pressure increase.

The total loss of load event was also analyzed assuming the plant to be initially operating at full power [Deleted] conditions with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 14.1-37 through 14.1-39 show the transients without credit for pressurizer spray or power-operated relief valves. The neutron flux remains essentially constant at full power until the reactor is tripped. The DNBR increases throughout the transient. In this case the pressurizer safety valves are actuated and maintain the system pressure below 110-percent of the design value.

Table 14.1-6 summarizes the sequence of events for the various transients considered for the total loss of load cases presented above.

The results of the complete loss of steam load from full power evaluation concluded that a loss of steam load without direct reactor trip on turbine trip below 35% of full power is bounded by the complete loss of flow event described in Section 14.1.6 with respect to the minimum DNBR condition reached during the transient and bounded by the loss of load (turbine trip) event from full power conditions with respect to peak overpressure RCS conditions.

14.1.8.4 Conclusions

The results of the analyses performed for a total loss of external electrical load without a direct or immediate reactor trip from full power conditions show that the plant design is such that there would be no challenge to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the design of the plant would be adequate to limit the maximum pressures to within the design limits. In addition, the integrity of the core would be maintained by operation of the reactor protection system; i.e., the DNBR would be maintained above the safety analysis limit value. Thus, no core safety limit would be violated. Furthermore, these results, in conjunction with the results for the complete loss of flow event from full power, bound the results for a complete loss of load from 50% power without a direct reactor trip on turbine trip.

14.1.9 Loss Of Normal Feedwater

14.1.9.1 Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If an alternate supply of feedwater were not furnished, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the reactor coolant system and possible core damage. Since the plant is tripped well before the steam generator heat transfer capacity would be reduced, the primary system variables never approach a departure from nucleate boiling condition.

The following events occur upon the loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

A. As the steam pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the steam generator power-operated relief valves are not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and reactor coolant pumps plus the residual decay heat produced in the reactor.

B. As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

Following the occurrence of a loss of normal feedwater, the reactor may be tripped by any of the following reactor protection system trip signals:

a. Low-low steam generator water level
b. Over-Temperature ΔT
c. High pressurizer pressure
d. High pressurizer water level
e. RCP undervoltage (if coincident with a LOOP signal)
f. Steam flow-feedwater flow mismatch in coincidence with low water level in any steam generator.

Auxiliary Feedwater (AFW) is supplied by actuation of two motor-driven auxiliary feedwater pumps, which are initiated by any of the following signals:

a. Low-low water level in any steam generator.
b. Automatic trip (not manual) of any main feed pump turbine.
c. Any safety injection signal.
d. Manual actuation.
e. Loss of offsite power concurrent with unit trip.

In addition, one turbine driven auxiliary feedwater pump starts on the following actuation signals although no automatic delivery of water to the steam generators occurs:

a. Low-low level in any two steam generators.

b. Loss of offsite power concurrent with unit trip and no safety injection signal.
c. Manual actuation.

The motor-driven auxiliary feedwater pumps are powered by the emergency diesel generators. The pumps take suction from the condensate storage tank for delivery to the steam generators. Each motor-driven pump is designed to supply the minimum required flow within 60 seconds of the initiating signal. The turbine-driven AFW pump is valved out during normal operation. Therefore, although it is automatically actuated, it is not available to deliver flow to the steam generators until an operator action is taken to align the turbine-driven train. Steam Generator Blowdown isolation is assumed starting from event initiation.

Backup in equipment and control logic is provided to ensure that reactor trip and automatic auxiliary feedwater flow will occur following any loss of normal feedwater, including that followed by loss of offsite power. An analysis of the system transient is presented below to show that following a loss of normal feedwater, the auxiliary feedwater system is capable of removing the stored and residual heat plus reactor coolant pump waste heat, thus preventing either overpressurization of the RCS or loss of water from the reactor core, and the plant returning to a safe condition.

14.1.9.2 Method of Analysis

A detailed analysis using the RETRAN computer code (Reference 21a) is performed to determine the plant transient following a loss of normal feedwater. The code simulates the core neutron kinetics, reactor coolant system, pressurizer, pressurizer power operated relief valves and safety valves, pressurizer heaters and spray, steam generators, main steam safety valves, and the auxiliary feedwater system, and computes pertinent variables, including pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

Assumptions made in the analysis are:

1. Initial steam generator level is at the nominal programmed value plus 10% narrow range span (NRS). Reactor trip occurs on steam generator low-low level at 0% of narrow range span.

2. The plant is initially operating at 102-percent of the NSSS power (3230 MWt) which bounds a nominal pump heat of 14MWt.

3. Conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 decay heat standard (ANS 5.1) plus 2 sigma uncertainty was used for calculation of residual decay heat levels.

4. The worst single failure in the AFW system occurs, i.e., failure of one of the motor-driven auxiliary feedwater pumps. The Auxiliary Feedwater System is assumed to automatically supply a total of 380 gpm to two steam generators from one motor-driven pump. Additional flow from the turbine-driven auxiliary feedwater pump is assumed available only following an operator action to align the turbine-driven pump.

5. The pressurizer sprays, heaters, and power operated relief valves are assumed operable. This maximizes the peak transient pressurizer water volume. If these control systems did not operate, the pressurizer safety valves would maintain peak RCS pressure at or below the actuation setpoint throughout the transient.

6. Secondary system steam relief is achieved through the steam generator safety valves. No credit is taken for the operation of steam dumps or power-operated relief valves.

7. Cases are analyzed assuming initial hot full power reactor vessel average coolant temperatures at the upper and lower ends of the uprated operating range with uncertainty applied in both the positive and negative direction. The vessel average temperature assumed at the upper end of the range is 572°F with an uncertainty of ±7.5°F. The average temperature assumed at the lower end of the range is 549°F with an uncertainty of ±7.5°F. Results for the limiting case are presented.

8. Initial pressurizer pressure is assumed to be 2250 psi with an uncertainty of +28/-37 psi. Cases are considered with the pressure uncertainty applied in both the positive and negative directions to conservatively bound potential operating conditions. Results for the limiting case are presented.

9. Cases are analyzed assuming initial feedwater temperatures at the upper and lower ends of the uprated operating feedwater temperature window (436.2°F and 390°F, respectively).

10. The high T_{avg} program cases assumes an initial pressurizer level of 71-percent (65% + 6% uncertainty). For the low T_{avg} program cases, an initial pressurizer level of 43-percent (37% + 6% uncertainty) is considered.

11. The enthalpy of the auxiliary feedwater is assumed to be 90.77 Btu/lbm corresponding to a condensate storage tank temperature of 120 ^{o}F.

12. Analyses with both minimum (0%) and maximum (10%) steam generator tube plugging were performed to conservatively bound potential operating conditions.

13. An auxiliary feedwater line purge volume of 268.8 ft^3 is assumed.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards systems (i.e., the auxiliary feedwater system). The analysis demonstrates the capability of the AFW system to remove long term decay heat, thus preventing RCS overpressurization or loss of RCS water by overfilling the pressurizer.

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above.

For the loss of normal feedwater transient, the reactor coolant volumetric flow remains at its normal value and the reactor trips via the low-low steam generator level trip. The reactor coolant pumps may be manually tripped at some later time to reduce heat addition to the RCS.

Normal reactor control systems are not required to function in this analysis. The reactor protection system is required to function following a loss of normal feedwater as analyzed herein. The auxiliary feedwater system is required to deliver a minimum auxiliary feedwater flow rate and no single active failure will prevent operation of any system required to function.

14.1.9.3 Results

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Sixty seconds following the initiation of the low-low level trip, at least one motor-driven auxiliary feedwater pump is automatically started and supplying the minimum required flow to reduce the rate of decrease in steam generator water level.

The capacity of one motor driven auxiliary feedwater pump is such that the rate of decrease of the water level in the steam generator being fed AFW flow is sufficiently slowed to provide an allowable time for the operator to align the turbine-driven train and prevent water relief from the RCS relief or safety valves.

The calculated sequence of events for this accident is listed in Table 14.1-7. Figure 14.1-43 (Sheet 1 through Sheet 5) shows the significant plant parameters following a loss of normal feedwater. The Figures show that the plant approaches a stabilized condition following reactor trip and auxiliary feedwater initiation. Figure 14.1-43 Sheet 1 shows the pressurizer water volume transient. As shown in Figure 14.1-43 Sheet 3, RCS subcooling is maintained since the RCS never reaches saturated conditions. Plant procedures may be followed to further stabilize and cool down the plant.

14.1.9.4 Conclusions

Results of the analysis show that, for a loss of normal feedwater event, all safety criteria are met. The AFW capacity is sufficient to prevent pressurizer filling and any subsequent water relief through the pressurizer relief and safety valves. This assures that the RCS is not overpressurized.

14.1.10 Excessive Heat Removal Due To Feedwater System Malfunctions

14.1.10.1 Description

Excessive heat removal due to feedwater system malfunctions is a means of increasing core power above full power and can result from a decrease in feedwater enthalpy or excessive feedwater additions. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower and overtemperature protection (high neutron flux, overtemperature ΔT, and overpower ΔT trips) prevent any power increase that could lead to a DNBR that is less than the DNBR limit.

An example of a feedwater control system malfunction that results in a decrease in feedwater enthalpy would be an inadvertent opening of the feedwater bypass valve which diverts flow around the low pressure feedwater heaters. The feedwater bypass valve was retired in place when operating experience proved that it was not required for its intended purpose of providing sufficient suction pressure at the feed pumps. The description of this event, however, including the method of analysis, results and conclusions, is being retained herein for informational

purposes. For this event, there would be a sudden reduction in inlet feedwater temperature to the steam generator. The increased subcooling of the secondary side would create a greater load demand on the primary side which can lead to reactor trip conditions.

An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, these occurrences could also cause a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater might cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Continuous excessive feedwater addition would be prevented by the steam generator high-high level trip, which closes the feedwater control valves.

14.1.10.2 Method of Analysis

The excessive heat removal due to feedwater system malfunction transients were analyzed using the RETRAN code (Reference 21a).

The decrease in feedwater enthalpy event is conservatively assumed to occur at hot full power initial conditions. As a result of opening the feedwater bypass valve and diverting the flow around the low-pressure feedwater heaters, the feedwater temperature at the inlet of the steam generator in the affected loop decreases from 430°F to 420°F. This results in a decrease in the feedwater enthalpy of less than 11 Btu/lbm. An evaluation shows that the reduction in feedwater enthalpy by 11 Btu/lbm is significantly less than that for excessive load increase events described in Section 14.1.11. Therefore, excessive load increase events (cases with manual reactor control at BOL and with automatic reactor control at EOL) bound the feedwater enthalpy cases as previously described.

For the excessive feedwater addition due to a control system malfunction or operator error that allows a feedwater control valve to open fully, three cases were analyzed as follows:

1. Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large moderator density coefficient characteristic of end-of-life conditions and the reactor in manual rod control.

2. Accidental opening of one feedwater control valve from full power initial conditions with the reactor in manual rod control.

3. Accidental opening of one feedwater control valve from full power initial conditions with the reactor in automatic rod control.

The reactivity insertion rate following a feedwater system malfunction was calculated with the following assumptions:

1. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction, resulting in a step increase to 130% of nominal feedwater flow to one steam generator.

2. For the feedwater control valve accident at zero load conditions, a feedwater valve malfunction occurs that results in a ramp increase in flow to one steam

generator from zero flow at time zero to 210% of the nominal full load value for one steam generator at 5 seconds.

3. For the zero load condition, a conservatively low feedwater enthalpy corresponding to a feedwater temperature of 100°F is assumed.

4. No credit is taken for the heat capacity of the RCS and steam generator metal in attenuating the resulting plant cooldown.

5. No credit is taken for the heat capacity of the steam and water in the unaffected steam generators.

14.1.10.3 Results and Conclusions

For the feedwater enthalpy reduction event, the reduction in feedwater enthalpy is less than the equivalent reduction in feedwater enthalpy from the excessive load increase incident as described in Section 14.1.11. Therefore, the results for the excessive load increase incident, which show considerable margin to the DNBR limit exist under these same conditions, bound the feedwater enthalpy reduction cases.

In the case of excessive feedwater flow resulting from an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the resulting transient is similar to, but less severe than the hypothetical steamline break transient described in Section 14.2.5. Because the excessive feedwater flow cases with the reactor at zero power is bounded by the analysis presented in Section 14.2.5, no transient results are given in this section. It should be noted that if the incident occurs with the unit just critical at no-load, the reactor may be tripped by the power range neutron flux trip (low setting).

For the full power cases, the results with automatic rod control are nearly identical to those with manual rod control assumed. This is because the small increase in feedwater flow (30% above nominal) results in a very small increase in RCS temperature. The rod control system actuates but rod movement is minimal due to the small RCS temperature change.

Transient results showing the core heat flux, pressurizer pressure, T_{avg}, and DNBR, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor are given in Figure 14.1-45 for the full power case with manual rod control. Steam generator water level rises until the feedwater is terminated as a result of the high-high steam generator water level trip. The DNBR does not fall below the safety analysis DNBR limit. The calculated sequence of events for the full power cases are shown in Table 14.1-8.

14.1.11 Excessive Load Increase Incident

14.1.11.1 Description

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step-load increase or a 5% per minute ramp load increase in the range of 15 to 100% of full power (the elimination of the automatic control rod withdrawal function could require the use of manual rod control to have the reactor respond to the turbine load change and to restore the coolant average temperature

to the programmed value). Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals: i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided that blocks the opening of the valves unless a large turbine load decrease or turbine trip has occurred.

14.1.11.2 Method of Analysis

Historically, four cases were analyzed to demonstrate plant behavior following a 10% step load increase from rated load. These cases are as follows:

1. Reactor control in manual with beginning-of-life minimum moderator reactivity feedback.
2. Reactor control in manual with end-of-life maximum moderator reactivity feedback.
3. Reactor control in automatic with beginning-of-life minimum moderator reactivity feedback.
4. Reactor control in automatic with end-of-life maximum moderator reactivity feedback.

For the beginning-of-life minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and the least negative Doppler only power coefficient curve; therefore the least inherent transient response capability. For the end-of-life maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value and the most negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A conservative limit on the turbine valve opening (equivalent to 120% turbine load) was assumed, and all cases were analyzed without credit being taken for pressurizer heaters.

This accident was analyzed using the Revised Thermal Design Procedure (RTDP).[22] Initial reactor power, RCS pressure, and temperature were assumed to be at their nominal values. Uncertainties in initial conditions were included in the limit DNBR as described in Section 14.0.2.1.

Normal reactor control systems and engineered safety systems were not required to function for this event. The reactor protection system was assumed to be operable; however, reactor trip was not encountered for most cases due to the error allowances assumed in the setpoints. No single active failure would prevent the reactor protection system from performing its intended function.

The cases which assume automatic rod control were analyzed to ensure that the worst case with respect to minimum DNBR is presented. The automatic rod control function is not required to mitigate the consequences of this event. The automatic control rod withdrawal feature in plant operation has been physically disabled, allowing only the automatic control rod insertion mode to be in effect when rod control is in automatic.

Given the non-limiting nature of this event with respect to the DNBR safety analysis criterion, an explicit analysis was not performed as part of the Stretch Power Uprate program. Instead, a detailed evaluation of this event was performed. The evaluation model consists of the generation of statepoints based on generic conservative data. The statepoints are then compared to the core thermal limits to ensure that the DNBR limit is not violated. Since automatic rod withdrawal has been disabled at Indian Point Unit 2, only cases assuming manual rod control are evaluated.

These cases are:
- Reactor in manual rod control with EOL (maximum moderator) reactivity feedback.

14.1.11.3 Results and Conclusions

An evaluation of this event was performed to support the Stretch Power Uprate program. The evaluation determined that the DNB design basis for a 10% step load increase continues to be met.

14.1.12 Loss of all AC Power to the Station Auxiliaries

14.1.12.1 Description

A complete loss of non-emergency AC power may result in the loss of all power to the plant auxiliaries: i.e., the RCPs, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite non-emergency AC distribution system.

The first few seconds of the transient would be almost identical to the four pump loss-of-flow case presented in Section 14.1.6 where the pump coastdown inertia along with the reactor trip prevent reaching the DNBR limit. After the trip, decay heat will be accommodated by the auxiliary feedwater system. This portion of the transient would be similar to that presented in Section 14.1.9 for the Loss of Normal Feedwater event.

The events following such a condition are described in the sequence listed below:

1. Plant vital instruments are supplied by emergency power sources (See Chapter 8).

2. As the steam system pressure rises following the trip, the steam system power-operated relief valves are automatically opened to the atmosphere. Steam bypass to the condenser is not available because of loss of the circulating water pumps. If the power-operated relief valves are not available, the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

3. As the no-load temperature is approached, the steam system power-operated relief valves (or the self-actuated safety valves, if the power-operated relief valve are not available) are used to dissipate the residual heat and to maintain the plant at the hot standby condition.

4. The emergency diesel generators are started on loss of voltage on the plant emergency buses and begin to supply plant vital loads.

The auxiliary feedwater system is started automatically as discussed in Section 14.1.9 for the loss of normal feedwater analysis. The two motor-driven AFW pumps are supplied by power from the emergency diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators. Each motor-driven pump is designed to supply the minimum required flow within 60 seconds of the initiating signal. Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops aided by the auxiliary feedwater in the secondary system. The analysis here will show that following a loss of AC power event, the natural circulation flow in the RCS is sufficient to remove residual heat from the core.

14.1.12.2 Method of Analysis

A detailed analysis using the RETRAN computer code (Reference 21a) is performed to determine the plant transient following a loss of AC power to the station auxiliaries. The code simulates the core neutron kinetics, reactor coolant system including natural circulation, pressurizer, pressurizer power operated relief valves and safety valves, pressurizer heaters and spray, steam generators, main steam safety valves, and the auxiliary feedwater system, and computes pertinent variables, including pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

Major assumptions differing from those in a loss of normal feedwater presented in Section 14.1.9 are:

1. No credit is taken for immediate response of control rod drive mechanisms caused by a loss of offsite power.
2. A heat transfer coefficient in the steam generator associated with RCS natural circulation is assumed following the reactor coolant pump coastdown.
3. The plant is initially operating at 102-percent of the NSSS power (3230 MWt). A nominal RCP heat of 14 MWt was assumed.

The complete loss of non-emergency AC power analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards systems (i.e., the auxiliary feedwater system). The analysis demonstrates the capability of the AFW system to remove long term decay heat, thus preventing RCS overpressurization or loss of RCS water by overfilling the pressurizer.

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as discussed in Section 14.1.9 for the assumptions in the loss of normal feedwater analysis.

14.1.12.3 Results

Figure 14.1-50 (Sheet 1 through Sheet 5) shows the plant parameters following a loss of offsite power to the station auxiliaries. The time sequence of events for this accident is given in Table 14.1-10.

After the reactor trip, stored and residual heat must be removed to prevent damage to either the RCS or the core. The RETRAN results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

14.1.12.4 Conclusions

Results of the analysis show that, for the loss of offsite power to the station auxiliaries event, all safety criteria are met. The AFW capacity is sufficient to prevent water relief through the pressurizer relief and safety valves; this assures that the RCS is not overpressurized.

The analysis also demonstrates that sufficient long-term heat removal capability exists by the natural circulation capability of the RCS following reactor coolant pump coastdown to prevent fuel or clad damage.

14.1.13 Likelihood And Consequences of Turbine-Generator Unit Overspeed

The assessment of turbine-generator overspeed prepared and submitted in the original 1968 Indian Pont Unit 2 FSAR (as part of the initial license application) assumed that all turbine missiles (i.e., fragments of turbine rotor disks) would be contained within the turbine casing. Subsequent to that submittal, a 1970 study was prepared by Westinghouse to document the results of additional analytical and experimental work performed regarding the likelihood and consequences of turbine overspeed. (See Reference 30). In response to an AEC request for further information, this study was provided as Appendix 14A in Supplement 12 to the original FSAR prior to initial plant operation. The results showed that the original position on the containment of low pressure turbine disk fragments within the turbine casing could no longer be maintained and a completely independent turbine electric overspeed detection and valve trip initiation system (i.e., IEOPS) was incorporated into the original Indian Point Unit 2 design.

In the late 1980s, Westinghouse and the Westinghouse Owners Group proposed and the NRC approved the generic application of a revised probabilistic methodology for turbine missile generation likelihood and the appropriate frequencies for inspection, testing, and maintenance of turbine rotors and control systems (See References 28, 31, 32, 33). The NRC concluded that maintaining a small probability of turbine missile generation through testing and inspection is a reliable means of ensuring safety-related structures, systems, and components are adequately protected from such missiles and that the revised approach simplifies and improves procedures for evaluation of turbine missile risks by eliminating from consideration factors such as missile trajectory and damage probability. The NRC's revised acceptance criteria for total turbine missile generation probabilities was established as less than 1E-4 per year for a favorably oriented turbine and less than 1E-5 per year for an unfavorably oriented turbine.

By letter dated February 8, 1994 (Reference 34), the NRC issued Amendment No. 168 to the Indian Point Unit 2 Operating License which approved the application of the revised generic methodology to Indian Point Unit 2, a revised surveillance interval for testing turbine stop and control valves, and the deletion of Technical Specification limiting conditions for operation and surveillance requirements for the Independent Electrical Overspeed Protection System (IEOPS). The IEOPS has been disabled and is out of service. This approval was based on the application of the generic methodology and data of Reference 28 as supplemented by Reference 29, and the Consolidated Edison commitment contained in Reference 35 to review and re-evaluate the turbine valve testing frequency probabilistic analysis any time major changes in the turbine system have been made or a significant upward trend in the valve failure rate is identified. This

commitment included the incorporation of information on valve failure rates in the UFSAR and the updating of that information at least once every three years (See Section 14.1.13.2).

14.1.13.1 Turbine Control and Protection

The likelihood of a turbine-generator unit overspeed condition is remote because of the reliability and redundancy of the turbine control and protection systems.

The turbine control and protection system is completely hydraulic. There are two low-pressure oil control systems: the auxiliary governor system and the emergency trip system. These two systems and the 300-psi system are interconnected through orifices. The control and protection system is fail-safe; any loss of oil pressure causes closure of the steam valves.

The main governor normally controls the unit. Should an overspeed take place, the auxiliary governor system will be actuated first, the auxiliary governor dome valve will open, the 300-psi pressure oil will drain, and the control valve will close.

Should the unit overspeed reach the mechanical overspeed trip setpoint, the overspeed trip valve will open, the 300-psi pressure oil will drain, and the throttle valves will close. At the same time, a second drain path will be provided for the 300-psi oil system that controls the first set of valves, so that the control valves will trip too, in case they did not trip.

Assuming, for the purpose of analysis, that a control valve and stop valve in the same steam path fail to close, a turbine runaway would occur.

Besides the provisions in the design of the turbine control and protection system during plant operation, valves are exercised on a periodic basis to preclude the possibility of a valve stem sticking. Analyses of oil samples are performed as required.

The turbine is periodically given an overspeed check to verify the trip speed. The remaining tripping devices are periodically checked.

14.1.13.2 Analysis and Results

Reference 28 documents the probabilistic analysis performed to determine the annual turbine missile ejection probability, as a function of turbine valve test frequency, for a group of nuclear power plants with Westinghouse turbines. Testing of turbine valves affects the probability that the valves will be incapable of closing given that the load on the turbine is lost. The failure or unavailability of the turbine valves contributes to the probability that the turbine will overspeed and eject a missile.

The analysis of turbine overspeed included a thorough identification of all faults and contributors to overspeed. Specific plant data was collected from the turbine owners in the effort. In addition, other systems which interface with the turbine were investigated to determine whether they have any impact on the probability of overspeed. The study quantified the total risk of turbine missile ejection at destructive overspeed (approximately 180-percent of rated turbine speed) and at lower speeds in the range of 120 to 136-percent at rated speed. The lower speeds were evaluated in two categories: design overspeed and intermediate overspeed.

The analysis performed used fault trees to determine the annual probability of overspeed for each of the three overspeed events. Failures of turbine valves and overspeed protection

components were modeled in the fault trees as a function of the valve test intervals as appropriate. The probability of overspeed was calculated for various test intervals. The probabilities of missile generation for the design and intermediate overspeed conditions were determined based on plant-specific low pressure rotor design information. The probability of missile generation for the destructive overspeed event was assumed to be 1.0. For each overspeed event, the probability of the overspeed event was combined with the probability of missile generation for that event. The resulting annual probabilities of missile generation for each event, for a given test interval, were summed to provide the total.

Subsequent to the issuance of Reference 28, a subgroup of plants with Westinghouse BB-95/96 turbines evaluated more recent valve failure data and modes. Reference 29 modeled the revised failure rates and modes using a fault tree for the destructive overspeed event. The destructive overspeed probability was calculated for various turbine valve test intervals. An allowance was defined for the missile ejection contributions of the design and intermediate overspeed. Reference 29 provided revised guidance for determining appropriate turbine valve test intervals. Using the destructive overspeed results in conjunction with the allowance, the results may be used to determine an appropriate turbine valve test interval which meets the NRC acceptance criterion of 1.0E-5 per year. Reference 36 contains the most recent assessment of turbine valve failure data and covers a period from January 1986 through December 1999. The valve failure data is presented in Figures 14.1-62 through 14.1-66, and the turbine valve test interval currently recommended by the vendor is presented in Figure 14.1-67.

Indian Point Unit No. 2 has fully integral low pressure turbine rotors. The fully integral design eliminates the disk bores and keyways of the earlier design, reducing peak stresses and transferring the location of peak stresses to the blade fastening locations on the rotor. Reference 16 (submitted to the NRC by Westinghouse) discusses the probabilities of crack initiation and missile generation. The probability of creating a disk-segment missile is significantly lower for the fully integral design than for the previous design. Based on the conclusions in Reference 16, Consolidated Edison notified the NRC (Reference 17) that periodic in-service inspections of the fully integral low pressure turbine rotors will not be required.

Because of the very large margin between the high pressure spindle bursting speed and the maximum speed at which the steam can drive the unit with all admission valves full open, the probability of spindle failure is practically zero. Therefore, no harmful missile is expected from the high pressure turbine rotor in case of a turbine runaway.

REFERENCES FOR SECTION 14.1

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32. Letter from A. C. Thandani, NRC, to D. M. Musolf, Northern States Power Company, dated November 2, 1989, forwarding an NRC Safety Evaluation and Supplemental Safety Evaluation regarding use of WCAP-11525 in future license amendment requests by Westinghouse Owners Group (Turbine Valve Test Frequency Subgroup Members).

33. Letter from D. S. Brinkman to S. B. Bram, Consolidated Edison, dated November 30, 1989, regarding approval for referencing of WCAP-11525 by members of the Westinghouse Owners Group (Turbine Valve Test Frequency Evaluation Subgroup).

34. Letter from F.J. Williams, Jr., NRC, to S.B. Bram, Consolidated Edison, forwarding License Amendment No. 168 to Facility Operating License No. DPR-26 and associated Safety Evaluation.

35. Letter from S.B. Bram, Consolidated Edison, to Document Control Desk, NRC, "Proposed Amendment to Technical Specifications Regarding Turbine Valve Test Frequency and Electrical Overspeed Protection System", dated August 9, 1991.

36. Letter from R. Sisk, Westinghouse Owners Group, to Turbine Valve Test Frequency Evaluation Subgroup, dated October 16, 2000, forwarding "Final Update and Evaluation of BB-95/96 Turbine Valve Failure Data Base – September 29, 2000."

37. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

TABLE 14.1-1
Uncontrolled RCCA Withdrawal From a Subcritical Condition
Time Sequence of Events

Event	Time (Seconds)
Start of the accident	0.0
High Neutron Flux Reactor Trip Setpoint (Low Setting) reached	9.8
Rods begin to fall	10.3
Minimum DNBR occurs	11.8
Peak Clad Average Temperature occurs	12.0
Peak Fuel Average Temperature occurs	12.3
Peak Fuel Centerline Temperature occurs	13.2

SUMMARY OF THE RESULTS

Peak Clad Average Temperature (°F)	701
Peak Fuel Average Temperature (°F)	1927
Peak Fuel Centerline Temperature (°F)	2286

TABLE 14.1-2
Uncontrollable RCCA Bank Withdrawal at Power
Time Sequence of Events

Accident	Event	Time (Sec)
Uncontrollable RCCA bank withdrawal at power		
1. Case A	Initiation of uncontrollable RCCA withdrawal at a high reactivity insertion rate (70 pcm/sec)	0
	Power range high neutron flux high trip point reached	1.6
	Rods begin to fall into core	2.1
	Minimum DNBR occurs	3.0
2. Case B	Initiation of uncontrollable RCCA withdrawal at a small reactivity insertion rate (1 pcm/sec)	0
	Overtemperature ΔT reactor trip signal initiated	100.2
	Rods begin to fall into core	102.2
	Minimum DNBR occurs	103.0

TABLE 14.1-3
Complete Loss of Flow (Undervoltage)
Time Sequence of Events

Event	Time (Seconds)
All the pumps begin to coastdown	0.
Reactor coolant pump undervoltage trip point reached at	0.
Rods begin to fall	1.5
Minimum DNBR occurs	3.3
Maximum RCS pressure occurs	15.0

SUMMARY OF THE RESULTS
COMPLETE LOSS OF FLOW (Undervoltage)

Maximum RCS Pressure (psia)	2349

Complete Loss of Flow (Underfrequency)
Time Sequence of Events

Event	Time (Seconds)
Frequency decay of 5 Hz/sec begins	0.
Reactor coolant pump underfrequency trip point reached and all the pumps begin to coastdown	0.6
Rods begin to fall	1.6
Minimum DNBR occurs	3.6
Maximum RCS pressure occurs	15.2

SUMMARY OF THE RESULTS
COMPLETE LOSS OF FLOW (Underfrequency)

Maximum RCS Pressure (psia)	2366

TABLE 14.1-4
Partial Loss of Flow
Time Sequence of Events

Event	Time (Seconds)
One pump begins to coastdown	0.
Reactor coolant low-flow trip setpoint (85%) reached	1.6
Rods begin to fall	2.6
Minimum DNBR occurs	3.4
Maximum RCS pressure occurs	14.4

SUMMARY OF THE RESULTS
PARTIAL LOSS OF FLOW

Maximum RCS Pressure (psia)	2331

TABLE 14.1-5
Locked Rotor Event – Hot Spot
Time Sequence of Events

Event	Time (Seconds)
Rotor in one pump seizes	0.
Reactor coolant low flow trip setpoint reached at	0.10
Rods begin to fall	1.10
Maximum clad temperature occurs	3.8
Maximum RCS pressure occurs	5.1

SUMMARY OF THE RESULTS
LOCKED ROTOR EVENT – HOT SPOT

Maximum Reactor Coolant System Pressure (psia)	2553
Maximum Clad Average Temperature (°F)	1813
% Zirconium Reacted	0.31%

TABLE 14.1-6
Loss of External Electrical Load
Time Sequence of Events

Event	Time of event, sec With Pressurizer Control – DNB Case	Time of event, sec Without Pressurizer Control – RCS Overpressurization Case
Loss of electrical load/turbine trip	0.0	0.0
Initiation of steam release from SG safety valves	11.97	8.30
High pressurizer pressure reactor trip point reached	9.81	6.33
Rods begin to fall	11.81	8.33
Minimum DNBR occurs (min. DNBR = 2.06)	13.10	N/A
Peak RCS pressure occurs (peak RCS pres. = 2676.83 psia)	N/A	8.69
Peak MSS pressure occurs (peak MSS pres. = 1158.68 psia)	N/A	15.67

TABLE 14.1-7
Loss of Normal Feedwater
Time Sequence of Events

Event	Time of event, sec
Main feedwater flow stops	20.0
Low-low steam generator water level reactor trip setpoint reached	64.0
Rods begin to drop	66.0
Automatic auxiliary feedwater from one motor driven auxiliary feedwater pump initiated	124.0
Operator action to establish auxiliary feedwater flow to remaining steam generators	666.0
Peak water level in the pressurizer occurs	925.0

TABLE 14.1-8
Feedwater Malfunction Event
Time Sequence of Events

Event	Time of event, sec	
	With Automatic Rod Control	Manual Rod Control
Feedwater flow to one SG increases to 130% of nominal	0.0	0.0
Peak pressurizer pressure occurs	99.5	100.0
Peak nuclear power occurs	98.0	98.0
Minimum DNBR occurs	97.5	98.0

TABLE 14.1-9
DELETED

TABLE 14.1-10
Loss of All AC Power to the Station Auxiliaries
Time Sequence of Events

Event	Time of event, sec
Main feedwater flow stops	20.0
Low-low steam generator water level reactor trip setpoint reached	64.0
Rods begin to drop	66.0
Reactor coolant pumps begin to coast down	68.1
Automatic auxiliary feedwater from one motor driven auxiliary feedwater pump initiated	124.0
Operator action to establish auxiliary feedwater flow to remaining steam generators	666.0
Peak water level in the pressurizer occurs	720.0

TABLE 14.1-11
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TABLE 14.1-12
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TABLE 14.1-13
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TABLE 14.1-14
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TABLE 14.1-15
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TABLE 14.1-19
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TABLE 14.1-20
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TABLE 14.1-21
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14.1 FIGURES

Figure 14.1-8	Uncontrolled RCCA Bank Withdrawal From Full Power With Minimum Reactivity Feedback (1 pcm/sec Withdrawal Rate)
Figure 14.1-9	Uncontrolled RCCA Bank Withdrawal From Full Power With Minimum Reactivity Feedback (1 pcm/sec Withdrawal Rate)
Figure 14.1-10	Uncontrolled RCCA Bank Withdrawal From Full Power With Minimum Reactivity Feedback (1 pcm/sec Withdrawal Rate)
Figure 14.1-11	Minimum DNBR Versus Reactivity Insertion Rate, Rod Withdrawal From 100 Percent Power
Figure 14.1-12	Minimum DNBR Versus Reactivity Insertion Rate, Rod Withdrawal From 60 Percent Power
Figure 14.1-13	Minimum DNBR Versus Reactivity Insertion Rate, Rod Withdrawal From 10 Percent Power
Figure 14.1-14	Dropped Rod Incident Manual Rod Control Nuclear Power and Core Heat Flux at BOL (Small Negative MTC) for Dropped RCCA of Worth - 400 PCM
Figure 14.1-15	Dropped Rod Incident Manual Rod Control Core Average and Vessel Inlet Temperature at BOL (Small Negative MTC) for Dropped RCCA of Worth - 400 PCM
Figure 14.1-16	Dropped Rod Incident Manual Rod Control Pressurizer Pressure at BOL (Small Negative MTC) for Dropped RCCA Worth of 400 PCM
Figure 14.1-16a	Deleted
Figure 14.1-17	Dropped Rod Incident Manual Rod Control Nuclear Power and Core Heat Flux at EOL (Large Negative MTC) for Dropped RCCA of Worth - 400 PCM
Figure 14.1-18	Dropped Rod Incident Manual Rod Control Core Average and Vessel Inlet Temperature at EOL (Large Negative MTC) for Dropped RCCA of Worth - 400 PCM
Figure 14.1-19	Dropped Rod Incident Manual Rod Control Pressurizer Pressure at EOL (Large Negative MTC)for Dropped RCCA Worth of 400 PCM
Figure 14.1-20	Loss of One Pump Out of Four Nuclear Power and Core Heat Flux vs. Time
Figure 14.1-21	Loss of One Pump Out of Four Total Core Flow and Faulted Loop Flow vs. Time
Figure 14.1-22	Loss of One Pump Out of Four Pressurizer Pressure and DNBR vs. Time
Figure 14.1-23	Four Pump Loss of Flow - Undervoltage Nuclear Power and Core Heat Flux vs. Time
Figure 14.1-24	Four Pump Loss of Flow - Undervoltage Total Core Flow and RCS Loop Flow vs. Time
Figure 14.1-25	Four Pump Loss of Flow - Undervoltage Pressurizer Pressure and DNBR vs. Time
Figure 14.1-26	Four Pump Loss of Flow - Underfrequency Nuclear Power and Heat Flux vs. Time
Figure 14.1-27	Four Pump Loss of Flow - Underfrequency Total Core Flow and RCS Loop Flow vs. Time

14.2 STANDBY SAFETY FEATURES ANALYSIS

Adequate provisions have been included in the design of the plant and its standby engineered safety features to limit potential exposure of the public to below the applicable limits for situations that could conceivably involve uncontrolled releases of radioactive materials to the environment. The following situations have been considered:

1. Fuel-handling accidents.
2. Accidental release of waste liquid.
3. Accidental release of waste gases.
4. Rupture of a steam-generator tube.
5. Rupture of a steam pipe.
6. Rupture of a control rod drive mechanism housing - rod cluster control assembly ejection.

14.2.1 Fuel-Handling Accidents

The possibility of a fuel-handling incident is very remote because of the many administrative controls and physical limitations imposed on fuel-handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety. Before any refueling operations begin, a verification of complete rod cluster control assembly insertion is obtained by opening the reactor trip breakers and observing the rod position indicators. Boron concentration in the coolant is raised to the refueling concentration and verified by sampling.

After the vessel head is removed, the rod cluster control drive shafts are disconnected from their respective assemblies using the manipulator crane and the shaft unlatching tool. A spring scale is used to indicate that the drive shaft is free of the control cluster as the lifting force is applied. The fuel-handling manipulators and hoists are designed so that fuel cannot be raised above a position that provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pit area. In the spent fuel pit, the design of storage racks and manipulation facilities is such that:

1. Fuel at rest is positioned by positive restraints in an eversafe, always subcritical, geometrical array. Even if an assembly is not placed in the correct region, sub-criticality is ensured because a minimum boron concentration of 2000 ppm is required at all times in the pool.

2. Fuel can be manipulated only one assembly at a time.

3. Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.

In addition, administrative controls do not permit the handling of heavy objects above the fuel racks under conditions specified in the Technical Requirements Manual.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pit.

Even if a spent fuel assembly becomes stuck in the transfer tube, the fuel assembly is completely immersed and natural convection will maintain adequate cooling to remove the decay heat. The fuel-handling equipment is described in detail in Section 9.5.

Two nuclear instrumentation system source range channels are continuously in operation and provide warning of any approach to criticality during refueling operations. This instrumentation provides a continuous audible signal in the containment and will annunciate a local horn and a horn and light in the plant control room if the count rate increases above a preset low level.

Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least 5-percent with all rod cluster control assemblies inserted. The refueling cavity is filled with water meeting the same boric acid specifications.

Special precautions are taken in all fuel-handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pit and during installation in the reactor. All handling operations on irradiated fuel are conducted under water. The handling tools used in the fuel-handling operations are conservatively designed, and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies from Unit 2 and Unit 3 are spaced in a pattern that prevents any possibility of a criticality accident. As required by 10 CFR 50.68, “Criticality Accident Requirements,” if the spent fuel pit takes credit for soluble boron, then “the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 per cent probability, 95 percent confidence level, if flooded with unborated water.” NET-173-01, “Criticality Analysis for Soluble Boron and Burnup Credit in the Con Edison Indian Point Unit 2 Spent Fuel Storage Racks” and NET-173-02, “Indian Point Unit 2 Spent Fuel Pool (SFP) Boron Dilution Analysis,” determined that 10 CFR 50.68(b)(4) will be met during normal SFP operation and all credible accident scenarios (including affects of boraflex degradation) if: a) spent fuel pit boron concentration is maintained within the Technical Specification limits and, b) fuel assembly storage location within the spent fuel pit is restricted based on the fuel assembly’s initial enrichment, burnup, decay of Pu^{241} (i.e., cooling time) and number of Integral Fuel Burnable Absorbers (IFBA) rods.

Northeast Technology Corporation report NET-173-01 evaluated non-accident conditions in the SFP including the affects of the projected boraflex degradation through the year 2006. Based upon BADGER testing in calendar years 2003, 2006 and 2010 and RACKLIFE code projections, the validity of the criticality and boron dilution analysis documented in NET-173-01 and NET-173-02 can be extended through the end of the current license (September 28, 2013). RACKLIFE calculation performed in 2012 allowed BADGER testing to be performed in 2013, to confirm the progression of localized Boraflex dissolution. The continued validity of the criticality and boron dilution analysis will be verified based on the Boraflex Monitoring Program as defined in the License Renewal Application.. This report determined that if storage location requirements in the Technical Specifications are met then the SFP will have a keff of ≤0.95 if filled with a soluble boron concentration of ≥786 ppm and will have a keff of <1.0 if filled with unborated water.

Northeast Technology Corporation report NET-173-01 also evaluated credible abnormal occurrences in accordance with ANSI/ANS-57.2-1983. This evaluation considered the effects of the following: a) a dropped fuel assembly or an assembly placed alongside a rack; b) a

misloaded fuel assembly; and, c) abnormal heat loads. Northeast Technology Corporation report NET-173-01 determined that the SFP will maintain a keff of ≤0.95 under the worst-case accident scenario if the SFP is filled with a soluble boron concentration of ≥ 1495 ppm.

Therefore, Northeast Technology Corporation report NET-173-01 confirmed that the requirements in 10 CFR 50.68, "Criticality Accident Requirements," will be met for both normal SFP operation and credible abnormal occurrences if:

a) Spent Fuel Pit boron concentration is maintained within the limits Technical Specifications, and;
b) Fuel assembly storage location within the spent fuel pit is restricted in accordance with Technical Specifications based on the fuel assembly's initial enrichment, burnup, decay of Plutonium-241 (i.e. cooling time), and number of Integral Fuel Burnable Absorbers (IFBA) rods.

Northeast Technology Corporation report NET-173-02 evaluated postulated unplanned SFP boron dilution scenarios assuming an initial SFP boron concentration within the Technical Specification limit. The evaluation considered various scenarios by which the SFP boron concentration may be diluted and the time available before the minimum boron concentration necessary to ensure subcriticality for the non-accident condition (i.e. it is not assumed an assembly is misloaded concurrent with the spent fuel pit dilution event). Northeast Technology Corporation report NET-173-02 determined that an unplanned or inadvertent event that could dilute the SFP boron concentration from 2000 ppm to 786 ppm is not a credible event because of the low frequency of postulated initiating events and because the event would be readily detected and mitigated by plant personnel through alarms, flooding, and operator rounds through the SFP area.

Northeast Technology Corporation report NET-173-01 and NET-173-02 are based on conservative projections of amount of Boraflex absorber panel degradation assumed in each sub-region. These projections are valid through the end of the year 2006. Based upon BADGER testing in calendar years 2003, 2006 and 2010 and RACKLIFE code projections, the validity of the criticality and boron dilution analysis documented in NET-173-01 and NET-173-02 can be extended through the end of the current license (September 28, 2013). RACKLIFE calculation performed in 2012 allowed BADGER testing to be performed in 2013, to confirm the progression of localized Boraflex dissolution. The continued validity of the criticality and boron dilution analysis will be verified based on the Boraflex Monitoring Program as defined in the License Renewal Application. These compensatory measures for boraflex degradation in the SFP were evaluated by the NRC in Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 227 to Facility Operating License No. DPR-26, May 29,2002. The design of the facility is such that it is not possible to carry heavy objects, such as a spent fuel transfer case, over the fuel assemblies in the storage racks. The design is such that only one fuel assembly can be handled at a given time.

The motions of the cranes that move the fuel assemblies are limited to a low maximum speed. Caution is exercised during fuel handling to prevent a fuel assembly from striking another fuel assembly or structures in the containment or fuel storage building.

The fuel-handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

All these safety features and precautions make the probability of a fuel handling incident very low. Nevertheless, since it is possible that a fuel assembly could be dropped during the handling operations, the radiological consequences of such an incident were evaluated.

Sections 14.2.1.1 through 14.2.1.3 specifically address evaluations performed for the following accidents:

1. Fuel-handling accident in the fuel-handling building.
2. Refueling accident inside containment.
3. Fuel-handling cask drop accident.

14.2.1.1 Fuel-Handling Accident in Fuel-Handling Building

As a design basis for equipment in the fuel-handling area, consideration has been given to the perforation of all rods in one assembly resulting from a dropped fuel assembly during refueling.

Provisions have been included in the fuel-handling building to give further assurance that the consequences of this fuel-handling accident involving a spent fuel assembly will be acceptable.

To show that the radiological consequences of the postulated accident in which all rods in an assembly are breached are acceptable, an investigation was made to determine what the expected situation would be in terms of iodine available for release to the spent fuel pool water, the retention of iodine in the pool water, and the resulting doses at the site boundary. These analyses do not take credit for either building holdup of the iodines or removal by charcoal filters. The activity released from the damaged assembly is assumed to be released to the environment over a two hour period.

The consequences of an accident in which all rods in an assembly are breached under water have been analyzed. This is a conservative design-basis case, in which factors are introduced to allow for uncertainties.

In the analysis, conservative assumptions regarding fission product inventories and species distribution were made as summarized in Table 14.2-2. The maximum offsite doses are 4.2 rem total effective dose equivalent (TEDE) at the site boundary and 2.0 rem TEDE at the low population zone. Thus the consequences of the postulated fuel-handling accident are well within (i.e., less than 25-percent of) the limits of 10 CFR 50.67 which is the dose acceptance limit identified in Regulatory Guide 1.183 (Reference 20). All reasonable measures are employed in the handling of irradiated fuel to ensure against the occurrence of fuel damage and the associated radiological hazard.

The accumulated dose of the control room operators following the postulated accident was calculated using the same release, removal and leakage assumptions as the offsite dose, using the control room model discussed in Section 14.3.6.5 and Tables 14.3-50 and 14.3-51. The calculated control room dose is presented in Table 14.3-52 and is less than the 5.0 rem TEDE control room dose limit values of 10 CFR 50.67.

14.2.1.1.1 Basis for Assumptions

The fuel handling accident considers the release of all fuel-to-clad gap activity from one fuel assembly. The radial peaking factor ($F\Delta H$) applied to this assembly is 1.7.

A value of 285 for the pool elemental iodine decontamination factor was conservatively assumed. A decontamination factor of 1.0 is modeled for organic iodine and noble gases. The iodine released from the assembly gap is assumed to be 99.85% elemental and 0.15% organic. The overall pool decontamination factor for iodine is 200.

No credit is taken for removal of iodine by filters nor is credit taken for isolation of the release path. The Fuel Storage Building Ventilation System will remain in operation and discharge through the plant stack as approved by Technical Specification Amendment 211. Although the containment purge will be automatically isolated on a purge line high radiation alarm, isolation is not modeled in the analysis. The analysis assumes that the equipment hatch and airlock doors will be open (no credit is given for the requirement to maintain outage management administrative controls in place for re-establishing containment closure consistent with plant conditions). The activity released from the damaged assembly is assumed to be released to the environment over a two hour period. Since no filtration or containment isolation is modeled, this analysis supports refueling operations in either the containment or the fuel handling building.

The decay time prior to fuel movement assumed in the fuel handling accident radiological consequences analysis is 84 hours.

14.2.1.1.2 Calculation of Offsite Exposure

In calculating offsite exposure, it is assumed that the incident occurs in either the spent fuel pit or in the containment building and that the activity is discharged to the atmosphere at the ground level.

The dispersion of this activity is computed using the Gaussian plume dispersion formula and taking credit for building wake dilution as included in the 2 hr dispersion factor developed in Section 14.3.

The dose calculations were performed for a two hour release of the fuel assembly gap activity to the spent fuel pit water. The dispersion factors (X/Q) used in the calculations are for the site boundary, and the low population zone.

The Total Effective Dose Equivalent (TEDE) dose is the sum of Committed Effective Dose Equivalent (CEDE) dose and Effective Dose Equivalent (EDE) dose for the duration of the exposure to the cloud.

14.2.1.1.2.1 Iodine Committed Effective Dose Equivalent (CEDE) Dose

A delay of at least 84 hours is required after shutdown before fuel movement. The iodine activity remaining in the fuel assembly gap at the end of this 84 hours is used as input to the calculation. The following equation is used to obtain the integrated CEDE dose at the site boundary or low population zone for each of the iodine isotopes.

$$D_I = \left[A_I \times \left(\frac{1}{DF} \right) \times DCF_I \times \frac{\chi}{Q} \times B \right]$$

where

A_I = activity of the Iodine isotope in fuel assembly gap after 100 hours of decay (Ci)

DF = decontamination factor for iodine in water

DCF_I = CEDE dose conversion factor for the Iodine isotope from EPA-520/1-88-020 (Reference 21) (rem/Ci) *[Note: See Table 14.2-2]*

χ/Q = atmospheric dilution factor (sec/m^3)

B = breathing rate (m^3/sec)

14.2.1.1.2.2 Effective Dose Equivalent (EDE) Dose

On an isotopic basis, the equations for the integrated doses at the site boundary or low population zone are given by:

$$D_I = 0.25\left[A_I \times \left(\frac{1}{DF}\right) \times E_I \times \frac{\chi}{Q}\right]$$

$$D_I = \left[A_I \times DCF_I \times \frac{\chi}{Q}\right]$$

where

A_I = activity of the isotope in fuel assembly gap after 84 hours of decay (Ci)

DF = pool decontamination factor (200 for iodines and 1.0 for noble gases)

DCF_I = EDE dose conversion factor for the nuclide from EPA 402-R-93-081 (Reference 22) (rem-m^3/Ci-sec) [*Note - **See Table 14.2-2*]

χ/Q = atmospheric dilution factor (sec/m^3)

14.2.1.2 Refueling Accident Inside Containment

Since no filtration or isolation of the release path is modeled in the analysis for the accident occurring in the fuel handling building, the analysis presented above (Section 14.2.1.1) is applicable to the accident occurring in containment.

14.2.1.3 Fuel Cask Drop Accident

As discussed in Section 9.5.6.4, Control of Heavy Loads Program, and Section 9.5.7.1, FSB 110-Ton Ederer Single Failure Proof Gantry Crane, the IP2 fuel storage building spent fuel cask handling operations are now conducted using a single-failure-proof 110-Ton Ederer Gantry Crane that conforms to the requirements in NUREG-0554 (Single-Failure-Proof Cranes for Nuclear Power Plants, May 1979). The Ederer Gantry Crane performs spent fuel cask handling activities without the necessity of having to postulate the drop of a spent fuel cask. With the Ederer crane's 110-ton main hoist qualified as single-failure-proof, the crane is used as part of a single-failure-proof handling system for critical lifts as discussed in Revision 1 of Section 9.1.5, Overhead Heavy Load Handling Systems, Sub-section III.4.C of NUREG-0800, and a cask drop accident is not a credible event and need not be postulated. Even though the IP2 fuel storage building 40-ton bridge crane is no longer used for spent fuel cask handling, the following fuel cask drop accident provisions and results are being retained since the analysis bounds other

drop accidents that may be postulated in the fuel storage building and cask loading pit even though a cask drop accident is no longer credible.

Performing an evaluation using the analysis assumptions for the fuel-handling accident shows that even with damage to a full core of recently discharged fuel assemblies by a fuel cask dropped into the spent fuel pool, the calculated fuel-handling accident doses would not be exceeded if 90 days had elapsed after shutdown. Since the fuel cask is handled by the single failure proof 110-ton gantry crane approved for use by the NRC in Technical Specification Amendment #244, this accident is not probable. Additional protections making this accident highly improbable are outlined below. In addition, Technical Requirements Manual Sections B 3.9.C and 3.9.E preclude movement of a spent fuel cask over any spent fuel storage racks.

During normal operation, if a spent fuel cask were placed in or removed from its position in the spent fuel pit, limit switches on the rails and logic built into the single-failure-proof control system would not allow for the cask to be moved any farther north or east of the spot reserved for the cask in the pit.

It is extremely improbable that the cask would be inadvertently or otherwise dropped during the process of transfer. This is due to the following provisions:

1. Conservative design margins used for the cask-related handling equipment (crane, rigging, hooks, etc.).
2. Periodic nondestructive equipment tests and inspection procedures.
3. Use of qualified crane operator and riggers.
4. Use of approved operating and administrative procedures.

These provisions will be rigorously met so that the inadvertent drop of the cask into the pool is highly improbable. However, should such a highly unlikely accident occur, the basic assumptions for analysis are as follows:

1. The drop would be from the highest position of the cask, which is 5-ft above the water surface and 43-ft above the bottom of the pool.
2. The cask is fully loaded and weighs 40 tons.

The results of the analysis indicate that the cask would hit the bottom of the pit with a velocity of approximately 40-ft/sec, assuming a conservative drag coefficient of 0.5. In comparison, the cask would have reached a velocity of 52-ft/sec if dropped through 43-ft in air.

Using the Ballistic Research Laboratories formula for the penetration of missiles in steel, the depth of penetration of the cask into the 1-in. wear plate covering the 1/4-in. pit liner plate would be 0.35-in., assuming the cask struck the wear plate while in a perfectly vertical position. In the event that the cask falls through the water at an angle, terminal velocity of the cask would be somewhat less because of the increased drag. However, the cask would strike the wear plate with an initial line contact and would penetrate the wear plate and the pit liner plate, causing some cracking of the concrete below. This reinforced concrete is a minimum of 3-ft thick and rests on solid rock.

Water would initially flow through the punctured liner plate and fill the cracks in the concrete. Since the pit is founded on solid rock and since the bottom of the pit is approximately 24 feet below the surrounding grade, very little water can be lost from the pit. The capacity of the

makeup demineralized water supply to the pit is 150 gpm. In addition, the spent fuel pit cooling system piping has a 4-in. blind flange connection for temporary cooling and/or makeup water.

Because the bottom of the spent fuel pit is 24-ft below grade and no equipment areas are in the vicinity, there can be no flooding of other areas with subsequent damage to equipment.

14.2.2 Accidental Release-Recycle Of Waste Liquid

Accidents that would result in the release of radioactive liquids are those which may involve the rupture or leaking of system pipe lines or storage tanks. The largest vessels are the three liquid holdup tanks of the chemical and volume control system, each sized to hold two-thirds of the reactor coolant liquid volume. The tanks are used to process the normal recycle of waste fluids produced. The contents of one tank will be passed through the liquid processing train while another tank is being filled.

All liquid waste components of the waste disposal system except the reactor coolant drain tank and the waste holdup tank are located in the auxiliary building, and any leakage from the tank or piping will be collected in the building sump to be pumped back into the liquid waste system. The waste holdup and the liquid holdup tanks are located in a thick concrete under-ground vault. The vault volume is sufficient to hold the full volume of any tank without overflowing to areas outside the vault. The reactor coolant drain tank is located in the containment building. Holdup tanks are equipped with safety pressure relief and designed to accept the established seismic forces at the site. Liquids in the chemical and volume control system flowing into and out of these tanks are controlled by manual valve operation and governed by prescribed administrative procedures.

The volume control tank design philosophy is similar in many respects to that applied to the holdup tanks. Level alarms, pressure relief valves, and automatic tank isolation and valve control ensure that a safe condition is maintained during system operation. Excess letdown flow is directed to the holdup tanks via the reactor coolant drain tank.

Piping external to the containment running between the containment and the auxiliary building and between the auxiliary building and liquid holdup tank vault is run below grade in concrete trenches. Any liquid spillage from pipe rupture or leaks in these trenches would drain to the sump and be pumped to the sump tank and to the waste holdup tank.

The incipient hazard from these process or waste liquid releases is derived only from the volatilized components. The releases are described and their effects summarized in Section 14.2.3.

A river diffusion analysis was performed to determine the concentrations that would result at the Chelsea reservoir if a release of waste liquid to the river was assumed. The results of the analysis show that even the instantaneous release of the entire primary coolant system maximum activity corresponding to operation with 1-percent defects would not result in peak concentrations at Chelsea in excess of 10 CFR 20 MPC limits. Drought conditions were assumed to exist at the time of and for a period following the spill limiting the total runoff flow to 4000 cfs. The mean longitudinal diffusion coefficient corresponding to this flow was 8.74 mi^2 per day. These data represent a drought similar to conditions existing in late summer of 1964, which can be verified by data in Section 2.5.

The unlikely event of a loss of water from a spent resin storage tank actuates a low-level alarm to warn the operator. Resin contained in the tank can then be cooled by periodically flushing water from the primary water storage tank through the resin. Two pathways are available for the water: (1) through the primary water storage injection pipeline used when resin is removed from the tank, or (2) through the primary water pipeline used when resin is sluiced from the demineralizers into the tank.

The following conservative assumptions are made to determine the frequency of flushing to cool the resin:

1. The tank contains resin from the letdown line mixed-bed demineralizers discharged to the spent resin storage tank following the operation of the plant for one cycle with 1-percent fuel defects. This assumption yields the maximum heat generation rate per unit volume of resin in the tank and the maximum level of radioactivity in the tank.

2. There are no heat losses through the tank walls.

3. Water is lost immediately following the discharge of a mixed-bed resin into the spent-resin storage tank. This yields a maximum heat generation rate due to fission product decay.

4. The heat generation rate and resin bed temperature equations are developed based on one cubic foot of resin.

5. The mean heat capacity of the resin is 0.31 Btu/lb-°F.

6. Resin specific gravity is 1.14 with a void fraction of 0.4 giving a resin density of 43 lb/ft^3.

7. The amount of radioactivity in the resin is:

	$\mu Ci/cm^3$
Br-84	6.4E-1
I-131	1.4E4
I-132	1.7E2
I-133	2.4E3
Rb-88	3.1E1
Cs-134	3.7E4
Cs-136	1.3E3
Ba-137m	2.6E4
Cs-138	1.3E1
Co-60	1.5E2
La-140	3.1E1

(There is assumed to be sufficient Cs-137 on the resin to maintain the inventory of Ba-137m at the above value.)

These assumptions result in the following relationships:

1. The heat generation rate, q (Btu/hr per cubic foot of resin), due to fission product decay is approximated closely as a function of time, t (hr), by

$$q = 0.247e^{-0.0578t} + 0.869e^{-0.0144t} + 4.65e^{-0.00144t} + 41.95$$

where the first term is due to nuclides with half lives less than 12 hours, the second term is due to nuclides with half-lives between 12 hours and 2 days, the third term is due to nuclides with half-lives between 2 and 20 days, and the last term is due to nuclides with half lives >20 days.

2. The resin bed temperature, T ($^\circ$F), as a function of time, t (hr), is

$$T = 0.32(1 - e^{-0.0578t}) + 4.52(1 - e^{-0.0144t}) + 242(1-e^{-0.00144t}) + 3.15t + T_o$$

where T_o is the initial resin temperature.

If T_o is assumed to be 90°F, it will take 14 hours for the resin temperature to rise to 140^{o}F, the normal resin operating limit. At or below a temperature of 140°F, the radioactivity will not be released from the resin. The actual time to heat to 140°F will be greater than 14 hours because of the conservative assumptions made in the calculation. With 100 cubic feet of resin in the tank, the heat accumulated in the resin through the initial 14 hours will be 66,650 Btu. The resin can be maintained at 140°F or less by back flushing the resin with primary water at appropriate intervals. Flush water will be collected by the floor drain system and be pumped to the waste holdup tank. If a 10°F rise is taken in the flush water, the total quantity of water required will be about 810 gal per backflush operation to remove the 66,650 Btu accumulated in the resin.

Hence, the loss of water from the spent resin storage tank presents no hazard offsite or onsite because means are available both to detect the situation occurring and to keep the resin temperature under control until the resin can be removed to burial facilities.

14.2.3 Accidental Release - Waste Gas

The leakage of fission products through cladding defects can result in a buildup of radioactive gases in the reactor coolant. Based on experience with other operational, closed-cycle, pressurized-water reactors, the number of defective fuel elements and the gaseous coolant activity is expected to be low. The shielding and sizing of components such as demineralizers and the waste handling system are based on activity corresponding to 1-percent defective fuel which is at least an order of magnitude greater than expected. Tanks accumulating significant quantities of radioactive gases during operation are the gas decay tanks, the volume control tank, and the liquid holdup tanks.

The volume control tank accumulates gases over a core cycle by stripping action of the entering spray. Gaseous activity for the tank based on operation with 1-percent defective fuel is given in Table 14.2-5. During a refueling shutdown, this activity is vented to the waste gas system and stored for decay. A rupture of this tank is assumed to release all of the contained noble gases plus a small fraction of the iodine in the tank (a partition factor of 0.01 is used). Also, the noble gas activity and a fraction of the iodine activity contained in the letdown flow would be released. A maximum letdown flow of 120 gpm, plus ten percent for uncertainty, is assumed. The noble gas activity in the primary coolant is based on operation with one percent fuel defects (see Table 9.2-4) and the iodine concentration is assumed to be 60 μCi/gm. The iodine

concentration is assumed to be reduced by a factor of ten by the demineralizer in the letdown line and ten percent of the remaining iodine activity is assumed to be released to the atmosphere. The letdown line is assumed to be isolated after 30 minutes.

The liquid holdup tanks receive reactor coolant, after passing through demineralizers, during the process of coolant purification. The contents of one tank are passed through the liquid processing train while another tank is being filled. In analyzing the consequence of rupture of a holdup tank, it is assumed that a single tank is filled to 80% of capacity using the letdown flow of 120 gpm (maximum purification flow) and the primary coolant noble gas concentrations are those for operation with one-percent fuel defects. The iodine concentration in the flow to the holdup tank is assumed to be 0.1 µCi/gm of dose-equivalent I-133 (this is ten-percent of the primary coolant equilibrium activity limit and this reduction is due to the 90% removal assumed to take place in the letdown line mixed-bed demineralizer). A major tank failure would be required to cause a release of all the contained noble gas. Since the tanks operate at low pressure, approximately 2 psig, a gas phase leak would result in an expulsion of approximately 12-percent of the contained gases and then the pressure would be in equilibrium with atmosphere. It is conservatively assumed that all of the contained noble gas activity and one-percent of the iodine activity are released. The tank pits are vented to the ventilation system so that any gaseous leakage would be discharged to the atmosphere by this route. Any liquid leaks from the tanks or piping will be collected in the tank sump pit to be pumped back into the liquid waste system.

The waste gas decay tanks receive the radioactive gases from the radioactive liquids from the various laboratories and drains processed by the waste disposal system. The maximum storage of waste gases occurs after a refueling shutdown, at which time the gas decay tanks store the radioactive gases stripped from the reactor coolant. A radiation monitor counts activity in a gas decay tank sample to the gas analyzer and an alarm is actuated if the activity approaches an administrative limit of 6000 Ci of dose-equivalent Xe-133. There is also an operating limit of 29,761 Ci of dose-equivalent Xe-133 in any tank. As discussed in Section 11.1, six shut-down gas decay tanks are provided in addition to the four gas decay tanks used during power operation to reduce the gaseous activity release as a result of an assumed rupture of one of the tanks during the decay period following a refueling shutdown.

The doses calculated for the tank failures are:

	Site Boundary WB Dose (rem)	Low Population WB Zone Dose (rem)	Control Room WB (rem)
Volume Control Tank	0.30	0.14	0.05
Gas Decay Tank	0.14	0.07	0.05
Holdup Tank	0.4	0.19	0.06

These doses are all less than 0.5 rem, whole body (RG 1.26).

[Deleted]

14.2.4 Steam-Generator Tube Rupture

Accident Description

The event examined is the complete severance of a single steam generator tube. The accident is assumed to take place during full power operation with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator power operated relief valves (and safety valves if their setpoint is reached).

The activity that is available for release from the secondary system is limited by:

1. Activities in the steam generator secondary that are a consequence of operational leakage prior to the complete tube rupture.
2. The activity concentration in the reactor coolant.
3. Operator actions to isolate the mixed primary and secondary leakage to atmosphere.

The steam generator tube material is highly ductile and it is considered that the assumption of a complete severance is conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continuous surveillance and an accumulation of minor leaks that cause the activity to exceed the limits in the Technical Specifications is not permitted during reactor operation.

For small leaks which will not result in a safety injection signal or containment isolation signal, the air ejector radiation monitor will alarm in the presence of activity in the air ejector discharge line. The air ejector discharge is automatically diverted back to the containment. The steam-generator liquid monitor will then alarm after a delay of about 2 minutes and the steam-generator blowdown/sampling lines will be isolated automatically. The main steamline nitrogen-16 (N-16) monitors and other main steamline monitors will also detect the presence of activity in the secondary system. See Sections 11.2.3.2.4, 11.2.3.2.8, 11.2.3.2.13, 11.2.3.2.19 and 11.2.3.4.3 for further information on the secondary system monitors provided.

The operator is expected to determine that a steam generator tube rupture (SGTR) has occurred, to identify and isolate the ruptured steam generator, and to complete the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. These actions should be performed on a restricted time scale in order to minimize the contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured steam generator. Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the recovery procedure can be carried out on a time scale that ensures that break flow to the ruptured steam generator is terminated before the water level in the affected steam generator rises into the main steam line. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Assuming normal operation of the various plant systems, the following sequence of events is initiated by a tube rupture:

1. Pressurizer low pressure and low level alarms are actuated, and prior to reactor trip, charging pump flow is increased in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip, and feedwater flow to the affected steam generator is reduced due to the additional break flow which is being supplied to that steam generator.

2. The main steamline N-16 monitor and air ejector radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system.

3. Decrease in reactor coolant system pressure due to a continued loss of reactor coolant inventory leads to a reactor trip signal on low pressurizer pressure or overtemperature ΔT. Resultant plant cool-down following reactor trip leads to a rapid decrease in pressurizer level and a safety injection signal, initiated by low pressurizer pressure, follows soon after reactor trip. The safety injection signal automatically terminates normal feedwater and initiates auxiliary feedwater.

4. The unit trip will automatically shut off steam flow through the turbine and will open steam bypass valves and bypass steam to the condenser if offsite power is available. In the event of a coincident loss of offsite power the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase, resulting in steam discharge to the atmosphere through the steam generator power operated relief valves (and the steam generator safety valves if their setpoint is reached).

5. Following reactor trip and safety injection actuation, the continued action of auxiliary feedwater supply and borated water injection flow provide a heat sink. Thus, steam bypass to the condenser, or in the case of a loss of offsite power, steam relief to atmosphere, is attenuated during the time in which the recovery procedure leading to isolation is being carried out.

6. Safety injection flow results in stabilization of the reactor coolant system (RCS) pressure and pressurizer water level and, if not for the operator's recovery actions, the RCS pressure trends towards the equilibrium value where the safety injection flow rate equals the break flow rate.

Recovery

In the event of an SGTR, the plant operators must diagnose the SGTR and perform the required recovery actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the Emergency Operating Procedures (EOPs). The EOPs are based on guidance in the Westinghouse Owner's Group Emergency Response Guidelines which address the recovery from a SGTR with and without offsite power available. The major operator actions include: identification of the ruptured steam generator, isolation of the ruptured steam generator, cooldown of the reactor coolant system using the intact steam generators to ensure subcooling at the ruptured steam generator pressure, controlled depressurization of the reactor coolant system to the ruptured steam generator pressure, and subsequent termination of safety injection flow to stop primary to secondary leakage.

These operator actions are described below.

1. Identify the ruptured steam generator.

   High secondary side activity, as indicated by the secondary side radiation monitors will typically provide the first indication of an SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level or high activity in any steam generator sample. For an SGTR that results in a reactor trip at high power, the steam generator water level as indicated on the narrow range will decrease significantly for all of the steam generators. The auxiliary feedwater flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary to secondary leakage adds additional inventory to the ruptured steam generator, the water level will increase more rapidly in that steam generator. This response, as displayed by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

   Once a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of filling the ruptured steam generator by (1) minimizing the accumulation of feedwater flow and (2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage.

3. Cooldown the Reactor Coolant System (RCS) using the intact steam generators.

   After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generators. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the power-operated atmospheric relief valves on the intact steam generators.

4. Depressurize the RCS to restore reactor coolant inventory.

   When the cooldown is completed, SI flow will tend to increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped.

   The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. However, if offsite power is lost or the RCPs are not running, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using a pressurizer power operated relief valve (PORV) or auxiliary spray.

5. Terminate SI to stop primary to secondary leakage.

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that the SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after the SI flow is stopped until the RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent re-pressurization of the RCS and re-initiation of leakage into the ruptured steam generator.

Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are performed to cooldown and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

Results

The analysis supports a Tavg window ranging from 549.0°F to 572.0°F. The analysis also supports a steam generator tube plug ranging from 0% to 10%. In estimating the mass transfer from the reactor coolant system through the broken tube, the following assumptions were made:

a. Plant trip occurs automatically as a result of low pressurizer pressure.

b. Following the safety injection signal, three high head safety injection pumps deliver flow for 30 minutes.

c. The ruptured steam generator pressure is maintained at the lowest steam generator safety valve reseat pressure of 885.4 psia (including 18% blowdown, which covers the 3% setpoint tolerance).

d. After reactor trip if the operators take no action to respond to the event the break flow will tend to equilibrate to the point where incoming safety injection flow is balanced by outgoing break flow as shown in Figure 14.2-0. In the accident analysis, this equilibrium break flow is assumed to persist from plant trip until 30 minutes after the accident initiation. The analysis does not require that the operators demonstrate the ability to terminate break flow within 30 minutes from the start of the event. It is recognized that the operators may not be able to terminate break flow within 30 minutes for all postulated SGTR events. The purpose of the calculation is to provide conservatively high mass-transfer rates for use in the radiological consequences analysis. This is achieved by assuming a constant break flow at the equilibrium flow rate for a relatively long time period. 30 minutes was selected for this purpose.

Sufficient indications and controls are provided at the control board to enable the operator to complete these functions satisfactorily within 60 minutes for the design-basis event even without offsite power. In order to demonstrate that releases calculated with the 30 minute equilibrium break flow assumption are indeed conservative, an evaluation was performed with a licensed thermal-hydraulic analysis code modeling the operator's response to the event. This evaluation modeled the operator's identification and isolation of the ruptured steam generator, cooldown of the RCS by dumping steam from the intact steam generators, depressurization of the RCS using the pressurizer PORV and subsequent termination of SI. This evaluation demonstrated

that although break flow was terminated at 60 minutes, the mass transfer data calculated with the assumption of a constant break flow at the equilibrium value for 30 minutes from reactor trip is limiting as input to the radiological consequences analysis.

In addition to the above assumptions, it is conservatively assumed that all stored energy and decay heat is removed by steaming until 30 hours from the start of the event at which point heat removal would be provided by the Residual Heat Removal System. These assumptions lead to the determination of the following:

Time of reactor trip	290 sec
Steam releases prior to reactor trip	1075.6 lb/sec per SG
Steam releases from ruptured SG after reactor trip (trip – 30 minutes)	77,300 lb
Steam releases from intact SGs after reactor trip	
Trip – 2 hours	542,000 lb
2 – 8 hours	1,090,000 lb
8 – 30 hours	1,760,000 lb
Ruptured SG break flow	
0 – trip	29,000 lb
trip – 30 minutes	99,000 lb
Break flow flashing fraction	
0 – trip	0.21
trip – 30 minutes	0.13

Radiological Consequences

The radiological consequences analysis considers both a pre-accident iodine spike and an accident initiated iodine spike.

In the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the SGTR and has raised the RCS iodine concentration to 60 μCi/gm of Dose Equivalent (DE) I-131 (60 times the assumed maximum coolant equilibrium concentration limit of 1.0 μCi/gm of Dose Equivalent I-131).

For the accident-initiated iodine spike case, the reactor trip associated with the SGTR creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 335 times greater than the release rate corresponding to the assumed maximum equilibrium RCS concentration of 1.0 μCi/gm of Dose Equivalent I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-cladding gap. Based on having 8-percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted within 4 hours and the analysis assumed that the spike is terminated at that time.

The noble gas activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level (see Table 9.2-4). The iodine activity concentration of the secondary coolant at the time the SGTR occurs is assumed to be equivalent to the Technical Specification limit of 0.15 μCi/gm of Dose Equivalent I-131.

The amount of primary to secondary steam generator tube leakage in the intact steam generators is assumed to be equal to 150 gpd per steam generator.

An iodine partition factor in the steam generators of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. Prior to reactor trip and concurrent loss of offsite power an iodine removal factor of 0.01 is taken for steam released to the condenser. All iodine contained in the fraction of the break flow that flashes to steam upon entering the secondary side of the steam generator is assumed to be immediately released to the outside atmosphere.

All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

At 30 hours after the accident, the Residual Heat Removal System is assumed to be capable of all decay heat removal and that there are thus no further steam releases to atmosphere from the secondary system.

The resultant site boundary doses are 3.3 rem TEDE for the pre-accident iodine spike and 1.2 rem TEDE for the accident-initiated iodine spike. The corresponding low population zone doses are 1.6 rem TEDE and 0.6 rem TEDE. These doses are calculated using the meteorological dispersion factors discussed in Section 14.3.6.2.1.

The offsite doses resulting from the accident with the assumed pre-accident iodine spike case are below than the limit values of 10 CFR 50.67 (25 rem TEDE) which is the dose acceptance limit identified in Regulatory Guide 1.183. The offsite doses resulting from the accident with the assumed accident-initiated iodine spike are less than 10-percent of the limit values of 10 CFR 50.67 (less than 2.5 rem TEDE) which is the dose acceptance limit identified in Regulatory Guide 1.183..

The accumulated doses to control room operators following the postulated accident were calculated using the same release, removal and leakage assumptions as the offsite doses, using the control room model discussed in Section 14.3.6.5 and Tables 14.3-50 and 14.3-51. The calculated control room doses are presented in Table 14.3-52 and are less than the 5.0 rem TEDE control room dose limit values of 10 CFR 50.67.

14.2.5 Rupture of a Steam Pipe

14.2.5.1 Description

A rupture of a steam pipe is assumed to include any accident that results in an uncontrolled steam release from a steam generator. The release can occur as a result of a break in a pipe line or a valve malfunction. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The removal of energy from the reactor coolant system causes a reduction of coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive control rod is assumed to be stuck in its fully withdrawn position, there is a possibility that the core may become critical and return to power even with the remaining control rods inserted. A return to power following a steam pipe rupture is a potential problem only because of the high hot-channel factors that may exist when the most reactive rod is assumed stuck in its fully withdrawn position. Even if the most pessimistic combination of circumstances that could lead to power generation following a steam line break was assumed, the core is ultimately shut down by the boric acid in the safety injection system.

The analysis of a steam pipe rupture was made to show that assuming the most reactive RCCA stuck in its fully withdrawn position and assuming the worst single failure in the engineered

safety features (ESFs), the core cooling capability is maintained and that offsite doses do not exceed applicable limits. In addition, the analysis considers conditions both with and without offsite power available.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis shows that DNB does not occur thus assuring clad integrity.

The following systems provide the necessary protection against a steam pipe rupture:

1. Safety injection system actuation from any one of the following:
   [***Note*** *- The details of the logic used to actuate safety injection are discussed in Section 7.2.*]

   a. Two-out-of-three channels of low pressurizer pressure signals.

   b. Two-out-of-three high differential pressure signals between steam lines.

   c. High steam flow in two-out-of-four lines (one-out-of-two per line) in coincidence with either low reactor coolant system average temperature (two-out-of-four) or low steam line pressure (two-out-of-four).

   d. Two-out-of-three high containment pressure signals.

   e. Manual actuation

2. The overpower reactor trips (nuclear flux and ΔT) and the reactor trip occurring upon actuation of the safety injection system.

3. Redundant isolation of the main feedwater lines. Sustained high feedwater flow would cause additional cooldown; however, in addition to the normal control action that will close the main feedwater valves, any safety injection signal will rapidly close all feedwater control valves and close the feedwater pump discharge valves, which in turn would trip the main feedwater pumps.

4. Closing of the fast-acting steam line stop valves (designed to close in less than 5 sec) on:

   a. High steam flow in any two steam lines (one-out-of-two per line) in coincidence with either low reactor coolant system average temperature (two-out-of-four) or low steam line pressure (two-out-of-four)
   b. Two sets of two-out-of-three high-high containment pressure signals.

Each main steam line has a fast-closing stop valve and a check valve. These eight valves prevent blowdown of more than one steam generator for any main steam line break location even if one valve fails to close. For example, for a main steam line break upstream of the stop valve in one line, a closure of either the check valve in that line or the stop valves in the other lines will prevent blowdown of the other steam generators.

Steam flow is measured by monitoring dynamic head in nozzles inside the steam pipes. The nozzles (16-in. ID versus a pipe diameter of 28-in. OD) are located inside the containment near

the steam generators and also serve to limit the maximum steam flow for any break further downstream. In particular, the nozzles limit the flow for all breaks outside the containment and those inside the containment which are downstream of the flow-measuring nozzles. A schematic showing the location of the stop valves, check valves, and nozzles is shown in Figure 14.2-1. In addition a flow limiting device (integral flow restrictor) consisting of seven (7) low pressure drop venturis is located in the steam outlet nozzle of each steam generator. This limits the flow of the postulated steam line break at the outlet nozzle to 1.4ft^2 (flow restrictor area).

14.2.5.2 Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

1. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. These conditions were determined using the RETRAN code.[4]

2. The thermal-hydraulic behavior of the core following a steam line break. A detailed thermal-hydraulic computer code, VIPRE[16] was used to determine if DNB occurs for the core conditions computed in the item 1 above.

The following conditions were assumed to exist at the time of a main steam line break accident:

1. The control rods give 1.3% shutdown reactivity margin at end-of-life (EOL), no-load conditions with equilibrium xenon. This is the EOL design value including design margins with the most reactive stuck rod in its fully withdrawn position. The actual shutdown capability is expected to be significantly greater.

2. The moderator reactivity coefficient corresponding to the EOL rodded core with the most reactive rod in its fully withdrawn position. The variation of the coefficient with temperature and pressure is included.

3. Minimum capability of the safety injection system, corresponding to two-out-of-three safety injection pumps in operation and degraded system performance.

4. Power peaking factors corresponding to one stuck RCCA and non-uniform core inlet temperatures are determined at EOL. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck RCCA during the return to power phase following the steam line break.

5. The Moody curve for L/D = 0 reported in Figure 3 of Reference 6 was used to calculate the steam flow through a steam line break.

6. The determination of the critical heat flux is based on local coolant conditions.

Two separate steam line rupture cases initiated from EOL, hot standby conditions were analyzed to determine the resulting core power and reactor coolant system transient conditions. These cases are:

Case A - Steam pipe rupture of a 1.4ft^2 break size (integral flow restrictor area) in faulted main steam line with offsite power available.

Case B - Steam pipe rupture 1.4ft^2 break size (integral flow restrictor area) in faulted main steam line with a loss of offsite power.

For the case with offsite power, it is assumed that within 12 seconds following receipt of an safety injection signal (including appropriate delays for the instrumentation, logic, and signal transport), the appropriate realignment of valves and actuations have been completed and that the high head safety injection pump is at full speed.

In the case where offsite power is not available, an additional 7 seconds delay is assumed to start the diesels and to load the necessary SI equipment on line.

14.2.5.3 Results

The results presented are a conservative indication of the events that would occur assuming a steam line rupture. The worst case assumes that the following occur simultaneously.

1. Minimum shutdown margin equal to 1.3% delta-K
2. The most negative moderator temperature coefficient for the rodded core at end of life.
3. The most reactive RCCA stuck in its fully withdrawn position.
4. One safety injection pump fails to function as designed.

The Time Sequence of the Events for both cases analyzed is reported in Table 14.2-6

14.2.5.4 Core Power and Reactor Coolant System Transients

Case A - Steam pipe rupture of a 1.4ft^2 break size (integral flow restrictor area) in faulted main steam line with offsite power available.

Figure 14.2-2 shows the reactor coolant system transient and core heat flux following a steam pipe rupture (complete severance of a pipe) at the initial no-load conditions. Should the core be critical at near zero power when the rupture occurs, a reactor trip signal from the safety injection signal initiated on high differential pressure between steam lines or by high steam flow signals in coincidence with either low reactor coolant system temperature or low steam line pressure would trip the reactor.

The break assumed is the largest break that could occur, i.e., assuming a double-ended rupture of the steamline, limited to 1.4ft^2 at the SG nozzle restrictor. Offsite power is assumed available such that full reactor coolant flow is maintained. Steam release out the break from the three intact steam generators would be prevented by the reverse flow check valve in the faulted loop or by the automatic closing of the fast-acting stop valves in the steam lines on a high steam flow signal in coincidence with low reactor coolant system temperature or low steam line pressure. Even with the failure of one valve, release from the three intact steam generators while the fourth steam generator blows down would be limited to the time required to obtain an isolation signal and to actuate steam line isolation via the fast-acting stop valves. The steam line stop valves are designed to be fully closed in less than 5 seconds with no flow through them. With the high flow that exists during a steam line rupture, the valves would close considerably faster.

For this case, a high steam flow condition in all four loops occurs almost immediately. A low-low average loop temperature condition (i.e., less than 537 degree F) is reached in 2 of 4 loops at 12.9 seconds. Seven seconds later, at 19.99 seconds, signals to initiate SI, steam line isolation,

and feedwater isolation are actuated. At 25.00 seconds, isolation of the 3 intact steam generators by closure of the main steam line isolation valves is completed. At 33.00 seconds, isolation of the main feedwater system is completed. The safety injection pumps which were started on the SI signal begin to deliver borated flow into the reactor core, after primary system pressure decreases below the SI pump head and the safety injection system lines are purged of unborated water.

As shown in Figure 14.2-2 the core becomes critical at 20.0 seconds. The peak core average heat flux of 15.5% of the nominal core power value (3216.0 MWt) is reached at 126.5 seconds.

Case B - Steam pipe rupture of a 1.4ft^2 break size (integral flow restrictor area) in faulted main steam line with a loss of offsite power.

For the case assuming a break with a loss of offsite power at time zero which results in a subsequent reactor coolant system flow coastdown, a high steam flow condition in all four loops occurs almost immediately. A low-low average loop temperature condition (i.e., less than 537°F) is reached in 2 of 4 loops at 14.65 seconds. Seven seconds later, at 21.65 seconds, signals to initiate SI, steam line isolation, and feedwater isolation are actuated. At 26.67 seconds, isolation of 3 intact steam generators is completed. At 34.67 seconds, isolation of the main feedwater system is completed. Following the appropriate safety injection system delay time required to start the safety injection pumps on the diesels, the safety injection pumps begin to deliver borated flow into the reactor core.

The peak core average heat flux of 9.20% of the nominal core power is reached at approximately 72 seconds.

14.2.5.5 Margin to Critical Heat Flux

Using the transients of Cases A and B, DNB analyses were performed for each steam line break cases. It was found that both cases have a minimum DNBR greater than the applicable safety analysis limit value.

14.2.5.6 Containment Peak Pressure for a Postulated Steam Line Break

The impact of steam line break mass and energy releases on containment pressure was addressed to assure the containment pressure remains below the design pressure of 47 psig. The LOFTRAN computer code was used to generate the mass and energy release to the containment for a large double-ended rupture at the discharge nozzle of the Model 44F replacement steam generator. A single failure of either the main or bypass feedwater control valve was assumed to occur concurrent with the break, which results in additional mass and energy release to containment from the feedwater system. The limiting case for the mass and energy releases is that which assumes offsite power is available at the hot full power conditions. The feedwater addition assumes a 2 second electronic delay, and a 5 second delay on tripping the main boiler feedwater pumps. The pumps are then assumed to coastdown in 10 seconds and the main boiler feedwater pump discharge valves (BFD-2) are assumed to close in 60 seconds. For the failure of the main feedwater control valve analysis, credit was taken for the main feedwater stop valves (BFD-5), with a closure time of 120 seconds. For the failure of the bypass feedwater control valve analysis, no credit was taken for the bypass feedwater stop valve (BFD-90).

An operator action to terminate auxiliary feed-water flow to the faulted steam generator was assumed at 15 minutes following receipt of the SI signal, which is more conservative than the 10 minute assumption previously used.

The COCO Code[17] was used to generate the containment response. The containment model was identical to that used for the Long Term LOCA Containment Integrity Analysis. The following assumptions were made:

-time dependent mass and energy release rates from LOFTRAN[5],
-an initial ambient pressure in the containment of 2 psig,
-an initial ambient temperature in the containment of 130°F,
-maximum safeguards of 5 fan coolers and two spray pumps.
-fan cooler initiation on SI signal with 60 second delay,
-containment spray initiation at 30 psig, with 60 second delay, and
-a containment spray temperature of 110°F.

The peak containment pressure was calculated to be 39.5 psig for failure of the main feedwater control valve, and 37.4 psig for failure of the bypass feedwater control valve. The calculated pressure time history is shown in Figure 14.2-7.

14.2.5.7 Dose Considerations

Assuming that a steam line break occurs when the steam generator is operating with a leak, the portion of reactor coolant activity discharged through the leak will be released to the steam generator. For the case in which the break is outside the containment and the leak occurs in the steam generator with the ruptured steam line, this activity is released to the atmosphere. In addition, the activity initially present in the steam generator will be released. Following the accident, the reactor coolant system would be cooled down and depressurized. The analysis assumes that the residual heat removal loop would be put into operation within 30 hours after the accident, and that there are no further steam releases to the atmosphere from the intact steam generators. Activity releases due to leakage of primary coolant to the faulted steam generator are assumed to continue until the primary coolant temperature is reduced to less than 212°F at 65 hours. At this point there would be no further release to the atmosphere. The radiological consequences analysis considers both a pre-accident iodine spike and an accident initiated iodine spike.

In the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the event and has raised the RCS iodine concentration to 60 µCi/gm of Dose Equivalent (DE) I-131 (60 times the assumed maximum coolant equilibrium concentration limit of 1.0 µCi/gm of Dose Equivalent I-131).

For the accident-initiated iodine spike case, the depressurization and reactor trip associated with the event creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the assumed maximum equilibrium RCS concentration of 1.0 µCi/gm of Dose Equivalent I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-cladding gap. Based on having 8-percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted within 3 hours and the analysis assumed that the spike is terminated at that time.

The noble gas activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level (see Table 9.2-4). The iodine activity concentration of the secondary coolant at the time the steam line break occurs is assumed to be equivalent to the Technical Specification limit of 0.15 μCi/gm of Dose Equivalent I-131.

The amount of primary to secondary steam generator tube leakage in the steam generators is 150 gpd per steam generator.

The steam generator connected to the broken steam line is assumed to boil dry within the initial five minutes following the steamline break. The entire liquid inventory of this steam generator is assumed to be steamed off and all of the iodine initially in the steam generator is assumed to be released to the environment. Also, the iodine carried over to the faulted steam generator by tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the steam generator.

For the intact steam generators an iodine partition factor in the steam generators of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. The concentration of iodine in the intact steam generators thus increases over the duration of the accident.

Prior to reactor trip and concurrent loss of offsite power an iodine removal factor of 0.01 could be taken for steam released to the condenser, but conservatively, the pre-trip condenser iodine removal is ignored.

All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

The resultant site boundary dose is 0.12 rem TEDE for both the pre-accident iodine spike case and the accident-initiated iodine spike case. The corresponding low population zone doses are 0.13 rem TEDE for the pre-accident spike and 0.33 rem TEDE for the accident-initiated spike. These doses are calculated using the meteorological dispersion factors discussed in Section 14.3.6.2.1.

The offsite doses resulting from the accident with the assumed pre-accident iodine spike case are below the limit values of 10 CFR 50.67 (25 rem TEDE) which is the dose acceptance limit identified in Regulatory Guide 1.183. The offsite doses resulting from the accident with the assumed accident-initiated iodine spike are less than 10-percent of the limit values of 10 CFR 50.67 (less than 2.5 rem TEDE) which is the dose acceptance limit identified in Regulatory Guide 1.183.

The accumulated doses to control room operators following the postulated accidents were calculated using the same release, removal and leakage assumptions as the offsite doses, using the control room model discussed in Section 14.3.6.5 and Tables 14.3-50 and 14.3-51. The calculated control room doses are presented in Table 14.3-52 and are less than the 5.0 rem TEDE control room dose limit values of 10 CFR 50.67.

14.2.6 Rupture of A Control Rod Mechanism Housing - Rod Cluster Control Assembly Ejection

14.2.6.1 Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Certain features are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design that lessens the potential ejection worth of rod cluster control assemblies and minimizes the number of assemblies inserted at high power levels.

14.2.6.2 Mechanical Design

Mechanical design and quality control procedures intended to preclude the possibility of a rod cluster control assembly drive mechanism housing failure are listed below:

1. Each control rod drive mechanism housing was completely assembled and shop tested at 4100 psi.

2. Each mechanism housing was individually hydro tested to 3105 psig as it was installed on the reactor vessel head adapters and checked during the hydro test of the completed reactor coolant system.

3. Stress levels in the mechanism are not affected by system transients at power, or by thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME code, Section III, for Class A components.

4. The latch mechanism housing and rod travel housing are each a single length of forged type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered. The joint between latch mechanism and head adapter is a threaded joint, reinforced using a canopy-type seal weld. The joint between the latch mechanism and rod travel housings is a Conoseal mechanical joint.

14.2.6.3 Nuclear Design

Even if a rupture of a rod cluster control assembly (RCCA) drive mechanism housing is postulated, the operation of a plant using chemical shim is such that the severity of an ejected RCCA is inherently limited. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of a RCCA ejection accident. Therefore, should a RCCA be ejected from its normal position during full-power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCA's above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCA's is continuously indicated in the control room. Alarms will occur if a bank of RCCA's approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration when a valid "APPROACHING ROD INSERTION LIMIT" alarm is received and emergency boration when a valid "ROD INSERTION LIMIT" alarm is received.

14.2.6.4 Reactor Protection

The protection for this accident is provided by high neutron flux trip (high and low setting).

14.2.6.5 Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a rod cluster control assembly mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings. However, even if damage is postulated, it would not be expected to lead to a more severe transient since rod cluster control assemblies are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that rod cluster control assembly not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

14.2.6.6 Limiting Criteria

This event is classified as an ANS Condition IV incident. Due to the extremely low probability of a rod cluster control assembly ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold or significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project by the Idaho Nuclear Corporation.[7] Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT results,[8] which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10-percent with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise) even for irradiated rods, did not occur below 300 cal/gm. In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are as follows:

1. Average fuel pellet enthalpy at the hot spot below 200 cal/gm.

2. Average clad temperature at the hot spot below 3000^{o}F and a Zirconium water reaction at the hot spot below 16%[23].

3. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

4. Fuel melting will be limited to less than 10-percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion (1) above.

Criteria 2 is a Westinghouse internal criterion established to address clad melting and embrittlement. However Criterion 1 was identified (Reference 23) as the limit which ensures that core cool ability is maintained.

14.2.6.7 <u>Method of Analysis</u>

The calculation of the rod cluster control assembly ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in Reference 9.

14.2.6.7.1 Average Core Analysis

The spatial kinetics computer code, TWINKLE,[10] is used for the average core transient analysis. This code solves the 2 group neutron diffusion theory kinetic equation in 1, 2, or 3 spatial dimensions (rectangular coordinates) for 6 delayed neutron groups and up to 8000 spatial points. The computer code includes a detailed multi-region, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feed-back effects. In this analysis, the code is used as one-dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and rod cluster control assembly movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described in the following) for calculating the ejected rod worth and hot-channel factor.

14.2.6.7.2 Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot-channel factor. During the transient, the heat flux hot-channel factor is linearly increased to the transient value in 0.1 sec, the time for full ejection of the rod. Therefore, the assumption is made that the hot spots before and after ejection are coincident. This is very conservative since the peak hot spot after ejection will occur in or adjacent to the assembly with the ejected rod, and before ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and cladding transient heat transfer computer code FACTRAN.[11] This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The

zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative pellet radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation[12] to determine the film boiling coefficient after DNB. The Bishop-Sandberg-Tong correlation is conservatively used assuming zero bulk fluid quality. The DNB ratio is not calculated; instead, the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full-power steady-state temperature distribution to agree with the fuel heat transfer design codes.

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 14.2-7 presents the parameters used in this analysis.

14.2.6.7.3 Ejected Rod Worths and Hot-Channel Factors

The values for ejected rod worths and hot-channel factors are calculated using either three-dimensional static methods or by a synthesis method employing one-dimensional and two-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Power distribution before and after ejection for a “worst case” can be found in Reference 9. During plant startup physics testing, rod worths and power distributions are measured in the zero- and full-power rodded configurations and compared to values used in the analysis. It has been found that the worth and power peaking factors are consistently over-predicted in the analysis.

14.2.6.7.4 Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks, occur in the channel where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which when applied to single-channel feedbacks correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one-dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension.

14.2.6.7.5 Moderator and Doppler Coefficient

The critical boron concentrations at the beginning-of-life and end-of-life are adjusted in the nuclear code in order to obtain moderator density coefficient curves, which are conservative

compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using a one-dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler weighting factor will increase under accident conditions, as discussed above.

14.2.6.7.6 Delayed Neutron Fraction, β

Calculation of the effective delayed neutron fraction (β_{eff}) yielded values no less than 0.500-percent at beginning-of-life and, 0.400-percent at end-of-life [Deleted].

14.2.6.7.7 Trip Reactivity Insertion

The trip reactivity insertion assumed includes the effect of one stuck rod cluster control assembly. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 sec after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 sec for the instrument channel to produce a signal, 0.15 sec for the trip breaker to open, and 0.15 sec for the coil to release the rods. A curve of trip rod insertion versus time was used, which assumed that insertion to the dashpot does not occur until 2.4 sec after the start of fall.

The minimum design shutdown available for this plant at hot zero power may be reached only at end-of-life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, adverse xenon distribution for calculational uncertainties. Physics calculations for this plant have shown that the effect of two stuck rod cluster control assemblies (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1-percent ΔK. Therefore, following a reactor trip resulting from a rod cluster control assembly ejection accident, the reactor will be subcritical when the core returns to hot zero power.

Depressurization calculations have been performed for a typical four-loop plant assuming the maximum possible size break (2.75-in. diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The safety injection system is actuated by the low pressurizer pressure trip within 1 min after the break. The reactor coolant pressure continues to drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about 2 to 3 min. Because of the large thermal inertia of the primary and secondary system, there has been no significant decrease in the reactor coolant system temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2-percent Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 min after the break. The addition of highly borated (2000-ppm) safety injection flow starting 1 min after the break is more than sufficient to ensure that the core remains subcritical during the cooldown.

As discussed previously, reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting). These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

14.2.6.8 Results

Cases are presented at zero and full power for both beginning-of-life and end-of-life.

1. Beginning-of-Life, Full Power - Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot-channel factor were conservatively calculated to be 0.17-percent Δk and 6.80, respectively. The maximum hot-spot clad average temperature was 2199°F. The maximum hot-spot fuel center temperature was 4958°F.

2. Beginning-of-Life, Zero Power - For this condition, control bank D was assumed to be fully inserted, and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.65-percent Δk and a hot-channel factor of 12.0. The maximum hot-spot clad average temperature reached 1881°F and the maximum fuel center temperature was 2812°F.

3. End-of-life, Full Power - Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot-channel factors were conservatively calculated to be 0.20-percent Δk and 7.10, respectively. This resulted in a maximum clad average temperature of 2132°F. The maximum hot-spot fuel center temperature reached 4861°F.

4. End-of-Life, Zero Power - The ejected rod worth and hot-channel factor for this case were obtained assuming control bank D to be fully inserted and banks C and B at their insertion limits. The results were 0.80-percent Δk and 20.00, respectively. The maximum clad average and fuel center temperatures were 2549°F and 3633°F, respectively.

A summary of the results for the cases presented above is given in Table 14.2-8. The nuclear power, fuel center, fuel average and clad temperature transients for all the cases are presented in Figures 14.2-11 through 14.2-18.
The calculated sequence of events for the rod ejection accident cases, are presented in Table 14.2-9. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously, the reactor will remain subcritical following a reactor trip.

The ejection of a rod cluster control assembly constitutes a break in the reactor coolant system boundary located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents are discussed in Section 14.3. Following the rod cluster control assembly ejection, the operator would follow the same emergency instructions as for any other loss-of-coolant accident to recover from the event.

14.2.6.9 Fission Product Release

As a result of the accident, fuel clad damage and a small amount of fuel melt are assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or the main steam safety valves. Iodine and alkali metals group activity is contained in the secondary coolant prior to the accident, and some of this activity is also released to the

atmosphere as a result of steaming the steam generators following the accident. Finally, radioactive reactor coolant is discharged to the containment via the spill from the opening in the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment.

As a result of the rod ejection accident, less than 10% of the fuel rods in the core undergo DNB. In determining the offsite doses following the rod ejection accident, it is conservatively assumed that 10% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released. Consistent with Regulatory Guide 1.183, a gap fraction of 10% is assumed for iodine and noble gas activity. Additionally, 12% of the alkali metal activity is assumed to be in the gap. The core activity is provided in Table 14.3-43 and it is assumed that the damaged fuel rods have all been operating at the maximum radial peaking factor of 1.70.

A small fraction of the fuel in the failed fuel rods is assumed to melt as a result of the rod ejection accident. This amounts to 0.25% of the core and the melting takes place in the centerline of the affected rods. Consistent with Regulatory Guide 1.183, for the containment leakage release pathway 25% of the iodine activity and 100% of the noble gas activity are assumed to enter the containment but for the secondary system release pathway 50% of the iodine activity and 100% of the noble gas activity are assumed. Additionally, for both pathways it is assumed that 100% of the alkali metal activity from the melted fuel is available for release.

The primary coolant iodine concentration is assumed to be at the equilibrium operating limit of 1.0 μCi/gm of dose equivalent 1-131 prior to the rod ejection accident. The alkali metals and noble gas activity concentrations in the RCS at the time the accident occurs are based on operation with a fuel defect level of one percent. The iodine activity concentration of the secondary coolant at the time the rod ejection accident occurs is assumed to be 0.15 μCi/gm of dose equivalent I-131.

Regulatory Guide 1.183 specifies that the iodine released from the fuel is 95% particulate (cesium iodide), 4.85% elemental, and 0.15% organic. These fractions are used for the containment leakage release pathway. However, for the steam generator steaming pathway the iodine in solution is considered to be all elemental and after it is released to the environment the iodine is modeled as 97% elemental and 3% organic.

Conservatively, all the iodine, alkali metals group and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment) when determining doses due to the primary to secondary steam generator tube leakage.

The primary to secondary steam generator tube leak used in-the analysis is 150 gpd per steam generator (total of 600 gpd).

When determining the doses due to containment leakage, all of the iodine, alkali metal and noble gas activity is assumed to be in the containment. The design basis containment leak rate of 0.1% per day is used for the initial 24 hours. Thereafter, the containment leak rate is assumed to be one-half the design value, or 0.05% per day. Releases are continued for 30 days from the start of the event.

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the

outside atmosphere. Secondary side releases are terminated when the primary pressure drops below the secondary side pressure.

An iodine partition factor in the steam generators of 0.01 curies/gm steam per curies/gm water is used. A partition factor of 0.0025 is used for the alkali metal activity in the steam generators.

For the containment leakage pathway, no credit is taken for sedimentation removal of aerosols. No credit is taken for elemental iodine deposition onto containment surfaces or for containment spray operation which would remove both airborne particulates and elemental iodine.

The resultant site boundary dose is 3.1 rem TEDE. The low population zone dose is 4.2 rem TEDE. These doses are calculated using the meteorological dispersion factors discussed in Section 14.3.6.2.1.

The offsite doses resulting from the accident are less than 25-percent of the limit values of 10 CFR 50.67 (less than 6.25 rem TEDE) which is the dose acceptance limit identified in Regulatory Guide 1.183.

The accumulated dose to control room operators following the postulated accident was calculated using the same release, removal and leakage assumptions as the offsite doses, using the control room model discussed in Section 14.3.6.5 and Tables 14.3-50 and 14.3-51. The calculated control room dose is presented in Table 14.3-52 and is less than the 5.0 rem TEDE control room dose limit values of 10 CFR 50.67.

14.2.6.10 Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning-of-life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits.[9] Since the severity of the present analysis does not exceed the worst case analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the reactor coolant system.

14.2.6.11 Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are undermoderated, and bowing will tend to increase the undermoderation at the hot spot. Since the 15 x 15 fuel design is also undermoderated, the same effect would be observed. In practice, no significant bowing is expected since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from the region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling sufficient to distort the lattice is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from

lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

14.2.6.12 Conclusions

Analyses indicate that the described fuel and cladding limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is not danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that the fission product release, as a result of a number of fuel rods entering departure from nucleate boiling, is limited to less than 10-percent of the fuel rods in the core. The radiological consequences of this event are within applicable limits.

REFERENCES FOR SECTION 14.2

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3. Deleted

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TABLE 14.2-1
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TABLE 14.2-2 (Sheet 1 of 2)
Fuel Handling Accident – Design Basis Case

Fuel Parameters		
Reactor power (including 2% uncertainty), MWt		3280.3
Number of assemblies		193
Fuel Rods per assembly		204
Normalized power, highest rated discharged assembly		1.70
Time from Reactor Shutdown to Accident, Hrs		84
Fission Product Release		
Fraction of Fuel Rod Activity in gap[1]		
I-131		0.12
Kr-85		0.30
Other iodines and noble gases		0.10
Decontamination factor for retention by pool water	Iodines	200
	Noble gases	1.00
Decontamination factor for Filters		Not Credited
Dispersion and Potential Exposure at Site Boundary		
Atmospheric dispersion factor (χ/Q, sec/m^3)		
Site Boundary		7.5×10^{-4}
Low Population Zone		3.5×10^{-4}
Receptor breathing rate (m^3/sec)		3.5×10^{-4}

TABLE 14.2-2 (Sheet 2 of 2)
Fuel Handling Accident – Design Basis Case

Nuclide	Shutdown Core Inventory after 84 hours, Curies (2)	CEDE Dose Conversion Factor, rem/Ci	EDE Dose Conversion Factor, rem-m^3/Ci-sec
I-130	3.44E+4	2.64E+3	3.848E-1
I-131	6.94E+7	3.29E+4	6.734E-2
I-132	6.39E+7	3.81E+2	4.144E-1
I-133	1.17E+7	5.85E+3	1.088E-1
I-134	0	1.31E+2	4.810E-1
I-135	2.62E+4	1.23E+3	2.953E-1
Kr-85M	0	-	2.768E-2
Kr-85	1.10E+6	-	4.403E-4
Kr-87	0	-	1.524E-1
Kr-88	0	-	3.774E-1
Xe-131M	9.85E+5	-	1.439E-3
Xe-133M	2.91E+6	-	5.069E-3
Xe-133	1.36E+8	-	5.772E-3
Xe-135M	4.20E+3	-	7.548E-2
Xe-135	7.83E+5	-	4.403E-2
Xe-138	0	-	2.135E-1

Notes:

1. The gap fractions are consistent with Regulatory Guide 1.25 except for I-131 for which the gap fraction was increased above the Regulatory Guide 1.25 value of 0.10 following the recommendations of NUREG/CR-5009. These values were selected for the analysis in place of the lower gap fraction values provided in Regulatory Guide 1.183 due to the expectation that the fuel would not meet the criteria of a peak rod average power of ≤6.3 kw/ft for some of the high burnup fuel rods.

2. Inventory of 0 means less than one Curie per fuel assembly.

TABLE 14.2-2A
[Deleted]

TABLE 14.2-3
[Deleted]

TABLE 14.2-4
[Deleted]

TABLE 14.2-5
Volume Control Tank Activity[1]

Nuclide	Volume Control Tank Inventory, Curies [1]
I-130	1.741E-2
I-131	5.916E-1
I-132	9.220E-1
I-133	1.468E+0
I-134	2.143E-1
I-135	6.884E-1
Kr-85M	1.521E+2
Kr-85	2.357E+3
Kr-87	4.640E+1
Kr-88	2.254E+2
Xe-131M	3.766E+2
Xe-133M	4.104E+2
Xe-133	2.991E+4
Xe-135M	7.292E+1
Xe-135	9.081E+2
Xe-138	6.303E+0

Notes:
1. Inventory is based on operation with one percent fuel defects. The reported activity reflects the combined vapor space and liquid space inventories.

TABLE 14.2-6

Time Sequence of Events for the Rupture of a Main Steamline		
Event	Case with Offsite Power Time (sec)	Case without Offsite Power Time (sec)
Double-Ended Steamline Rupture in Loop 1 ($1.4ft^2$)	0.00	0.00
High Steamline Flow Setpoint Reached (2/4 loops)	0.29	0.27
Loss of Offsite Power (RCPs begin coasting down)	--	2.99
High Steamline Flow Signal Generated (2/4 loops)	9.29	9.27
Low-Low T_{avg} Setpoint Reached in Loop 1	9.92	10.66
Low-Low T_{avg} Setpoint Reached in Loop 2	12.99	14.65
Low-Low T_{avg} Signal Generated in Loop 1	16.92	17.66
Low-Low T_{avg} Signal Generated in Loop 2	19.98*	21.64*
Safety Injection SLI and FWI Actuation due to Coincidence of Low-Low T_{avg} (2/4 loops) / High Steam Flow (2/4 loops) ESF	19.99	21.66
MSIV Closure Initiated in Loops 1, 2, 3, and 4	24.89(1)	26.56(1)*
MSIV Closure Completed in Loops 1, 2, 3, and 4	25.00	26.67
MFIV Closure Initiated in Loops 1, 2, 3, and 4	32.89(1)*	34.56(1)*
MFIV Closure Completed in Loops 1, 2, 3, and 4	33.00	34.67
Maximum Heat Flux Reached	126.49	71.99

Note:
*additional modeling delay (round off) not included
1. An additional 0.1 second allowance for valve closure time.

TABLE 14.2-7
Parameters Used in the Analysis of the Rod Cluster Control Assembly Ejection Accident

	BOL-HFP	BOL-HZP	EOL-HFP	EOL-HZP
Power level, percent	102	0	102	0
Ejected rod worth, percent Δk	0.17	0.65	0.20	0.80
Delayed neutron fraction, percent	0.50	0.50	0.40	0.40
Feedback reactivity weighting	1.46	2.16	1.50	2.96
Trip reactivity, percent Δk	4.0	2.0	4.0	2.0
F_Q after rod ejection	6.8	12.0	7.1	20.0
Number of operational pumps	4	2	4	2

Key:
BOL Beginning of Life
EOL End-of-Life
HFP Hot full Power
HZP Hot zero Power

TABLE 14.2-8
Results of the Analysis of the Rod Cluster Control
Assembly Ejection Accident

	BOL-HFP	BOL-HZP	EOL-HFP	EOL-HZP
Maximum fuel pellet average temperature, °F	3971	2472	3860	3300
Maximum fuel center temperature, °F	4958	2812	4861	3633
Maximum clad average temperature, °F	2199	1881	2132	2549
Maximum fuel stored energy, Btu/lb	311.0	178.0	300.7	249.6
Percent fuel melt	3.46	0	3.53	0

Key: BOL Beginning-of-Life
EOL End-of-Life
HFP Hot full Power
HZP Hot zero Power

TABLE 14.2-9
Time Sequence of Events for Rod Cluster Control Assembly Ejection

RCCA Ejection	Time of Event, sec			
Event	BOL-HFP	BOL-HZP	EOL-HFP	EOL-HZP
Initiation of rod ejection	0.0	0.0	0.0	0.0
Power range neutron flux set point reached (HFP, High / HZP, Low)	0.06	0.34	0.04	0.17
Peak nuclear power occurs	0.14	0.40	0.13	0.20
Rods begin to fall into core	0.56	0.84	0.54	0.67
Peak fuel average temperature occurs	2.17	2.45	2.24	1.67
Peak clad temperature occurs	2.29	2.36	2.35	1.51
Peak heat flux occurs	2.34	2.46	2.41	1.55

14.2 FIGURES

Figure No.	Title
Figure 14.2-0	Steam Generator Tube Rupture, Break Flow and Safety Injection Flow vs. Reactor Coolant System Pressure
Figure 14.2-1	Steam Line Valve Arrangement Schematic
Figure 14.2-2 Sh. 1	Steam Line Rupture Offsite Power Available, EOL, Core Heat Flux and Core Reactivity vs. Time
Figure 14.2-2 Sh. 2	Steam Line Rupture Offsite Power Available, EOL, Reactor Coolant Pressure and RV Inlet Temperature vs. Time
Figure 14.2-2 Sh. 3	Steam Line Rupture Offsite Power Available, EOL, Steam Flow and Steam Generator Pressure vs. Time
Figure 14.2-2 Sh. 4	Steam Line Rupture Offsite Power Available, EOL, Core Boron Concentration vs. Time
Figure 14.2-3 Sh. 1	Deleted
Figure 14.2-3 Sh. 2	Deleted
Figure 14.2-3 Sh. 3	Deleted
Figure 14.2-4 Sh. 1	Deleted
Figure 14.2-4 Sh. 2	Deleted
Figure 14.2-4 Sh. 3	Deleted
Figure 14.2-5 Sh. 1	Deleted
Figure 14.2-5 Sh. 2	Deleted
Figure 14.2-5 Sh. 3	Deleted
Figure 14.2-6 Sh. 1	Deleted
Figure 14.2-6 Sh. 2	Deleted
Figure 14.2-7	Containment Pressure Time History (Double - Ended Main Steam Line Break Main FCV Failure Maximum Containment Safeguards)
Figure 14.2-8 Through 14.2-10	Deleted
Figure 14.2-11	Rod Ejection Accident, BOL-HFP, Nuclear Power vs. Time
Figure 14.2-12	Rod Ejection Accident, BOL-HFP, Fuel Temperatures vs. Time
Figure 14.2-13	Rod Ejection Accident, BOL-HZP, Nuclear Power vs. Time
Figure 14.2-14	Rod Ejection Accident, BOL-HZP, Fuel Temperatures vs. Time
Figure 14.2-15	Rod Ejection Accident, EOL-HZP, Nuclear Power vs. Time
Figure 14.2-16	Rod Ejection Accident, EOL-HZP, Fuel Temperatures vs. Time
Figure 14.2-17	Rod Ejection Accident, EOL-HFP, Nuclear Power vs. Time
Figure 14.2-18	Rod Ejection Accident, EOL-HFP, Fuel Temperatures vs. Time
Figure 14.2-19 Thru Figure 14.2-22	Deleted

14.3 LOSS-OF-COOLANT ACCIDENTS

14.3.1 Identification of Causes And Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the reactor coolant system pressure boundary. A major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0-ft^2. This event is considered a limiting fault, an ANS Condition IV event, in that it is not expected to occur during the lifetime of the plant, but is postulated as a conservative design basis.

A minor pipe break (small break) is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area less than 1.0-ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered an ANS Condition III event in that it is an infrequent fault that may occur during the life of the plant.

The acceptance criteria for the loss-of-coolant accident are described in 10 CFR 50 Paragraph 46 (Reference 1), as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met by demonstrating that the PCT does not exceed 2200°F, the maximum local oxidation does not exceed 17%, and the seismic and LOCA forces are not sufficient to distort the fuel assemblies to the extent that the core cannot be cooled.

5. After any successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

These criteria were established to provide significant margin in emergency core cooling system performance following a LOCA. Reference 2 presents a study in regard to the probability of occurrence of RCS system pipe ruptures.

14.3.2 Sequence Of Events And Systems Operations

Should a major break occur, the depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safety injection actuation signal is

generated when the appropriate setpoint is reached. These countermeasures limit the consequences of the accident in the following two ways:

1. Reactor trip and borated water injection complement void formation in causing a rapid reduction of power to a residual level corresponding to fission product decay heat.
2. The injection of borated water provides for heat transfer from the core, prevents excessive clad temperatures, and maintains subcriticality.

14.3.2.1 Description of Large-Break LOCA Transient

The RCS is assumed to be operating normally at full power. A break is assumed to open nearly instantaneously in one of the main coolant pipes. Calculations where the location and size of the break have been varied indicate that a break in the cold leg between the reactor coolant pump and the reactor vessel leads to the most severe transient. For this break location, a rapid depressurization occurs, along with a core flow reversal as subcooled liquid flows out of the vessel into the broken cold leg. Boiling begins in the core, and the reactor core begins to shut down. With in approximately 2 seconds, the core is highly voided, and core fission is terminated. The cladding temperature rises rapidly as heat transfer to the fuel rods is reduced.

Within approximately 6 seconds, the pressure in the pressurizer has fallen to the point where the protection systems initiate safety injection. Along with the safety injection, containment isolation is also initiated.

In the first few seconds, the coolant in all regions of the vessels begins to flash. In addition, the break flow becomes saturated and is substantially reduced. This reduces the depressurization rate, and may also lead to a period of positive core flow as the RCS pumps in the intact loops continue to supply water to the vessel, and as flashing continues in the vessel lower plenum and downcomer. Cladding temperatures may be reduced, and some portions of the core may rewet during this period.

This positive core flow period ends as two-phase conditions occur in the pumps, reducing their effectiveness. Once again, the core flow reverses as most of the vessel mass flows out through the broken cold leg. Core cooling occurs as a result of the reverse flow.

At approximately 10 seconds after the break, the pressure falls to the point where the accumulators begin injecting cold water into the cold legs. Because the break flow is still high, much of the injected ECCS water, which flows from the cold legs into the downcomer of the vessel, is assumed to be carried out to the break.

Approximately 28 seconds after the break, most of the original RCS inventory has been ejected or boiled off. The system pressure and break flow are reduced and the ECCS water from the accumulator, which has been filling the downcomer, begins to fill the lower plenum of the vessel.

During this time, core heat transfer is relatively poor and cladding temperatures increase.

Approximately 40 seconds after the break, the lower plenum has re-filled. ECCS water from the refueling water storage tank (RWST) begins to flow into the vessel and enters the core. The flow into the core is oscillatory, as cold water rewets hot fuel cladding, generating steam. This steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generator, and the pump before it can be vented out the break. The resistance of this flow path

to the steam flow is balanced by the driving force of water filling the downcomer. Shortly after reflood begins, the accumulators exhaust their inventory of water, and begin to inject the nitrogen gas, which was used to pressurize the accumulators. This results in a short period of improved heat transfer as the nitrogen forces water from the downcomer into the core. When the accumulators have exhausted their supply of nitrogen the reflood rate may be reduced and peak cladding temperatures may again rise. This heatup may continue until the core has reflooded to several feet. Approximately 3 minutes after the break, all locations in the core begin to cool. The core is completely quenched within 5 minutes, and long term cooling and decay heat removal begin. Long term cooling for the next several minutes is characterized by continued boiling in the vessel as decay power and heat stored in the reactor structures is removed.

Continued operation of the emergency core cooling system pumps supplies water during long-term cooling. Core temperatures would be reduced to long-term steady-state levels associated with the dissipation of residual heat generation. After the water level of the refueling water storage tank reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold-leg recirculation mode of operation in which spilled borated water is drawn from the recirculation sump by the recirculation pumps and returned to the reactor coolant system cold legs. The containment spray pumps continue to operate drawing water from the refueling water storage tank for further reduction of containment pressure. Approximately 6.5 hours after initiation of the LOCA, the emergency core cooling system is realigned to supply water to the reactor coolant system hot legs in order to control the boric acid concentration in the reactor vessel.

The sequence of events for the large break LOCA is summarized in Table 14.3-1.

14.3.2.2 Description of Small-Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for the small break LOCA there are only three characteristic stages, i.e., a gradual blowdown in which the water level decreases, core recovery, and long-term recirculation.

For small break LOCAs, the most limiting single active failure is the one that results in the minimum ECCS flow delivered to the RCS. This has been determined to be the loss of an emergency power train, which results in the loss of one complete train of ECCS components. This means that credit can be taken for two out of three high head safety injection pumps, and one RHR (low head) pump. During the small break transient, two high head pumps are assumed to start and deliver flow into all four loops. The flow to the broken loop was conservatively assumed to spill to RCS in accordance with Reference 93 for a four-loop plant.

For the limiting break location analyzed (cold leg), the depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. Loss-Of-Offsite-Power (LOOP) is assumed to occur coincident with reactor trip. A safety injection signal is generated when the appropriate setpoint (pressurizer low pressure SI) is reached. After the safety injection signal is generated, an additional delay ensues. This delay accounts for the instrumentation delay, the diesel generator start time, plus the time necessary to align the appropriate valves and bring the pumps up to full speed. The safety features described will limit the consequences of the accident in two ways:

1) Reactor trip and borated water injection supplement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the small break LOCA analysis for the boron content of the injection water. In addition, in the small break LOCA analysis, credit is taken for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the RCCA at the most reactive location is stuck in the full out position, and

2) Injection of (borated) water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs, the plant is assumed to be in normal plant operation at 102% of hot full power, i.e., the heat generated in the core is being removed via the secondary system. During the earlier phase of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as the pumps coast down following LOOP. Upward flow through the core is maintained. However, depending on the break size, the core flow is not sufficient to prevent a partial core uncovery. Subsequently, the ECCS provides sufficient core flow to cover the core, adequately removing decay heat.

During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of heat transfer from the RCS to the secondary, heat addition to the secondary results in increased secondary system pressure which leads to steam relief via the safety valves. Makeup to the secondary is automatically provided by the auxiliary feedwater pumps. The safety injection signal isolates normal feedwater flow by closing the main feedwater control and bypass valves. Auxiliary feedwater flow is initiated by the reactor trip signal with coincident LOOP. In the Small Break LOCA analysis, flow from a single motor driven auxiliary feedwater pump is assumed to begin 60 seconds after the generation of a reactor trip signal coincident with LOOP. The secondary flow aids in the reduction of RCS pressure. Also, due to the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the blowdown analysis.

The cold leg accumulators will inject borated water into the reactor coolant loops if the RCS depressurizes to the nitrogen cover gas pressure.

14.3.3 Core And System Performance

14.3.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in 10 CFR 50.46 (Reference 1).

14.3.3.1.1 Large Break LOCA Evaluation Model

The evaluation model used to comply with the requirements of 10 CFR50.46 (Reference 1), Revisions to the Acceptance Criteria (Reference 3), and USNRC Regulatory Guide 1.157 (Reference 74), is described in this section. The analytical techniques used for the large break LOCA analysis are in compliance with 10 CFR 50.46 (Reference 1) as amended in Reference 3, and are described in References 69 and 75.

In 1988, the NRC staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models" to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best estimate codes is provided in Regulatory Guide 1.157 (Reference 74).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 77). This method outlined an approach for defining and qualifying a best estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A Westinghouse LOCA evaluation methodology for three- and four-loop Pressurized Water Reactor (PWR) plants based on the revised 10 CFR 50.46 rules was developed with the support of EPRI and Consolidated Edison. The methodology is documented in WCAP-12945-P-A, "Code Qualification Document (CQD) for Best-Estimate LOCA Analysis" (Reference 75). [Deleted]

More recently, Westinghouse developed an alternate methodology called ASTRUM (Reference 69). This method is still based on the CQD methodology and follows the steps in the CSAU methodology. However, the uncertainly analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the responses surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case.

The three 10 CFR 50.46 criteria (peak clad temperature, maximum local oxidation and core-wide oxidation) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size). In particular, the statistical theory predicts that 124 calculations are required to simultaneously bound the 95 percentile of three parameters with a 95-percent confidence level.

The thermal-hydraulic computer code, which was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a large break LOCA is WCOBRA/TRAC Version MOD7A, Revision 6 (Reference 69).

WCOBRA/TRAC combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ. Dividing the liquid phase into two fields is a convenient and physically accurate way of handling flows where the liquid can appear in both film and droplet form. The droplet field permits more accurate modeling of thermal-hydraulic phenomena such as entrainment, de-entrainment, fallback, liquid pooling, and flooding.

WCOBRA/TRAC also features a two-phase, one-dimensional hydrodynamics formulation. In this model, the effect of phase slip is modeled indirectly via a constitutive relationship, which provides the phase relative velocity as a function of fluid conditions. Separate mass and energy conservation equations exist for the two-phase mixture and for the vapor.

The reactor vessel is modeled with the three-dimensional, three-field model, while the loop, major loop components, and safety injection points are modeled with the one-dimensional model.

All geometries modeled using the three-dimensional model are represented as a matrix of cells. The number of mesh cells used depends on the degree of detail required to resolve the flow field, the phenomena being modeled, and practical restrictions such as computing costs and core storage limitations.

The basic building block for the mesh is the channel, a vertical stack of single mesh cells. Several channels can be connected together by gaps to model a region of the reactor vessel. Regions that occupy the same level form a section of the vessel. Vessel sections are connected axially to complete the vessel mesh by specifying channel connections between sections. Heat transfer surfaces and solid structures that interact significantly with the fluid can be modeled with rods and unheated conductors.

The noding diagram for Indian Point Unit 2 is shown in Figures 14.3-1 and 14.3-2. The vessel channel layout is shown in Figure 14.3-1. Figure 14.3-2 shows the one-dimensional component layout for the loops. Within the channels and components, additional subdivisions into cells are present, as described in Reference 75.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state, initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown refill, and reflood follows continuously, using the same computer code.

WCAP-16009-P-A (Reference 69) provides ASTRUM methodology and also includes a description of the code models and their implementation. Volumes II and III of the CQD (Reference 75) presented a detailed assessment of the computer code WCOBRA/TRAC through comparisons to experimental data. From this assessment, a quantitative estimate was obtained of the code's ability to predict peak clad temperatures (PCTs) in a PWR large-break loss-of-coolant accident (LOCA). Modeling of a PWR introduced additional uncertainties, which were identified and discussed in Section 21 of the CQD Volume IV (Reference 75). A list of key LOCA parameters was compiled as a result of these studies. Models of several PWRs were used to perform sensitivity studies and establish the relative importance of these parameters. The final step of the best-estimate methodology, in which all the important uncertainties of the LOCA parameters are accounted for to estimate a PCT, local maximum oxidation (LMO) and core-wide oxidation (CWO) at 95-percent probability, is described in the following sections. The methodology is summarized below:

Plant Model Development

In this step, a WCOBRA/TRAC model of the plant is developed. A high level of noding detail is used to insure an accurate simulation of the transient. Specific guidelines are followed to assure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, with some plant-specific modeling dictated by hardware differences such as in the upper plenum of the reactor vessel or the Emergency Core cooling System (ECCS) injection configuration.

Determination of Plant Operating Conditions

In this step, the expected or desired operating range of the plant to which the analysis is to be applied is established using information supplied by the utility. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the “initial transient." Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range. Because certain parameters are not included in the uncertainty analysis, these parameters are set at their bounding condition. This analysis is commonly referred to as the confirmatory analysis. Section 1.2.11 of Reference 79 describes the parameters of interest for the confirmatory analysis. The most limiting input conditions, based on these confirmatory runs, are then combined into the model that will represent the limiting state for the plant, which is the starting point for the assessment of uncertainty.

Assessment of Uncertainty

The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of WCAP-16009-P-A (Reference 69). The determination of the PCT uncertainty, LMO uncertainty, and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA/TRAC calculations are necessary to assess against the three 10 CFR 50.46 criteria (PCT, LMP, and CWO).

The uncertainty contributors are sampled randomly from their respective distribution for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each WCOBRA/TRAC calculation, include initial conditions, power distributions, and model uncertainties. The time in the cycle, break type (split or double-ended guillotine), and break size for the split break are also sampled as uncertainty conditions within the ASTRM methodology.

Results from the 124 calculations are tallied by ranking the PCT from highest to lowest. A similar procedure is repeated for LMO and CWO. The highest rank of PCT, LMO, and CWO will bound 95 percent of their respective populations with 95-percent confidence level.

Plant Operating Range

The plant operating range over which the uncertainty evaluation applies is shown in Table 14.3-5A. If operation in maintained within these ranges, the large break LOCA analysis developed in Reference 79 using [Deleted] WCOBRA/TRAC is valid.

14.3.3.1.2 Small Break LOCA Evaluation Model

For loss-of-coolant accidents due to small breaks less than 1 ft^2, the NOTRUMP (References 15, 16 and 93) computer code is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of flow through the break.

Clad thermal analyses are performed with the LOCTA-IV code (Reference 7), which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from NOTRUMP hydraulic calculations as input. The LOCTA-IV code version used for the clad thermal analysis of the small break LOCA includes the clad swelling and rupture model of NUREG-0630.

For these analyses, the safety injection delivery considers pumped injection flow, which is depicted in Figure 14.3-3 as a function of RCS pressure. This figure represents injection flow from the high head safety injection pumps based on performance curves degraded 7 percent from the design head. A 25 second delay was assumed from the time that the SI signal is generated to the time that the pumps are at full speed and capable of injecting water into the system. The effect of the low head safety injection pumps (Residual Heat Removal pump) flow is not considered since their shutoff head is lower than the Reactor Coolant System pressure during the time period of the transient. Also, minimum Emergency Core Cooling System capability has been assumed in these analyses. The small break LOCA analysis also assumes that the rod drop time is 2.7 seconds.

Figure 14.3-53 presents the hot rod power shape utilized as input to perform the small break analysis presented here. This power shape was chosen because it represents a distribution with power concentrated in the upper regions of the reactor core. Such a distribution is limiting for SBLOCA since it minimizes coolant swell while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations.

The small break analysis was performed with the Westinghouse ECCS small break Evaluation Model using the NOTRUMP code, approved for this use by the Nuclear Regulatory Commission in May 1985 (Reference 16) and in August 1996 (Reference 93).

14.3.3.2 Input Parameters and Initial Conditions

14.3.3.2.1 Large-Break Input Parameters and Initial Conditions

Table 14.3-3 and the following summarize key plant and model parameters whose range and uncertainty are considered in the large break LOCA analysis. The assumed initial condition for the initial and reference case calculations in Reference 79 is also given.

1.0 Plant Physical Description

1.0a **Dimensions:** Nominal geometry is assumed. Nominal geometry input is accounted for in the code uncertainty, since experiments were also subject to thermal expansion and dimensional uncertainty effects.

1.0b **Flow Resistance:** Best estimate values of loop flow resistance are assumed. Variations in this parameter are accounted for in the uncertainty analysis.

1.0c **Pressurizer Location:** On an intact loop, which was confirmed to be the limiting location or to have a small effect on the results.

1.0d **Hot Assembly Location:** The location assumed for the hot assembly is that which reduces the direct flow of water from the upper head or upper plenum. This location is described in Section 3.2.1 of Reference 79.

1.0e **Hot Assembly Type:** The hot assembly is a fresh 15 x 15 upgraded fuel reload assembly with ZIRLO™ cladding. Variations in cycle burnup are accounted for in the uncertainty analysis.

1.0f **Steam Generator Tube Plugging (SGTP) Level:** The maximum value of SGTP level is used for the initial transient. The limiting value over the expected range is discussed in Section 4.3.1 of Reference 79.

2.0 Plant Initial Operating Conditions

2.1 Reactor Power

2.1a **Initial Core Average Linear Heat Rate:** Maximum power without measurement uncertainties is assumed. Uncertainties are accounted for as part of the uncertainty analysis.

2.1b **Hot Rod Peak Linear Heat Rate:** The hot rod peak linear heat rate is assumed to be the median expected value, without uncertainties, between the desired Tech Spec limit and the maximum value for steady-state depletion. The value of F_Q assumed in the initial transient is therefore substantially higher than the value likely to be measured during normal scheduled surveillance. Variations in this parameter are accounted for as part of the uncertainty analysis.

2.1c **Hot Rod Average Linear Heat Rate:** The hot rod average linear heat rate is derived from Tech Spec value. The value of $F_{\Delta H}$ assumed in the reference transient is therefore substantially higher than the value likely to be measured during most of the fuel cycle. Variations in this parameter are accounted for as part of the uncertainty analysis.

2.1d **Hot Assembly Average Linear Heat Rate:** The power generated in the hot assembly rod is 4 percent lower than that generated in the hot rod. Variations in this parameter are accounted for as part of the uncertainty analysis.

2.1e **Hot Assembly Peak Linear Heat Rate:** Consistent with the average linear heat rates, the peaking factor used to calculate the peak nuclear energy generated in the hot assembly average rod is 4 percent lower than the value assumed in the hot rod. Variations in this parameter are accounted for as part of the uncertainty analysis.

2.1f **Axial Power Distribution:** A shape with a top skewed power distribution (Figure 14.3-20) within the expected range is assumed. Variations in axial power distribution due to transient operation are accounted for as part of the uncertainty analysis.

2.1g **Low Power Region (PLOW):** A relative power of 30 percent of the core average is assumed for the low power region. The limiting value over the expected operating range for this parameter is discussed in Section 4.3.3 of Reference 79.

2.1h **Hot Assembly Burnup:** Beginning of Life (BOL) conditions in the hot assembly are assumed in the initial transient. The time in cycle is a sampled attribute in the ASTRUM methodology.

2.1i **Prior Operating History:** The reactor is assumed to have been operating at full power. When a given axial power distribution is considered, it is assumed to have existed since this startup time. This means that the distribution of fission products coincides with the steady-state fission rate distribution. This assumption conservatively places both the initial fission rate and stored energy, and the subsequent decay heat production, at the same axial location.

2.1j **Moderator Temperature Coefficient:** A value greater than or equal to the maximum specified in Technical Specifications is assumed, to conservatively estimate core reactivity and fission power.

2.1k **Hot Full Power (HFP) Boron Concentration:** A low value typical of those used in current cores at BOL conditions is assumed.

2.2 Fluid Conditions

2.2a **Average Fluid Temperature (T_{avg}):** T_{avg} is assumed at the maximum expected value during normal full power operation. Minimum T_{avg} is analyzed as part of the confirmatory calculations in Section 4.3.2 of Reference 79. Variations in the uncertainty of this parameter are included in the uncertainty analysis.

2.2b **Pressurizer Pressure:** The nominal operating value of pressurizer pressure is assumed. Uncertainties associated with this parameter are accounted for in the uncertainty analysis.

2.2c **Loop Flowrate:** The thermal design loop flowrate is assumed.

2.2d **Upper Head Temperature (T_{UH}):** The appropriate best estimate value of T_{UH} is assumed. Since variation in this parameter is small, uncertainties are not included.

2.2e **Pressurizer Level:** The nominal value of pressurizer level is assumed. Because the pressurizer level is automatically controlled and the effect on PCT is small, uncertainties are not included.

2.2f **Accumulator Water Temperature:** A nominal value is assumed, with variations treated as part of the [Deleted] uncertainty analysis.

2.2g **Accumulator Pressure:** A nominal value is assumed with variations treated as part of the uncertainty analysis.

2.2h **Accumulator Water Volume:** A nominal value is assumed with variations treated as part of the uncertainty analysis.

2.2i **Accumulator Line Resistance:** A best estimate value of accumulator line resistance is assumed. Uncertainty in line resistance is included in the [Deleted] uncertainty analysis.

2.2j **Accumulator Boron Concentration:** The minimum value is assumed.

3.0 Accident Boundary Conditions

3.0a **Break Location:** A break near the mid-point in the cold leg is assumed. Scoping studies reported in the CQD (Reference 75) show that the cold leg remains the limiting location for large LOCA.

3.0b **Break Type:** A Double Ended Guillotine Cold Leg (DEGGL) break is assumed in the initial and reference transient. The effect of variations in break type is accounted for in the uncertainty analysis.

3.0c **Break Size:** A nominal cold leg area is assumed. The effect of variations in the break area is accounted for in the uncertainty analysis.

3.0d **Offsite Power:** No loss of offsite power is assumed for the initial transient. Calculations assuming loss of offsite power are performed as part of the confirmatory analysis in Section 4.3.4 of Reference 79 to confirm the limiting condition. Note that it was determined in the confirmatory analysis that loss of offsite power is limiting; therefore, loss of offsite power is modeled in the uncertainty analysis.

3.0e **Safety Injection (SI) Flow:** Minimum SI flow is assumed (see Section 3.2.3 of Reference 79, Emergency Core Cooling and Safety Injection Model). Scoping studies reported in the CQD (Reference 75) indicate that increased SI flow reduces PCT. This parameter is therefore bounded. The primary reason for this choice is that using best estimate values for this important parameter, while producing more realistic results, may also require additional testing and surveillance to verify the assumed flow uncertainty.

3.0f **Safety Injection (SI) Temperature:** Nominal values are assumed. Variations are accounted for in the uncertainty analysis.

3.0g **Safety Injection (SI) Delay:** Maximum values consistent with the offsite power assumption are used for the initial transient (offsite power available) and the confirmatory runs (loss of offsite power).

3.0h **Containment Pressure:** The containment pressure curve shown in Figure 14.3-22 is calculated using the approved containment model (References 6 and 8), the raw data in Table 14.3-2 and the Mass and Energy releases in Table 14.3-2A. Note that a conservative (lower) containment pressure curve than the containment pressure curve shown in Figure 13.2-22 is used for the WCOBRA/TRAC initial and confirmatory calculations in Section 4 and the ASTRUM calculations in Section 5 of Reference 79.

3.0i **Single Failure Assumption:** The worst single failure is assumed to be the loss of a full train of SI, consistent with the recommended scenario outlined in the CQD (Reference 75) and the RMR (Reference 76).

3.0j **Rod Drop Time:** Consistent with the current design basis for this plant, [Deleted] control rods are assumed not to insert during the LBLOCA.

4.0 Model parameters

All model parameters are used at their best estimate or as coded values in the initial transient. [Deleted]

Table 14.3-3 summarizes the initial transient assumptions described above. For those parameters where a best estimate or nominal value was used, the corresponding uncertainty treatment is also given.

14.3.3.2.2 Small-Break Input Parameters and Initial Conditions

Table 14.3-11 lists important input parameters and initial conditions used in the Indian Point Unit 2 small break LOCA analysis.

The small break LOCA analysis was performed with the upper head fluid temperature equal to the Reactor Coolant System hot leg fluid temperature. In addition, this analysis has included the effects of a 10% uniform steam generator tube plugging.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (References 17-18). In addition, the requirements of 10 CFR 50 Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time the LOCA occurs and include such items as the core peaking factors and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

14.3.3.3 Large Break Results

14.3.3.3.1 Large Break LOCA Reference Transient Description

Based on the results of the confirmatory analysis (discussed in Section 14.3.3.3.2), the reference transient models the following conditions:

- High steam generator tube plugging (SGTP) (10%)
- High nominal RCS T_{avg} (572°F)
- Maximum power fraction in the low power/peripheral assemblies (0.8)
- Loss-of-Offsite Power (LOOP)

These limiting conditions are then assumed for the ASTRUM uncertainty analysis discussed in Section 14.3.3.3.3.

The LOCA transient can be conveniently divided into a number of time periods in which specific phenomena are occurring. For a typical large break, the blowdown period can be divided into the critical heat flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflooding and long term cooling phases. The important phenomena occurring during each of these phases are discussed for the reference transient. The results are shown in Figures 14.3-6 and 14.3-19.

I. Critical Heat Flux (CHF) Phase (0 to ~ 2 seconds)

Immediately following the cold leg rupture, the break discharge rate is subcooled and high, the core flow reverses, the fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up while core power shuts down. Figure 14.3-6 shows the maximum cladding temperature in the core, as a function of time. The hot water in the core and the upper plenum begins to flash to steam during this period. The phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture swells and the intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

II. Upward Core Flow Phase (~2 to 12 seconds)

Heat transfer is improved as the two-phase mixture is pulled into the core. This phase may be enhanced if the pumps are not degraded, and the break discharge rate is reduced because the fluid is saturated at the break. Figures 14.3-7 and 14.3-8 show the break flowrate for the vessel side and pump side, respectively, for the reference transient. This phase ends as lower plenum mass is depleted, the loops become two-phase, and the pump head degrades. If pumps are highly degraded or the break flow is large, the cooling effect due to upward flow may not be significant. Figure 14.3-9 [Deleted] shows the void fraction [Deleted] for one intact loop pump and the broken loop pump. The intact loop pump remains in single-phase liquid flow for several seconds, while the broken loop pump is in two-phase and steam flow soon after the break.

III. Downward Core Flow Phase (~12 to 28 seconds)

The loop flow is pushed into the vessel by the intact loop pumps and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core. Figures 14.3-10 and 14.3-11 show the vapor flow at the mid-core of channels 17 and 19, respectively. While liquid and entrained liquid flows also provide core cooling, the vapor flow entering the core best illustrates this phase of core cooling. This period is enhanced by flow from the upper head. As the system pressure continues to fall, the break flow and consequently the core flow, are reduced. The core begins to heat up as the system reaches containment pressure and the vessel begins to fill with Emergency Core Cooling System (ECCS) water.

IV. Refill Phase (~28 to 36 seconds)

The core [Deleted] experiences a nearly adiabatic heatup as the lower plenum fills with ECCS water. Figure 14.3-12 shows the lower plenum liquid level. This phase ends when the ECCS water enters the core and entrainment begins, with a resulting improvement in heat transfer. Figures 14.3-13 and 14.3-14 show the liquid flows from the accumulator and the safety injection, respectively, from an intact loop (Loop 2).

V. [Deleted] Reflood Phase (~36 to end)

The accumulators begin to empty and nitrogen enters the system (Figure 14.3-13). This forces water into the core which then boils as the lower core region begins to quench, causing repressurization. The repressurization is best illustrated by the increase in downcomer liquid level (Figure 14.3-16). During this time, core cooling may be increased.

The system then settles into a gravity driven reflood. Figures 14.3-15 and 14.3-16 show the core and downcomer liquid levels, respectively. Figure 14.3-17 shows the vessel fluid mass. As the quench front progresses further into the core, the peak cladding temperature (PCT) location moves higher in the top core region. Figure 14.3-18 shows the movement of the PCT location. As the vessel continues to fill, the PCT location is cooled and the PCT heatup is terminated on all fuel rods (Figures 14.3-18 and 14.3-19).

VI. Long Term Core Cooling

At the end of the WCOBRA/TRAC calculation, the core and [Deleted] downcomer levels are increasing as the pumped safety injection flow exceeds the break flow. The core and downcomer levels would be expected to continue to rise, until the downcomer mixture level approaches the loop elevation. At that point, the break flow would increase, until it roughly matches the injection flowrate. The core would continue to be cooled until the entire core is eventually quenched as shown in Figure 14.3-19.

The reference transient resulted in a blowdown PCT of 1506°F and a limiting reflood PCT of 1747°F.

14.3.3.3.2 Confirmatory Sensitivity Studies

A number of sensitivity calculations were carried out to investigate the effect of the key LOCA parameters, and to determine the reference transient. In the sensitivity studies performed, LOCA parameters were varied one at a time. For each sensitivity study, a comparison between the base case and the sensitivity case transient results was made.

The results of the sensitivity studies are summarized in Table 4.3-1 of Reference 79. A full report of the results for all confirmatory sensitivity study results is included in Section 4.3 of Reference 79. The results of these analyses lead to the following conclusions:

1. Modeling maximum steam generator tube plugging (10%) results in a higher PCT than minimum steam generator tube plugging (0%).

2. Modeling loss-of-offsite-power (LOOP) results in a higher PCT than no loss-of-offsite-power (no-LOOP).

3. Modeling the maximum value of vessel average temperature (T_{avg} = 572°F) results in a higher PCT than minimum value of vessel average temperature (T_{avg} = 549°F).

4. Modeling the maximum power fraction (P_{LOW} = 0.8) in the low power / periphery channel of the core results in a higher PCT than minimum power fraction (P_{LOW} = 0.3).

14.3.3.3.3 Uncertainty Evaluation and Results

The ASTRUM methodology requires the execution of 124 transients to determine a bounding estimate of the 95th percentile of the Peak Clad Temperature (PCT), Local Maximum Oxidation (LMO), and Core Wide Oxidation (CWO) with 95% confidence level. The results [Deleted] are given in Table 14.3-4, which shows the limiting peak clad temperature of 1962°F, the limiting local maximum oxidation of 2.39% and the limiting corewide oxidation of 0.35%. The sequence of events for the large break LOCA limiting PCT transient is summarized in Table 14.3-1.

The effect of the integral fuel burnable absorber (IFBA) fuel on PCT and LMO was analyzed as part of the ASTRUM methodology. The IFBA rods are treated as having a local effect, i.e., their presence in the core does not contribute to the global thermal-hydraulic response during a large-break LOCA. The analysis results indicate that as far as the PCT and LMO limit, IFBA fuel is bounded by the non-IFBA fuel. Note that IFBA has no effect on the calculation of the CWO value, which is based on global parameters and relies on WCOBRA/TRAC results only.

The detail discussion of the ASTRUM analysis results and the ranking with regard to PCT, LMO AND CWO is presented in Section 5.3 of Reference 79.

14.3.3.3.4 Evaluation

The base analysis discussed in Sections 14.3.3.3.1 to 14.3.3.3.3 was performed assuming a full core of upgraded fuel. For [Deleted] large break LOCA analysis, additional calculations were performed to assess the effect of missing fuel assembly alignment pins.

[Deleted]

Missing Fuel Assembly Alignment Pins Evaluation

Operation [Deleted] with missing fuel assembly alignment pins at peripheral core location A8 has been evaluated. Detailed assessment results in a conservative 5 ^{o}F PCT penalty.

[Deleted]

14.3.3.3.5 Plant Operation Range

The expected PCT and its uncertainty developed above is valid for a range of plant operation conditions. [Deleted] The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Table 14.3-5A summarizes the operating ranges [Deleted]. If operation is maintained within these ranges, the LOCA analyses developed in this section using WCOBRA/TRAC are valid.

14.3.3.3.6 Large Break LOCA Conclusions

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met [Deleted] is as follows:

1. There is a high level of probability that the peak cladding temperature (PCT) shall not exceed 2200 ^{o}F. The results presented in Table 14.3-4 indicate that this regulatory limit has been met with a calculated limiting PCT of 1962^{o}F, which is a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. The PCT including Assessments is reported annually to the NRC as per the requirements of 10 CFR 50.46.

2. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount (or 1 percent) that would be generated if all of the metal in the cladding cylinders surrounding the fuel were to react. The results presented in Table 14.3-4 indicate that this regulatory limit has been met with a calculated maximum core-wide oxidation of 0.35 percent.

3. The calculated maximum local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. The results presented in Table 14.3-4 indicate that this regulatory limit has been met with a calculated maximum local oxidation of 2.39 percent.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met by demonstrating that the PCT does not exceed 2200 ^{o}F, the maximum local oxidation does not exceed 17%, and the seismic and LOCA forces are not sufficient to distort the fuel assemblies to the extent that the core cannot be cooled. The approved methodology (Reference 75) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the lower power channel as defined in the WCOBRA/TRAC model. This situation has not been previously calculated to occur [Deleted]. Therefore, this regulatory limit is met.

5. 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. The approved position on this criterion is that this requirement is satisfied if a coolable core geometry is maintained, and the core remains subcritical following the LOCA (Reference 78). This position is unaffected by the use of the best-estimate LOCA methodology.

14.3.3.4 Small-Break Results

This section presents the results of the small break LOCA analysis for a range of break sizes and fuel with ZIRLO™ cladding. NUREG-0737 (Reference 80), Section II.K.3.31, requires a plant specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC Generic Letter 83-35 (Reference 81), generic analyses using NOTRUMP (References 15, 16, and 93) were performed and are presented in WCAP-11145 (Reference 59). Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break location is limiting. The limiting break for Indian Point Unit 2 was found to be a 3-inch cold leg break. Also, in compliance with 10 CFR50.46 Section (a)(1)(i), additional cases were analyzed to ensure that the 3-inch diameter break was limiting. Calculations were run assuming breaks of 2 inches and 4 inches for ZIRLO™ clad fuel.

A list of input assumptions used in the small break analysis is provided in Table 14.3-11. The results of a spectrum analysis (three break sizes) performed for the upgraded ZIRLO™ clad fuel are summarized in Table 14.3-13, while the key transient event times are listed in Table 14.3-12.

For the limiting 3-inch break transient, Figures 14.3-54 through 14.3-61 depict the following parameters:

- RCS Pressure
- Core mixture level
- Hot rod cladding temperature
- Core steam flow rate
- Hot assembly rod surface heat transfer coefficient
- Hot spot fluid temperature
- Cold leg break mass flow rate
- Safety injection mass flow rate

In addition, the following transient parameters are presented for the non-limiting 2-inch and 4 inch breaks:

- RCS Pressure
- Core mixture level
- Hot rod cladding temperature

Figures 14.3-62 through 14.3-64 are for the 2-inch break transient, while Figures 14.3-65 through 14.3-67 show the above parameters for the 4-inch break.

During the initial period of the small break transient, the effect of the break flow rate is not strong enough to overcome the flow rate maintained by the reactor coolant pumps as they coast down following Loss-Of-Offsite-Power (LOOP). At the low heat generation rates following reactor trip, the fuel rods continue to be well cooled as long as the core is covered by a two-phase mixture. From the cladding temperature transients for the limiting break calculation shown in Figure 14.3-56, it can be seen that the peak cladding temperature occurs near the time of minimum core mixture level (1308 seconds) when the top of the core is steam cooled. This time is accompanied by the highest vapor superheating above the mixture level. The peak cladding temperature during the transient was 1028°F. At the time the transient was terminated, the safety injection flow rate that was delivered to the RCS exceeded the mass flow rate out the break. The decreasing RCS pressure results in greater safety injection flow as well as reduced break flow. As the RCS inventory continues to gradually increase, the reactor mixture level will continue to increase and the fuel cladding temperatures will continue to decline.

The maximum calculated peak cladding temperature for all small breaks analyzed is 1028°F, which is less than the 10 CFR 50.46 ECCS Acceptance Criteria limit of 2200°F. The maximum local metal water reaction is below the embrittlement limit of 17-percent as required by 10 CFR 50.46. The total metal-water reaction is less than 1 percent, as compared with the 1 percent, criterion of 10 CFR 50.46, and the cladding temperature transient is terminated at a time when the core geometry is still coolable. As a result, the core temperature will continue to drop and the ability to remove decay heat for an extended period of time will be provided. The PCT results provided in Table 14.3-13 relate to the small break LOCA Analysis of Record and do not reflect any individual PCT assessments made relative to the Analysis-of-Record and the accepted SBLOCA Evaluation Model which are reported separately, pursuant to 10CFR 50.46 and Reference 82.

An additional feature of the 15 X 15 upgraded fuel, the Integral Fuel Burnable Absorber (IFBA), has been generically evaluated for its impact on small break LOCA Analyses. This feature was previously discussed in the Reload Safety Evaluation for Cycle 11, (Reference 85) for 15 x 15 VANTAGE+ fuel. The evaluation has determined that the magnitude of SBLOCA PCT differences between IFBA and non IFBA Fuel is negligible and remains valid for 15 x 15 upgraded fuel. Therefore the small break LOCA transient was analyzed assuming upgraded fuel without IFBA.

14.3.3.4.1 Conclusions

Analyses presented in this section show that the high head safety injection of the Emergency Core Cooling System (the low head safety injection pumps were not modeled in the Indian Point Unit 2 small break LOCA analysis), provides sufficient core flooding to keep the calculated peak cladding temperature below the required limit of 10 CFR 50.46.

The results of this analysis demonstrate that, for a small break LOCA, the Emergency Core Cooling System will meet the acceptance criteria as presented in 10 CFR 50.46 (Reference 1). These criteria are as follows:

1) The calculated peak fuel element cladding temperature is below the requirement of 2200°F.

2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent by weight of the total amount of zircaloy in the reactor.

3) The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17-percent by weight are not exceeded during or after quenching.

4) The core remains amenable to cooling during and after the break.

5) The core temperature is reduced and decay heat is removed for an extended period of time as required by the long-lived radioactivity remaining in the core.

14.3.4 Core And Internals Integrity Analysis

14.3.4.1 Design Criteria

The basic requirement of any LOCA (Loss-Of-Coolant-Accident), including the double- ended severance of a reactor coolant pipe, is that sufficient integrity be maintained to permit the safe and orderly shutdown of the reactor. This implies that the core must remain essentially intact and the deformations of the internals must be sufficiently small so that primary loop flow, and particularly, adequate safety injection flow, is not impeded. The ability to insert control rods, to the extent necessary, to provide shutdown following the accident must be maintained. Maximum allowable deflection limitations are established for those regions of the internals that are critical for plant shutdown. The allowable and no loss of function deflection limits under dead weight loads plus the maximum potential earthquake and/or blowdown excitation loads are presented in Table 14.3-14.

With the acceptance of Leak-Before-Break by USNRC, Reference 20, (see Section 14.3.5.4.3.2) the dynamic effects of main coolant loop piping no longer have to be considered in the design basis analysis. Only the dynamic effects of the next most limiting breaks of auxiliary lines need to be considered; and consequently the components will experience considerably less loads and deformations than those from the main loop line breaks.

14.3.4.2 Internals Evaluation

The horizontal and vertical forces exerted on reactor internals and the core, following a LOCA are computed by employing MULTIFLEX (3.0), Reference 21, NRC accepted for similar applications, Reference 19, computer code developed for the space-time dependent analysis of nuclear power plants.

14.3.4.2.1 LOCA Forces Analysis

MULTIFLEX (3.0), Reference 21, is a digital computer program for calculation of pressure, velocity, and force transients in reactor primary coolant systems during the subcooled, transition, and early saturation portion of blowdown caused by a LOCA. During this phase of the accident, large amplitude rarefaction waves are propagated through the system with the velocity of sound causing large differences in local pressures. As local pressures drop below saturation, causing formation of steam, the amplitudes and velocities of these waves drastically decrease. Therefore, the largest forces across the reactor internals due to wave propagation occur during the subcooled portions of the blowdown transient. MULTIFLEX includes mechanical structure models and their interaction with the thermal-hydraulic system.

14.3.4.2.2 MULTIFLEX

The thermal-hydraulic portion of MULTIFLEX (3.0), Reference 21, is based on the 1-dimensional homogeneous flow model which is expressed as a set of mass, momentum, and energy conservation equations. These equations are quasi-linear first order partial differential equations, which are solved by the method of characteristics. The numerical method employed is the explicit scheme, consequently time steps for stable numerical integration are restricted by sonic propagation.

In MULTIFLEX, the structural walls surrounding a hydraulic path may deviate from their neutral positions depending on the force differential on the wall. The wall displacements are represented by those of 1-dimensional mass points, which are described by the mechanical equations of vibration.

MULTIFLEX is a generalized program for analyzing and evaluating thermal-hydraulic-structure system dynamics. The thermal-hydraulic system is modeled with an equivalent pipe network consisting of 1-dimensional hydraulic legs, which define the actual system geometry. The actual system parameters of length, area, and volume are represented with the pipe network.

MULTIFLEX computes the pressure response of a system during a decompression transient. The asymmetric pressure field in the downcomer annulus region of a PWR can be calculated. This pressure field is integrated over the core support barrel area to obtain total dynamic load on the core support barrel. The pressure distributions computed by MULTIFLEX can also be used to evaluate the reactor core assembly and other primary coolant loop component support integrity.

MULTIFLEX evaluates the pressure and velocity transients for locations throughout the system. The pressure and velocity transients are made available to the programs LATFORC and FORCE-2 (described in Reference 24, Appendix A and B), which used detailed geometric descriptions to evaluate hydraulic loadings on reactor internals.

14.3.4.2.3 Horizontal / Lateral Forces - LATFORC

LATFORC, described in Reference 24 Appendix A, calculates the lateral hydraulic loads on the reactor vessel wall, core barrel, and thermal shield, resulting from a postulated loss-of-coolant accident in the primary reactor coolant system. A variation of the fluid pressure distribution in the downcomer annulus region during the blowdown transient produces significant asymmetrical loading on the reactor vessel internals. The LATFORC computer code is used in conjunction

with MULTIFLEX, which provides the transient pressures, mass velocities, and other thermodynamic properties as a function of time.

14.3.4.2.4 Vertical Forces - FORCE2

FORCE-2, described in Reference 24 Appendix B, determines the vertical hydraulic loads on the reactor vessel internals. Each reactor component for which force calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element.
2. Flow stagnation on, and unrecovered orifice losses across, the element.
3. Friction losses along the element.

Input to the code, in addition to the MULTIFLEX pressure and velocity transients, includes the effective area of each element on which acts the force due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

14.3.4.3 Structural Response of Reactor Vessel Internals During LOCA and Seismic Conditions

14.3.4.3.1 Structural Model and Method of Analysis

The response of reactor vessel internals components due to an excitation produced by a complete severance of auxiliary loop piping is analyzed. With the acceptance of Leak-Before-Break by USNRC, Reference 20, the dynamic effects of main coolant loop piping no longer have to be considered in the design basis analysis. Only the dynamic effects of the next most limiting breaks of auxiliary lines need to be considered; and consequently the components will experience considerably less loads than those from the main loop line breaks.

The required break locations are defined in Reference 25. Aside from 8 locations on the primary coolant loop piping, the 3 largest auxiliary line breaks are also postulated. These are the accumulator line, the pressurizer surge line, and the RHR line. In accordance with Reference 25, the auxiliary line break is postulated to occur at the safe-end junction between the branch connection and the branch piping. In practice, this has been conservatively represented in these applicable MULTIFLEX analyses as a break location 1 foot from the main coolant loop piping, with a branch line nozzle flow area equivalent to the main coolant loop piping, although a longer branch line connection (nozzle) with a smaller flow area is justified by the approved methodology described in Reference 24 and 25. Branch line nozzles with thermal shields are conservatively assumed to have no thermal shield, since the thermal shield could be postulated to be lost with the ruptured branch line piping.

Assuming that such a pipe break on the cold leg occurs in a very short period of time (1 ms), the rapid drop of pressure at the break produces a disturbance that propagates through the reactor vessel nozzle into the downcomer (vessel and barrel annulus) and excites the reactor vessel and the reactor internals. The characteristics of the hydraulic excitation combined with those of the structures affected present a unique dynamic problem. Because of the inherent gaps that exist at various interfaces of the reactor vessel/reactor internals/fuel, the problem becomes that of nonlinear dynamic analysis of the RPV system. Therefore, nonlinear dynamic analyses

(LOCA and Seismic) of the RPV system includes the development of LOCA and seismic forcing functions which are also discussed here.

14.3.4.3.2 Structural Model

Figure 14.3-101 is schematic representation of the reactor pressure vessel system. In this figure, the major components of the system are identified. The RPV system finite element model for the nonlinear time history dynamic analysis consists of three concentric structural sub-models connected by nonlinear impact elements and linear stiffness matrices. The first sub-model, shown in Figure 14.3-102 represents the reactor vessel shell and its associated components. The reactor vessel is restrained by four reactor vessel supports (situated beneath alternate nozzles) and by the attached primary coolant piping. Also shown in Figure 14.3-102 is a typical RPV support mechanism.

The second sub-model, shown in Figure 14.3-103a represents the reactor core barrel, thermal shield, lower support plate, tie plates, and the secondary support components. These sub-models are physically located inside the first, and are connected to them by stiffness matrices at the vessel/internals interfaces. Core barrel to reactor vessel shell impact is represented by nonlinear elements at the core barrel flange, upper support plate flange, core barrel outlet nozzles, and the lower radial restraints.

The third and innermost sub-model, shown in Figure 14.3.103b represents the upper support plate assembly consisting of guide tubes, upper support columns, upper and lower core plates, and the fuel. The fuel assembly simplified structural model incorporated in to the RPV system model preserves the dynamic characteristics of the entire core. For each type of fuel design the corresponding simplified fuel assembly model is incorporated in to the system model. The third sub-model is connected to the first and second by stiffness matrices and nonlinear elements. Finally, Figure 14.3-104 shows the RPV system model representation.

14.3.4.3.3 Analysis Technique

The WECAN Computer Code (Westinghouse Electric Computer Analysis), Reference 22, which is used to determine the response of the reactor vessel and its internals, is a general purpose finite element code. In the finite element approach, the structure is divided into a finite number of discrete members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformations, the element global matrices and arrays are assembled into global structural matrices and arrays, and used for dynamic solution of the differential equation of motion for the structure.

The WECAN Code solves equation of motions using the nonlinear modal superposition theory. Initial computer runs such as dead weight analysis and the vibration (modal) analyses are made to set the initial vertical interface gaps and to calculate eigenvalues and eigenvectors. The modal analysis information is stored on magnetic tapes, and is used in a subsequent computer runs which solves equations of motions. The first time step performs the static solution of equations to determine steady state solution under normal operating hydraulic forces. After the initial time step, WECAN calculates the dynamic solution of equations of motions and nodal displacements and impact forces are stored on tape for post-processing

The fluid-solid interactions in the LOCA analysis are accounted through the hydraulic forcing functions generated by MULTIFLEX Code, Reference 21. Following a postulated LOCA pipe

rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wave front with low pressure on one side and high pressure on the other.

Depressurization waves propagate from the postulated break location into the reactor vessel through either a hot leg or a cold leg nozzle. After a postulated break in the accumulator branch line on the cold leg, the depressurization path for waves entering the reactor vessel is through the nozzle which contains the broken pipe and into the region between the core barrel and the reactor vessel (i.e., downcomer region). The initial wave propagates up, around, and down the downcomer annulus, then up through the region circumferentially enclosed by the core barrel, that is, the fuel region. In the case of a break in a branch line on the cold leg, the region of the downcomer annulus close to the break depressurizes rapidly but, because of the restricted flow areas and finite wave speed (approximately 3000 feet per second), the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and the reactor vessel. As the depressurization wave propagates around the downcomer annulus and up through the core, the core barrel differential pressure reduces and, similarly, the resulting hydraulic forces drop.

In the case of a postulated auxiliary branch line break on the hot leg piping (such as the RHR line or Pressurizer surge line), the wave follows a similar depressurization path, passing through the outlet nozzle and directly into the upper internals region depressurizing the core and entering the downcomer annulus from the bottom exit of the core barrel. Thus, after a branch line break, on the hot leg, the downcomer annulus would be depressurized with very little difference in pressure forces across the outside the diameter of the core barrel. A branch line break on the hot leg produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the downcomer annulus is not directly involved) and internal differential pressures are not as large as for a cold leg break of the same size. Since the differential pressure is less for a branch line break, the horizontal force applied to the core barrel is less for a hot leg break than for a branch line break on the cold leg. For breaks in branch line piping on both the hot leg and cold leg, the depressurization waves continue to propagate by reflection and translation through the reactor vessel and loops.

The MULTIFLEX computer code, Reference 21, calculates the hydraulic transients within the entire primary coolant system. It considers subcooled, transition, and early two-phase (saturated) blowdown regimes. The MULTIFLEX code employs the method of characteristics to solve the conservation laws, and assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture. As mentioned earlier, the MULTIFLEX code considers a coupled fluid-structure interaction by accounting for the deflection of constraining boundaries, which are represented by separate spring-mass oscillator system. A beam model of the core support barrel has been developed from the structural properties of the core barrel; in this model, the cylindrical barrel is vertically divided into equally spaced segments and the pressure as well as the wall motions are projected onto the plane parallel to the broken loop inlet nozzle. Horizontally, the barrel is divided into 10 segments; each segment consists of four separate walls. The spatial pressure variation at each time step is transformed into 10 horizontal forces which act on the 10 mass points of the beam model. Each flexible wall is bounded on either side by a hydraulic flow path. The motion of the flexible wall is determined by solving the global equations of motions for the masses representing the forced vibration of an undamped beam.

In order to obtain the response of reactor pressure vessel system (vessel/internals/fuel), the LOCA horizontal and vertical forces obtained from the LATFORC and FORCE2 Codes, which

were described earlier, are applied to the finite element system model. The transient response of the reactor internals consists of time history nodal displacements and time history impact forces.

14.3.4.3.4 Seismic Analysis

The basic mathematical model for seismic analysis is essentially similar to the LOCA model except for some minor differences. In LOCA model, as mentioned earlier, the fluid-structure interactions are accounted though the MULTIFLEX Code; whereas in the seismic model the fluid-structure interactions are included through the hydrodynamic mass matrices in the downcomer region. Another difference between the LOCA and seismic models is the difference between in loop stiffness matrices. The seismic model uses the unbroken loop stiffness matrix, whereas the LOCA model uses the broken loop stiffness matrix. Except for these two differences, the RPV system seismic model is identical to that of LOCA model.

The horizontal fluid-structure or hydroelastic interaction is significant in the cylindrical fluid flow region between the core barrel and the reactor vessel annulus. Mass matrices with off-diagonal terms (horizontal degrees-of-freedom only) attach between nodes on the core barrel, thermal shield and the reactor vessel. The mass matrices for the hydroelastic interactions of two concentric cylinders are developed using the work of reference 23. The diagonal terms of the mass matrix are similar to the lumping of water mass to the vessel shell, thermal shield, and core barrel. The off-diagonal terms reflect the fact that all the water mass does not participate when there is no relative motion of the vessel and core barrel. It should be pointed out that the hydrodynamic mass matrix has no artificial virtual mass effect and is derived in a straight-forward, quantitative manner.

The matrices are a function of the properties of two cylinders with the fluid in the cylindrical annulus, specifically, inside and outside radius of the annulus, density of the fluid and length of the cylinders. Vertical segmentation of the reactor vessel and the core barrel allows inclusion of radii variations along their heights and approximates the effects beam mode deformation. These mass matrices were inserted between the selected nodes on the core barrel, thermal shield, and the reactor vessel as shown in Figure 14.3-104. The seismic evaluations are performed by including the effects of simultaneous application of time history accelerations in three orthogonal directions. The WECAN computer code, which is described earlier, is also used to obtain the response for the RPV system under seismic excitations.

14.3.4.3.5 Results and Acceptance Criteria

The reactor internals behave as a highly nonlinear system during horizontal and vertical oscillations of the LOCA forces. The nonlinearities are due to the coulomb friction at the sliding surfaces and due to gaps between components causing discontinuities in force transmission. The frequency response is consequently a function not only of the exciting frequencies in the system but also of the amplitude. Different break conditions excite different frequencies in the system. This situation can be seen clearly when the response under LOCA forces is compared with the seismic response. Under seismic excitations, the system response is not as nonlinear as LOCA response because various gaps do not close during the seismic excitations.

The results of the nonlinear LOCA and seismic dynamic analyses include the transient displacements and impact loads for various elements of the mathematical model. These displacements and impact loads, and the linear component loads (forces and moments) are

then used for detailed component evaluations to assess the structural adequacy of the reactor vessel, reactor internals, and the fuel.

14.3.4.3.6 Structural Adequacy of Reactor Internals Components

The reactor internal components of Indian Point Unit 2 are not ASME Code components, because Sub-section NG of the ASME Boiler and Pressure Code edition applicable to this unit did not include design criteria for the reactor internals since its design preceded Subsection NG of the ASME Code. However, these components were originally designed to meet the intent of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code with addenda through the Winter 1971. As mentioned earlier, that with the acceptance of Leak-Before-Break (LBB) by USNRC, Reference 20, the dynamic effects of the main reactor coolant loop piping no longer have to be considered in the design basis analysis. Only the dynamic effects of the next most limiting breaks of the auxiliary lines (Accumulator line and Pressurizer Surge or RHR line) are considered. Consequently, the components experience considerably less loads and deformations than those from the main loop breaks which were considered in the original design of the reactor internals.

14.3.4.3.7 Allowable Deflection and Stability Criteria

The criteria for acceptability in regard to mechanical integrity analyses is that adequate core cooling and core shutdown must be ensured. This implies that the deformation of reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the reactor internals are concerned principally with the maximum allowable deflections and stability of the components.

For faulted conditions, deflections of critical internals structures are limited to values given in Table 14.3-14. In a hypothesized vertical displacement of internals, energy-absorbing devices limit the displacement to 1.25 inches by contacting the vessel bottom head.

Core Barrel Response Under Transverse Excitations

In general, there are two possible modes of dynamic response of the core barrel during LOCA conditions: a) during a cold leg break the inside pressure of the core barrel is much higher than the outside pressure (downcomer), thus subjecting the core barrel to outward deflections, and b) during hot leg break the pressure outside the core barrel (downcomer) is greater than the inside pressure thereby subjecting the core barrel to compressive loadings. Therefore this condition requires the dynamic stability check of the core barrel during hot leg break.

(1) To ensure shutdown and cooldown of the core during cold leg blowdown, the basic requirement is a limitation on the outward deflection of the barrel at the locations of the inlet nozzles connected to unbroken lines. A large outward deflection of the upper barrel in front of the inlet nozzles, accompanied with permanent strains, could close the inlet area and restrict the cooling water coming from the accumulators. Consequently, a permanent barrel deflection in front of the unbroken inlet nozzles larger than a certain limit, called “no loss of function” limit, could impair the efficiency of the ECCS.

(2) During the hot leg break, the rarefaction wave enters through the outlet nozzle into the upper internals region and thus depressurizes the core

and then enters the downcomer annulus from the bottom exit of the core barrel. This depressurization of the annulus region subjects the core barrel to external pressures and this condition requires a stability check of the core barrel during hot leg break. Therefore, to ensure rod insertion and to avoid disturbing the control rod cluster guide structure, the barrel should not interfere with the guide tubes.

Table 14.3-14 summarizes the allowable and no loss of function deflection limits of the core barrel for both the cold leg and hot leg breaks postulated in the main line loop piping. With the acceptance of LBB, the reactor internal components such as core barrel will experience much less loads and deformations than those obtained from main loop piping.

Control Rod Cluster Guide Tubes

The deflection limits of the guide tubes, which were established from the test data, and for fuel assembly thimbles, cross-section distortion (to avoid interference between the control rod and the guides) are given in Table 14.3-14.

Upper Package

The local vertical deformation of the upper core plate, where a guide tube is located, shall be below 0.100 inch. This deformation will cause the plate to contact the guide tube since the clearance between the plate and the guide tube is 0.100 inch. This limit will prevent the guide tubes from undergoing compression. For a plate local deformation of 0.150 inch, the guide tube will be compressed and deformed transversely to the upper limit previously established. Consequently, the value of 0.150 inch is adopted as the no loss function local deformation with an allowable limit of 0.100 inch. These limits are given in Table 14.3-14.

14.3.4.4 Evaluations of Effects of Loss-of-Coolant and Safety Injection on the Reactor Vessel

The effects of Safety Injection on the Reactor Vessel following a loss of coolant accident were generically evaluated after the Three Mile Island – 2 accident as part of NUREG-0737, Item II.K.2.13, and determined to be acceptable as documented in NRC's June 15, 1984 Safety Evaluation Report (SER) from Steven A. Varga (NRC) to John D. O'Toole (Consolidated Edison). The potential for thermal shock of reactor vessels was later broadened in scope to include all over cooling events, the evaluation of which is currently required by the Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61. As described in Section 4.2.5, NRC's February 27, 1987 Safety Evaluation Report (SER) from M. Slosson (NRC) to M. Selman (Consolidated Edison) concluded that the Indian Point Unit 2 evaluations were acceptable and meet the requirements of the PTS Rule.

14.3.5 Containment Integrity Analysis

14.3.5.1 Containment Structure

14.3.5.1.1 Design Bases

The design and analysis of the Indian Point 2 containment structure are described in Chapter 5. The design bases and design criteria are discussed in Section 5.1.1.1.6 and 5.1.2.2, respectively. The discussion contained in this Section pertains to containment response to Loss

of Coolant Accidents. Containment response to secondary system pipe ruptures is discussed in Section 14.2.5.6.

Sources and amounts of energy that may be available for release to the containment are discussed in Section 14.3.5.3. To obtain a conservative pressure, energy is added to the containment in the manner most detrimental to peak pressure response for the containment response analysis.

Systems for removing energy from within the containment include the safety injection system (Section 6.2), the containment fan cooler system (Section 6.4), and the containment spray system (Section 6.3). The containment fan coolers remove energy from the containment atmosphere. Containment spray is used for rapid pressure reduction and for containment airborne activity removal. During the recirculation phase, the recirculation system removes heat from the reactor fuel via containment sump water. Heat removal by containment spray during the recirculation phase, which is part of the engineered safety features, is not assumed in the containment response analyses.

Engineered safety features systems are redundant and independent such that any single active failure in the engineered safety features system during the injection phase or any single active or passive failure during recirculation (See Section 6.2.3.3) will not affect the ability to mitigate containment pressure as discussed in Sections 14.3.5.3.7 and 14.3.5.5.

Reference 61 has provided the basis for the loss-of-coolant accident spectrum that is analyzed to provide limiting containment pressures and temperatures. These results are bounded by the transient used for design as discussed in Section 5.1.2.2. Results are provided for a Double-Ended Pump Suction (DEPS) break with minimum and maximum safeguards and a Double-Ended Hot Leg (DEHL) break. These analyses were performed at a reactor power level of 3216 MWt. Analyses, assumptions, and results are presented in sections 14.3.5.1.3 through 14.3.5.3.9 for the break spectrum analyzed.

To summarize the break cases, Tables 14.3-16 through 14.3-30 show mass and energy release information, Tables 14.3-15 and 14.3-37 show systems and containment assumptions, and the assumed containment safeguards equipment. Tables 14.3-35 and 14.3-36 show the containment passive heat sink information assumed, and Figure 14.3-115 shows the heat removal capability assumed for one RCFC. Results of the break cases are shown in Figures 14.3-109 through 14.3-114, and are summarized in Table 14.3-34. The break cases show that a reactor coolant system double-ended pump suction (DEPS) rupture, assuming operation of the minimum emergency cooling system equipment, three RCFC units, and one containment spray pump consistent with the assumption of a single failure of one diesel generator, results in the highest containment pressure after a LOCA. The chronology of events for the DEPS minimum safeguards case is shown in Table 14.3-31. (See Section 5.1.1.1.6 for a discussion of the structural containment evaluation based on the limiting case.) The selection of the limiting case, based on both the sensitivity cases and the generic conclusions of the mass and energy release topical report (Reference 61), remains valid for reanalysis of the Indian Point Unit 2 containment transients at stretch power uprate conditions.

The analysis of the limiting case, a DEPS rupture with minimum safeguards, has been performed at an NSSS power of 3230 MWt (core power of 3216 MWt). The analysis was performed using the models and assumptions presented in Reference 61 and Table 14.3-15. The mass and energy release models include the Model 44F replacement steam generator input (including the conservative assumption of 0% tube plugging) and the containment

response model includes the release of the accumulator nitrogen gas to containment. Tables 14.3-16 through 14.3-18 show the mass and energy releases for the blowdown phase, the reflood phase, and the post-reflood phase respectively. Tables 14.3-19 and 14.3-20 show the mass and energy balance data, while Table 14.3-21 shows the principal parameters during the reflood phase. Table 14.3-37 shows the assumed containment safeguards equipment, and Tables 14.3-35 and 14.3-36 show the containment passive heat sink information assumed. Calculation of containment pressure and temperature transients is accomplished by use of the COCO (Reference 6) computer code.

Fan cooler (RCFC) heat removal performance assumed in the analysis is shown in Figure 14.3-115. The critical parameter as regards RCFC capability in the calculated containment pressure is the available total RCFC heat removal capacity. In the containment pressure response analysis, three RCFCs, each with the heat removal capability presented in Figure 14.3-115, are modeled. Any RCFC configuration that assures that heat removal greater than or equal to three times that of Figure 14.3-115 is available post-LOCA via the RCFC system is equally acceptable. Service water flow rate and temperature, fouling factor, and the number of RCFCs available under accident conditions may be modified as long as the required total RCFC heat removal capability exists.

The reanalysis of the limiting containment pressure case, a DEPS rupture with minimum safeguards, has been performed at an NSSS power level of 3230 MWt (core power level of 3216 MWt). The chronology of events is shown in Table 14.3-31. Quantities of heat removed by structures, fan coolers and containment spray are shown in Figures 14.3-105, 14.3-106, and 14.3-107 respectively. The structural heat transfer coefficient is shown in Figure 14.3-108. Results in Figures 14.3-109 and 14.3-110 show that the calculated maximum pressure and temperature are 45.71 psig and 266.81°F, respectively. This indicates margin to the containment design pressure of 47 psig.

Heat removal by recirculation spray is not credited in the analysis. Therefore, the increase in the duration of the recirculation spray flow has no impact on the containment integrity design basis LOCA analysis.

14.3.5.1.2 System Design

Structural design of the containment and containment internal structures is discussed in Chapter 5.

14.3.5.1.3 Design Evaluation

The results of the transient analysis of the containment for the loss-of-coolant accidents are shown in Figures 14.3-105 through 14.3-114. A series of cases were performed in this analysis illustrating the sensitivity to break location. Subsection 14.3.5.3 documented the mass and energy release (LOCA) for the minimum and maximum safeguards cases for a Double-Ended Pump Suction (DEPS) break and the releases from the blowdown of a Double-Ended Hot Leg (DEHL) break. All of these design basis cases show that the containment pressure will remain below design pressure with margin without taking credit for the recirculation spray. After the peak pressure is attained, the performance of the minimum safeguards system reduces that containment pressure. At the end of the first day following the accident, the containment pressure has been reduced to a low value. The peak pressures are shown in Table 14.3-34 for a variety of containment safeguards availability assumptions.

Calculation of containment pressure and temperature transients is accomplished by use of the digital computer code, COCO[6]. Transient phenomena within the reactor coolant system affect containment conditions by means of mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into systems. The first system consists of the air-steam phase; the second consists of the water phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. Thermodynamic equations of state and conditions may vary during the transient. The equations have been derived for all possible cases of superheated or saturated steam and subcooled or saturated water. Switching between states is handled automatically within the COCO code. The following are the major assumptions made in the containment analysis:

1. Discharge mass and energy flow rates through the reactor coolant system break are established from the analysis in Section 14.3.5.3.

2. For the LOCA containment response analysis, the discharge flow from either end of the break separates into steam and water phases upon entry to the containment atmosphere. For each input set of tables of break effluent mass and energy, the COCO code assumes that the saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment.

3. Homogeneous mixing is assumed. The steam-air mixture and the water phase each have uniform properties. More specifically, thermal equilibrium between the air and steam is assumed. This does not imply thermal equilibrium between the steam-air mixture and the water phase, which may be at different temperature.

4. Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

14.3.5.1.4 Initial Conditions

The pressure, temperature, and humidity of the containment atmosphere prior to the postulated reactor coolant system rupture are conservatively specified in the analysis. Also, conservative values for the temperature of the service water and refueling water storage tank water solution are assumed. All of these values are as shown in Table 14.3-37.

In each of the transients, the safeguards systems shown in Table 14.3-37 are assumed to operate with a 60 second delay in startup. The assumed spray flow rate is based on one of two trains of the containment spray system operating.

14.3.5.1.5 Heat Removal

The significant heat removal source during the early portion of the transient are structural heat sinks. Provision is made in the containment pressure transient analysis for heat transfer through, and heat storage in, both interior and exterior walls. Every wall is divided into many nodes; for each node, a conservation of energy equation expressed in finite- difference form accounts for transient conduction into and out of the node and temperature rise of the node.

Tables 14.3-35 and 14.3-36 are summaries of the containment structural heat sinks used in the analysis.

The heat transfer coefficient to the containment structure is calculated by the code based primarily on the work of Tagami[31]. From this work, it was determined that the value of the heat transfer coefficient increases parabolically to peak value at the end of blowdown for LOCA. The value then decreases exponentially to a stagnant heat transfer coefficient, which is a function of steam-to-air-mass ratio.

Tagami presents a plot of the maximum value of h as a function of "coolant energy transfer speed," defined as follows:

$$\frac{\text{total coolant energy transferred into containment}}{\text{(containment volume) x (time interval to peak pressure)}}$$

From this, the maximum h of steel is calculated:

$$h_{max} = 75\left[\frac{E}{t_p V}\right]^{0.60} \qquad (14.3-1)$$

where:

75 = material coefficient for steel

h_{max} = maximum value of h (Btu/hr ft ^{2} °F).

t_p = time from start of accident to end of blowdown (sec)

V = containment volume (ft^3).

E = coolant energy discharge (Btu).

The parabolic increase to the peak value is given by:

$$h_s = h_{max}\left(\frac{t}{t_p}\right)^{0.5}, 0 \le t \le t_p \qquad (14.3-2)$$

where

h_s = heat transfer coefficient for steel (Btu/hr ft^2 °F).

t = time from start of accident (sec).

For concrete, the heat transfer coefficient is taken as 40-percent of the value calculated for steel.

The exponential decrease of the heat transfer coefficient is given by:

$$h_s = h_{stag} + \left(h_{max} - h_{stag}\right) e^{-0.05\left(t - t_p\right)}, t > t_p \qquad (14.3-3)$$

where

h_{stag} = $2 + 50X, 0 < X < 1.4.$

h_{stag} = h for stagnant conditions (Btu/hr ft^2 ^oF).

X = steam-to-air mass ratio in containment.

For a large break, the engineered safety features are quickly brought into operation. Because of the brief period of time required to depressurize the reactor coolant system, the containment safeguards do not influence the blowdown peak pressure; however, they significantly reduce the containment pressure after the blowdown and maintain a low long-term pressure. Also, although the containment structure is not a very effective heat sink during the initial reactor coolant system blowdown, it still contributes significantly as a form of heat removal.

14.3.5.2 Engineered Safety Features

During the injection phase of post-accident operation, the emergency core cooling system pumps water from the refueling water storage tank (RWST) into the reactor vessel (the containment spray pumps also inject RWST water into the containment). Since this water enters the vessel at refueling water storage tank temperature, which is less than the temperature of the water in the vessel, it can absorb heat from the core until saturation temperature is reached. During the recirculation phase of operation, water is taken from the containment sump and cooled in the residual heat removal heat exchanger. The cooled water is then pumped back to the reactor vessel to absorb more decay heat. The heat is removed from the residual heat exchanger by component cooling water and from the component cooling heat exchanger by service water.

14.3.5.2.1 Containment Spray

Another containment heat removal system is the containment spray. During the injection phase of operation, the containment spray pumps draw water from the RWST and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray from the RWST temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase of post-accident operation, water can be drawn from the residual heat removal heat exchanger outlet and sprayed into the containment atmosphere via the recirculation spray system. However, no-credit was taken for recirculation spray in the analysis in calculating the peak containment pressure.

When a spray drop enters the hot, saturated, steam-air containment environment following a loss-of-coolant accident, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the drop. This mass flow will carry energy to the drop. Simultaneously, the temperature difference between the atmosphere and the drop will cause the drop temperature and vapor pressure to rise. The vapor pressure of the drop will eventually become equal to the partial pressure of the steam, and the condensation will cease. The temperature of the drop will equal the temperature of the steam-air mixture.

The equations describing the temperature rise of a falling drop are as follows:

$$\frac{d}{dt}(Mu) = mh_g + q \qquad (14.3-4)$$

$$\frac{d}{dt}(M) = m \qquad (14.3-5)$$

where

$q = h_c A\ (T_s - T).$

$m = k_g A\ (P_s - P_v).$

The coefficients of heat transfer (h_c) and mass transfer (k_g) are calculated from the Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu'.

Both Nu and Nu' may be calculated from the equations of Ranz and Marshall.[38]

$$Nu = 2 + 0.6\ (Re)^{1/2}\ (Pr)^{1/3} \qquad (14.3\text{-}6)$$

$$Nu' = 2 + 0.6\ (Re)^{1/2}\ (Sc)^{1/3} \qquad (14.3\text{-}7)$$

Thus, Equations 14.3-4 and 14.3-5 can be integrated numerically to find the internal energy and mass of the drop as a function of time as it falls through the atmosphere. Analysis shows that the temperature of the (mass) mean drop produced by the SPRACO 1713A spray nozzles rises to a value within 99-percent of the bulk containment temperature in less than 2 seconds. Drops of approximately 1000 micron average size (as discussed in Chapter 6) will reach temperature equilibrium with the steam-air containment atmosphere after falling through less than half the available spray fall height. Detailed calculations of the heatup of spray drops in post-accident containment atmospheres by Parsly[43] show that drops of all sizes encountered in the containment spray reach equilibrium in a fraction of their residence time in a typical pressurized water reactor containment. These results confirm the assumption that the containment spray will be 100-percent effective in removing heat from the atmosphere. Nomenclature in this section is as follows:

A = area

h_c = coefficient of heat transfer

h_g = steam enthalpy

k_g = coefficient of mass transfer

M = droplet mass

m = diffusion rate

Nu = Nusselt number for heat transfer

Nu' = Nusselt number for mass transfer

P_s = steam partial pressure

P_v = droplet vapor pressure

Pr = Prandtl number

q = heat flow rate

Re = Reynolds number
Sc = Schmidt number

T_s = droplet temperature

T = steam temperature

t = time

u = internal energy

14.3.5.2.2 Reactor Containment Fan Coolers (RCFCs)

The reactor containment fan coolers are a principal means of post-accident containment heat removal. The fans draw the dense atmosphere through banks of finned cooling coils and mix the cooled steam/air mixture with the rest of the containment atmosphere. The coils are kept at a low temperature by maintaining the required flow of cooling water from the service water system. Since the RCFCs do not use water from the RWST, the mode of operation remains the same before and after the containment spray and emergency core cooling systems are changed to the recirculation mode.

The ability of the containment air recirculation coolers to function properly in the accident environment is demonstrated by the coil vendor's analysis. This analysis determines the plate-fin cooling coil heat removal rate when operating in a saturated steam-air mixture.

In the heat removal analysis of the RCFC coils, a mass flow rate of cooling water is first established. This determines the inside film coefficient of the tube. Next, the resistance to heat transfer between the cooling water and the outside of the fin collars is computed, including inside film coefficient, fouling factor, tube radial conduction, fin-collar interface resistance, and conduction across the fin collars. [***Note*** *- A fouling factor of 0.001 hr-ft^2-$^\circ$F/Btu, under both normal and design basis accident conditions, has been assumed for cooling coil design purposes. This value is conventionally used in sizing heat exchangers cooled by river water at 95°F or less and with tube water velocity greater than 3-ft/sec*[33] *and is considered sufficiently conservative for this application. Computer analysis of the coils selected shows that the required post-accident heat removal rate can be achieved even with a slight increase in fouling.*] The analysis now becomes iterative. One assumes an overall heat transfer rate Q_{tot} and the temperature at the outside of the fin collars is determined from Q_{tot} and the sum of the resistances cited above.

A second iterative procedure is now established. The variable whose value is assumed is the effective film coefficient between the fins and the gas stream, which involves the effect of convective heat transfer and mass transfer. With this value of $h_{effective}$, fin efficiency and the fin temperature distribution can be determined. It is assumed that a condensate film exists on the

vertical fins. An analysis is performed, which relates this film thickness to the rate of removal due to gravity and shear and the rate of addition of condensate by mass transfer from the bulk gas. In the process, from an energy balance the temperature of the interface between the bulk gas and the condensate can be determined; this is necessary for determining the mass transfer rate from the gas. Now that the thickness of the condensate film is known, the value of the assumed $h_{effective}$ is checked from the relation $h_{eff} = K_{water}/\delta_{film}$. If the assumed and computed values are not the same, a new value is selected and calculations repeated until the assumed and computed values are equal.

When this occurs, the heat transfer rate from the fins and fin collar is computed, using the standard equations for fin and fin collar heat transfer and the values of $h_{effective}$ and film-bulk gas interface temperature. If this value is not the same as Q_{tot} initially assumed in order to determine fin collar temperature, the whole analysis is repeated with a new estimate of Q_{tot}. When, finally, the heat transfer rate to the cooling water from the fin collar equals the resulting computed rate to the fin collar and fins from the gas, the effect of this heat transfer rate on the cooling water is computed. The water exit temperature is established, and this value is used as the inlet temperature for the next heat exchanger pass. Also, the effect of convective heat transfer and condensate mass transfer is determined relative to the gas composition and thermodynamic state. The updated gas state is used as inlet conditions for the next pass. The process is repeated for the second, third, etc., passes until the gas exits the heat exchanger.

The mass transfer coefficients used in the computer code were derived from analyses and reports of experimental data.[33, 34, 35] From Reference 34, the mass flow rate of condensate is defined by [***Note*** - *Nomenclature used is given at the end of this discussion.*]

$$\dot{m} = \bar{h}_D \left(\rho_{sg} - \rho_{sw}\right) \qquad (14.3\text{-}8)$$

From Reference 34, pp. 471-473, experimental data for mass and heat transfer correlate well with the expression

$$\frac{\bar{h}_D}{u_s}\left(Sc\right)^{-2/3} = St\left(Pr\right)^{-2/3} \qquad (14.3 - 9)$$

as shown in Figure 16-10 of Reference 34. Thus,

$$\bar{h}_D = u_s \text{ x } St\left(\frac{Sc}{Pr}\right)^{2/3} \qquad (14.3 - 10a)$$

$$\text{Substituting : } St = \frac{h}{\rho C u_s} \text{ thus we get,}$$

$$\bar{h}_D = \frac{u_s \text{ x } h}{\rho C u_s} \text{ x } \left(\frac{Sc}{Pr}\right)^{2/3} \qquad (14.3 - 10b)$$

As Reference 34 points out, for large partial pressures of the condensing components, Equation 14.3-10b must be corrected by a factor P_t/P_{am}. Thus,

$$\bar{h}_D = \frac{h}{\rho C} x \frac{P_t}{P_{am}} x \left(\frac{Sc}{Pr}\right)^{2/3} \qquad (14.3\text{-}11)$$

This is essentially the same result as reported by Reference 35, p. 343 and Reference 33.

Reference 34 states that experiments show Equation (14.3-8) to be valid when the Schmidt number does not differ greatly from 1.0. Equations (14.3-8) and (14.3-11) are combined to give the mass transfer rate, which is

$$\dot{m} = \frac{h}{\rho C} x \frac{P_t}{P_{am}} x \left(\frac{Sc}{Pr}\right)^{2/3} x \left(\rho_{sg} - \rho_{sw}\right) \qquad (14.3 - 12)$$

An approximation was made in assuming that $(Sc/Pr)^{2/3} \cong 1.0$, thus the local mass transfer rate was computed from

$$\dot{m} = \frac{h}{\rho C} x \frac{P_t}{P_{am}} x \left(\rho_{sg} - \rho_{sw}\right) \qquad (14.3 - 13)$$

The heat transfer rate due to condensation is computed from

$$q_1 = \frac{\lambda h P_t}{\rho C P_{am}} x \left(\rho_{sg} - \rho_{sw}\right) \qquad (14.3 - 14)$$

where

ρ_{sg}	is evaluated at the local bulk gas temperature
ρ_{sw}	is evaluated at the local gas-condensate interface temperature
λ	is evaluated at the local gas-condensate interface temperature
P_t	is evaluated at the local bulk gas temperature
C	is evaluated at the local bulk gas temperature

The heat transfer coefficient, h, was determined from experiments on the same geometry used in this application.

The heat transfer rate, locally, is computed from

$$q_2 = h \text{ x } (T_g - T_i) \qquad (14.3\text{-}15)$$

The basis for selecting these values is that the authorities cited as references have shown, through analyses and through cited experiments, that the methods used are accurate.

The air side pressure drop across the cooling coils at a conservative design-basis accident condition of 47 psig is estimated to be approximately 3.2-in. of water, or 0.115 psi. This will have a negligible effect on the heat removal capability of the cooling coils.

The pressure of noncondensible gases is taken into consideration because the theory behind the analysis assumed that the condensible vapor must diffuse through a noncondensible gas.

The nomenclature is as follows:

$\dot{m}$	mass flow rate of condensate, lbm/hr-ft^2
$\bar{h}_D$	mass transfer coefficient, ft/hr
ρ_{sg}	density of saturated steam at local bulk gas temperature, lbm/ft^3
ρ_{sw}	density of saturated steam at local condensate-gas interface temperature, lbm/ft^3
U_S	free steam gas velocity, ft/min
Sc	Schmidt number, $\mu/\rho D$, dimensionless
μ	viscosity of bulk gas, lbm/ft-hr
ρ	bulk gas density, lbm/ft^3
D	gas-air diffusion coefficient, ft^2/hr
St	Stanton number, $h/\rho C u_s$, dimensionless
h	convective heat transfer coefficient, Btu/hr-ft^2-^{o}F
C	specific heat of bulk gas, Btu/lbm-^{o}F
Pr	Prandtl number, $\mu c/k$, dimensionless
k	thermal conductivity of bulk gas, Btu/hr-ft-^{o}F
P_t	total gas pressure, lbf/ft^2
P_{am}	air log-mean $\dfrac{P_{aw} - P_{ag}}{\ln \dfrac{P_{aw}}{P_{ag}}}$, lbf/ft^2
P_{aw}	partial pressure of air at the local gas-condensate interface, lbf/ft^2
P_{ag}	partial pressure of air at the local bulk gas temperature, lbf/ft^2
λ	latent heat of vaporization (or condensation) at the local gas-condensate interface temperature, Btu/lbm

q_1 local heat transfer rate due to condensation, Btu/hr-ft^2

q_2 local heat transfer rate due to convection, Btu/hr-ft^2

T_g local bulk gas temperature, °F

T_i local gas-condensate interface temperature, °F

δ_{film} water film thickness, ft

A similar heat removal analysis of the currently installed RCFC coils results in the fan-cooler heat removal rate per fan as presented in Figure 14.3-115.

14.3.5.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

This analysis presents the mass and energy releases to the containment subsequent to a hypothetical loss-of-coolant accident (LOCA) at 3216 MWt. The release rates are calculated for pipe failure at three distinct locations:

1. Hot leg (between vessel and steam generator)
2. Pump suction (between steam generator and pump)
3. Cold leg (between pump and vessel)

The LOCA transient is typically divided into four phases:

1. Blowdown - which includes the period from accident occurrence (when the reactor is at steady state operation) to the time when the total break flow stops.

2. Refill - the period of time when the lower plenum is being filled by accumulator and safety injection water. (This phase is conservatively neglected in computing mass and energy releases for containment evaluations.)

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-Reflood - describes the period following the reflood transient. For the pump suction and cold leg breaks, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the flow out of the break becomes two phase.

During the reflood phase, these breaks have the following different characteristics. For a cold leg pipe break, all of the fluid, which leaves the core must vent through a steam generator and becomes locally superheated. However, relative to breaks at the other locations, the core flooding rate (and therefore the rate of fluid leaving the core) is low, because all the core vent paths include the resistance of the reactor coolant pump. For a hot leg pipe break, the vent path resistance is relatively low, which results in a high core flooding rate, but the majority of the fluid, which exits the core bypasses the steam generators in venting to the containment. The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and steam generator heat addition, as in the cold leg break. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period, thereby bounding the hot leg breaks.

The spectrum of breaks analyzed includes the largest pump suction and hot leg breaks. Because of the phenomena of reflood as discussed above, the pump suction break location is the worst case for long term containment depressurization. Smaller hot leg breaks have been shown on similar plants to be less severe than the double-ended hot leg. Cold leg breaks, however, are lower both in the blowdown peak and in the reflood pressure rise and therefore have not been analyzed.

14.3.5.3.1 Mass and Energy Release Data

Blowdown Mass and Energy Release Data

Tables 14.3-16, 14.3-22 and 14.3-28 present the calculated mass and energy releases for the blowdown phase of the various breaks analyzed.

The mass and energy releases for the double-ended pump suction break, given in Table 14.3-16, terminate 26.4 seconds after the postulated accident for the minimum ECCS case. The DEPS maximum ECCS case has a blowdown time of 26.0 seconds and the mass and energy release are given in Table 14.3-22.

Reflood Mass and Energy Release Data

Tables 14.3-17 and 14.3-23 present the calculated mass and energy releases for the reflood phase of the various breaks analyzed along with the corresponding safety injection assumption (minimum and maximum).

Two Phase Post-Reflood Mass and Energy Release Data

Tables 14.3-18 and 14.3-24 present the two phase (froth) mass and energy release data for a double-ended pump suction break using minimum and maximum safety injection assumptions, respectively.

Equilibrium and Depressurization Energy Release Data

The equilibrium and depressurization energy release has been incorporated in the post-reflood mass and energy release data. This eliminates the need to determine additional releases due to the cooling of steam generator secondary and primary metal.

14.3.5.3.2 Mass and Energy Sources

The sources of mass considered in the LOCA mass and energy release analysis are given in the mass balance Tables 14.3-19, 14.3-25, and 14.3-29. These sources are the reactor coolant system, accumulators and pumped injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Tables 14.3-20, 14.3-26, and 14.3-30. The energy sources include:

1. Reactor coolant system
2. Accumulators
3. Pumped injection
4. Decay heat

5. Core stored energy
6. Primary metal energy
7. Secondary metal energy
8. Steam generator secondary energy
9. Secondary transfer of energy (feedwater into and steam out of the steam generator secondary), main feedwater coastdown following reactor trip and SI signal generation.

The inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time that broken loop secondary energy is removed.
6. Time that intact loops secondary energy is removed.
7. Time that the secondary side is assumed to equilibrate to 14.7 psia and 212°F.

The methods and assumptions used to release the various energy sources are given in NRC-approved WCAP-10325[61].

The following items ensure that the core energy release is conservatively analyzed for maximum containment pressure:

1. Maximum expected operating temperature of the reactor coolant system
2. Allowance in operating temperature for instrument error and deadband (+7.5°F)
3. Margin in volume (1.4-percent)
4. Allowance in volume for thermal expansion (1.6-percent)
5. A core power level of 3216 MWt was assumed
6. Allowance for calorimetric error (2-percent of 3216 MWt)
7. Appropriately modified coefficients of heat transfer
8. Allowance in core stored energy for effect of fuel densification
9. Margin in core stored energy (+15-percent)

14.3.5.3.3 Blowdown Model Description

The computer code used to calculate the mass and energy release in the blowdown phase is SATAN-VI. The model is described in WCAP-9220[12] and WCAP-8302[4]. WCAP–10325[61] provides the method by which the model is used.

14.3.5.3.4 Refill Model Description

At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively model the refill period for the purpose of containment mass and energy releases, this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. Thus, the time required for refill is conservatively neglected.

14.3.5.3.5 Reflood Model Description

The computer code used for the reflood phase is WREFLOOD. The model is described in WCAP-9220[12] and WCAP-8170[5]. WCAP-10325[61] describes the method by which this model is used and the modifications. A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the use and application of the M&E release evaluation model (Reference 61) in recent analyses, for example, D. C. Cook Docket (Reference 60). Even though the WCAP-10325-P-A (Reference 61) model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 60). Transients of the principle parameters during reflood are given in Tables 14.3-21 and 14.3-27 for the double-ended pump suction break with minimum and maximum safety injection.

14.3.5.3.6 Post-Reflood Model Description

Two-Phase (FROTH)

The transient model (FROTH), along with its method of use, is described in WCAP-8312-A[62]. The mass and energy rates calculated by FROTH are utilized in the containment analysis to the time of containment depressurization.

Long Term (Dry Steam)

After depressurization, the mass and energy release from decay heat for 3216 MWt is based on ANSI/ANS-5.1- 1979 and the following input:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239. The highest decay heat release rates come from the fission of the U-238 nuclei. Thus, to maximize the decay heat rate a maximum value (8%) has been assumed for the U-238 fission fraction.

2. The second highest decay heat release rate comes from the fission of the U-235 nuclei. Therefore, the remaining fission fraction (92%) has been assumed for U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Table 10 of ANS (1979).

5. The fuel has been assumed to be at full power for 10^8 seconds.

6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

7. Two sigma uncertainty has been applied to the fission product decay.

14.3.5.3.7 Single Failure Analysis

The effect of single failures of various ECCS components on the mass and energy releases is included in these data. Two analyses bound this effect for the pump suction double-ended rupture.

No failure of any ECCS component is assumed in determining the mass and energy releases for the maximum safeguards case. For the maximum safeguards case, the single failure assumed is the loss of one containment spray pump. For the minimum safeguards case, the single failure assumed is the loss of one emergency diesel generator, which results in the loss of the pumped safety injection (i.e., one residual heat removal pump and one safety injection pump) and the loss of the containment safeguards on that diesel. For further conservatism, an additional containment fan cooler unit is assumed to be unavailable, thus limiting the assumed available containment safeguards to three fan cooler and one spray pump. The analysis of both maximum and minimum safeguards cases ensure that the effect of all credible single failures on mass and energy releases is bounded.

A single failure analysis is not performed for the hot leg ruptures since the ECCS has no effect on the maximum containment pressure, which occurs at the end of blowdown.

14.3.5.3.8 Metal-Water Reaction

In the mass and energy release data presented, no zirconium-water reaction heat was considered because the clad temperature did not rise high enough for the rate of reaction to be of any significance.

14.3.5.3.9 Additional Information

System parameters needed to perform confirmatory analyses are provided in Tables 14.3-37, 14.3-38, 14.3-39, and 14.3-40. The chronology of events for the DEPS breaks are presented in Tables 14.3-31 and 14.3-32.

14.3.5.4 <u>Evaluation of Containment Internal Structures</u>

14.3.5.4.1 Previous Design Basis

The containment internal structures such as the reactor coolant loop compartments and the reactor shield wall are designed for the pressure build-up that could occur following a loss of coolant. If a LOCA were to occur in these relatively small volumes, the pressure would build up at a rate faster than the overall compartments.

A digital computer code, COMCO, was developed to analyze the pressure build-up in the reactor coolant loop compartments. The COMCO code is largely an extension of the COCO code in that a separation of the two-phase blowdown into steam and water is calculated and the pressure build-up of the steam-air mixture in the compartment is determined. Each compartment has a vent opening to the free volume of the containment.

The main calculation performed is a mass energy balance within the control volume of a compartment. The pressure builds up in the compartment until a mass and energy relief through the vent exceeds the mass and energy entering the compartment from the break. The

reactor coolant loop compartments are designed for the maximum calculated differential pressure resulting from an instantaneous double-ended rupture of the reactor coolant pipe.

There are two reactor coolant loop compartments (i.e., crane wall areas) with two loops in each compartment. The total free volume of each compartment is 113,500-ft^3 with a vent area of 1000-ft^2. The calculated differential pressure across the wall of the compartment is 6.4 psi.

The primary shield around the reactor vessel is designed for a pressure of 1000 psi to provide missile protection against the highly unlikely failure of the reactor vessel by longitudinal splitting or by various modes of circumferential cracking.

14.3.5.4.2 Current Design Basis

Additional analyses for initial conditions, including prior operation parameters and 3216 MWt power operation parameters, were evaluated relative to short term subcompartment pressurization effects. The mass and energy releases from postulated full double-ended Reactor Coolant System (RCS) breaks were determined with the SATAN-V computer program, reference 62. The TMD computer program, reference 64, was used to evaluate the subcompartment containment response to the hypothetical pipe ruptures. The results of the evaluation indicate that for the full double-ended breaks the peak calculated differential pressure across the wall of the loop compartment was conservatively calculated to be greater that the current design basis of 6.4 psi, as discussed in Section 14.3.5.4.1.

References 65 and 66 demonstrate that RCS primary loop pipe breaks need not be considered in the structural design basis of the Indian Point 2 Plant. Therefore, implementation of Leak-Before-Break (LBB) Technology has eliminated the large RCS breaks from dynamic consideration. For the LOCA event, the break locations and the break sizes are significantly less severe than the previously mentioned RCS double-ended breaks. The previously calculated subcompartment pressure of 6.4 psi is discussed in 14.3.5.4.1. The subcompartment pressure loadings have been evaluated and it has been determined that the loadings, including LBB and operation at 3216 MWt, are less than 6.4 psi. The peak differential pressure across the primary shield wall is bounded by the design pressure of 1000 psi, as discussed in Section 14.3.5.4.1. The effects of the differential operating parameters at 3216 MWt do not result in a challenge to the subcompartment designs.

14.3.5.5 <u>Evaluation of Long Term Fan Cooler Capability</u>

The ability of the fan coolers to limit containment pressure following loss of the component cooling system has been examined. If the component cooling loop were lost for any reason during long-term recirculation, core subcooling could be lost and boiling in the core would begin. Since the cooling units of the fans are cooled by service water, the energy from the core would be removed from the containment via the fans.

The model employed in this analysis does not consider recirculation spray to operate and conservatively considers decay heat from the core to enter the containment as steam during the entire LOCA long-term transient. Therefore, the pressures calculated are not affected with a postulated component cooling system failure, because core energy is already postulated to enter the containment as boil off. Containment pressure at various times for the DEPS case with minimum safeguards is shown below:

Time After Accident Occurs	3 Fans (psig)	2 Fans (psig)
Deleted	Deleted	Deleted
At 1 day	17	25.6
At 1 week	12.6	17.9

14.3.5.6 Radiolytic Hydrogen Formation

Radiolytic hydrogen formation is discussed in Section 6.8.3.

14.3.6 Environmental Consequences Of A Loss-Of-Coolant Accident

Chapters 5 and 6 describe the protection systems and features that are specifically designed to limit the consequences of a major LOCA. The capability of the safety injection system for preventing melting of the fuel clad and the ability of the containment and containment cooling systems to absorb the blowdown resulting from a major loss of coolant are discussed in Section 14.3.4. The capability of the safeguards in meeting dose limits set forth in 10 CFR 50.67 was demonstrated as documented in this section.

For the Large Break Loss-of-Coolant Accident radiological consequences, an abrupt failure of the main reactor coolant pipe is assumed to occur. It is assumed that the emergency core cooling features fail to prevent the core from experiencing significant degradation (i.e. melting). A portion of the activity that is released to the containment is assumed to be released to the environment due to the containment leaking at its design rate.

In the following sections, the expected activity is described and the containment and isolation features are discussed. Sodium tetraborate is used to control pH in the recirculation solutions, as described in Sections 6.3.2.1.2 and 6.3.2.2.12.

14.3.6.1 Effectiveness of Containment and Isolation Features in Terminating Activity Release

The reactor containment serves as a boundary limiting activity leakage. The containment is steel lined and designed to withstand internal pressure in excess of that resulting from the design-basis LOCA (Chapter 5). All weld seams and penetrations are designed with a double barrier to inhibit leakage. In addition, the weld channel and penetration pressurization system supplies a pressurized nitrogen seal, at a pressure above the containment design pressure, between the double barriers so that if leakage occurred it would be into the containment (Section 6.5). The containment isolation system, Section 5.2, provides a minimum of two barriers in piping penetrating the containment. The isolation valve seal-water system, Section 6.6, provides a water seal at a pressure above containment design pressure in the piping lines that could be a source of leakage and is actuated on the containment isolation signal within 1 min to terminate containment leakage. The containment is designed to leak at a rate of less than 0.1-percent per day at design pressure without including the benefit of either the isolation valve seal-water system or the weld channel and penetration pressurization system. The weld seams and penetrations are pressurized continuously during reactor operation causing zero outleakage through these paths.

14.3.6.1.1 Effectiveness of Spray System for Removal of Airborne Activity

One train of the containment spray system is assumed to operate following the LOCA. The containment sprays are an effective means for removing airborne activity existing as aerosols or

as elemental iodine. As discussed in Appendix 6A, the following spray removal coefficients have been determined for Indian Point Unit 2:

Aerosol Removal

Injection spray mode of operation	4.4 hr^{-1}
Recirculation spray mode of operation	2.25 hr^{-1}

Once a DF of 50 is attained (i.e., when the airborne activity is reduced to 2% of the total activity released to the containment atmosphere), the spray removal coefficient is reduced by a factor of 10. For the Indian Point 2 analysis it is assumed that the sprays are terminated after the DF of 50 is reached for aerosols and that aerosol removal continues after that time due to sedimentation only. Consistent with this assumption, sprays are credited for 3.4 hours following the event.

Elemental Iodine Removal

Injection spray mode of operation	20 hr^{-1}
Recirculation spray mode of operation	5.0 hr^{-1}

Once a DF of 200 is attained (i.e., when the airborne activity is reduced to 0.5% of the total activity released to the containment atmosphere), no additional removal of elemental iodine is assumed.

14.3.6.1.2 [Deleted]

14.3.6.1.3 [Deleted]

14.3.6.1.4 Sedimentation Removal of Particulates

During spray operation credit is taken for sedimentation removal only in the unsprayed portion of the containment. It is assumed that containment spray operation is terminated at 3.4 hours as discussed in Section 14.3.6.1.1. After spray operation is terminated sedimentation is credited throughout the containment.

Based on the Containment Systems Experiments (CSE) which examined the air cleanup experienced through natural transport processes, it was found that a large fraction of the aerosols were deposited on the floor rather than on the walls indicating that sedimentation was the dominant removal process for the test (Reference 86). The CSE tests determined that there was a significant sedimentation removal rate even with a relatively low aerosol concentration. From Reference 86, even at an air concentration of 10 $\mu g/m^3$, the sedimentation removal coefficient was above 0.3 hr^{-1}. With 2.0-percent of particulates remaining airborne at the end of crediting spray removal, there would be in excess of 10,000 $\mu g/m^3$ and an even higher sedimentation rate would be expected. The sedimentation removal coefficient is conservatively assumed to be only 0.1 hr^{-1}. It is also conservatively assumed that sedimentation removal does not continue beyond a DF of 1000.

14.3.6.2 Source Term

The reactor coolant activity is assumed to be released over the first 30 seconds of the accident. However, the activity in the coolant is insignificant compared with the release from the core and is not included in the analysis.

The use of NUREG-1465 (Reference 87) and Regulatory Guide 1.183 (Reference 92) source term modeling results in several major departures from the assumptions used in previous LOCA dose analyses from TID-14844 (Reference 88). Instead of assuming instantaneous melting of the core and release of activity to the containment, the release of activity from the core occurs over a 1.8 hour interval. Also, instead of considering only the release of iodines and noble gases, a wide spectrum of nuclides is taken into consideration. Table 14.3-43 lists the nuclides being considered for the LOCA with core melt (eight groups of nuclides). Table 14.3-43a provides the fission product release fractions and the timing/duration of releases to the containment as assumed in the analysis based on Regulatory Guide 1.183.

Instead of the iodine being primarily in the elemental form, the iodine is mainly in the particulate form (cesium iodide) and the fraction that is in the organic form is much smaller than in the earlier model. The iodine characterization from NUREG-1465 and Regulatory Guide 1.183 is 4.85% elemental, 0.15% organic and 95% particulate. The other groups of nuclides (other than the noble gases) all occur as particulates only.

14.3.6.2.1 Atmosphere Dispersion

The offsite dispersion factors were calculated with the following meteorology and the model described in Reg. Guide 1.4 (Reference 67).

a. Pasqual Type F, 1 m/sec wind speed, nonvarying wind direction, and volumetric building wake correction factor with C = 0.5 and the cross-sectional area of the containment structure for the first 8 hr.

b. From 8 to 24 hr, Pasqual Type F, 1 m/sec wind speed with plume meander in a 22.5-degree sector.

c. From 1 to 4 days, Pasqual Type F and 2 m/sec wind speed with a frequency of 60-percent Pasqual Type D and 3 m/sec wind speed with a frequency of 40-percent, with a meander in the same 22.5-degree sector.

d. From 4 to 30 days, Pasqual Types C, D, and F each occurring 33-1/3-percent of the time with wind speeds of 3 m/sec, 3 m/sec, and 2 m/sec, respectively, with a meander in the same 22.5-degree sector 33-1/3-percent of the time.

The radiological consequences analysis employs the dispersion factors listed in Table 14.3-46 for the site boundary and low population zone.

14.3.6.3 Method of Analysis

The activity leaking from the containment following the accident is calculated for each isotope as a function of time taking into account the core activity, release fractions, removal in containment via sprays and sedimentation (as described previously) and the containment leak rate. The major assumptions and parameters used to determine the doses due to containment leakage

are given in Table 14.3-49. To evaluate the ability to meet the 10 CFR 50.67 limits, the total effective dose equivalent (TEDE) dose was calculated at the site boundary and at the low population zone. Onsite exposure is evaluated in the control room. The TEDE dose is equivalent to the committed effective dose equivalent (CEDE) dose from inhalation of activity plus the effective dose equivalent (EDE) dose from submersion in the activity cloud for the duration of the exposure to the cloud.

14.3.6.3.1 Offsite CEDE Dose

The CEDE dose resulting from activity leaking from the reactor containment following an accident is computed from:

$$D(I,T) = Q(I,T) \cdot DCF(I) \cdot B(T) \cdot \frac{\chi}{Q(x,T)}$$

Where:

- D(I,T) = CEDE dose from isotope I during period T (rem)
- Q(I,T) = activity of isotope I released in time period T (curies)
- DCF(I) = CEDE dose conversion factor for isotope I (rem/curie) (See Table 14.3-45)
- B(T) = breathing rate (m^3/sec)
- χ/Q(x,T)= atmospheric dispersion factor at distance x and during period T (sec/m^3)

14.3.6.3.2 Offsite EDE Dose

For the computation of the offsite EDE doses from cloud immersion, the following equation was used:

$$D(I,T) = Q(I,T) \cdot DCF(I) \cdot \frac{\chi}{Q(x,T)}$$

Where:

- D(I,T) = EDE dose from isotope I during period T (rem)
- Q(I,T) = activity of isotope I released in time period T (curies)
- DCF(I) = EDE dose conversion factor for isotope I (rem-m^3/curie-sec) (See Table 14.3-45)
- χ/Q(x,T) = atmospheric dispersion factor (sec/m^3) at distance x and during period T

14.3.6.4 <u>Containment Leakage NUREG-1465 Core Release Doses</u>

The resultant site boundary dose is 17.8 rem TEDE. The low population zone dose is 12.1 rem TEDE.

The total large break LOCA offsite dose is the combination of the dose for the containment leakage pathway discussed above and the dose for the ECCS recirculation leakage pathway discussed in Section 14.3.6.6.

14.3.6.5 <u>Control Room Dose Evaluations</u>

The control room is modeled as a discrete volume. The filtered and unfiltered inflow to the control room are used to calculate the activity in the control room. The control room parameters modeled in the analysis are presented in Table 14.3-50.

The control room CEDE dose from each isotope for each time period is:

$$D(I,T) = CONC(I,T) \cdot DCF(I) \cdot B(T)$$

Where:

D(I,T) = CEDE dose from isotope I during period T (rem)
CONC(I,T) = concentration of isotope I in the control room (Ci-sec/m^3)
DCF(I) = CEDE dose conversion factor for isotope I (rem/curie)
B(T) = breathing rate (m^3/sec)

The control room EDE dose from each nuclide for each time period is:

$$D(I,T) = \frac{1}{GF} CONC(I,T) \cdot DCF(I)$$

Where:

D(I,T) = inhalation dose from isotope I during period T (rem)
DCF(I) = inhalation dose conversion factor for isotope I (rem-m^3/curie-sec)
CONC(I,T) = concentration of isotope I in the control room (Ci-sec/m^3)
GF = geometry factor, calculated based on Reference 89 using the following equation, where V is the control room volume

$$GF = \frac{1173}{V^{0.338}}$$

The ARCON96 computer code was utilized to analyze the X/Q (atmospheric dispersion factor) values at the control room intake for releases at Indian Point 2. This code was developed by Pacific Northwest National Laboratory for the United States Nuclear Regulatory Commission.

The ARCON96 analysis for Indian Point 2 required calculation of X/Q values for five locations: a containment surface leak, the side of the auxiliary boiler feedwater building, vent stacks on the roof of the auxiliary boiler feedwater building, the containment vent, and the refueling water storage tank. These correspond to potential release points for various accident scenarios. Additional conservatisms were added to the calculations:

1. The initial plume standard deviations used were equal to one-sixth of the width and available height of the containment.
2. The initial horizontal plume dimension for vent releases is the equivalent vent diameter divided by six
3. All vertical velocities were set to zero

The X/Q values calculated for release of activity from the event-specific release point to the control room intake are used to determine the activity available at the control room intake, and are presented in Table 14.3-51.

The accumulated dose to control room operators following the postulated accidents were calculated using the same release, removal and leakage assumptions as the offsite doses. The control room personnel dose calculations includes the direct dose from the radiation cloud outside the control room as well as the inhalation and acute doses from the activity introduced inside the control room. That direct dose takes into account the shielding afforded by the control room walls.

In addition to the dose from activity released from containment, the large break LOCA control

room dose includes a conservative calculation of the direct whole-body gamma dose in the control room from the activity inside the containment. The activity is assumed to be homogeneously distributed within the free volume of the reactor containment. The source intensity as a function of time after the accident is determined considering decay and removal by processes described in Section 14.3.6.1. The direct dose rate in the control room due to the activity dispersed within the containment is calculated based on a point kernel attenuation model. The source region is divided into a number of incremental source volumes and the associated attenuation, gamma ray buildup, and distance through regions between each source point and the control room are computed. The summation of all point source contributions gives the total direct dose rate at the control room.

The control room doses calculated for each of the events are presented in table 14.3-52 and in all cases are less than the 5.0 rem TEDE control room dose limit values of 10 CFR 50.67.

14.3.6.6 External Recirculation

The Indian Point Unit 2 design includes internal recirculation which is to be maintained for the first 6.5 hours following a LOCA. An analysis has been performed to calculate the dose resulting from leakage from the ECCS outside containment after external recirculation is established at 6.5 hours. The analysis models the same core iodine release model as the containment leakage releases discussed in Section 14.3.6.2, the same dose calculation method as discussed in Section 14.3.6.3 and the control room model discussed in Section 14.3.6.5. The offsite dose is calculated using the meteorological dispersion factors discussed in Section 14.3.6.2.1. The analysis considered a leak rate of 4.0 gph. This is double the allowable limit as required by Regulatory Guide 1.183 (Reference 92). This is more than 15 times greater than the estimated SI and RHR system design leakage of 999 cm^3/hr discussed in Section 6.2.3.8 and Table 6.2-9. The leakage is assumed to start at 6.5 hours and continues until 30 days from accident initiation. A conservatively low sump water volume is modeled to maximize the iodine concentration in the leakage.

The calculations were performed using the approach in Regulatory Guide 1.183 (Reference 92) guidance that if the calculated flash fraction is less than 10% or if the water is less than 212^{o}F, then an amount of iodine smaller than 10% of the iodine in the leakage may be used if justified based upon actual sump pH history and ventilation rates. Iodine release fractions have been specifically calculated for external leakage sources (ECCS leakage post LOCA) beginning at 6.5 hours post accident when ECCS flow is directed by procedure to go to portions of the external safety injection system. These calculations are based upon calculated post accident fluid temperatures and pH in sump water, flows and volumes in the primary auxiliary building (PAB), and ventilation flow rates in various areas of the PAB. Leakage is assumed to be at 4 gph. The calculation was performed both with and without the boundary layer effect. The boundary layer effect credits the iodine concentration gradient across the boundary layer at the liquid-gas interface, thus lowering the equilibrium iodine concentration in the gas phase. The calculated values are:

	Fraction of Incoming Iodine Released	
Time Period	With Boundary Layer Effect	Without Boundary Layer Effect
6.5 to 8 hours	0.012	0.12
8 to 24 hours	0.00855	0.0855
1 to 4 days	0.00523	0.0523
4 to 30 days	0.003	0.03

The releases would be subject to filtration by the filtered ventilation system provided for the primary auxiliary building which houses the portions of the ECCS located outside containment. However, filtration of the releases is not credited in the analysis.

In addition to the ECCS leakage to the Primary Auxiliary Building, the potential ECCS back-leakage to the refueling water storage tank (RWST) is also considered during the external recirculation phase of a large break LOCA. The analysis uses the same dose calculation method as discussed above. The analysis considered an ECCS back-leakage rate of 20.0 gpm at 6.5 hours into a large break LOCA.

Since the leakage is initiated at 6.5 hours after the LOCA, it does not contribute to the 2 hour site boundary dose. When boundary layers effects are considered the 30 day low population zone dose is 0.15 rem TEDE and the 30 day control room dose is 0.14 rem TEDE. When the boundary layer effects are neglected the doses increase to 1.5 rem at the low population zone and 1.36 rem in the control room.

The total large break LOCA offsite and control room doses are the combination of the doses for the containment leakage pathway discussed in Sections 14.3.6.4 and 14.3.6.5 and the doses for the ECCS recirculation leakage pathway discussed above.

The remainder of this section discusses the analysis performed prior to implementation of the Regulatory Guide 1.183 dose methodology and is retained for historical purposes.

Indian Point Unit 2 has an internal spilled coolant and injection water recirculation system incorporating two pumps for return of water to the reactor core for decay heat removal after a LOCA. The residual heat removal pumps serve as a backup to these pumps. The residual heat removal compartment and piping is surrounded by 2-ft-thick concrete shield walls. In addition, each residual heat removal compartment is shielded from its adjacent residual heat removal compartment and piping by 2-ft of concrete. Figure 14.3-129 shows the results of an evaluation of direct radiation levels surrounding a 14-in. residual heat removal pipe. The evaluation was based on gap activity, except noble gases, being diluted in the reactor coolant and refueling water volume, which is being recirculated through the pipes. With the 24-in. of concrete provided, the dose levels would be an order of magnitude less than shown for 12-in. of concrete.

As discussed in Section 6.2, design leakage for the external recirculation system was less that 1000 cm^3/hr. Westinghouse performed experiments in which solutions of iodine in sodium hydroxide of pH that would exist in the containment after a loss of coolant were evaporated to dryness. The result was that less that 10^{-3} of the iodine was released. For purpose of conservatism, it was assumed that for a period of 1 hr, 10-percent of the iodine in the leakage was released to atmosphere. Assuming gap iodine activity immediately after the loss of coolant was present in the sump water being recirculated, the offsite thyroid dose for the period was less than 2 mrem. Protection from inhalation dose in the auxiliary building following an accident can be attained by the use of self-contained breathing apparatus during those periods when access is required.

14.3.6.7 Small Break LOCA Radiological Consequences

The radiological consequences resulting from a small break LOCA which is large enough to result in actuation of the containment spray system would be bounded by the Large Break

LOCA analysis. This is true because a small break releases less activity to the containment than that assumed in the large break, but the spray system would function in an identical manner.

An analysis was performed to determine the radiological consequences for a small break LOCA that does not actuate the containment sprays. As a result of the accident, fuel clad damage is assumed to occur. Due to the potential for leakage between the primary and secondary systems, radioactive reactor coolant is assumed to leak from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or the main steam safety valves. Radioactive reactor coolant is also discharged to the containment via the break. A portion of this radioactivity is released through containment leakage to the environment.

In determining the offsite doses following the accident, it is conservatively assumed that all of the fuel rods in the core suffer sufficient damage that all of their gap activity is released. Five percent of the core activity of iodines, noble gases, and alkali metals is assumed to be contained in the pellet-clad gap. The iodine released from the fuel is assumed to be 95% particulate (cesium iodide), 4.85% elemental, and 0.15% organic. These fractions are used for the containment leakage release pathway. However, for the steam generator steaming pathway the iodine in solution is considered to be all elemental and after it is released to the environment the iodine is modeled as 97% elemental and 3% organic.

Conservatively, all the iodine, alkali metals group and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment) when determining doses due to the primary to secondary steam generator tube leakage.

The primary to secondary steam generator tube leak used in the analysis is 150 gpd per steam generator (total of 600 gpd).

When determining the doses due to containment leakage, all of the iodine, alkali metal and noble gas activity is assumed to be in the containment. The design basis containment leak rate of 0.1% per day is used for the initial 24 hours. Thereafter, the containment leak rate is assumed to be one-half the design value, or 0.05% per day. Releases are continued for 30 days from the start of the event.

No credit for activity partitioning is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere. Secondary side releases are terminated when the primary pressure drops below the secondary side pressure.

An iodine partition factor in the steam generators of 0.01 curies/gm steam per curies/gm water is used. This partition factor is also used for the alkali metal activity in the steam generators. This conservatively overstates the release of alkali metal activity via this pathway since their release would be limited by the moisture carryover fraction of 0.0025.

For the containment leakage pathway, no credit is taken for containment spray operation which would remove airborne particulates and elemental iodine. Credit is taken for sedimentation of particulates and deposition of elemental iodine onto containment surfaces. The sedimentation coefficient is assumed to be 0.1 hr^{-1}, the same as credited in the large break LOCA analysis

(see Section 14.3.6.1.4). Deposition removal of elemental iodine is determined using the model described in SRP Section 6.5.2 (Reference 68). The first order deposition removal rate constant for elemental iodine is written as follows:

$$\lambda_e = kA / V$$

where λ_e = Elemental removal rate constant due to deposition, hr^{-1}
k = Mass transfer coefficient = 4.9 m/hr
A = Area available for deposition, ft^2
V = Containment volume, ft^3

Parameters for Indian Point Unit 2 are:

A = 250,000 ft^2
V = 2.61×10^6 ft^3

The resulting deposition removal coefficient is 1.5 hr^{-1}. Consistent with SRP Section 6.5.2, removal of elemental iodine is terminated when a DF of 200 is reached.

The resultant 2 hour site boundary dose is 7.8 rem TEDE. The 30 day low population zone dose is 10.8 rem TEDE. These doses are calculated using the meteorological dispersion factors discussed in Section 14.3.6.2.1. The offsite doses resulting from the accident are less than the 25 rem TEDE limit value of 10 CFR 50.67.

The accumulated dose to the control room operators following the postulated accident was calculated using the same release, removal and leakage assumptions as the offsite dose, and using the control room model discussed in Section 14.3.6.5 and Tables 14.3-50 and 14.3-51. The calculated central control room doses are presented in Table 14.3-52 and are less than the 5.0 rem TEDE control room dose limit values of 10CFR 50.67.

14.3.6.8 Summary and Conclusions

The total large break LOCA offsite and control room doses are the combination of the doses for the containment leakage pathway discussed in Sections 14.3.6.4 and 14.3.6.5 and the doses for the ECCS recirculation leakage pathway discussed in Section 14.3.6.6. With boundary layer effects considered in the ECCS recirculation leakage analysis the total LOCA doses are 17.8 rem TEDE for the limiting 2 hour site boundary dose, 12.25 rem TEDE for the 30 day low population zone dose and 3.68 rem TEDE for the 30 day control room dose. Neglecting boundary layer effects in the ECCS leakage analysis has no impact on the limiting 2 hour site boundary dose, but increases the 30 day low population zone and control room doses to 13.6 rem TEDE and 4.9 rem TEDE, respectively.

The small break LOCA doses are 7.8 rem TEDE for the limiting 2 hour site boundary dose, 10.8 rem TEDE for the 30 day low population zone dose and 3.5 rem TEDE for the 30 day control room dose.

Table 14.3-52 lists the calculated control room doses for all the analyzed accidents and demonstrates that the large break LOCA results in the highest control room doses.

Thus, the doses resulting from large break and small break LOCA the accidents are less than the 25 rem TEDE offsite dose limit and 5.0 rem TEDE control room dose limit values of 10 CFR

50.67. It is concluded that even with very pessimistic assumptions that do not take full credit for the safeguards systems provided, doses after a loss of coolant accident would be within the 10 CFR 50.67 limits.

REFERENCES FOR SECTION 14.3

1. U. S. Nuclear Regulatory Commission, Code of Federal Regulations - Energy, Title 10, Chapter 1, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors."

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TABLE 14.3-1
Large Break LOCA Sequence of Events for Limiting PCT Transient (LOOP)

Event	**Time (sec)**
Start of Transient	0.0
Safety Injection Signal (Pressurizer Pressure)	6.0
Accumulator Injection Begins	10.0
Containment Spray Heat Removal System Starts (Offsite Power Available)*	20.0
End of Blowdown	28.0
Containment Fan Cooler Heat Removal System Starts (Offsite Power Available)*	30.0
Accumulator Empty	39.0
Bottom of core Recovery	40.0
Safety Injection Begins	51.0
PCT Occurs	123.0
PCT Elevation Quench	330.0
End of Transient	500.0

* Offsite power available is conservatively assumed in containment modeling, as this minimizes containment pressure.

TABLE 14.3-2
Large-Break Containment Data

Net free volume	2.61 x 10^6-ft^3
Initial conditions	
Pressure	14.7 psia
Temperature	80°F
Refueling water storage tank temperature	35°F
Service water temperature	28°F
Outside temperature	-20°F
Spray system	
Number of pumps operating	2
Total flow rate	6712 gpm
Actuation time	20 sec
Safeguards fan coolers	
Number of fan coolers operating	5
Fastest postaccident initiation of fan coolers	30 sec

.

TABLE 14.3-2 (Cont.)
Large-Break Containment Data

Structural heat sinks		
	Thickness (in.)	Area (ft^2)
1.	0.007 paint, 0.375 steel, 54.0 concrete	38,584
2.	0.007 paint, 0.5 steel, 42.0 concrete	28,613
3.	12.0 concrete	15,000
4.	0.375 stainless steel, 12.0 concrete	10,000
5.	12.0 concrete	61,000
6.	0.5 steel	68,792
7.	0.007 paint, 0.375 steel	81,704
8.	0.25 steel	27,948
9.	0.007 paint, 0.1875 steel	69,800
10.	0.125 steel	3,000
11.	0.138 steel	22,000
12.	0.0625 steel	10,000
13.	0.019 stainless steel, 1.25 insulation, 0.75 steel, 54.0 concrete	785
14.	0.019 stainless steel, 1.25 insulation 0.5 steel, 54.0 concrete	6,849
15.	0.025 stainless steel, 1.5 insulation 0.5 steel, 54.0 concrete	3,792
16.	0.025 stainless steel, 1.5 insulation 0.375 steel, 54.0 concrete	4362
17.	0.007 paint, 0.375 steel, 54.0 concrete	7,100
18.	0.025 stainless steel, 1.5 insulation, 0.5 steel, 54.0 concrete	24
19.	0.0334 stainless steel	53457

TABLE 14.3-2A
Best-Estimate Large Break LOCA Mass and Energy Releases from BCL Used for COCO Calculation at Selected Time Points for Indian Point Unit 2

	M&E from Vessel Side BCL		M&E from Loop Side BCL	
Time (seconds)	**Mass Flow (lbm/sec.)**	**Energy Flow (BTU/sec)**	**Mass Flow (lbm/sec)**	**Energy Flow (BTU/sec)**
0.0	8623	4591667	-8	0
0.5	26190	13828421	54917	29053714
1.0	25923	13848634	51797	27373685
1.5	24961	13659035	47428	25064545
2.0	22548	12660066	42978	22737052
4.0	11656	7400663	27704	14808982
6.0	7208	5577153	21746	11977783
8.0	5777	4809423	17896	10226020
10.0	4630	4000111	13209	8027556
12.0	3700	3204590	9758	6010368
14.0	2714	2398163	10389	4983225
16.0	1517	1519547	8428	3404399
18.0	854	909753	6549	2103497
20.0	458	513981	6494	1569218
25.0	104	127461	1487	244532
50.0	108	137218	842	349322
75.0	51	64361	156	132775
100.0	40	51027	94	74727
125.0	48	60750	164	119909
150.0	46	58576	175	130207
175.0	43	54090	146	114152
200.0	48	60379	174	141615
225.0	46	58331	160	128531
250.0	43	53809	173	112595
275.0	48	58978	165	137987
300.0	51	62270	250	141408
325.0	53	63195	227	152226
350.0	74	81605	248	145791
375.0	49	59722	221	134873
400.0	53	61745	170	119837
425.0	46	55114	154	126105
450.0	42	50763	138	114950
475.0	79	82319	171	136287
500.0	69	73949	125	96929

TABLE 14.3-3
Key LOCA Parameters and Initial Transient Assumptions for
Indian Point Unit 2

Parameter	Initial Transient	Range / Uncertainty
1.0 Plant Physical Description		
a. Dimensions	Nominal	Sample (3)
b. Flow resistance	Nominal	Sample (3)
c. Pressurizer location	Opposite broken loop	Bounded
d. Hot assembly location	Under limiting location	Bounded
e. Hot assembly type	15 x 15 upgraded, ZIRLO™ clad and Non-IFBA	Bounded
f. SG tube plugging level*	High (10%)	Bounded(1)
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
a. Core average linear heat rate (AFLUX)	Nominal – Based on 100% of uprated power (3216 MWt)	Sample (3)
b. Hot Rod Peak Linear heat rate (PLHR)*	Derived from desired Tech Spec (TS) limit F_Q = 2.5 and maximum baseload F_Q = 2.0	Sample (3)
c. Hot rod average linear heat rate (HRFLUX)*	Derived from TS $F_{\Delta H}$ = 1.7	Sample (3)
d. Hot assembly average heat rate (HAFLUX)	HRFLUX/1.04	Sample (3)
e. Hot assembly peak heat rate (HAPHR)	PLHR/1.04	Sample (3)
f. Axial power distribution (PBOT, PMID)	Figure 14.3-20	Sample (3)
g. Low power region relative power (PLOW)	0.3	Bounded (2)
h. Cycle burnup	~100 MWD/MTD	Sample (3)
i. Prior operating history	Equilibrium decay heat	Bounded
j. Moderator Temperature Coefficient (MTC)	Tech Spec Maximum (0)	Bounded
k. HFP boron	800 ppm	Generic

TABLE 14.3-3 (Cont.)
Key LOCA Parameters and Initial Transient Assumptions for
Indian Point Unit 2

Parameter	Initial Transient	Range / Uncertainty
2.2 Fluid Conditions		
a. T_{avg}	High Nominal T_{avg} =572°F	Bounded [1], Sample [3]
b. Pressurizer pressure	Nominal (2250.0 psia)	Sample [3]
c. Loop flow	80,700 gpm	Bounded [4]
d. T_{UH}	T_{Hot}	0
e. Pressurizer level	Nominal at high Tavg	0
f. Accumulator temperature	Nominal (105 °F)	Sample [3]
g. Accumulator pressure	Nominal (656.2 psia)	Sample [3]
h. Accumulator liquid volume	Nominal (795 ft^3)	Sample [3]
i. Accumulator line resistance	Nominal	Sample [3]
j. Accumulator boron	Minimum (2000 ppm)	Bounded
3.0 Accident Boundary Conditions		
a. Break location	Cold leg	Bounded
b. Break type	Guillotine (DEGCL)	Sample [3]
c. Break size	Nominal (cold leg area)	Sample [3]
d. Offsite power	Available (RCS pumps running)	Bounded[2]
e. Safety injection flow	Minimum (Table 14.3-5B)	Bounded
f. Safety injection temperature	Nominal (725 °F)	Sample [3]
g. Safety injection delay	Max delay (38.0 sec)	Bounded
h. Containment pressure	Bounded – Lower (conservative) than pressure curve shown in figure 14.3-22	Bounded
i. Single failure	ECCS: Loss of 1 SI train; Containment pressure: all trains operational	Bounded
j. Control rod drop time	No control rods	Bounded

TABLE 14.3-3 (Cont.)
Key LOCA Parameters and Initial Transient Assumptions for
Indian Point Unit 2

Parameter	Initial Transient	Range / Uncertainty
4.0 Model Parameters		
a. Critical Flow	Nominal (C_D = 1.0)	Sample [3]
b. Resistance uncertainties in broken loop	Nominal (as coded)	Sample [3]
c. Initial stored energy/fuel rod behavior	Nominal (as coded)	Sample [3]
d. Core heat transfer	Nominal (as coded)	Sample [3]
e. Delivery and bypassing of ECC	Nominal (as coded)	Conservative
f. Steam binding/entrainment	Nominal (as coded)	Conservative
g. Noncondensable gases/accumulator nitrogen	Nominal (as coded)	Conservative
h. Condensation	Nominal (as coded)	Sample [3]

Notes:
1. Confirmed to be limiting
2. High PLOW of 0.8 confirmed to be limiting; Loss-Of-Offsite-Power confirmed to be limiting
3. Sampling distribution defines in Table 5.2-1 of Reference 79
4. Assumed to be result of loop resistance uncertainty

* Fuel pellet thermal conductivity degradation evaluations esulted in a reduction of the maximum steam generator tube plugging from 10% to 5%, a reduction of the Fq from 2.5 to 2.3, and a reduction of Fdh from 1.70 to 1.65.

TABLE 14.3-4
Limiting Large Break PCT and Oxidation Results for Indian Point Unit 2

Parameter	Result
95/95 Peak Clad Temperature (PCT)	1,962°F*
95/95 Maximum Cladding Oxidation (LMO)	<2152°F_1
95/95 Maximum Core-wide Oxidation (CWO)	<13%

* The PCT result provided do not reflect any individual PCT assessments discussed in Section 14.3.3.3.4

TABLE 14.3-5A
Plant Operating Range Allowed by the Best-Estimate Large Break LOCA Analysis for Indian Point Unit 2

Parameter		Operating Range
1.0	Plant Physical Description	
	a) Dimensions	No in-board assembly grid deformation during LOCA + SSE
	b) Flow resistance	N/A
	c) Pressurizer location	N/A
	d) Hot assembly location	Anywhere in core interior (149 locations)[(1)]
	e) Hot assembly type	15 X 15 Upgraded fuel design
	f) SG tube plugging level*	≤ 10%
	g) Fuel assembly type	15 X 15 upgraded fuel with ZIRLO™ cladding, non-IFBA or IFBA [(2)]
2.0	Plant Initial Operating Conditions	
	2.1 Reactor Power	
	a) Core average linear heat rate	Core power $\leq$ 102% of 3216 MWt
	b) Peak linear heat rate*	$F_Q \leq 2.5$
	c) Hot rod average linear heat rate*	$F_{\Delta H} \leq 1.70$
	d) Hot assembly average linear heat rate*	$P_{HA} \leq 1.7 / 1.04$
	e) Hot assembly peak linear heat rate*	$F_{Q(HA)} \leq 2.5 / 1.04$
	f) Axial power distribution (PBOT, PMID)	Figure 14.3-21
	g) Low power region relative power (PLOW)	$0.3 \leq$ PLOW ≤ 0.8
	h) Hot assembly burnup	$\leq$ 75,000 MWD/MTU, lead rod
	i) Prior operating history	All normal operating histories
	j) MTC	$\leq$ 0 at hot full power (HFP)
	k) HFP boron (minimum)	800 ppm (at BOL)

TABLE 14.3-5A (Cont.)
Plant Operating Range Allowed by the Best-Estimate Large Break LOCA Analysis
for Indian Point Unit 2

Parameter		Operating Range
	2.2 Fluid Conditions	
	a) T_{avg}	$549 - 3.3°F \leq T_{avg} \leq 572 + 3.3°F$ (2)
	b) Pressurizer pressure	$2250 - 25$ psia $\leq P_{RCS} \leq 2250 + 25$ psia (3)
	c) Loop flow	≥ 80,700 gpm/loop
	d) T_{UH}	Current upper internals, T_{Hot} UH
	e) Pressurizer level	Normal level, automatic control
	f) Accumulator temperature	$80°F \leq T_{ACC} \leq 130°F$
	g) Accumulator pressure	612.7 psia $\leq P_{ACC} \leq$ 699.7 psia
	h) Accumulator liquid volume	723 $ft^3 \leq V_{acc} \leq$ 875-ft^3
	i) Accumulator fL/D	Current line configuration
	j) Minimum ECC boron	≥ 2000 ppm
3.0	Accident Boundary Conditions	
	a) Break location	N/A
	b) Break type	N/A
	c) Break size	N/A
	d) Offsite power	Available or Loss-Of-Offsite-Power (LOOP)
	e) Safety injection flow	Table 14.3-5B
	f) Safety injection temperature	35°F ≤ SI Temp ≤110°F
	g) Safety injection delay	≤ 38 seconds (with offsite power) ≤ 45 seconds (with LOOP)
	h) Containment pressure	Figure 14.3-22, raw data in Table 14.3-2 and M&E releases in Table 14.3-2A
	i) Single failure	Loss of one ECCS train
	j) Control rod drop time	N/A

NOTE:
(1) 44 peripheral locations (Figure 3.2-8 from Reference 79) will not physically be lead power assembly.
(2) Include -3 (bias); Bias sign correction: "+" means indicated value is higher than actual and "-" means indicated value is lower than actual.

(3) Include -3, + 12 (bias); Bias sign correction: "+" means indicated value is higher than actual and "-" means indicated value is lower than actual.

* Fuel pellet thermal conductivity degradation evaluations resulted in a reduction of the maximum steam generator tube plugging from 10% to 5%, a reduction of the Fq from 2.5 to 2.3, and a reduction of Fdh from 1.70 to 1.65.

TABLE 14.3-5B
Total Minimum Injected Safety Injection Flow Used in Best-Estimate Large Break LOCA Analysis for Indian Point Unit 2

RCS Pressure (psig)	**Flow Rate (gpm)**
0	2330.03
10	1962.54
20	1636.46
30	1333.48
40	1041.40
50	770.92
60	691.73
100	678.21
200	641.35
300	603.25
400	564.49
500	525.07
600	469.87
700	396.91
800	335.54
900	304.38
1000	235.22
1100	134.33
1200	23.56
1300	0.0

TABLE 14.3-6
Broken Loop Accumulator and Safety Injection Spill to Containment During Blowdown
Deleted

TABLES 14.3-7 through 14.3-10
Deleted

TABLE 14.3-11
Initial Parameters For Small Break LOCA Analysis

Licensed Core Power (MWt) (includes 2% calorimetric uncertainty)	3281
Total Peaking Factor, Fq	2.5
Axial Offset, %	13
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.70
Maximum Assembly Average Power, PHA	1.51
Fuel Assembly Array	15 x 15 upgraded with IFMs
Nominal Accumulator Water Volume, ft^3	795
Accumulator Tank Volume, ft^3	1100
Minimum Accumulator Gas Pressure, psia	613
Loop Flow (gpm)	80700
Vessel Inlet Temperature, °F	537.451
Vessel Outlet Temperature, °F	606.549
RCS Pressure with Uncertainty, psia	2310
Steam Pressure, psia	735.645
Steam Generator Tube Plugging, %	10
Maximum Refueling Water Storage Tank Temperature, ºF	110
Maximum Condensate Storage Tank Temperature, ºF	120
Non-IFBA Fuel Backfill Pressure, psig	275
Reactor Trip Setpoint, psia	1860
Safety Injection Signal Setpoint, psia	1715
Safety Injection Delay Time, sec.	25
Signal Processing Delay and Rod Drop Time, sec.	4.7 (2.0 + 2.7)
Feedwater Trip Processing Delay Time, sec.	2
Time for Main Feedwater Flow Coastdown, sec.	8
Auxiliary Feedwater Flow – gpm (1)	380
Auxiliary Feedwater Pump Start Delay Time, sec.	60
Maximum Loop Specific Purge Volume, ft^3	268.8

NOTE:
(1) The flow from one motor-driven Auxiliary Feedwater Pump is modeled.

TABLE 14.3-12
Small - Break LOCA Time Sequence of Events

EVENT	Break Size		
	2.0 Inch	3.0 Inch	4.0 Inch
Break Initiation, sec.	0.0	0.0	0.0
Reactor Trip Signal, sec.	43.3	18.2	10.4
Safety Injection Signal, sec.	61.0	26.4	14.9
Top of Core Uncovered, sec.	1711	629	692
Accumulator Injection Begins, sec.	NA	1689	850
Peak Clad Temperature Occurs, sec.	1967	1308	955
Top of Core Recovered, sec.	3854	1924	1170

TABLE 14.3-13
Small Break LOCA Analysis Results

RESULT	Break Size		
	2.0 Inch	3.0 Inch	4.0 Inch
Peak Clad Temperature, °F	938	1028	878
Peak Clad Temperature Location, ft.	10.75	11.00	11.00
Local Zr/H_2O Reaction (max), %	<17	<17	<17
Local Zr/H_2O Reaction Location, ft.	11.25	11.00	11.25
Total Zr/H_2O Reaction, %	<1.0	<1.0	<1.0
Hot Rod Burst Time, seconds	NA	NA	NA
Hot Rod Burst Location, ft.	NA	NA	NA

TABLE 14.3-14
Internals Deflections Under Abnormal Operation

	Allowable Limit[1] (inches)	No Loss-of-Function Limit[1] (inches)
Upper barrel, expansion/compression (to ensure sufficient inlet flow area/and to prevent the barrel from touching any guide tube to avoid disturbing the rod cluster control guide structure)	Inward - 4.1 Outward - 1.0	8.2 1.0
Upper package, axial deflection (to maintain the control rod guide structure geometry)[2,3]	0.100	0.150
Rod cluster control guide tube, deflection as a beam (to be consistent with conditions under which ability to trip has been tested)[3]	1.0	1.60/1.75
Fuel assembly thimbles, cross-section distortion (to avoid interference between the control rods and the guides)[3]	0.036	0.072

Notes:

1. The deflection limit values given above correspond to stress levels for the internals structure well below the limiting criteria given by the collapse curves in WCAP-5890 (Reference 30). Consequently, for the internals the geometric limitations established to ensure safe shutdown capability are more restrictive than those given by the failure stress criteria.
2. See Reference 26.
3. See Reference 27.

TABLE 14.3-15
SYSTEM PARAMETERS FOR 3216 MWt

Parameters	Value
RCS Pressure (psia) (with 60 psi uncertainty)	2310
Core Thermal Power (MWt) (without uncertainties)	3216
Reactor Coolant System Total Flowrate (lbm/sec)	34,250
Vessel Outlet Temperature (°F) (with uncertainty)	613.3
Core Inlet Temperature (°F) (with uncertainty)	545.7
Vessel Average Temperature (°F)	579.5
Initial Steam Generator Steam Pressure (psia)	788
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	104,300.1
Assumed Maximum Containment Backpressure (psia)	61.7
Accumulator	
Water Volume (ft^3) per accumulator (including line volume)	770
N_2 Cover Gas Pressure (psia)	700
Temperature (°F)	130
Safety Injection Delay, total (sec) (from beginning of event) (Minimum ECCS case)	49.1
(Maximum ECCS case)	45

TABLE 14.3-16 (Sheet 1 of 4)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	BREAK PATH NO.1*		BREAK PATH NO.2**	
SECONDS	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC
.00000	.0	.0	.0	.0
.00102	84849.3	45618.7	38460.5	20632.5
.00204	40171.1	21550.8	39870.6	21387.9
.00312	40159.2	21545.3	39615.5	21250.0
.101	39922.0	21501.5	19727.2	10572.6
.201	40974.9	22258.1	22318.8	11974.5
.302	44538.4	24465.2	23349.8	12534.5
.402	44682.9	24869.9	23496.8	12619.0
.502	43717.2	24675.3	23121.9	12423.6
.601	44127.3	25230.0	22645.7	12173.7
.701	43629.7	25226.5	22265.5	11975.3
.801	42265.6	24676.5	22065.9	11873.3
.902	40921.3	24115.0	21964.7	11823.2
1.00	39778.0	23661.6	21916.4	11800.6
1.10	38622.2	23215.4	21868.9	11777.7
1.20	37319.2	22690.8	21834.5	11761.1
1.30	35944.7	22103.1	21808.2	11748.2
1.40	34695.2	21548.2	21802.4	11746.0
1.50	33741.2	21128.7	21829.4	11761.1
1.60	32976.7	20806.2	21824.2	11758.5
1.70	32268.2	20512.2	21705.1	11693.9
1.80	31535.6	20200.6	21555.6	11612.9
1.90	30716.4	19828.7	21402.4	11530.0
2.00	29976.9	19501.2	21256.3	11451.0
2.10	29196.0	19140.5	21112.8	11373.8
2.20	28344.1	18723.1	20957.2	11290.1
2.30	27335.3	18191.3	20784.2	11197.1
2.40	26135.3	17523.8	20600.6	11098.5
2.50	24599.3	16613.7	20405.2	10993.7
2.60	22535.4	15316.1	20199.6	10883.6
2.70	21036.9	14396.0	19989.1	10771.0
2.80	20639.2	14205.3	19798.0	10668.9
2.90	19987.2	13800.9	19602.5	10564.7
3.00	19522.8	13516.0	19407.4	10460.8
3.10	19376.6	13446.0	19204.5	10352.8
3.20	19201.3	13342.5	18988.4	10237.7
3.30	18810.9	13089.4	18759.2	10115.5
3.40	18480.2	12877.9	18531.2	9994.0
3.50	18122.3	12639.8	18312.4	9877.7
3.60	17692.9	12347.4	18094.2	9761.8
3.70	17211.2	12017.4	17873.1	9644.3
3.80	16724.6	11684.5	17658.8	9530.6
3.90	16264.9	11369.3	17458.8	9424.7

TABLE 14.3-16 (Sheet 2 of 4)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
SECONDS	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC
4.00	15820.9	11062.9	17267.5	9323.6
4.20	14978.6	10480.2	16898.7	9128.8
4.40	14290.6	10002.9	16558.4	8949.6
4.60	13707.5	9594.2	16243.5	8784.0
4.80	13236.1	9259.0	15953.9	8632.0
5.00	12820.2	8956.5	15678.5	8487.5
5.20	12481.3	8705.1	15423.5	8353.9
5.40	12234.8	8509.6	15184.4	8228.7
5.60	12013.2	8329.5	14948.5	8105.1
5.80	11845.2	8184.2	14736.2	7994.2
6.00	11685.0	8040.9	14523.9	7883.0
6.20	11581.6	7932.9	14331.5	7782.7
6.40	11553.1	7869.2	14292.3	7767.5
6.60	12293.5	8327.7	14710.1	8000.1
6.80	11952.9	8084.8	14733.3	8016.3
7.00	10706.7	7749.8	14578.9	7935.9
7.20	9189.8	7146.8	14522.0	7909.0
7.40	8892.9	6975.6	14368.8	7828.6
7.60	8891.4	6931.8	14262.7	7774.8
7.75	8873.2	6896.3	14156.0	7719.6
7.80	8859.4	6880.4	14108.8	7694.8
8.00	8797.6	6810.4	13887.9	7577.7
8.20	8769.7	6749.4	13661.1	7457.0
8.40	8796.7	6713.6	13513.7	7379.2
8.60	8840.2	6682.9	13477.8	7360.1
8.80	8856.5	6634.0	13353.8	7289.4
9.00	8859.7	6593.2	13187.4	7194.4
9.20	8823.9	6536.6	13058.2	7120.2
9.40	8768.8	6472.5	12930.8	7047.5
9.60	8705.1	6398.9	12780.7	6962.8
9.80	8632.0	6315.4	12629.7	6878.0
10.0	8548.7	6227.9	12492.6	6801.4
10.2	8442.2	6132.4	12355.1	6724.5
10.401	8328.6	6042.5	12216.2	6646.9
10.402	8328.1	6042.1	12215.6	6646.6
10.403	8327.4	6041.6	12214.9	6646.2
10.6	8200.0	5951.8	12080.4	6571.1
10.8	8065.0	5865.6	11944.4	6495.3
11.0	7923.0	5780.9	11806.4	6418.6
11.2	7779.9	5700.5	11668.3	6342.2
11.4	7631.7	5620.8	11530.0	6265.9
11.6	7480.7	5541.8	11389.2	6188.4
11.8	7327.0	5463.6	11250.5	6112.2

TABLE 14.3-16 (Sheet 3 of 4)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
SECONDS	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC
12.0	7176.5	5390.8	11111.5	6036.0
12.2	7026.3	5316.4	10968.4	5957.5
12.4	6883.8	5243.8	10826.9	5880.1
12.6	6747.1	5172.2	10685.1	5802.7
12.8	6615.8	5101.2	10544.3	5726.0
13.0	6488.9	5030.5	10403.6	5649.5
13.2	6366.6	4960.4	10263.5	5573.5
13.4	6249.0	4891.3	10125.0	5498.4
13.6	6135.2	4822.6	9987.3	5423.8
13.8	6026.2	4755.4	9853.5	5351.4
14.0	5919.5	4688.6	9716.7	5277.2
14.2	5816.8	4623.7	9595.4	5206.8
14.4	5713.0	4557.7	9502.1	5137.0
14.6	5602.6	4486.4	9411.0	5054.8
14.8	5476.2	4402.2	9352.2	4978.2
15.0	5328.9	4296.1	9293.2	4893.3
15.2	5176.5	4174.4	9247.0	4809.1
15.4	5038.0	4049.0	9247.1	4747.4
15.6	4932.8	3939.5	9212.9	4671.6
15.8	4851.8	3846.7	9144.7	4586.4
16.0	4778.1	3768.8	9017.4	4479.5
16.2	4704.7	3700.7	8935.1	4400.9
16.4	4630.2	3639.4	8875.9	4338.7
16.6	4554.4	3583.5	8757.9	4251.9
16.8	4479.1	3533.9	8641.2	4168.4
17.0	4403.7	3489.6	8576.6	4112.2
17.2	4326.9	3449.9	8497.4	4051.1
17.4	4248.8	3415.1	8359.4	3964.2
17.6	4169.2	3386.2	8225.0	3880.8
17.8	4088.8	3361.4	8059.9	3783.8
18.0	4003.3	3338.9	8074.0	3771.4
18.2	3911.2	3320.1	7968.5	3702.9
18.4	3816.0	3306.6	7735.9	3574.5
18.6	3711.0	3295.4	7557.1	3468.0
18.8	3599.3	3288.9	7489.2	3409.1
19.0	3476.9	3281.6	7426.3	3351.2
19.2	3339.5	3274.3	7291.6	3261.6
19.4	3112.8	3201.9	7025.2	3115.2
19.6	2872.0	3112.9	6504.2	2859.2
19.8	2656.5	3017.4	5888.0	2563.0
20.0	2468.0	2906.1	5468.7	2349.1
20.2	2308.0	2789.5	5535.0	2328.4

TABLE 14.3-16 (Sheet 4 of 4)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
SECONDS	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC
20.4	2123.5	2598.3	6023.0	2465.5
20.6	1956.7	2408.2	6647.5	2649.6
20.8	1812.0	2238.1	6252.1	2446.3
21.0	1685.8	2087.7	5729.5	2214.5
21.2	1569.8	1948.2	5441.1	2075.4
21.4	1456.6	1811.1	5220.4	1957.8
21.6	1354.4	1686.8	5013.5	1843.9
21.8	1255.2	1565.8	4785.1	1723.3
22.0	1167.3	1458.1	4545.6	1601.6
22.2	1080.0	1351.1	4308.3	1484.2
22.4	1007.1	1261.7	4080.5	1374.5
22.6	925.4	1160.7	3859.9	1271.9
22.8	867.4	1089.4	3646.6	1176.7
23.0	825.3	1037.2	3444.0	1089.3
23.2	789.4	992.7	3240.0	1005.3
23.4	746.7	939.5	3027.4	922.4
23.6	698.4	879.3	2818.3	844.1
23.8	652.2	821.7	2632.9	775.9
24.0	605.7	763.5	2411.8	700.2
24.2	558.3	704.1	2167.0	620.5
24.4	510.1	643.7	1892.7	535.3
24.6	461.8	583.1	1579.6	442.0
24.8	413.7	522.6	1221.0	338.7
25.0	364.5	460.6	822.5	226.8
25.2	313.9	396.9	427.8	117.6
25.4	261.3	330.6	101.7	28.0
25.6	207.1	262.2	.0	.0
25.8	156.1	197.8	.0	.0
26.0	102.9	130.6	.0	.0
26.2	40.0	50.9	.0	.0
26.4	.0	.0	.0	.0

Notes:
- * mass and energy exiting the SG side of the break
- ** mass and energy exiting the pump side of the break

TABLE 14.3-16a (Sheets 1, 2, 3, & 4 of 4)
Deleted

TABLE 14.3-16b (Sheets 1, 2, 3, 4 & 5 of 5)
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TABLE 14.3-16c (Sheets 1 & 2 of 2)
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TABLE 14.3-16d
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TABLE 14.3-16e
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TABLE 14.3-16f (Sheets 1 & 2 of 2)
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TABLE 14.3-16g
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TABLE 14.3-16h
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TABLE 14.3-16i
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TABLE 14.3-17 (Sheet 1 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
REFLOOD MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
SECONDS	FLOW LBM/SEC	ENERGY THOUSAND SBTU/SEC	FLOW LBM/SEC	ENERGY THOUSAND S BTU/SEC
26.4	.0	.0	.0	.0
26.9	.0	.0	.0	.0
27.1	.0	.0	.0	.0
27.2	.0	.0	.0	.0
27.3	.0	.0	.0	.0
27.4	.0	.0	.0	.0
27.4	.0	.0	.0	.0
27.5	31.9	37.5	.0	.0
27.6	13.9	16.4	.0	.0
27.7	12.9	15.2	.0	.0
27.8	18.5	21.8	.0	.0
27.9	23.4	27.6	.0	.0
28.0	29.8	35.1	.0	.0
28.1	35.0	41.2	.0	.0
28.2	39.7	46.7	.0	.0
28.3	44.2	52.1	.0	.0
28.4	47.1	55.5	.0	.0
28.5	51.2	60.3	.0	.0
28.7	54.5	64.2	.0	.0
28.8	58.2	68.6	.0	.0
28.84	60.4	71.1	.0	.0
28.9	61.2	72.1	.0	.0
29.0	64.6	76.1	.0	.0
29.1	67.4	79.4	.0	.0
29.2	70.6	83.1	.0	.0
29.3	73.2	86.2	.0	.0
29.4	75.6	89.1	.0	.0
30.4	97.2	114.5	.0	.0
31.4	115.0	135.6	.0	.0
32.4	130.5	153.9	.0	.0
33.4	144.3	170.2	.0	.0
33.7	148.5	175.2	.0	.0
34.4	279.2	329.8	2803.8	441.0
35.5	382.8	453.1	4084.0	671.4
36.5	380.7	450.7	4057.5	673.4
37.5	374.3	443.1	3986.8	665.2
38.5	367.8	435.4	3914.7	656.5

TABLE 14.3-17 (Sheet 2 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
REFLOOD MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
SECONDS	FLOW LBM/SEC	ENERGY THOUSAND SBTU/SEC	FLOW LBM/SEC	ENERGY THOUSAND S BTU/SEC
38.7	366.6	433.8	3900.4	654.8
39.5	361.5	427.8	3843.7	647.9
40.5	355.4	420.5	3774.2	639.4
41.5	349.5	413.5	3706.6	631.0
42.5	343.8	406.7	3640.9	622.9
43.5	338.3	400.2	3577.1	614.9
44.5	333.0	393.9	3515.2	607.2
44.7	332.0	392.6	3503.0	605.7
45.5	327.9	387.8	3455.1	599.8
46.5	412.6	487.9	233.8	276.9
47.5	566.9	673.5	318.8	379.0
48.5	551.8	655.3	310.6	369.1
49.5	470.2	557.7	326.3	288.5
50.3	446.6	529.3	314.3	272.8
50.5	444.3	526.6	313.1	271.3
51.5	434.2	514.5	307.9	264.6
52.5	424.5	502.9	302.9	258.3
53.5	415.1	491.7	298.1	252.1
54.5	405.9	480.7	293.3	246.1
55.5	397.0	470.1	288.7	240.3
56.5	388.2	459.7	284.2	234.6
56.6	387.4	458.6	283.8	234.0
57.5	379.7	449.5	279.8	229.0
58.5	371.4	439.5	275.6	223.6
59.5	363.2	429.8	271.4	218.4
60.5	355.2	420.3	267.3	213.2
61.5	347.4	411.0	263.3	208.2
62.5	339.8	401.9	259.4	203.3
63.5	332.3	393.0	255.5	198.5
64.5	324.9	384.3	251.8	193.8
65.5	317.8	375.7	248.2	189.2
66.5	310.8	367.4	244.6	184.8
67.5	303.9	359.3	241.1	180.4
68.5	297.2	351.3	237.7	176.2
69.5	290.7	343.5	234.4	172.0
70.5	284.3	336.0	231.1	168.0
71.5	278.1	328.6	228.0	164.1
72.4	272.6	322.1	225.2	160.7

TABLE 14.3-17 (Sheet 3 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
REFLOOD MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
SECONDS	FLOW LBM/SEC	ENERGY THOUSAND SBTU/SEC	FLOW LBM/SEC	ENERGY THOUSAND S BTU/SEC
72.5	272.0	321.4	224.9	160.3
73.5	266.1	314.3	221.9	156.6
74.5	260.3	307.5	219.0	153.0
75.5	254.7	300.9	216.2	149.5
76.5	249.3	294.4	213.5	146.2
77.5	244.0	288.2	210.8	142.9
78.5	238.9	282.0	208.2	139.7
79.5	233.9	276.1	205.7	136.6
80.5	229.0	270.3	203.3	133.6
81.5	224.2	264.7	200.9	130.7
82.5	219.6	259.2	198.6	127.9
83.5	215.1	253.9	196.4	125.2
84.5	210.8	248.8	194.3	122.5
85.5	206.6	243.8	192.2	120.0
86.5	202.5	239.0	190.2	117.5
87.5	198.6	234.3	188.2	115.1
89.5	191.0	225.4	184.5	110.6
91.5	184.0	217.0	181.0	106.4
93.5	177.4	209.3	177.8	102.4
94.6	174.0	205.2	176.2	100.4
95.5	171.3	202.0	174.9	98.8
97.5	165.6	195.3	172.1	95.5
99.5	160.3	189.1	169.6	92.4
101.5	155.5	183.3	167.2	89.5
103.5	151.0	178.0	165.1	86.9
105.5	146.8	173.1	163.1	84.5
107.5	143.0	168.6	161.3	82.3
109.5	139.5	164.5	159.7	80.3
111.5	136.4	160.8	158.2	78.5
113.5	133.5	157.4	156.8	76.9
115.5	130.8	154.3	155.6	75.4
117.5	128.5	151.4	154.5	74.0
119.5	126.3	148.9	153.5	72.8
121.5	124.4	146.6	152.6	71.7
123.5	122.6	144.6	151.8	70.8
125.0	121.5	143.2	151.2	70.1
125.5	121.1	142.8	151.1	69.9
127.5	119.7	141.1	150.4	69.1

TABLE 14.3-17 (Sheet 4 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
REFLOOD MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
	FLOW	ENERGY	FLOW	ENERGY
SECONDS	LBM/SEC	THOUSAND SBTU/SEC	LBM/SEC	THOUSAND S BTU/SEC
129.5	118.5	139.7	149.8	68.4
131.5	117.4	138.4	149.3	67.8
133.5	116.4	137.2	148.9	67.3
135.5	115.6	136.3	148.5	66.8
137.5	114.9	135.4	148.2	66.4
139.5	114.2	134.6	147.9	66.0
141.5	113.7	134.0	147.6	65.7
143.5	113.2	133.4	147.4	65.4
145.5	112.8	133.0	147.2	65.2
147.5	112.5	132.6	147.0	65.0
149.5	112.3	132.4	146.9	64.9
151.5	112.1	132.2	146.8	64.8
153.5	112.0	132.0	146.8	64.7
155.5	111.9	131.9	146.7	64.6
157.5	111.8	131.8	146.7	64.6
159.5	111.8	131.8	146.6	64.5
161.1	111.8	131.7	146.6	64.5
161.5	111.8	131.8	146.6	64.5
163.5	111.8	131.8	146.6	64.5
165.5	111.8	131.8	146.6	64.5
167.5	111.9	131.9	146.7	64.6
169.5	112.0	132.0	146.7	64.6
171.5	112.1	132.2	146.7	64.6
173.5	112.2	132.3	146.8	64.7
175.5	112.4	132.5	146.8	64.8
177.5	112.5	132.6	146.9	64.8
179.5	112.7	132.8	146.9	64.9
181.5	112.9	133.0	147.0	65.0
183.5	113.0	133.3	147.1	65.1
185.5	113.2	133.5	147.1	65.1
187.5	113.4	133.7	147.2	65.2
189.5	113.6	133.9	147.3	65.3
191.5	113.8	134.2	147.4	65.4
193.5	114.0	134.4	147.4	65.5
195.5	114.3	134.7	147.5	65.6
197.5	114.5	134.9	147.6	65.7
199.5	114.7	135.2	147.7	65.8
201.5	114.9	135.4	147.8	65.9

TABLE 14.3-17 (Sheet 5 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
REFLOOD MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
SECONDS	FLOW LBM/SEC	ENERGY THOUSAND SBTU/SEC	SECONDS	FLOW LBM/SEC
203.5	115.1	135.7	147.9	66.0
205.5	115.3	135.9	147.9	66.1
207.5	115.5	136.1	148.0	66.2
209.5	115.7	136.4	148.1	66.3
211.5	115.9	136.6	148.2	66.4
213.5	116.1	136.9	148.3	66.5
215.5	116.3	137.1	148.4	66.6
217.5	116.5	137.4	148.4	66.7
219.5	116.8	137.6	148.5	66.8
221.5	117.0	137.9	148.6	66.9
223.5	117.2	138.1	148.7	67.0
225.5	117.4	138.4	148.8	67.1
227.5	117.6	138.7	148.9	67.2
229.5	117.9	138.9	149.0	67.4
231.5	118.1	139.2	149.1	67.5
233.5	118.3	139.5	149.1	67.6
235.5	118.5	139.7	149.2	67.7
237.5	118.8	140.0	149.3	67.8
239.5	119.0	140.3	149.4	67.9
239.7	119.0	140.3	149.4	67.9

Notes:
- * mass and energy exiting the SG side of the break
- ** mass and energy exiting the pump side of the break

TABLE 14.3-18
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
POST-REFLOOD MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
	FLOW	ENERGY	FLOW	ENERGY
SECONDS	LBM/SEC	THOUSANDS BTU/SEC	LBM/SEC	THOUSANDS BTU/SEC
239.8	218.7	272.7	208.8	137.2
244.8	218.2	272.0	208.5	136.8
249.8	217.7	271.4	208.2	136.4
254.8	217.1	270.7	207.9	136.0
259.8	216.5	269.9	207.5	135.6
264.8	216.6	270.0	207.2	135.2
269.8	215.9	269.2	206.9	134.8
274.8	215.2	268.3	206.5	134.4
279.8	215.2	268.3	206.2	134.0
284.8	214.4	267.3	205.9	133.6
289.8	214.3	267.1	205.5	133.2
294.8	213.4	266.1	205.2	132.9
299.8	213.2	265.8	204.9	132.5
304.8	212.9	265.4	204.5	132.1
309.8	211.9	264.2	204.2	131.7
314.8	211.5	263.7	203.8	131.3
319.8	211.0	263.1	203.5	130.9
324.8	210.5	262.4	203.2	130.4
329.8	210.5	262.4	202.8	130.0
334.8	209.8	261.5	202.5	129.6
339.8	209.6	261.3	202.1	129.2
344.8	208.7	260.2	201.8	128.8
349.8	208.3	259.7	201.4	128.4
354.8	207.8	259.1	201.1	128.0
359.8	207.2	258.3	200.7	127.6
364.8	207.0	258.0	200.4	127.2
369.8	206.6	257.6	200.0	126.8
374.8	206.0	256.9	199.7	126.3
379.8	205.3	255.9	199.3	125.9
384.8	204.8	255.3	199.0	125.5
389.8	204.6	255.0	198.6	125.1
394.8	204.0	254.3	198.3	124.7
399.8	203.5	253.8	197.9	124.2
404.8	202.8	252.8	197.5	123.8
409.8	202.4	252.4	197.2	123.4
414.8	202.3	252.2	196.8	123.0
419.8	201.6	251.4	196.5	122.6
424.8	201.1	250.7	196.1	122.2
429.8	205.4	256.1	199.9	126.6

TABLE 14.3-18 (Cont.)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
POST-REFLOOD MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATH NO.1***		**BREAK PATH NO.2****	
	FLOW	ENERGY		FLOW
SECONDS	LBM/SEC	THOUSANDS BTU/SEC	SECONDS	LBM/SEC
434.8	205.1	255.7	199.5	126.1
439.8	204.6	255.1	199.1	125.7
444.8	204.0	254.3	198.8	125.3
449.8	203.4	253.6	198.4	124.8
454.8	85.7	106.9	310.3	154.0
627.6	85.7	106.9	310.3	154.0
627.7	87.5	108.4	308.4	147.8
629.8	87.5	108.3	308.5	147.6
1262.4	87.5	108.3	308.5	147.6
1262.5	74.8	86.1	321.1	31.0
1500.5	71.4	82.2	324.5	31.6
1500.6	71.4	82.2	167.3	63.8
2334.0	64.4	74.1	174.3	65.0
2334.1	64.4	74.1	174.3	65.0
3600.0	57.2	65.8	181.5	66.3
3600.1	54.3	62.5	184.3	48.7
10000.0	39.5	45.5	199.2	52.6
23400.0	31.9	36.7	206.8	54.6
23400.1	31.9	36.7	73.3	19.4
100000.0	21.1	24.3	84.1	22.2
1000000.0	9.1	10.4	96.1	25.4
10000000.0	2.8	3.3	102.4	27.0

* mass and energy exiting the SG side of the break
** mass and energy exiting the pump side of the break

TABLE 14.3-19
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
MASS BALANCE FOR 3216 MWt

		Mass Balance						
Time (Seconds)		.00	26.40	26.4+δ	239.71	627.68	1262.40	3600.00
		Mass (Thousand lbm)						
Initial	In RCS and ACC	714.25	714.25	714.25	714.25	714.25	714.25	714.25
Added Mass	Pumped Injection	.00	.00	.00	74.24	227.83	479.16	1074.54
	Total Added	.00	.00	.00	74.24	227.83	479.16	1074.54
*** TOTAL AVAILABLE ***		714.25	714.25	714.25	788.50	942.08	1193.41	1788.79
Distribution	Reactor Coolant	524.25	58.26	84.53	144.26	144.26	144.26	144.26
	Accumulator	190.00	126.74	100.47	.00	.00	.00	.00
	Total Contents	714.25	185.00	185.00	144.26	144.26	144.26	144.26
Effluent	Break Flow	.00	529.24	529.24	644.22	801.15	1052.40	1647.79
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	529.24	529.24	644.22	801.15	1052.40	1647.79
***** TOTAL ACCOUNTABLE *****		714.25	714.24	714.24	788.48	945.41	1196.65	1792.04

TABLE 14.3-20
DOUBLE-ENDED PUMP SUCTION BREAK GUILLOTINE MIN SI
ENERGY BALANCE FOR 3216 MWt

		Energy Balance						
Time (Seconds)		.00	26.40	26.4+δ	239.71	627.68	1262.40	3600.00
		Energy (Million Btu)						
Initial Energy	In RCS, ACC, Steam Gen	781.41	781.41	781.41	781.41	781.41	781.41	781.41
Added Energy	Pumped Injection	.00	.00	.00	5.79	17.78	37.38	177.08
	Decay Heat	.00	7.52	7.52	30.60	63.24	107.96	235.94
	Heat From Secondary	.00	8.54	8.54	8.54	8.54	8.54	8.54
	Total Added	.00	16.07	16.07	44.94	89.56	153.89	421.56
*** TOTAL AVAILABLE ***		781.41	797.48	797.48	826.35	870.98	935.30	
Distribution	Reactor Coolant	305.58	13.36	15.98	38.17	38.17	38.17	38.17
	Accumulator	18.95	12.64	10.02	.00	.00	.00	.00
	Core Stored	27.00	13.89	13.89	3.95	3.78	3.55	2.71
	Primary Metal	166.68	158.73	158.73	131.10	92.65	69.95	53.20
	Secondary Metal	40.99	41.26	41.26	38.20	29.44	20.05	15.21
	Steam Generator	222.23	237.68	237.68	217.15	162.06	107.20	80.60
	Total Contents	781.41	477.56	477.56	428.56	326.09	238.91	189.88
Effluent	Break Flow	.00	319.44	319.44	389.59	537.57	680.69	998.74
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	319.44	319.44	389.59	537.57	680.69	998.74
***** TOTAL ACCOUNTABLE *****		781.41	797.01	797.01	818.15	863.66	919.61	1188.62

TABLE 14.3-21

DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI

PRINCIPLE PARAMETERS DURING REFLOOD FOR 3216 MWt

Time	Flooding		Carry-over	Core	Downcomer	Flow	Injection			
	Temp	Rate	Fraction	Height	Height	Fraction	Total	Accum	Spill	Enthalpy
Seconds	(°F)	(in/sec)	(-----)	(Feet)	(Feet)	(-----)	(Pounds Mass Per Second)			Btu/Lbm
26.4	190.0	.000	.000	.00	.00	.250	.0	.0	.0	.00
27.1	188.6	20.878	.000	.50	1.17	.000	6713.2	6713.2	.0	99.50
27.4	187.2	24.643	.000	1.09	1.23	.000	6647.4	6647.4	.0	99.50
27.8	186.9	2.520	.126	1.34	2.09	.225	6528.7	6528.7	.0	99.50
28.1	187.0	2.568	.184	1.39	2.78	.288	6457.6	6457.6	.0	99.50
28.8	187.3	2.426	.300	1.50	4.37	.322	6325.9	6325.9	.0	99.50
29.4	187.5	2.365	.373	1.58	5.66	.333	6214.4	6214.4	.0	99.50
33.7	189.4	2.656	.616	2.00	14.86	.353	5534.7	5534.7	.0	99.50
35.5	190.2	4.041	.665	2.18	16.12	.552	4887.5	4887.5	.0	99.50
37.5	191.2	3.817	.693	2.39	16.12	.548	4666.9	4666.9	.0	99.50
38.7	191.8	3.708	.703	2.51	16.12	.545	4554.1	4554.1	.0	99.50
44.7	195.4	3.354	.726	3.00	16.12	.529	4072.9	4072.9	.0	99.50
45.5	195.9	3.319	.727	3.06	16.12	.527	4017.0	4017.0	.0	99.50
46.5	196.5	3.905	.732	3.14	16.05	.638	.0	.0	.0	.00
47.5	197.3	4.688	.732	3.24	15.59	.640	.0	.0	.0	.00
50.3	199.4	3.866	.735	3.51	14.51	.608	358.5	.0	.0	78.02
56.6	204.5	3.371	.738	4.00	13.08	.603	367.3	.0	.0	78.02
64.5	211.7	2.865	.737	4.54	11.70	.596	375.5	.0	.0	78.02
72.4	219.2	2.447	.735	5.00	10.70	.587	381.4	.0	.0	78.02
83.5	229.1	1.994	.731	5.54	9.81	.571	386.6	.0	.0	78.02
94.6	236.8	1.674	.727	6.00	9.37	.554	389.8	.0	.0	78.02
109.5	244.9	1.407	.723	6.52	9.23	.532	392.0	.0	.0	78.02
125.0	251.7	1.265	.721	7.00	9.42	.516	393.0	.0	.0	78.02

TABLE 14.3-21 (Cont.)

DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
PRINCIPLE PARAMETERS DURING REFLOOD FOR 3216 MWt

Time	Flooding		Carry-over Fraction	Core Height	Downcomer Height	Flow Fraction	Injection			
	Temp	Rate					Total	Accum	Spill	Enthalpy
Seconds	(°F)	(in/sec)	(-----)	(Feet)	(Feet)	(-----)	(Pounds Mass Per Second)			Btu/Lbm
143.5	258.3	1.196	.724	7.52	9.83	.508	393.4	.0	.0	78.02
161.1	263.8	1.177	.728	8.00	10.30	.506	393.5	.0	.0	78.02
173.5	267.2	1.174	.732	8.33	10.64	.507	393.4	.0	.0	78.02
181.5	269.2	1.175	.734	8.54	10.86	.508	393.4	.0	.0	78.02
199.5	273.4	1.179	.740	9.00	11.35	.510	393.3	.0	.0	78.02
219.5	277.5	1.183	.747	9.50	11.89	.513	393.3	.0	.0	78.02
239.7	281.2	1.188	.755	10.00	12.42	.515	393.2	.0	.0	78.02

TABLE 14.3-22 (Sheet 1 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MAXIMUM ECCS FLOWS
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	BREAK PATH NO.1*		BREAK PATH NO.2**	
	FLOW	ENERGY	FLOW	ENERGY
SECONDS	LBM/SEC	THOUSANDS BTU/SEC	LBM/SEC	THOUSANDS BTU/SEC
.00000	.0	.0	.0	.0
.00102	84849.3	45618.7	38460.5	20632.5
.00204	40171.1	21550.8	39870.6	21387.9
.00312	40159.2	21545.3	39615.5	21250.0
.101	39922.8	21502.1	19730.1	10574.2
.201	40979.6	22261.5	22314.8	11972.4
.301	44573.7	24485.7	23347.7	12533.3
.401	44733.5	24903.4	23497.2	12619.1
.502	43754.1	24707.6	23117.1	12420.9
.602	44173.0	25273.0	22640.6	12170.9
.701	43682.7	25278.5	22266.0	11975.5
.802	42297.2	24725.0	22065.4	11873.0
.901	40947.3	24164.9	21967.8	11824.8
1.00	39767.1	23698.6	21919.4	11802.2
1.10	38584.1	23240.8	21873.8	11780.4
1.20	37249.5	22702.1	21840.4	11764.4
1.30	35816.5	22086.0	21815.1	11752.1
1.40	34574.8	21534.6	21810.6	11750.5
1.50	33613.4	21111.2	21839.1	11766.5
1.60	32853.5	20790.6	21835.5	11764.7
1.70	32154.3	20501.2	21717.1	11700.6
1.80	31432.8	20195.4	21569.0	11620.3
1.90	30640.2	19839.4	21417.8	11538.5
2.00	29904.6	19514.1	21273.0	11460.3
2.10	29142.5	19163.4	21131.5	11384.2
2.20	28311.2	18759.2	20976.6	11300.9
2.30	27339.3	18251.3	20805.2	11208.8
2.40	26109.9	17561.4	20624.4	11111.9
2.50	24628.0	16684.9	20436.4	11011.2
2.60	22707.6	15483.7	20235.9	10904.1
2.70	21119.5	14495.7	20024.8	10791.3
2.80	20677.2	14277.7	19830.0	10687.6
2.90	20092.1	13921.6	19641.2	10587.2
3.00	19618.0	13629.3	19452.2	10486.9
3.10	19439.5	13538.0	19253.4	10381.6

TABLE 14.3-22 (Sheet 2 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MAXIMUM ECCS FLOWS
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	BREAK PATH NO.1*		BREAK PATH NO.2**	
	FLOW	ENERGY		FLOW
SECONDS	LBM/SEC	THOUSANDS BTU/SEC	SECONDS	LBM/SEC
3.20	19277.2	13446.3	19038.5	10267.5
3.30	18887.7	13195.0	18811.0	10146.7
3.40	18582.5	13000.8	18590.3	10029.7
3.50	18234.9	12768.3	18376.3	9916.6
3.60	17794.6	12467.4	18158.9	9801.7
3.70	17321.2	12144.4	17944.9	9688.6
3.80	16848.6	11822.4	17737.4	9579.2
3.90	16396.6	11512.6	17538.0	9474.3
4.00	15955.2	11206.7	17348.5	9374.9
4.20	15135.8	10639.0	16986.7	9185.4
4.40	14450.3	10162.8	16655.3	9012.6
4.60	13877.1	9759.9	16344.2	8850.7
4.80	13404.7	9422.8	16060.8	8703.8
5.00	13011.8	9135.4	15789.2	8563.0
5.20	12732.5	8922.1	15540.3	8434.4
5.40	12478.1	8723.0	15299.7	8310.1
5.60	12276.0	8558.7	15078.5	8196.2
5.80	12100.2	8407.6	14860.7	8083.8
6.00	11970.8	8286.4	14662.3	7982.1
6.20	11897.3	8198.2	14466.7	7881.5
6.40	11941.5	8184.6	14308.5	7801.8
6.60	12573.4	8577.2	14836.8	8099.2
6.80	12267.5	8335.5	14923.1	8152.2
7.00	11115.9	8022.4	14778.1	8077.9
7.20	9516.4	7395.4	14698.4	8039.6
7.40	9167.6	7210.5	14540.2	7957.4
7.60	9127.7	7152.4	14376.2	7872.8
7.80	9064.7	7090.9	14232.0	7799.4
8.00	8976.5	7009.1	13995.3	7674.5
8.20	8938.7	6944.0	13762.8	7551.6
8.40	8944.4	6890.1	13558.1	7442.5
8.60	8964.5	6847.8	13401.7	7357.6
8.80	8988.7	6815.2	13346.4	7326.0
9.00	8943.5	6738.5	13265.6	7276.9

TABLE 14.3-22 (Sheet 3 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MAXIMUM ECCS FLOWS
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	BREAK PATH NO.1*		BREAK PATH NO.2**	
	FLOW	ENERGY		FLOW
SECONDS	LBM/SEC	THOUSANDS BTU/SEC	SECONDS	LBM/SEC
9.20	8873.3	6665.8	13116.3	7188.5
9.40	8771.5	6589.3	12977.4	7107.1
9.60	8643.0	6499.3	12853.6	7035.2
9.80	8522.7	6406.2	12713.2	6954.8
10.0	8413.6	6305.9	12565.4	6870.7
10.2	8325.3	6213.6	12425.6	6791.3
10.4	8236.9	6121.7	12287.2	6712.8
10.6	8147.1	6034.8	12149.6	6634.7
10.8	8049.5	5950.5	12013.2	6557.5
11.0	7943.5	5868.5	11876.1	6480.1
11.2	7827.4	5787.9	11739.4	6403.4
11.4	7703.1	5708.8	11602.8	6327.0
11.6	7571.4	5630.8	11466.2	6251.0
11.8	7432.6	5552.9	11327.9	6174.2
12.0	7287.4	5475.4	11190.3	6098.1
12.2	7143.9	5404.4	11053.9	6022.8
12.4	6998.6	5332.3	10913.4	5945.4
12.6	6857.7	5261.2	10773.8	5868.7
12.8	6722.4	5191.7	10634.2	5792.3
13.0	6590.5	5122.9	10494.3	5715.9
13.2	6461.5	5054.6	10355.0	5640.1
13.4	6336.3	4987.1	10215.8	5564.4
13.6	6213.6	4920.1	10076.1	5488.5
13.8	6095.2	4854.0	9939.9	5414.5
14.0	5981.4	4789.2	9804.5	5341.0
14.2	5870.0	4725.2	9667.8	5266.6
14.4	5762.4	4663.0	9548.5	5195.8
14.6	5651.0	4597.9	9450.6	5121.3
14.8	5528.1	4523.8	9346.8	5029.6
15.0	5387.0	4432.0	9292.2	4951.8
15.2	5233.8	4321.4	9221.7	4856.4
15.4	5084.6	4200.2	9195.7	4779.9
15.6	4953.8	4082.5	9185.0	4710.7
15.8	4851.5	3981.3	9150.7	4634.3
16.0	4763.9	3896.2	9065.1	4540.5

TABLE 14.3-22 (Sheet 4 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MAXIMUM ECCS FLOWS
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	BREAK PATH NO.1*		BREAK PATH NO.2**	
	FLOW	ENERGY		FLOW
SECONDS	LBM/SEC	THOUSANDS BTU/SEC	SECONDS	LBM/SEC
16.2	4679.7	3824.4	8924.7	4427.4
16.4	4594.8	3761.2	8849.9	4352.3
16.6	4509.0	3704.4	8774.1	4281.8
16.8	4423.0	3654.2	8624.3	4179.2
17.0	4338.2	3611.9	8508.6	4095.2
17.2	4249.9	3574.2	8449.4	4040.2
17.4	4161.1	3542.1	8334.7	3961.4
17.6	4070.4	3517.3	8134.0	3843.5
17.8	3977.6	3498.7	7692.7	3611.5
18.0	3879.6	3483.8	7925.8	3695.0
18.2	3766.4	3467.4	8408.4	3898.6
18.4	3647.5	3459.9	7772.1	3588.3
18.6	3523.9	3459.4	6787.4	3115.0
18.8	3316.3	3392.0	6792.5	3085.4
19.0	3075.5	3296.5	6861.7	3076.2
19.2	2851.5	3194.6	6591.0	2917.6
19.4	2653.8	3085.2	6186.4	2705.8
19.6	2481.6	2968.9	5760.6	2488.9
19.8	2319.3	2822.8	5425.6	2309.7
20.0	2157.3	2647.7	5521.7	2301.1
20.2	1997.9	2463.6	5978.1	2425.4
20.4	1849.6	2288.4	6479.4	2560.5
20.6	1720.9	2134.5	6024.8	2338.2
20.8	1601.7	1991.3	5615.7	2151.8
21.0	1491.6	1857.6	5333.2	2015.1
21.2	1391.2	1735.8	5105.3	1895.5
21.4	1296.0	1619.3	4883.9	1777.2
21.6	1207.7	1511.3	4646.3	1654.6
21.8	1116.9	1399.1	4398.9	1531.3
22.0	1047.8	1314.8	4158.4	1414.7
22.2	968.7	1216.9	3926.8	1305.8
22.4	922.7	1160.5	3713.1	1207.8
22.6	883.2	1111.8	3507.5	1116.8
22.8	840.3	1058.5	3309.9	1032.7
23.0	800.0	1008.2	3105.1	950.2

TABLE 14.3-22 (Sheet 5 of 5)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MAXIMUM ECCS FLOWS
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	BREAK PATH NO.1*		BREAK PATH NO.2**	
	FLOW	ENERGY		FLOW
SECONDS	LBM/SEC	THOUSANDS BTU/SEC	SECONDS	LBM/SEC
23.2	755.0	952.2	2893.1	869.5
23.4	706.2	891.1	2726.2	805.7
23.6	656.7	829.2	2523.1	734.2
23.8	606.2	765.9	2294.3	658.1
24.0	555.7	702.5	2035.7	576.5
24.2	504.9	638.5	1740.4	487.3
24.4	454.2	574.8	1396.2	387.3
24.6	403.2	510.5	1003.6	276.4
24.8	350.6	444.2	589.7	161.7
25.0	296.1	375.3	226.7	62.1
25.2	239.7	303.9	.0	.0
25.4	183.3	232.7	.0	.0
25.6	130.6	165.9	.0	.0
25.8	75.7	96.4	.0	.0
26.0	.0	.0	.0	.0

* mass and energy exiting the SG side of the break
** mass and energy exiting the pump side of the break

TABLE 14.3-23

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TABLE 14.3-24

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TABLE 14.3-25

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TABLE 14.3-26

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TABLE 14.3-27 (Sheet 1 of 2)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MAXSI
PRINCIPLE PARAMETERS DURING REFLOOD FOR 3216 MWt

Time Seconds	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total	Injection Accum	Spill	Enthalpy Btu/lbm
	Temp °F	Rate in/sec					(Pounds Mass per Second)			
26.0	189.3	.000	.000	.00	.00	.250	.0	.0	.0	.00
26.7	187.9	20.997	.000	.50	1.18	.000	6759.0	6759.0	.0	99.50
27.0	186.5	24.771	.000	1.10	1.24	.000	6692.1	6692.1	.0	99.50
27.4	186.3	2.530	.127	1.34	2.12	.228	6571.3	6571.3	.0	99.50
27.7	186.4	2.571	.171	1.38	2.64	.279	6514.3	6514.3	.0	99.50
28.4	186.6	2.434	.294	1.50	4.35	.321	6369.9	6369.9	.0	99.50
29.1	186.9	2.368	.379	1.59	5.82	.333	6242.7	6242.7	.0	99.50
33.3	188.8	2.663	.616	2.00	14.96	.353	5562.9	5562.9	.0	99.50
35.2	189.6	4.048	.667	2.19	16.12	.553	4902.2	4902.2	.0	99.50
37.2	190.6	3.818	.694	2.40	16.12	.549	4684.2	4684.2	.0	99.50
38.3	191.2	3.718	.703	2.50	16.12	.546	4579.7	4579.7	.0	99.50
44.3	194.8	3.360	.726	3.00	16.12	.530	4093.2	4093.2	.0	99.50
45.2	195.4	3.530	.727	3.08	16.12	.553	4511.9	3864.4	.0	96.42
46.2	196.1	2.828	.729	3.15	16.12	.410	1858.2	1183.4	.0	91.70
47.2	196.8	3.885	.733	3.23	16.00	.581	629.7	.0	.0	78.02
50.4	199.3	3.715	.737	3.50	15.54	.579	635.3	.0	.0	78.02
56.9	205.1	3.407	.741	4.00	14.78	.573	647.7	.0	.0	78.02
64.2	212.5	3.145	.743	4.52	14.17	.566	658.0	.0	.0	78.02
71.7	220.5	2.923	.745	5.00	13.75	.560	666.2	.0	.0	78.02
80.2	229.4	2.716	.747	5.51	13.47	.552	674.5	.0	.0	78.02
89.1	237.4	2.544	.749	6.00	13.38	.544	682.4	.0	.0	78.02

TABLE 14.3-27 (Sheet 2 of 2)
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MAXSI
PRINCIPLE PARAMETERS DURING REFLOOD FOR 3216 MWt

Time Seconds	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total	Injection Accum	Spill	Enthalpy Btu/lbm
	Temp °F	Rate in/sec					(Pounds Mass per Second)			
99.2	245.0	2.401	.751	6.52	13.46	.537	688.5	.0	.0	78.02
109.1	251.4	2.305	.754	7.00	13.67	.531	692.3	.0	.0	78.02
121.2	258.0	2.230	.757	7.56	14.04	.527	695.0	.0	.0	78.02
131.2	262.8	2.194	.761	8.00	14.41	.525	696.1	.0	.0	78.02
143.2	267.8	2.170	.765	8.52	14.90	.524	696.7	.0	.0	78.02
147.2	269.3	2.165	.767	8.69	15.06	.524	696.7	.0	.0	78.02
154.7	272.0	2.179	.770	9.00	15.37	.528	695.5	.0	.0	78.02
165.2	275.4	2.200	.773	9.44	15.70	.536	693.0	.0	.0	78.02
167.2	276.0	2.200	.774	9.52	15.74	.538	692.6	.0	.0	78.02
178.9	279.3	2.168	.778	10.00	15.95	.544	692.1	.0	.0	78.02

TABLE 14.3-28 (Sheet 1 of 5)
DOUBLE-ENDED HOT LEG GUILLOTINE
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATHNO.1***		**BREAK PATH NO.2****	
	FLOW	**ENERGY**	**FLOW**	**ENERGY**
SECONDS	**LBM/SEC**	**THOUSANDS BTU/SEC**	**LBM/SE**	**THOUSANDS BTU/SEC**
.00000	.0	.0	.0	.0
.00107	45431.7	28592.9	45428.6	28589.4
.00213	44959.4	28295.7	44682.5	28114.2
.102	45793.5	29109.1	25686.4	16129.1
.201	33239.6	21496.0	22659.2	14136.4
.301	32619.6	21083.9	20264.4	12470.7
.401	31885.0	20590.9	19026.0	11515.6
.502	31493.9	20329.6	18206.6	10840.9
.601	31480.4	20318.8	17625.0	10345.0
.701	31467.6	20326.8	17235.4	9988.6
.801	31177.0	20177.2	16891.0	9682.4
.902	30811.3	19992.6	16655.5	9455.9
1.00	30442.3	19816.4	16457.2	9266.0
1.10	30178.5	19718.7	16334.0	9128.3
1.20	29985.2	19679.7	16264.2	9030.1
1.30	29765.5	19624.5	16277.2	8984.8
1.40	29462.3	19513.9	16329.9	8966.8
1.50	29085.4	19347.4	16416.8	8972.9
1.60	28711.1	19176.2	16524.2	8994.9
1.70	28394.9	19041.5	16647.5	9030.0
1.80	28092.0	18914.9	16773.3	9071.1
1.90	27710.5	18731.3	16894.8	9114.1
2.00	27250.2	18485.7	17006.0	9155.3
2.10	26796.9	18239.2	17106.1	9193.9
2.20	26405.9	18037.0	17195.7	9229.7
2.30	26026.9	17844.1	17274.6	9262.1
2.40	25597.1	17606.7	17340.4	9289.5
2.50	25129.9	17331.2	17393.6	9311.9
2.60	24686.1	17068.7	17437.0	9330.3
2.70	24290.3	16839.1	17470.2	9344.3
2.80	23893.3	16603.8	17493.4	9353.7
2.90	23492.6	16354.8	17505.6	9358.0
3.00	23104.3	16107.1	17508.7	9357.8
3.10	22737.0	15867.2	17504.4	9354.2
3.20	22389.3	15634.4	17493.1	9347.1
3.30	22077.7	15422.8	17476.1	9337.2
3.40	21783.4	15217.4	17454.0	9324.9
3.50	21490.5	15003.4	17426.7	9310.0
3.60	21223.8	14801.9	17394.4	9292.6
3.70	20991.8	14622.4	17358.7	9273.7

TABLE 14.3-28 (Sheet 2 of 5)
DOUBLE-ENDED HOT LEG GUILLOTINE
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME SECONDS	BREAK PATHNO.1* FLOW LBM/SEC	BREAK PATHNO.1* ENERGY THOUSANDS BTU/SEC	BREAK PATH NO.2** FLOW LBM/SEC	BREAK PATH NO.2** ENERGY THOUSANDS BTU/SEC
3.80	20770.1	14444.9	17319.3	9253.0
3.90	20560.5	14270.0	17275.4	9230.1
4.00	20383.2	14116.6	17227.8	9205.6
4.20	20059.6	13822.3	17120.7	9150.8
4.40	19813.2	13575.6	16995.4	9087.4
4.60	19604.3	13349.2	16852.0	9015.4
4.80	19455.3	13162.8	16689.9	8934.5
5.00	19357.0	13009.0	16509.6	8845.2
5.20	19365.9	12932.0	16313.3	8748.6
5.40	19406.3	12882.9	16099.0	8643.5
5.60	19468.9	12860.5	15867.0	8530.1
5.80	19556.2	12870.9	15615.7	8407.4
6.00	19707.4	12910.1	15356.3	8281.5
6.20	11028.8	8300.7	15067.3	8140.3
6.40	14455.2	10617.0	14672.0	7941.4
6.60	14508.9	10566.7	14237.6	7722.0
6.80	14629.0	10544.4	13845.1	7525.9
7.00	14807.5	10579.1	13465.1	7335.8
7.20	14990.4	10601.2	13051.2	7125.5
7.40	15147.5	10624.3	12651.4	6922.1
7.60	15324.6	10652.1	12283.8	6735.6
7.80	15406.3	10588.1	11920.3	6549.8
8.00	15644.1	10638.4	11545.0	6356.1
8.20	15556.0	10536.5	11181.0	6167.9
8.40	15808.7	10588.4	10841.7	5992.4
8.60	16038.3	10635.9	10516.7	5824.0
8.80	16243.1	10677.0	10199.7	5659.1
9.00	16434.6	10716.8	9899.8	5502.5
9.20	16619.4	10756.8	9613.9	5352.9
9.40	16808.2	10802.1	9342.8	5210.8
9.60	17018.1	10862.2	9081.5	5073.5
9.80	17302.6	10968.3	8833.8	4943.2
10.0	17347.4	10951.5	8587.4	4813.4
10.2	17122.8	10772.8	8345.5	4686.2
10.4	15365.1	9795.4	8104.7	4559.7
10.602	14402.9	9265.0	7870.3	4437.2
10.604	14399.0	9262.5	7866.6	4435.2
10.607	14396.3	9260.9	7864.2	4433.9
10.609	14394.2	9259.5	7862.2	4432.9

TABLE 14.3-28 (Sheet 3 of 5)
DOUBLE-ENDED HOT LEG GUILLOTINE
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	BREAK PATHNO.1*		BREAK PATH NO.2**	
SECONDS	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC	FLOW LBM/SEC	ENERGY THOUSANDS BTU/SEC
10.8	14314.8	9194.1	7642.1	4318.4
11.0	14303.2	9168.8	7427.1	4207.3
11.2	14286.7	9144.1	7231.0	4106.7
11.4	14268.5	9117.2	7039.7	4008.3
11.6	14235.3	9078.9	6850.8	3911.2
11.8	14133.9	9001.0	6665.1	3816.0
12.0	13860.2	8831.1	6483.7	3723.3
12.2	13361.0	8543.6	6304.3	3632.2
12.4	12862.1	8261.1	6126.6	3542.6
12.6	12547.5	8079.7	5955.6	3457.0
12.8	12329.5	7951.6	5788.3	3373.8
13.0	12138.1	7840.4	5628.4	3294.8
13.2	11932.6	7723.9	5474.2	3219.1
13.4	11697.7	7593.9	5327.5	3147.3
13.6	11418.6	7442.6	5182.9	3076.7
13.8	11117.0	7281.9	5044.5	3009.5
14.0	10808.3	7119.7	4908.8	2944.0
14.2	10513.3	6967.5	4777.7	2881.1
14.4	10232.1	6825.1	4650.0	2820.0
14.6	9944.6	6681.6	4522.3	2759.0
14.8	9642.7	6534.6	4391.9	2696.5
15.0	9312.3	6377.9	4248.5	2627.8
15.2	8962.9	6217.0	4092.0	2554.0
15.4	8607.5	6059.7	3918.8	2473.7
15.6	8230.6	5898.9	3734.6	2389.4
15.8	7823.9	5728.2	3547.8	2303.4
16.0	7408.1	5558.9	3361.8	2216.0
16.2	7003.5	5399.7	3189.1	2131.5
16.4	6605.4	5247.9	3034.3	2052.1
16.6	6205.0	5096.2	2898.8	1978.6
16.8	5809.6	4945.4	2784.2	1913.1
17.0	5418.6	4794.4	2687.3	1855.1
17.2	5037.9	4645.9	2604.3	1803.6
17.4	4654.9	4468.2	2532.0	1757.9
17.6	4317.7	4214.3	2465.5	1715.5
17.8	4092.9	4020.9	2404.2	1676.5
18.0	3912.3	3859.9	2346.2	1640.3
18.2	3471.5	3595.3	2289.7	1606.7
18.4	3010.9	3305.0	2234.0	1575.2

TABLE 14.3-28 (Sheet 4 of 5)
DOUBLE-ENDED HOT LEG GUILLOTINE
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	BREAK PATHNO.1*		BREAK PATH NO.2**	
	FLOW	ENERGY	FLOW	ENERGY
SECONDS	LBM/SEC	THOUSANDS BTU/SEC	LBM/SEC	THOUSANDS BTU/SEC
18.6	2713.6	3107.4	2178.1	1544.9
18.8	2476.1	2875.7	2123.0	1515.5
19.0	2277.1	2699.9	2073.9	1487.3
19.2	2102.0	2522.1	2025.1	1455.8
19.4	1987.8	2406.6	1972.2	1422.9
19.6	1858.1	2262.5	1904.6	1390.6
19.8	1732.1	2117.2	1821.2	1366.2
20.0	1620.5	1990.4	1718.2	1345.1
20.2	1508.6	1860.9	1595.9	1316.2
20.4	1405.8	1743.3	1474.1	1276.4
20.6	1309.4	1633.1	1369.7	1240.5
20.8	1222.1	1531.5	1267.3	1214.7
21.0	1146.3	1445.3	1143.8	1196.4
21.2	1074.3	1361.7	1014.4	1163.9
21.4	1011.3	1284.7	856.4	1034.6
21.6	959.8	1221.2	744.5	912.0
21.8	916.3	1166.8	642.3	790.3
22.0	877.2	1117.9	575.6	710.2
22.2	825.8	1053.1	520.8	643.3
22.4	774.8	989.3	442.9	548.5
22.6	726.6	929.0	403.6	500.8
22.8	654.6	838.1	374.4	464.9
23.0	611.8	785.3	360.8	448.2
23.2	580.1	746.3	343.4	427.4
23.4	545.2	702.3	325.1	404.9
23.6	504.4	650.4	317.1	395.2
23.8	509.9	654.9	295.6	368.6
24.0	522.0	673.2	269.9	337.2
24.2	514.7	663.5	275.4	344.2
24.4	509.9	657.6	278.5	348.3
24.6	504.6	650.7	274.1	342.8
24.8	493.8	636.8	256.7	321.2
25.0	484.3	624.5	254.1	318.3
25.2	474.5	611.6	241.9	303.2
25.4	470.6	606.4	238.5	299.0
25.6	462.9	596.3	227.1	285.0
25.8	450.9	580.8	189.6	238.2
26.0	423.4	545.3	167.1	210.4

TABLE 14.3-28 (Sheet 5 of 5)
DOUBLE-ENDED HOT LEG GUILLOTINE
BLOWDOWN MASS AND ENERGY RELEASES FOR 3216 MWt

TIME	**BREAK PATHNO.1***		**BREAK PATH NO.2****	
	FLOW	**ENERGY**	**FLOW**	**ENERGY**
SECONDS	**LBM/SEC**	**THOUSANDS BTU/SEC**	**LBM/SEC**	**THOUSANDS BTU/SEC**
26.2	388.3	500.6	178.8	225.3
26.4	381.0	487.6	153.7	193.7
26.6	407.2	521.4	136.6	172.6
26.8	395.8	499.9	156.5	197.7
27.0	430.6	530.6	174.5	220.1
27.2	408.0	513.6	186.8	235.5
27.4	450.7	557.4	175.5	221.5
27.6	432.8	543.9	200.1	252.3
27.8	436.3	542.5	181.9	229.3
28.0	513.0	630.5	183.0	230.9
28.2	537.6	661.0	216.2	272.5
28.4	552.1	677.6	224.9	283.2
28.6	492.9	615.3	203.0	256.0
28.8	316.7	407.5	201.8	254.4
29.0	89.2	116.7	60.2	76.2
29.2	.0	.0	.0	.0

→

* mass and energy exiting from the reactor vessel side of the break
** mass and energy exiting from the SG side of the break

TABLE 14.3-29
DOUBLE-ENDED HOT LEG GUILLOTINE
MASS BALANCE FOR 3216 MWt

Time (Seconds)		.00	29.20	29.20+δ
		Mass (Thousand lbm)		
Initial	In RCS and ACC	731.97	731.97	731.97
Added Mass	Pumped Injection	.00	.00	.00
	Total Added	.00	.00	.00
TOTAL AVAILABLE		731.97	731.97	731.97
Distribution	Reactor Coolant	524.25	96.33	123.07
	Accumulator	207.72	129.91	103.17
	Total Contents	731.97	226.24	226.24
Effluent	Break Flow	.00	505.71	505.71
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	505.71	505.71
TOTAL ACCOUNTABLE		731.97	731.95	731.95

TABLE 14.3-30
DOUBLE-ENDED HOT LEG GUILLOTINE
ENERGY BALANCE FOR 3216 MWt

	Time (Seconds)	.00	29.20	29.20+δ
			Energy (Million Btu)	
Initial Energy	In RCS, ACC, Steam Gen	784.57	784.57	784.57
Added Energy	Pumped Injection	.00	.00	.00
	Decay Heat	.00	8.37	8.37
	Heat From Secondary	.00	-.23	-.23
	Total Added	.00	8.14	8.14
	TOTAL AVAILABLE	784.57	792.72	792.72
Distribution	Reactor Coolant	305.58	22.91	25.57
	Accumulator	20.67	12.93	10.27
	Core Stored	27.00	10.30	10.30
	Primary Metal	166.68	155.81	155.81
	Secondary Metal	40.99	40.76	40.76
	Steam Generator	223.66	222.24	222.24
	Total Contents	784.57	464.94	464.94
Effluent	Break Flow	.00	327.28	327.28
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	327.28	327.28
	TOTAL ACCOUNTABLE	784.57	792.23	792.23

TABLE 14.3-31
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MIN SI
SEQUENCE OF EVENTS FOR 3216 MWt

Time (sec)	Event Description
0.0	Break occurs, reactor trip and LOOP power are assumed
0.604	Reactor trip on pressurizer low pressure of 1860 psia
1.75	Containment HI-1 pressure setpoint reached
4.1	Low pressurizer pressure SI setpoint @ 1695 psia reached (SI begins coincident with low pressurizer pressure SI setpoint)
7.75	Main Feedwater Flow Control Valve closed
11.43	Containment HI-3 pressure setpoint reached
13.9	Broken-loop accumulator begins injecting water
14.2	Intact-loop accumulator begins injecting water
26.40	End-of-blowdown phase
45.806	Broken-loop accumulator water injection ends
46.256	Intact-loop accumulator water injection ends
49.1	SI begins
61.75	Reactor containment air recirculation fan coolers actuate
71.48	Containment spray pump(s) (RWST) start
239.706	End-of-reflood for MIN SI Case
1264.1	Peak pressure and temperature occur
1500.46	RHR/HHSI alignment for recirculation
2354	Containment spray is terminated due to RWST LO-LO signal
23400	Hot leg recirculation
1.0×10^7	Transient modeling terminated

TABLE 14.3-32
DOUBLE-ENDED PUMP GUILLOTINE MAX SI
SEQUENCE OF EVENTS FOR 3216 MWt

Time (sec)	Event Description
0.0	Break Occurs, and Loss of Offsite Power are assumed
1.75	Containment HI-1 Pressure Setpoint Reached
4.1	Low Pressurizer Pressure SI Setpoint - 1695 psia reached in blowdown
11.32	Containment HI-3 Pressure Setpoint Reached
14.0	Broken Loop Accumulator Begins Injecting Water
14.3	Intact Loop Accumulator Begins Injecting Water
26.0	End of Blowdown Phase
45.0	Safety Injection Begins
45.6	Broken Loop Accumulator Water Injection Ends
46.15	Intact Loop Accumulator Water Injection Ends
61.75	Reactor Containment Air Recirculation Fan Coolers Actuate
71.32	Containment Spray Pump(s) (RWST) start
178.9	End of Reflood Phase
319.0	Peak Pressure and Temperature Occur
1085.	Cold Leg Recirculation Begins
1536.	Containment Spray is terminated due to RWST LO-LO Signal
23400.	Hot Leg Recirculation Begins
1.0E+07	Transient Modeling Terminated

TABLE 14.3-33
DOUBLE-ENDED HOT LEG GUILLOTINE
SEQUENCE OF EVENTS FOR 3216 MWt

Time (sec)	Event Description
0.0	Break Occurs, and Loss of Offsite Power are assumed
1.86	Containment HI-1 Pressure Setpoint Reached
3.8	Low Pressurizer Pressure SI Setpoint - 1695 psia reached in blowdown
9.87	Containment HI-3 Pressure Setpoint Reached
14.0	Broken Loop Accumulator Begins Injecting Water
14.2	Intact Loop Accumulator Begins Injecting Water
23.5	Peak Pressure and Temperature Occur
29.2	End of Blowdown Phase and Transient Modeling Terminated

TABLE 14.3-34
CONTAINMENT PEAK PRESSURE AND TEMPERATURE
FOR 3216 MWt

Case	Peak Press. (psig)	Peak Steam Temp. (°F)	Pressure (psig) @ 24 hours	Steam Temperature (°F) @ 24 hours
Double-Ended Pump Suction Min SI	45.71 @ 1264.1 sec	266.81 @ 1264.1 sec	17.05 @ 24 hrs	204.97 @ 24 hrs
Double-Ended Pump Suction Max SI	39.67 @ 319 sec	257.596 @ 319 sec	21.38 @ 24 hrs	216.192 @ 24 hrs
Double-Ended Hot Leg	40.62 @ 23.50 sec	259.98 @ 23.49 sec	NA	NA

TABLE 14.3-35

CONTAINMENT HEAT SINKS

NO.	MATERIAL	HEAT TRANSFER AREA (FT^2)	THICKNESS IN
1.	Carbon Steel Concrete	41530	0.375 54.0
2.	Carbon Steel Concrete	26012	0.5 42.0
3.	Concrete	13636	12.0
4.	Concrete	55454	12.0
5.	Stainless Steel Concrete	9091	0.375 12.0
6.	Carbon Steel	62538	0.5
7.	Carbon Steel	74276	0.375
8.	Carbon Steel	25407	0.25
9.	Carbon Steel	63454	0.1875
10.	Carbon Steel	2727	0.125
11.	Carbon Steel	20000	0.138
12.	Carbon Steel	9090	0.0625
13.	Stainless Steel PVC Insulation Carbon Steel Concrete	714	0.019 1.25 0.75 54.0

TABLE 14.3-35 (CONT.)

CONTAINMENT HEAT SINKS

NO.	MATERIAL	HEAT TRANSFER AREA (FT^2)	THICKNESS IN
14.	Stainless Steel	6226	0.019
	PVC Insulation		1.25
	Carbon Steel		0.5
	Concrete		54.0
15.	Stainless Steel	3469	0.025
	Foam Insulation		1.5
	Carbon Steel		0.5
	Concrete		54.0
16.	Stainless Steel	3965	0.025
	Foam Insulation		1.5
	Carbon Steel		0.375
	Concrete		54.0

Note:

1. All carbon steel exterior surfaces are modeled with 0.00033-ft layer of paint on top of a 0.000258-ft layer of carbozinc primer.
2. Approximately 25-ft² of the PVC insulation was replaced with fiberglass. As described in Section 14.3.5.1.1, modeling the PVC insulation, instead of the fiberglass insulation was determined to be conservative and bounding.
3. Approximately 7100-ft² of the liner top coat material (Phenoline 305) was replaced with Carboline 890. As described in Section 14.3.5.1.1, modeling the Phenoline 305 top coat material, instead of the Carboline 890 top coat material was determined to be conservative
   and bounding.
4. Installation of the Sump Strainer Modification and Vortex Suppression Modification resulted in an overall increase in metal mass in the Containment. For the containment pressure analyses it is conservative not to include this.

TABLE 14.3-36
THERMOPHYSICAL PROPERTIES OF CONTAINMENT HEAT SINKS

Material	Thermal Conductivity (Btu/hr-ft - °F)	Volumetric Heat Capacity (Btu/ft^3 - °F)
Paint layer 1, Phenoline	0.08	28.8
Paint layer 2, Carbozinc	0.9	28.8
Carbon Steel	26.0	56.35
Stainless Steel	8.6	56.35
Concrete	0.8	28.8
PVC Insulation	0.0208	1.20
Foam Insulation	0.0417	1.53

TABLE 14.3-37
LOCA CONTAINMENT RESPONSE ANALYSIS PARAMETERS

Service water temperature (°F)	95
RWST water temperature (°F)	110
Initial containment temperature (°F)	130
Initial containment pressure (psia)	16.7
Initial relative humidity (%)	20
Net free volume (ft^3)	2.61×10^6
Reactor Containment Air Recirculation Fan Coolers	
Total	5
Analysis maximum	4
Analysis minimum	3
Containment Hi-1 setpoint (psig)	10.0
Delay time (sec)	
With Offsite Power	NA
Without Offsite Power	60.0
Containment Spray Pumps	
Total	2
Analysis maximum	1
Analysis minimum	1
Flowrate (gpm)	
Injection phase (per pump)- see Table 14.3-40	2180
Recirculation phase (total)	0
Containment Hi-3 setpoint (psig)	30.
Delay time (sec)	
With Offsite Power (delay after High High setpoint)	NA
Without Offsite Power (total time from t=0)	60.0
ECCS Recirculation Switchover, sec	
Minimum Safeguards	1500
Maximum Safeguards	1085
Containment Spray Termination on LO-LO RWST Level, (sec)	
Minimum Safeguards	2345
Maximum Safeguards	1536

TABLE 14.3-37 (CONT.)
LOCA CONTAINMENT RESPONSE ANALYSIS PARAMETERS

Emergency Core Cooling System (ECCS) Flows (GPM)	
Minimum ECCS	
Injection alignment	2871.2
Recirculation alignment	1864.0
Maximum ECCS	
Injection alignment	5394.5
Recirculation alignment	6320.5
Residual Heat Removal System	
RHR Heat Exchangers	
Modeled in analysis *	1
Recirculation switchover time, sec	
Minimum Safeguard	1500
Maximum Safeguard	1085
UA, 10^6 *	
BTU/hr-°F	0.767
Flows - Tube Side and Shell Side - gpm	
Minimum Safeguard	4936
Maximum Safeguard	9871
Component Cooling Water Heat Exchangers	
Modeled in analysis	2
UA, 10^6 *	
BTU/hr-°F	2.40
Flows - Shell Side and Tube Side - gpm	
Shellside *	4936
Tubeside *	
(service water)	5000
Additional heat loads, (BTU/hr)	19.675×10^6

*Minimum safeguard data representing 1 EDG

TABLE 14.3-38
SAFETY INJECTION FLOW MINIMUM SAFEGUARDS

RCS Pressure (psia)	**Total Flow (gpm)**
INJECTION MODE (Reflood Phase)	
14.7	3250.0
34.7	3097.8
54.7	2932.7
61.7	2871.2
74.7	2753.6
94.7	2558.3
114.7	2330.3
214.7.	872.1
INJECTION MODE (Post-Reflood Phase)	
61.7	2871.2
COLD LEG RECIRCULATION MODE	
61.7	1864.0
HOT LEG RECIRCULATION MODE	
61.7	822.0

TABLE 14.3-39
SAFETY INJECTION FLOW MAXIMUM SAFEGUARDS

RCS Pressure (psia)	Total Flow (gpm)
INJECTION MODE (Reflood Phase)	
14.7	6320.50
34.7	5996.18
54.7	5652.86
74.7	5283.84
94.7	4862.22
114.7	4389.80
174.7	1865.02
214.7	1651.00
INJECTION MODE (Post-Reflood Phase)	
61.7	5523.7
COLD LEG RECIRCULATION MODE	
61.7	6320.5
HOT LEG RECIRCULATION MODE	
61.7	6320.5

TABLE 14.3-40
CONTAINMENT SPRAY PERFORMANCE

Containment Pressure (psig)	with 1 Pump(gpm)
0 - 47	2180
Containment Pressure (psig)	with 2 Pumps(gpm)
0 - 47	4200

TABLE 14.3-41

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TABLE 14.3-42

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TABLE 14.3-43
Core Fission Product Inventory

Nuclide	Inventory (Ci)
I-130	3.80E+06
I-131	9.16E+07
I-132	1.33E+08
I-133	1.88E+08
I-134	2.06E+08
I-135	1.75E+08
Kr-85M	2.43E+07
Kr-85	1.10E+06
Kr-87	4.66E+07
Kr-88	6.56E+07
Xe-131M	1.01E+06
Xe-133M	5.87E+06
Xe-133	1.80E+08
Xe-135M	3.68E+07
Xe-135	4.77E+07
Xe-138	1.55E+08
Cs-134	2.06E+07
Cs-136	6.01E+06
Cs-137	1.19E+07
Cs-138	1.71E+08
Rb-86	2.38E+05
Te-127	9.84E+06
Te-127M	1.29E+06
Te-129	2.92E+07
Te-129M	4.30E+06
Te-131M	1.33E+07
Te-132	1.31E+08
Sb-127	9.95E+06
Sb-129	2.97E+07
Sr-89	8.83E+07
Sr-90	8.75E+06
Sr-91	1.11E+08
Sr-92	1.20E+08
Ba-139	1.67E+08
Ba-140	1.61E+08
Ru-103	1.40E+08
Ru-105	9.62E+07

Nuclide	Inventory (Ci)
Ru-106	4.89E+07
Rh-105	8.86E+07
Mo-99	1.75E+08
Tc-99M	1.53E+08
Ce-141	1.52E+08
Ce-143	1.42E+08
Ce-144	1.20E+08
Pu-238	4.13E+05
Pu-239	3.50E+04
Pu-240	5.23E+04
Pu-241	1.18E+07
Np-239	1.88E+09
Y-90	9.11E+06
Y-91	1.14E+08
Y-92	1.20E+08
Y-93	1.39E+08
Nb-95	1.56E+08
Zr-95	1.54E+08
Zr-97	1.55E+08
La-140	1.73E+08
La-141	1.53E+08
La-142	1.48E+08
Nd-147	6.11E+07
Pr-143	1.37E+08
Am-241	1.41E+04
Cm-242	3.52E+06
Cm-244	3.82E+05

TABLE 14.3-43a
Core Fission Product Release Fractions

	Gap Release (1)	**Early In-Vessel** (2)
Noble gases	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium group	0	0.05
Barium, Strontium	0	0.02
Noble Metals (Ruthenium group)	0	0.0025
Cerium group	0	0.0005
Lanthanides	0	0.0002

Note:
(1) Release is initiated at 30 seconds and is terminated at 0.5 hours.
(2) Released over a 1.3 hour period starting at the end of the gap release phase.

TABLE 14.3-44
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TABLE 14.3-45
Data Used in Evaluating Offsite Doses
(Isotope Dependent Data)

COMMITTED EFFECTIVE DOSE EQUIVALENT DOSE CONVERSION FACTORS

Isotope	DCF (rem/curie)	Isotope	DCF (rem/curie)
I-130	2.64E3	Cs-138	1.01E2
I-131	3.29E4	Cs-134	4.63E4
I-132	3.81E2	Cs-136	7.33E3
I-133	5.85E3	Cs-137	3.19E4
I-134	1.31E2	Rb-86	6.62E3
I-135	1.23E3		
		Ru-103	8.95E3
Kr-85m	N/A	Ru-105	4.55E2
Kr-85	N/A	Ru-106	4.77E5
Kr-87	N/A	Rh-105	9.55E2
Kr-88	N/A	Mo-99	3.96E3
Xe-131m	N/A	Tc-99m	3.26E1
Xe-133m	N/A		
Xe-133	N/A	Y-90	8.44E3
Xe-135m	N/A	Y-91	4.89E4
Xe-135	N/A	Y-92	7.81E2
Xe-138	N/A	Y-93	2.15E3
		Nb-95	5.81E3
Te-127	3.18E2	Zr-95	2.36E4
Te-127m	2.15E4	Zr-97	4.33E3
Te-129m	2.39E4	La-140	4.85E3
Te-129	8.95E1	La-141	5.81E2
Te-131m	6.4E3	La-142	2.53E2
Te-132	9.44E3	Nd-147	6.85E3
Sb-127	6.03E3	Pr-143	8.10E4
Sb-129	6.44E2	Am-241	4.44E8
		Cm-242	1.73E7
Ce-141	8.95E3	Cm-244	2.48E8
Ce-143	3.39E3		
Ce-144	3.74E5	Sr-89	4.14E4
Pu-238	3.92E8	Sr-90	1.3E6
Pu-239	4.29E8	Sr-91	1.66E3
Pu-240	4.29E8	Sr-92	8.07E2
Pu-241	8.25E6	Ba-139	1.7E2
Np-239	2.51E3	Ba-140	3.74E3

TABLE 14.3-45 (Cont.)
Data Used in Evaluating Offsite Doses
(Isotope Dependent Data)

EFFECTIVE DOSE EQUIVALENT DOSE CONVERSION FACTORS			
Nuclide	DCF (rem-m^3/Ci-sec)	Nuclide	DCF (rem-m^3/Ci-sec)
I-130	3.848E-01	Cs-134	2.801E-01
I-131	6.734E-02	Cs-136	3.922E-01
I-132	4.144E-01	Cs-137[(1)]	1.066E-01
I-133	1.088E-01	Cs-138	4.477E-01
I-134	4.810E-01	Rb-86	1.780E-02
I-135	2.953E-01		
		Ru-103	8.325E-02
Kr-85m	2.768E-02	Ru-105	1.410E-01
Kr-85	4.403E-04	Ru-106	0.00E+00
Kr-87	1.524E-01	Rh-105	1.376E-02
Kr-88	3.774E-01	Mo-99	2.694E-02
Xe-131m	1.439E-03	Tc-99m	2.179E-02
Xe-133m	5.069E-03		
Xe-133	5.772E-03	Y-90	7.030E-04
Xe-135m	7.548E-02	Y-91	9.620E-04
Xe-135	4.403E-02	Y-92	4.810E-02
Xe-138	2.135E-01	Y-93	1.776E-02
		Nb-95	1.384E-01
Te-127	8.954E-04	Zr-95	1.332E-01
Te-127m	5.439E-04	Zr-97	3.337E-02
Te-129m	5.735E-03	La-140	4.329E-01
Te-129	1.018E-02	La-141	8.843E-03
Te-131m	2.594E-01	La-142	5.328E-01
Te-132	3.811E-02	Nd-147	2.290E-02
Sb-127	1.232E-01	Pr-143	7.770E-05
Sb-129	2.642E-01	Am-241	3.027E-03
		Cm-242	2.105E-05
Ce-141	1.269E-02	Cm-244	1.817E-05
Ce-143	4.773E-02		
Ce-144	3.156E-03	Sr-89	2.860E-04
Pu-238	1.806E-05	Sr-90	2.786E-05
Pu-239	1.569E-05	Sr-91	1.277E-01
Pu-240	1.758E-05	Sr-92	2.512E-01
Pu-241	2.683E-07	Ba-139	8.029E-03
Np-239	2.845E-02	Ba-140	3.175E-02

Note:
1. Decay of Cs-137 does not result in gamma radiation. The EDE DCF listed for Cs-137 is actually the value associated with the decay of the short-lived daughter product Ba-137m.

TABLE 14.3-46
Input Values for Doses

Atmospheric Dilution

Time Period (hr)	χ/Q (520m) (sec/m^3)	χ/Q (1100m) (sec/m^3)
0-8	7.5×10^{-4}	3.5×10^{-4}
8-24	--	1.2×10^{-4}
24-96	--	4.2×10^{-5}
96-720	--	9.3×10^{-6}

Containment Leakage

Time Period (hr)	Leak Rate (percent/day)
0-24	0.1
24-720	0.05

Breathing Rate Offsite

Time Period (hr)	Breathing Rate (m^3/sec)
0-8	3.5×10^{-4}
8-24	1.8×10^{-4}
24-720	2.3×10^{-4}

TABLE 14.3-47
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TABLE 14.3-48
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TABLE 14.3-49
ASSUMPTIONS USED FOR LARGE LOCA DOSE ANALYSIS

Iodine Chemical Species	
Elemental	4.85%
Methyl	0.15%
Particulate	95%
Iodine Removal in Containment	
Containment Spray	
Spray Start Delay	60 sec
Injection spray flowrate	2135 gpm
Injection spray duration	37.8 min
Recirculation spray flowrate	1080 gpm
Recirculation spray duration	Note 1
Iodine removal coefficient	
Elemental, λ_s	
during spray injection	20.0/hr DF < 200
during spray recirculation	5.0/hr DF < 200
Particulate λ_p	
during spray injection	4.4/hr DF $\leq$ 50
during spray recirculation	2.25/hr DF $\leq$ 50
Sedimentation Particulate Removal	0.1/hr DF < 1000 (Note 2)
Fan Cooler Units Containment Filters	
Start Delay Time	60 sec
Number of Units	3
Flow Rate per Unit	64,500 cfm
Containment Free Volume	2.61×10^6-ft^3
Containment Leak Rate	
0-24 hr	0.10%/day
> 24 hr	0.05%/day

Notes:
1. Total spray duration assumed is 3.4 hours following the initiation of the event.
2. Credit for sedimentation removal is limited to the unsprayed portion of the containment until sprays are terminated (at 3.4 hours).

TABLE 14.3-49a
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TABLE 14.3-50
ASSUMPTIONS USED FOR ANALYSIS OF
CONTROL ROOM DOSES

Volume	102,400-ft^3
Unfiltered Inleakage	700 cfm
Filtered Makeup	1800 cfm
Filtered Recirculation	0 cfm
Filter Efficiency	
Elemental	95%
Organic	90%
Particulate	99%
Breathing Rate	3.5 x 10^{-4} m^3/sec
Atmospheric Dispersion Factors	See Table 14.3-51
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

TABLE 14.3-51
ATMOSPHERIC DISPERSION FACTORS USED
FOR ANALYSIS OF CONTROL ROOM DOSES

Release Point	Atmospheric Dispersion Factors (sec/m^3)
Containment Surface Leak (1)	
0-2 hr	3.82×10^{-4}
2-8 hr	2.81×10^{-4}
8-24 hr	1.05×10^{-4}
24-96 hr	8.31×10^{-5}
96-720 hr	7.04×10^{-5}
Side of the Auxiliary Boiler Feedwater Building (2)	
0-2 hr	1.09×10^{-3}
2-8 hr	1.02×10^{-3}
8-24 hr	4.99×10^{-4}
24-96 hr	3.86×10^{-4}
96-720 hr	2.99×10^{-4}
Vent Stacks on the Roof of the Auxiliary Boiler Feedwater Building (3)	
0-2 hr	9.49×10^{-4}
2-8 hr	8.65×10^{-4}
8-24 hr	4.17×10^{-4}
24-96 hr	3.30×10^{-4}
96-720 hr	2.54×10^{-4}
Containment Vent (4)	
0-2 hr	6.44×10^{-4}
2-8 hr	4.69×10^{-4}
8-24 hr	1.72×10^{-4}
24-96 hr	1.37×10^{-4}
96-720 hr	1.17×10^{-4}
Refueling Water Storage Tank Vent (5)	
0-2 hr	5.62×10^{-4}
2-8 hr	3.72×10^{-4}
8-24 hr	1.35×10^{-4}
24-96 hr	1.10×10^{-4}
96-720 hr	9.02×10^{-4}

Notes:

1. Used for Containment Leakage Releases in Rod Ejection (14.2.6.9), Large Break LOCA (14.3.6.5) and Small Break LOCA (14.3.6.7).

2. Used for Steamline Break (14.2.5.7).
3. Used for Locked Rotor (14.1.6.5.3) and Steam Generator Tube Rupture (14.2.4) and Secondary Side Releases for Rod Ejection (14.2.6.9) and Small Break LOCA (14.3.6.7).
4. Used for Fuel Handling Accident (14.2.1.1) and Large Break LOCA ECCS Recirculation Leakage (14.3.6.6).
5. Used for Large Break LOCA ECCS Recirculation Leakage (14.3.6.6).

TABLE 14.3-52
CALCULATED CONTROL ROOM DOSES

Event	TEDE Dose (rem)
Large Break LOCA	
Containment Leakage	3.5
Direct Dose from Activity in Containment	0.02
ECCS Recirculation Leakage With Boundary Layer Effects	0.14
ECCS Recirculation Leakage Without Boundary Layer Effects	1.36
Total With Boundary Layer Effects	3.68
Total Without Boundary Layer Effects	4.90
Small Break LOCA	3.5
Locked Rotor	0.65
Rod Ejection	1.4
Fuel Handling Accident	3.0
Steamline Break	
Pre-Existing Iodine Spike	0.18
Accident-Initiated Iodine Spike	0.52
Steam Generator Tube Rupture	
Pre-Existing Iodine Spike	1.4
Accident-Initiated Iodine Spike	0.48

14.3 FIGURES

Figure No.	Title
Figure 14.3-1	Indian Point Unit 2 WCOBRA/TRAC Vessel Noding Diagram
Figure 14.3-2	Indian Point Unit 2 WCOBRA/TRAC Vessel Model Loop Layout
Figure 14.3-3	High Head Safety Injection Flow Rate
Figure 14.3-3a	Safety Injection Flow vs. RCS Pressure
Figure 14.3-4	Deleted
Figure 14.3-5	Deleted
Figure 14.3-6	Peak Cladding Temperature For Reference Case
Figure 14.3-6a	Deleted
Figure 14.3-6b	Deleted
Figure 14.3-7	Vessel Side Break Flow For Reference Transient
Figure 14.3-7a	Deleted
Figure 14.3-7b	Deleted
Figure 14.3-8	Loop Side Break Flow For Reference Transient
Figure 14.3-8a	Deleted
Figure 14.3-8b	Deleted
Figure 14.3-9	Void Fraction At The Intact And Broken Loop Pump Inlet For Reference Transient
Figure 14.3-9a	Deleted
Figure 14.3-9b	Deleted
Figure 14.3-10	Vapor Flow Rate Per Assembly At Mid-Core Average Channel 17 During Blowdown For Reference Transient
Figure 14.3-10a	Deleted
Figure 14.3-10b	Deleted
Figure 14.3-11	Vapor Flow Rate Per Assembly At Mid-Core Average Channel 19 During Blowdown For Reference Transient
Figure 14.3-11a	Deleted
Figure 14.3-11b	Deleted
Figure 14.3-12	Collapsed Liquid Level Plenum For Reference Transie
Figure 14.3-12a	Deleted
Figure 14.3-12b	Deleted
Figure 14.3-13	Intact Loop 2 Accumulator Flow For Reference Transient
Figure 14.3-13a	Deleted
Figure 14.3-13b	Deleted
Figure 14.3-14	Intact Loop 2 Safety Injection Flow For Reference Transient
Figure 14.3-14a	Deleted
Figure 14.3-14b	Deleted
Figure 14.3-15	Collapsed Liquid Level In Core Average Channel 17 For Reference Transient
Figure 14.3-15a	Deleted
Figure 14.3-15b	Deleted
Figure 14.3-16	Collapsed Liquid Level In Intact Loop Downcomer For Reference Transient
Figure 14.3-16a	Deleted
Figure 14.3-16b	Deleted
Figure 14.3-17	Vessel Fluid Mass For Reference Transient

Figure 14.3-17a	Deleted
Figure 14.3-17b	Deleted
Figure 14.3-18	Peak Cladding Temperature Elevation For Reference Transient
Figure 14.3-18a	Deleted
Figure 14.3-18b	Deleted
Figure 14.3-19	Peak Cladding Temperature Comparison For Five Rods For Reference Transient
Figure 14.3-19a	Deleted
Figure 14.3-19b	Deleted
Figure 14.3-20	Indian Point Unit 2 Axial Power Distribution For Initial And Reference Transient
Figure 14.3-20a	Deleted
Figure 14.3-20b	Deleted
Figure 14.3-21	Indian Point Unit 2 PBOT/PMID Analysis And Operating Limits
Figure 14.3-22	Indian Point Unit 2 Lower Bound COCO Calculated Containment Pressure
Figure 14.3-23	Deleted
Figure 14.3-24	Deleted
Figure 14.3-25	Deleted
Figure 14.3-26	Deleted
Figure 14.3-27 Through 14.3-52	Deleted
Figure 14.3-53a Through 14.3-58b	Deleted
Figure 14.3-53	Small Break LOCA Axial Power Shape
Figure 14.3-54	3.0” Small Break LOCA RCS Pressure
Figure 14.3-55	3.0” Small Break LOCA Core Mixture Level
Figure 14.3-56	3.0” Small Break LOCA Hot Rod Clad Average Temperature
Figure 14.3-57	3.0” Small Break LOCA Core Outlet Steam Flow
Figure 14.3-58	3.0” Small Break LOCA Heat Transfer Coefficient
Figure 14.3-59	3.0” Small Break LOCA Hot Spot Fluid Temperature
Figure 14.3-60	3.0” Small Break LOCA Break Flow
Figure 14.3-61	3.0” Small Break LOCA Safety Injection Mass Flow Rate
Figure 14.3-62	2.0” Small Break LOCA RCS Pressure
Figure 14.3-63	2.0” Small Break LOCA Core Mixture Level
Figure 14.3-64	2.0” Small Break LOCA Hot Rod Clad Average Temperature
Figure 14.3-65	4.0” Small Break LOCA RCS Pressure
Figure 14.3-66	4.0” Small Break LOCA Core Mixture Level
Figure 14.3-67	4.0” Small Break LOCA Hot Rod Clad Average Temperature
Figure 14.3-68 Through 14.3-100	Deleted
Figure 14.3-101	Reactor Vessel Internals
Figure 14.3-102	RPV Shell And Support System
Figure 14.3-103	Deleted
Figure 14.3-103a	Reactor Vessel Internals Core Barrel Assembly
Figure 14.3-103b	Reactor Internals and Fuel
Figure 14.3-104	RPV System Model

14.4 ANTICIPATED TRANSIENTS WITHOUT SCRAM

An anticipated transient without scram (ATWS) is an anticipated operational occurrence (such as loss of feedwater, loss of load, or loss of offsite power) that is assumed to be accompanied by a failure of the reactor trip system to shut down the reactor. As presented in Reference 1, the reactor is adequately protected against anticipated plant transients by the reactor protection system in the Westinghouse design, which is both redundant and diverse. As a result, failure to trip was not considered a credible event and the effects of ATWS were not considered part of the design basis for transients analyzed for Westinghouse plants. Nevertheless, in response to an AEC request for further information at the time of Indian Point Unit 2 initial licensing, the hypothetical effects of anticipated transients with no credit taken for reactor trip were provided in Supplement 6 to the original Indian Point Unit 2 FSAR. Those assessments were historical and were superseded by a later series of generic studies on ATWS (References 2 & 3) that showed acceptable consequences would result for Westinghouse designed plants provided that the turbine trips and auxiliary feedwater flow is initiated in a timely manner. The final USNRC ATWS Rule (Reference 4) requires that all US Westinghouse-designed plants install ATWS Mitigation System Actuation Circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater independent of the reactor trip system. The Indian Point Unit 2 AMSAC is described in Section 7.10.

REFERENCES FOR SECTION 14.4

1. T. W. T. Burnett, et al., Reactor Protection System Diversity in W PWRs, WCAP-7306, Westinghouse Electric Corporation, April 1969.

2. Burnett, T.W.T, et al., "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

3. Letter from T.M. Anderson (Westinghouse) to S.H. Hanauer (USNRC), "ATWS Submittal," NS-TMA-2182, December 1979.

4. ATWS Final Rule, Code of Federal Regulations 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants.

TABLE 14.4-1
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14.4 FIGURES

APPENDIX 14A
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