ATTACHMENT II TO IPN-97-149

PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 13 EFFECTIVE FULL POWER YEARS

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

1.16 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR 50.

1.17 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Plant operation within these operating limits is addressed in individual specifications.

1.18 SHUTDOWN MARGIN

SHUTDOWN MARGIN (SDM) is the instantaneous amount of negative reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

1.19 PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.1.7. Plant operation within these operating limits is addressed in Sections 3.1.A.1, 3.1.A.8, 3.1.B and 4.3.A.

1.20 OVERPRESSURIZATION PROTECTION SYSTEM (OPS) ENABLE TEMPERATURE

The OPS ENABLE TEMPERATURE is the temperature at or below which the Overpressurization Protection System is required to be in service. This temperature is controlled and defined in the PTLR.

d. When the reactor coolant system T_{avg} is less than 200°F, but not in the refueling operation condition, and as permitted during special plant evolutions, at least one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation. This RHR pump may be out of service for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

e. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

f. The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation.

g. If the requirements of 3.1.A.1.e and 3.1.A.1.f above cannot be satisfied, the reactor shall be brought to the hot shutdown condition within 1 hour.

h. A reactor coolant pump (RCP) may not be started (or jogged) when the RCS cold leg temperature (T_{cold}) is at or below the OPS enable temperature, unless RCS make up is not in excess of RCS losses, and one of the following requirements is met:

(1) The OPS is <u>operable</u>, steam generator pressure is not decreasing, and the temperature of each steam generator is less than or equal to the coldest T_{cold}. Following the start of one or more RCPs and prior to reaching the OPS enable temperature, the RCS pressure shall not exceed that given by Figure 2.3 in the PTLR;

Or

(2) The OPS is <u>operable</u>, the temperature of the hottest steam generator exceeds the coldest T_{cold} by no more than 64°F, pressurizer level is at or below 73 percent, and T_{cold} is as per Figure 2.4 in the PTLR;

Or

(3) The OPS is <u>inoperable</u>, steam generator pressure is not decreasing, the temperature of each steam generator is less than or equal to the coldest T_{cold}, pressurizer level is at or below 73 percent, and the RCS pressure does not exceed that given by Figure 2.3 in the PTLR. The pressurizer level must be restricted per Figures 2.5 and 2.6 of the PTLR if operation below the OPS enable temperature exceeds 8 hours.

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7. REACTOR VESSEL HEAD VENTS

Whenever the reactor coolant system is above 350°F, two reactor vessel head vent paths consisting of two valves in series with power available from emergency buses shall be OPERABLE.

a. If one of the above reactor vessel head vent paths is inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path. Restore the inoperable vent path to operable status within 90 days, or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

b. With both reactor vessel head vent paths inoperable restore one vent path to operable status within 7 days or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

8. OVERPRESSURE PROTECTION SYSTEM (OPS)

a. When the RCS temperature is below the OPS enable temperature,

1. the OPS shall be armed and operable. Both OPS PORVs shall have lift settings not to exceed those given in Figure 2.3 of the PTLR, or

2. the RCS must be vented with an equivalent opening of 2.0 square inches.

b. The requirements of 3.1.A.8.a may be modified to allow one PORV and/or its series block valve to be inoperable for a maximum of seven (7) consecutive days.

c. If the requirements of 3.1.A.8.a or 3.1.A.8.b cannot be met, then one of the following actions shall be completed within 8 hours.

(1) The RCS must be depressurized and vented with an equivalent opening of at least 2.00 square inches;

Or

(2) The RCS must be heated in accordance with the restrictions of Specification 3.1.A.1.h(3) and maintained above the maximum OPS threshold temperature defined in Figure 2.3 of the PTLR;

Or

(3) Restrict pressurizer level as per the curves on Figures 2.5 and 2.6 of the PTLR.

d. In the event the PORV's or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.j within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor vessel head vent path ensures that capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

The OPS is designed to relieve the RCS pressure for certain unlikely incidents to prevent the peak RCS pressure from exceeding the limits established in Reg. Guide 1.99, Revision 2. The OPS is considered to be operable when the minimum number of required channels (per Table 3.5-3) are available to open the PORVs upon receipt of a high pressure signal which is based upon RCS T_{cold}, as shown in Figure 2.3 of the PTLR. The OPS setpoint is based upon a comparative analysis of References 5 and 9, with allowances for metal/fluid temperature differences (as described below) and for the static head due to elevation differences and dynamic head effect of the operation of four reactor coolant pumps. "Arming" means that the motor operated block valve (MOV) is in the open position. This can be accomplished either automatically by the OPS when the RCS temperature is less than or equal to the OPS enable temperature or manually by the control room operator.

The start of an RCP is allowed when the steam generators' temperature does not exceed the RCS and the OPS is operable. During all modes of operation, the steam generator temperature may be measured using the Control Room instrumentation or, as a backup, from a contact reading off the steam generator's shells.

Most start-ups will satisfy these requirements as provided in Specification 3.1.A.1.h (1). In order to allow start of an RCP when the steam generators are hotter than the RCS, requirements for a pressurizer bubble (gas or steam) are developed (technical specification value for pressurizer level includes an allowance for instrument uncertainty). During this Heat Input initiation event the RCS fluid temperature rise is considerably more rapid than the reactor vessel metal temperature rise. Since OPS utilizes a setpoint curve (Figure 2.3 in the PTLR) and the temperature measured is the fluid temperature, and not the reactor vessel metal, it is necessary to shift to the right the OPS setpoint curve by 50°F to ensure the pressure does not exceed the allowable values for the vessel. For the conditions when the OPS is inoperable, additional requirements are developed for the pressurizer bubble, RCS pressure and temperature.

Due to the rate of energy transferred to the RCS, when the RCP is started, the resultant rate of temperature rise and the pressure increase are strongly dependent on the temperature difference between the RCS and the steam generators. The presence of a pressurizer bubble provides for a more moderate pressure increase. The bubble size is sufficient to prevent the RCS from going water solid for 10 minutes during which time operator action will terminate the pressure transient. Pressurizer level refers to indicated level and includes instrument uncertainty. The preventive measures for a Mass Input initiating event (i.e., up to three charging pumps or one SI pump) as well as the Heat Input initiating event are described in References (3), (4) and (5). (Also refer to Specifications 3.3.A.8, 3.3.A.9, and 3.3.A.10). The OPS need not be operable when the RCS temperature is less than the OPS enable temperature if the RCS is depressurized and vented with an equivalent opening of at least 2.00 square inches. One PORV, blocked fully open, also satisfies this vent area requirement. This opening is adequate to relieve the worst case analyzed. It should be noted that the analysis of record (Reference 5) is based upon a minimum vent area of 1.4 square inches, which for the sake of conservatism has been rounded up to 2.0 square inches.

The OPS enable temperature permits the performance of an RCS hydrostatic test (see Fig. 2.7 in the PTLR) without activating the OPS.

Upon OPS inoperability, the RCS may be heated above the temperature that is the value for which the RCS heatup and cooldown curves (Figures 2.1 and 2.2 in the PTLR) permit pressurization to the setting of the pressurizer safety valves. At these conditions, the pressurizer safety valves will preclude violation of the 10 CFR 50, Appendix G, curves. In addition, the OPS need not be operable upon satisfying the conditions of Specification 3.1.A.8.c(3) which requires the presence of a pressurizer bubble to preclude RCS overpressurization during inadvertent mass inputs. Specification 3.1.A.8.c(3) also places restrictions on the number of charging and SI pumps capable of feeding the RCS (see Specifications 3.3.A.8, 3.3.A.9, and 3.3.A.10). Any pump can be rendered incapable of feeding the RCS if, for example, its switch is in the trip pull-out position, or if at least one valve in the flow path to the RCS is closed and locked (if manual) or de-energized (if motor operated). This section has also been revised in accordance with the results of tests conducted on the capsule T, Y, and Z specimens (References 6, 7 and 8).

References

1) FSAR Section 14.1.6
2) FSAR Section 14.1.8
3) Letter dated 10/25/78 "Summary of Changes to IP-3 Plant Operating Procedures in Order to Preclude RCS Overpressurization"
4) Letter dated 2/28/76 "Conceptual Design of the Reactor Coolant Overpressure Protection System" and response to NRC questions.
5) IP-3 Low Temperature Overpressurization Protection System Analysis, NYPA Report dated 8/24/84.
6) WCAP-9491 "Analysis of Capsule T from IP-3 Reactor Vessel Radiation Surveillance Program", J.A. Davidson, S.L. Anderson, W.T. Kaiser, April 1979.
7) WCAP-10300-1, "Analysis of Capsule Y from the Power Authority of the State of New York Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, March 1993.
8) WCAP-11815, "Analysis of Capsule Z from the New York Power Authority Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, L. Albertin, March 1988.
9) ASME Code Case N-514, "Low Temperature Overpressure Protection," February 12, 1992.

FIGURE 3.1.A-1

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Amendment No. 67, 121,

Figure 3.1.A-2

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FIGURE 3.1.A-3

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Amendment No. 67, 121, 154,

FIGURE 3.1.A-4

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FIGURE 3.1.A-5

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FIGURE 3.1.A-6

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B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figures 2.1 and 2.2 in the PTLR. The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr respectively.

   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.

2. The limit lines shown in PTLR Figures 2.1 and 2.2 shall be recalculated periodically using methods discussed in the Basis and results of surveillance specimens as covered in Specification 4.2. The order of specimen removal may be modified based on the results of testing of previously removed specimens.

3. 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.

4. 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.

5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3.

Basis

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code [6] and ASTM E185 [5] and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code [1], and the calculation methods described in WCAP-7924 [2].

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by Westinghouse Corporation and the results have been evaluated and reported [7]. Similar reports were prepared for the surveillance capsules [10, 8] removed in 1982 and 1987. Based on the Westinghouse evaluation, heatup and cooldown curves were developed for up to the service life identified in all Figures of the PTLR.

Generic Letter 88-11 requested that licensees use the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A. of 10 CFR part 50, Appendix G. Capsule Z was analyzed [8] and new pressure-temperature curves were developed using this methodology.

The maximum shift in RT_{NDT} for the analyzed service life is identified in Figures 2.1 and 2.2 of the PTLR. The limiting plate in the reactor vessel is Plate B2803-3, which is also more limiting than all reactor vessel welds.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of the analyzed years of service life. The service life period is chosen such that the limiting RT_{NDT} at the 1/4 T location in the core region is higher than the RT_{NDT} of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table Q4.2-1 [3].

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point by point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve becomes mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at 1/4 T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4 T location and compressive stresses at the 3/4 T position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel I.D. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 2.2 of the PTLR represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculations are provided in WCAP-7924 [2]. Information on the specific calculations used to develop the current heatup-cooldown curves can be found in Reference 9.

FIGURE 3.1-1

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FIGURE 3.1-2

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d. One pressure and one level transmitter shall be operating per accumulator.

e. Three safety injection pumps together with their associated piping and valves are operable.

f. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

g. Two recirculation pumps together with the associated piping and valves are operable.

h. Valves 856B and 856G in the Safety Injection discharge headers shall be closed and their power supplies de-energized.

i. Valve 1810 in the suction line of the high-level SI pumps and valves 882 and 744 in the suction and discharge lines, respectively, of the residual heat removal pumps shall be open and their power supplies de-energized.

j. Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the RWST are de-energized in the open position.

k. The refueling water storage tank low level alarms are operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

l. Valve 883 in the RHR return line to the RWST is de-energized in the closed position.

m. Valves 1870 and 743 in the miniflow line for the Residual Heat Removal Pumps shall be open and their power supplies de-energized.

n. The RHR system is in the ESF alignment with the normal RHR suction line isolated from the RCS.

4. The requirements of 3.3.A.3 may be modified to allow any one of the following components to be inoperable at any one time:

2) RCS temperature and the source range detectors are monitored hourly;

and

3) no operations are permitted which would reduce the boron concentration of the reactor coolant system.

8. When the RCS average cold leg temperature (T_{cold}) is below the OPS enable temperature, or when RHR is in service (i.e., not isolated from the RCS), no safety injection pumps shall be energized and aligned to feed the RCS.

9. The requirements of 3.3.A.8 may be relaxed to allow one safety injection pump energized and aligned to feed the RCS under the following circumstances:

   a. emergency boration; OR

   b. for pump testing, for a period not to exceed 8 hours; OR

   c. loss of RHR cooling.

10. The requirements of 3.3.A.8 may be further relaxed when the RCS is < 200°F, such that two safety injection pumps may be energized and aligned to feed the RCS under the following circumstances:

   a. the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety valve flange, OR

   b. indicated pressurizer level is at 0%. (Alternate methods and instrumentation may be used to confirm actual RCS elevation.)

B. Containment Cooling and Iodine Removal Systems

1. The reactor shall not be brought above the cold shutdown condition unless the following requirements are met:

   a. The spray additive tank contains a minimum of 4000 gallons of solution with a sodium hydroxide concentration ≥35% and ≤38% by weight.

   b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. The requirements of 3.3.B.1 may be modified to allow any one of the following components to be inoperable at one time:

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With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.[3] The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of their performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.[13]

During operating modes in the temperature range between 200°F and 350°F, a sufficient decay heat removal capability is provided by a reactor coolant pump with a steam generator heat sink or a residual heat removal loop. This redundancy ensures that a single failure will not result in a complete loss of decay heat removal. Above 350°F, the normal RHR suction line is isolated from the RCS to protect RHR piping from overpressurization due to inadvertent SI pump actuation.

During operating modes when the reactor coolant T_{avg} is less than 200°F, but not in the refueling operation condition, a sufficient decay heat removal capability is provided by a residual heat removal loop.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95°F).[4] In the event of a Design Basis Accident, any one of the following configurations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also in the event of a Design Basis Accident, any one of three configurations of fan-cooler units (with charcoal filters) and/or containment spray pumps (with sodium hydroxide addition) will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values.[5] Any one of these three configurations constitutes the minimum safeguards for iodine removal.

The combination of three fan-coolers and one containment spray pump is capable of being operated on emergency power with one diesel generator failing to start. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

These toxic gas monitoring systems are designed to alarm in the control room upon detection of the short term exposure limit (STEL) value. The operability of the toxic gas monitoring systems provides assurance that the control room operators will have adequate time to take protective action in the event of an accidental toxic gas release. Selection of the gases to be monitored are based on the results described in the Indian Point Unit 3 Habitability Study for the Control Room, dated July, 1981. The alarm setpoints will be in accordance with industrial ventilation standards as defined by the American Conference of Governmental Industrial Hygienists.[16]

The RHR suction line is required to be isolated from the RCS when temperature is above 350°F. This protects the RHR system from overpressurization when the SI system is required to be in service. The requirement to prevent safety injection pumps from being able to feed the RCS under specific conditions prevents overpressurization of the RHR system or the RCS beyond the capacity of the OPS to mitigate. These conditions include when OPS is required to be in service and when RHR is in service. Special allowances are made for pump testing, loss of RHR cooling (during which time an SI pump may be required to recirculate coolant to the core), or emergency boration. Two SI pumps may be energized and aligned to feed the RCS when situations prevail that could not result in overpressurization. This is satisfied when the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety flange or when the pressurizer level is low enough (indicating 0%) to ensure at least a ten minute operator response time on inadvertent SI actuation without the pressurizer completely filling. Alternate methods and instrumentation may be used to confirm actual RCS elevation. Methods to ensure that an SI pump is unable to feed the RCS include placing the SI pump switches in the trip pull-out position, or by closing and locking (if manual) or de-energizing (if motor operated) at least one valve in the flow path from these pumps to the RCS.

References

1) FSAR Section 9
2) FSAR Section 6.2
3) FSAR Section 6.2
4) FSAR Section 6.3
5) FSAR Section 14.3.5
6) FSAR Section 1.2
7) FSAR Section 8.2
8) FSAR Section 9.6.1
9) FSAR Section 14.3
10) FSAR Section 6.8
11) FSAR Section 6.5
12) Response to Question 14.6, FSAR Volume 7
13) FSAR Appendix 14C
14) Response to Question 9.35, FSAR Volume 7
15) WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95° at IP-3"
16) American Conference of Governmental Industrial Hygienists 1982 Industrial Ventilation, 19th Edition
17) NYPA calculation IP3-CALC-SI-00725, Rev. 0, "Instrument Loop Accuracy/Setpoint Calc./RWST Level."
18) Nuclear Safety Evaluation 93-3-162-SI, Rev. 0, Adequate Post-LOCA Coolant Inventory.

4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

A. Reactor Coolant System Integrity Testing

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following refueling, repair, replacement or modification.

Specification

a) 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 ASME Section XI.

b) Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of ASME Section XI.

c) The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 2.7 in the PTLR for the applicable service period. This figure will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure 2.2 in the PTLR.

Basis

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.

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.

The inservice leak test temperatures are shown on Figure 2.7 of the PTLR. The temperatures are calculated in accordance with ASME Code Section III, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

Amendment No. 28, 101, 100, 121, 170, 171,

RT_{NDT} predictions for the applicable service period are shown on PTLR Figure 2.7, with the accompanying margins to safety limits.

The heatup limits specified in Figure 2.7 of the PTLR, must not be exceeded while the reactor coolant system is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of PTLR Figure 2.2 must not be exceeded. PTLR Figures 2.2 and 2.7 are recalculated periodically, using methods discussed in the Basis for Specification 3.1.B and results of surveillance specimens, as covered in Specification 4.2.

Reference

1. FSAR, Section 4.

FIGURE 4.3-1

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3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety limits are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 Pressure and Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. OPS limits for specification 3.1.A.8,

2. Heatup and cooldown limits for specification 3.1.B, and

3. Pressure/temperature limits for the RCS leak test for specification 4.3.A.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. "Final Report on the Pressure-Temperature Limits for Indian Point 3 Nuclear Power Plant," July 1990. (Approved as part of Amendment 109.)
(Methodology for specifications 3.1.A.8, 3.1.B, and 4.3.B)

2. ASME Code Case N-514, "Low Temperature Overpressure Protection," February 1992. (Approved by NRC for use at IP3 through an exemption to 10 CFR 50.60.)
(Methodology for specification 3.1.A.8.)

c. The PTLR shall be provided to the NRC within 45 days of issuance for each reactor vessel fluence period and for any revision or supplement thereto.

Amendment No. 10, 11, 12, 31, 44, 51, 59, 63, 66, 67, 83, 88, 103, 108, 116, 117, 157,

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator-Region 1 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification;

a. Sealed source leakage on excess of limits (Specification 3.9)

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Seismic event analysis (Specification 4.10)

d. Inoperable plant vent sampling, main steam line radiation monitoring or effluent monitoring capability (Table 3.5-4, items 5, 6 and 7)

e. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

f. Deleted.

g. Release of radioactive effluents in excess of limits (Appendix B Specifications 2.3, 2.4, 2.5, 2.6)

h. Inoperable containment high-range radiation monitors (Table 3.5-5, Item 24)

i. Radioactive environmental sampling results in excess of reporting levels (Appendix B Specification 2.7, 2.8, 2.9)

j. Operation of Overpressure Protection System (Specification 3.1.A.8.d)

k. Operation of Toxic Gas Monitoring Systems (Specification 3.3.H.3.)

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspection, repair and replacements of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all source material of record.

i. Records of reactor tests and experiments.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transient cycles.

g. Records of training and qualifications for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the SRC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of secondary water sampling and water quality.

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and records showing that these procedures were followed.

o. Records of service lives of all safety-related hydraulic snubbers including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION AND RESPIRATORY PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure so as to maintain exposures as far below the limits specified in 10 CFR Part 20 as reasonably achievable. Pursuant to 10 CFR 20.103, allowance shall be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this plant in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1 of 10 CFR 20.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203 (c) (2) of 10-CFR 20, each high radiation area in which the intensity of radiation is 1000 mrem/hr or less and 100 mrem/hr or greater shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

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* Health Physics Personnel shall be exempt from the RWP issuance requirements for entries into high radiation areas during the performances of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1 above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the plant Radiological and Environmental Services Manager or his designee.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 Environmental qualification of electric equipment important to safety shall be in accordance with the provisions of 10 CFR 50.49. Pursuant to 10 CFR 50.49, Section 50.49 (d), the EQ Master List identifies electrical equipment requiring environmental qualification.

6.13.2 Complete and auditable records which describe the environmental qualification method used, for all electrical equipment identified in the EQ Master List, in sufficient detail to document the degree of compliance with the appropriate requirements of 10 CFR 50.49 shall be available and maintained at a central location. Such records shall be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995" as modified by the following exception:

a. ANS 56.8 - 1994, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.39 psig. The minimum test pressure is 42.42 psig.

Amendment No. 11, 39 (Order dated October 24, 1980), 88, 101, 103, 116, 117, 162, 174,

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

Leakage acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and C tests and $\leq 0.75 L_a$ for Type A tests;

b. Air lock testing acceptance criteria are :

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

   2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

c. Isolation valves sealed with the service water system leakage rate into containment acceptance criterion is ≤ 0.36 gpm per fan cooler unit

d. Isolation Valve Seal Water System leakage rate acceptance criterion is 14,700 cc/hr at $1.1P_a$.

The provisions of Specification 1.12 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program. The provisions of Specification 4.1, "Applicability," as they relate to delay of 24 hours in applying an LCO following the discovery of a surveillance test not performed, are applicable to the Primary Containment Leakage Rate Testing Program.