ATTACHMENT I TO IPN-96-063

PROPOSED TECHNICAL SPECIFICATION CHANGE

REGARDING MAXIMUM ALLOWABLE LEAKAGE FROM

RESIDUAL HEAT REMOVAL AND SAFETY INJECTION

SYSTEMS OUTSIDE CONTAINMENT

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

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 cold leg temperature (T_{cold}) is at or below 332°F, no more than one safety injection pump shall be energized and aligned to feed the RCS.

9. During power operation, total leakage from RHR and SI systems outside containment shall not exceed 1.09 gallons per hour. If this requirement is not met, reduce the total leakage to within the allowable limit within 7 days. If leakage cannot be reduced to within the allowable limit within 7 days, the reactor shall be in hot shutdown within 4 hours and in cold shutdown within the next 24 hours.

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 $\geq$35% and $\leq$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:

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.

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.

During power operation, the total leakage from the RHR and SI systems outside containment is limited to 1.09 gallons per hour to ensure that the control room habitability design basis is met. The allowed time of 7 days to reduce the total leakage within the maximum allowable limit is based on a probabilistic risk analysis using results from the Individual Plant Examination.

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.

I. Residual Heat Removal System

1. Test

a. (1) The portion of the Residual Heat Removal System that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 350 psig at the interval specified below.

(2) The piping between the residual heat removal pumps suctions and the containment isolation valves in the residual heat removal pump suction line from the containment sump shall be hydrostatically tested at no less than 100 psig at the interval specified below.

b. Visual inspection shall be made for excessive leakage during these tests from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

2. Leakage Acceptance Criterion

The total leakage from RHR and SI systems outside containment shall not exceed 1.09 gallons per hour.

3. Corrective Action

Repairs or isolation shall be made as required by specification 3.3.A.9 to maintain leakage within the acceptance criterion.

4. Test Frequency

Tests of the Residual Heat Removal System shall be conducted at least once per 24 months.

A full pressure test of the air lock will be periodically performed at 6-month intervals to detect any unanticipated leakage.

The containment isolation valve leakage and sensitive leakage rate measurements obtained periodically, periodic inspection of accessible portions of the containment wall to detect possible damage to the liner plates, combined with the leakage monitoring afforded by the weld Channel and Penetration Pressurization System, [5] and IVSWS [6] provide assurance that the containment leakage is within design limits.

The testing of containment isolation valves in Table 4.4-1 either individually or in groups, utilizes the WC & PPS [5] or IVSWS [6] (where appropriate), and is in accordance with the requirements of Type C tests in Appendix J (issue effective date March 16, 1973) to 10CFR50, except for the surveillance frequency. The 25% increase in surveillance frequency allowed (from a maximum of 24 months to a maximum of 30 months) was compensated for by a proportionate increase in the margin between the specified allowable leakage and the maximum allowable leakage. (The specified allowable leakage was decreased from 0.6 L_a to 0.5 L_a.) The specified test pressures are greater than the peak calculated accident pressure. Sufficient water is available in the Isolation Valve Seal Water System, Primary Water System, Service Water System, Residual Heat Removal System, and the City Water System to assure a sealing function for at least 30 days. The leakage limit for the Isolation Valve Seal Water System is consistent with the design capacity of the Isolation Valve Seal Water supply tank.

The acceptance criterion of 0.5 L_a for the combined leakage of isolation valves subject to gas or nitrogen pressurization, the air lock, containment penetrations and double-gasketed seals accounts for possible degradation of the containment leakage barriers for a 30 month test interval.

The 350 psig test pressure, achieved either by normal Residual Heat Removal System operation or hydrostatic testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the containment sump return line of 100 psig gives an adequate margin over the highest pressure within the line after a design basis accident. The maximum allowable leakage of 1.09 gallons per hour includes leakage from the Residual Heat Removal (RHR) and the Safety Injection (SI) systems outside containment. Both the RHR and SI systems could be exposed to highly radioactive fluids during the recirculation phase after a design basis loss of coolant accident. The maximum allowable leakage limit ensures that the potential overall post-loss of coolant accident doses to the control room personnel and offsite receptors are within the regulatory limits.

ATTACHMENT II TO IPN-96-063

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGE REGARDING MAXIMUM ALLOWABLE LEAKAGE FROM RESIDUAL HEAT REMOVAL AND SAFETY INJECTION SYSTEMS OUTSIDE CONTAINMENT

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

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGE REGARDING MAXIMUM ALLOWABLE LEAKAGE FROM THE RESIDUAL HEAT REMOVAL (RHR) AND SAFETY INJECTION (SI) SYSTEMS LOCATED OUTSIDE CONTAINMENT

Section I - Description of Changes

This application for amendment to the Indian Point 3 Technical Specifications (TS) proposes to amend Sections 3.3 and 4.4 and the Bases of Appendix A of the Operating License. The proposed change would add a Limiting Condition for Operation (LCO) for the maximum allowable leakage from the Residual Heat Removal (RHR) and Safety Injection (SI) systems outside containment, and revise the maximum allowable leakage from 2 gallons per hour (gph) to 1.09 gph. This TS change would ensure that the control room habitability design basis for post accident leakage outside containment is met.

On page 3.3-5a: Add item 9, to read,"During power operation, total leakage from RHR and SI systems ----------cold shutdown within the next 24 hours."

On page 3.3-17: Add a new paragraph after the third paragraph to read, "During power operation, the total leakage from----------
----------results from the Individual Plant Examination."

On page 4.4-6: Revise item 2 to read, "Leakage Acceptance Criterion," "The total leakage from ----------exceed 1.09 gallons per hour."
Revise item 3 to read, "Corrective Action,"Repairs or isolation shall be made as required by ----------acceptance criterion."

On page 4.4-9: Revise the last paragraph to delete, "A recirculation system----------
----------design basis accident, and add, " The maximum allowable leakage of 1.09 gallons per hour----------
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----------regulatory limits."

Section II - Evaluation of Changes

Current TS 4.4.I.2 states that the maximum allowable leakage from the RHR system components located outside of the containment shall not exceed 2 gallons per hour. The Power Authority has interpreted this specification to intend that the maximum leakage from components of the RHR and the SI systems outside containment, that would be used in the recirculation phase after a design basis loss of coolant accident, shall not exceed this limit (Reference 3). During the design basis reconstitution effort to determine the operability

requirements of the Primary Auxiliary Building (PAB) Ventilation and Filtration system, the Authority determined that a control room habitability calculation from 1981 included a value of 0.7 gph for leakage, which was the actual leakage for post accident external recirculation system leakage at the time, and did not take into account the TS limit of 2 gph (Reference 4). The original calculation, using 0.7 gph, concluded that post accident operation of the PAB Ventilation and Filtration system was not required for control room habitability. In order to resolve the above discrepancy, a new control room habitability calculation was performed and the Power Authority determined that in order to ensure post accident control room habitability without the use of the PAB Ventilation and Filtration system, the external recirculation system leakage should be limited to 1.34 gph (Reference 5).

Therefore, in order to ensure the control room habitability design basis is met, the Authority proposes to change the TS leakage acceptance criterion for RHR and SI system components outside containment from 2 gph to a lower value, namely 1.09 gph. The procedure for tabulating RHR and SI system leakage has already been revised to provide an allowable leakage limit that meets the requirements of the proposed TS change. With a postulated post-LOCA RHR and SI leakage at this new limit, the overall dose to the critical receptor/organ (control room/thyroid) due to both vapor containment leakage and unfiltered RHR and SI leakage would be approximately 27 rem. This provides a 10% margin below the regulatory limit of 30 rem for control room dose to accommodate potential post-LOCA releases via other pathways. The Authority conducted a probabilistic risk analysis calculation using results from the Individual Plant Examination (IPE) to determine an LCO that would establish how long the plant can operate at power before shutting down due to total RHR and SI leakage in excess of the TS limit. The radiological release frequency associated with LOCA-induced core damage sequences from the IPE yielded the LCO results (Reference 6). Based on the result of this calculation an LCO has been added to the TS to reduce the leakage from RHR and SI system components within the TS limit within 7 days. If leakage cannot be reduced to within the TS limit within this time period, the reactor shall be in hot shutdown within 4 hours and cold shutdown within the following 24 hours.

Section III - No Significant Hazards Evaluation

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change would make the RHR and SI system leakage limit more restrictive to ensure control room habitability after a design basis loss of coolant accident. This will ensure

no increase in the consequences of an accident. The proposed change has added an LCO when the total RHR and SI system leakage outside containment exceeds the TS limit. The allowed time when the leakage exceeds the proposed TS limit is based on a probabilistic risk analysis using results from the IPE. The radiological release frequency associated with LOCA-induced core damage sequences from the IPE yielded the LCO results (Reference 6). Based on the result of this calculation an LCO has been added to the TS to reduce the leakage from RHR and SI system components within the TS limit within 7 days. If leakage cannot be reduced to within the TS limit within this time period, the reactor shall be in hot shutdown within 4 hours and cold shutdown within the following 24 hours. The maximum allowed leakage has a margin of 10% below the regulatory limit for control room dose to accommodate potential post-LOCA releases via other pathways. The proposed change involves a TS limit for systems used to mitigate the consequences of an accident. No new modes of operation are being proposed. Therefore, the proposed change would have no effect on the probability of an accident previously evaluated.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. This proposed change involves a TS limit for systems used to mitigate the consequences of a previously evaluated accident. The proposed change would make the leakage limit of the RHR and SI system components outside containment more restrictive. No new modes of operation for any system or equipment are being introduced. Therefore, the proposed change to reduce the TS limit would not create the possibility of a new or different kind of accident.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed amendment does not involve a significant reduction in a margin of safety. The TS change would make the total leakage limit of the RHR and SI system components located outside containment more restrictive. This will ensure the regulatory limit for control room dose is not exceeded with a margin of 10%. This change will add an LCO and provide an allowed time to reduce the leakage within allowable limit. If this cannot be accomplished, the reactor would be brought to hot shutdown condition in 4 hours and cold shutdown condition in 24 hours and this change is based on probabilistic risk analysis. This analysis used radiological release frequency associated with LOCA-induced core damage sequences from the IPE to yield the LCO results. Therefore, the proposed change would not involve a significant

reduction in a margin of safety.

Section IV - Impact Changes

These changes will not adversely affect the following:
ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of these changes: a) will not involve a significant increase in the probability or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not create the possibility of a new or different kind of accident from any accident previously evaluated in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any technical specification; and d) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

1) IP3 FSAR
2) IP3 SER
3) IP3-TSI-014
4) IP3 Licensee Event Report No. 95-016-00
5) IP3-CALC-RAD-0021
6) IP3-CALC-MULT-01589