B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) averaged over one hour shall be limited in accordance with Figure 3.1.B-1 and Figure 3.1.B-2 for the service period up to 25 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

    b. Figure 3.1.B-1 and Figure 3.1.B-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figure 3.1.B-1 and Figure 3.1.B-2 shall be recalculated periodically using NRC approved methods.

3. DELETED

4. 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.

5. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

6. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3 of the Technical Specifications.

Basis

All components in the Reactor Coolant System are designed to withstand the effects of the cyclic loads due to reactor system temperature and pressure changes[(1)]. These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the UFSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation[(2)].

Heatup and cooldown limit curves define acceptable regions for normal operation. The heatup and cooldown limit curves establish operating limits that provide a margin to non-ductile failure of the reactor coolant pressure boundary. The reactor vessel is the most limiting component most subject to non-ductile failure. Therefore, the reactor vessel limits control the heatup and cooldown limits provided in Figures 3.1.B-1 and 3.2.B-2. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

References

(1) Indian Point Unit No. 2 UFSAR, Section 4.1.5.
(2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.
(3) Indian Point Unit No. 2 UFSAR, Section 4.2.5.
(4) WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, S.L. Anderson, S.E. Yanichko, April 1975.
(5) ASME Boiler and Pressure Vessel Code, Section XI, 1996 Edition, Appendix G.
(6) DELETED
(7) DELETED
(8) Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, June 30, 1977.
(9) Supplement to Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, December 1980.
(10) Final Report - SWRI Project No. 02-5212 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y," E.B. Norris, November 1980.
(11) Final Report - SWRI Project No. 06-7379 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Z," E.B. Norris, April 1984.
(12) Final Report - SWRI Project No. 17-2108 (Revised)- "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule V," F.A. Iddings - SWRI, March, 1990.
(13) WCAP-15629, Revision 1, Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation
(14) WCAP-14040-NP-A, Rev.2, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.

4.2 INSERVICE INSPECTION AND TESTING

Applicability

Applies to the inservice inspection of Quality Group* A, B, and C components and the inservice testing of pumps and valves whose function is required for safety.

Objective

To provide assurance of the continued integrity and/or operability of those structures, systems, and components to which this specification is applicable.

Specifications

4.2.1 Inservice Testing

Inservice testing of pumps and valves whose function is required for safety shall be performed in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50, Section 50.55a(f), except where specific written relief pursuant to 10 CFR 50, Section 50.55a has been granted.

4.2.2 Inservice Inspection

Inservice inspection of Quality Group* (* Quality Group classification is in accordance with Revision 3 of Regulatory Guide 1.26.) A, B, and C components shall be performed in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code as Required by 10 CFR 50, Section 50.55a(g), except where specific written relief pursuant to 10 CFR 50, Section 50.55a has been granted.

4.2.3 Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods. The examinations schedules are shown in Table 4.2-1.

C. CONTAINMENT SYSTEMS

1. The containment vessel has an internal spray system which is capable of providing a distributed borated water spray of at least 2200 gpm.[3]

2. The containment vessel has an internal air recirculation system which includes five fan-cooler units (centrifugal fans and water cooled heat exchangers), with a total heat removal capability of at least 308.5 MBtu/hr under conditions following a loss-of-coolant accident and at service water temperature of 95°F.[4]

References

(1) UFSAR Section 5.1.2.2
(2) UFSAR Section 5.1.4
(3) UFSAR Section 6.3
(4) UFSAR Section 6.4

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall facility activities and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Plant Manager shall be responsible for facility operations and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

6.2.1 Facility Management and Technical Support

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Program Description (QAPD).

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The corporate officer with direct responsibility for the plant shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Operation Manager's and the Assistant Operation Manager's SRO license requirement which shall be in accordance with Technical Specification 6.2.2.h, and, (2) the Radiation Protection Manager who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 The Plant Manager shall meet or exceed the minimum qualifications specified for Plant Manager in ANSI N18.1-1971.

6.3.3 The Watch Engineer shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971.

6.4.2 DELETED

6.5 REVIEW AND AUDIT

6.5.1 The review and audit functions of the Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC) are described in the Quality Assurance Program Description (QAPD).

6.6 REPORTABLE EVENT ACTION

6.6.0 A Reportable Event is defined as any of the conditions specified in 10 CFR 50.73a(2).

6.6.1 The following actions shall be taken in the event of a Reportable Event: