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James Knubel
Senior Vice President and
Chief Nuclear Officer

February 4, 2000
IPN-00-008

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Proposed Change to the Technical Specifications
Regarding Revised Main Steam Line Break Accident

Dear Sir:

This application for an amendment to Indian Point 3 Technical Specifications (TS) proposes to revise TS to reflect the cycle 11 main steam line break (MSLB) analysis. The cycle 11 MSLB analysis was revised to correct the assumption for non-isolatable feedwater and also includes revised assumptions regarding boron in the safety injection piping and shutdown margin. The proposed revision adds a limiting condition for operation and associated action statements to TS 3.3.A for boron in safety injection line, a surveillance test to Table 4.1-2 to verify the presence of boron, a change to the non-conservative shutdown reactivity requirement in TS 3.10.1.1, and a revision to TS 6.14 to reflect an associated change in the peak calculated containment pressure following a postulated MSLB.

Enclosed for filing is the signed original of a document entitled, "Application for Amendment to Operating License." Attachment I to the application for amendment contains the proposed changes to the Technical Specifications. Attachment II to the application contains the associated safety evaluation. Attachments III and IV contain, for information only, a markup to show the Technical Specification change and a markup of the changes to the Improved Technical Specification submittal.

No new commitments are made by the Power Authority in this submittal.

In accordance with 10 CFR 50.91, a copy of this application and the associated attachments are being submitted to the designated New York State official.

A001

If you have any questions, please contact Mr. K. Peters.

Very truly yours,

HARRY P. SALMON JR.
VP ENGINEERING
for J. Knubel
Senior Vice President and
Chief Nuclear Officer

Attachments

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BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
POWER AUTHORITY OF THE STATE OF NEW YORK	)	**Docket No. 50-286**
Indian Point 3 Nuclear Power Plant	)	

APPLICATION FOR AMENDMENT TO THE OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the Nuclear Regulatory Commission, the Power Authority of the State of New York, as holder of the Facility Operating License No. DPR-64, hereby applies for an amendment to the Technical Specifications contained in Appendix A of the license.

This application for an amendment to Indian Point 3 Technical Specifications (TS) proposes to revise TS to reflect the cycle 11 main steam line break (MSLB) analysis. The cycle 11 MSLB analysis was revised to correct the assumption for non-isolatable feedwater and also includes revised assumptions regarding boron in the safety injection piping and shutdown margin. The proposed revision adds a limiting condition for operation and associated action statements to TS 3.3.A for boron in safety injection line, a surveillance test to Table 4.1-2 to verify the presence of boron, a change to the non-conservative shutdown reactivity requirement in TS 3.10.1.1, and a revision to TS 6.14 to reflect an associated change in the peak calculated containment pressure following a postulated MSLB.

Attachment I to the application for amendment contains the proposed changes to the Technical Specifications. Attachment II to the application contains the associated safety evaluation. Attachments III and IV contain, for information only, a markup to show the Technical Specification change and a markup of the changes to the Improved Technical Specification submittal.

POWER AUTHORITY OF THE
STATE OF NEW YORK

VP ENGINEERING
for J. Knubel
Senior Vice President and
Chief Nuclear Officer

STATE OF NEW YORK
COUNTY OF WESTCHESTER
Subscribed and sworn to before me

this 4th day of February, 2000.

Geraldine Strand
Notary Public

GERALDINE STRAND
Notary Public, State of New York
No. 4991272
Qualified in Westchester County
Commission Expires Jan. 27, 2002

ATTACHMENT I TO IPN-00-008

PROPOSED TECHNICAL SPECIFICATION CHANGE

REGARDING REVISED MAIN STEAM LINE BREAK ACCIDENT

Revise Appendix A as follows:

Remove Page	Insert Page
3.1-9	3.1-9
3.3-5	3.3-5
3.3-5a	3.3-5a
3.3-6	3.3-6
3.3-15	3.3-15
3.3-18	3.3-18
3.3-21	3.3-21
3.10-1	3.10-1
3.10-15	3.10-15
Table 4.1-2 (Sheet 1 of 2)	Table 4.1-2 (Sheet 1 of 2)
Table 4.1-2 (Sheet 2 of 2)	Table 4.1-2 (Sheet 2 of 2)
4.4-7	4.4-7
4.4-10	4.4-10
6-22	6-22

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

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 3.1.A-2) 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 and/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.9, 3.3.A.10, and 3.3.A.11.) The OPS need not be operable when the RCS temperature is less than 319°F 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 2.00 square inches.

The OPS enable temperature of 319°F permits the performance of an RCS hydrostatic test (see Fig. 4.3-1) without activating the OPS.

Upon OPS inoperability, the RCS may be heated above 411°F. This temperature is that value for which the RCS heatup and cooldown curves (Figures 3.1-1 and 3.1-2) permit pressurization to the setting of the pressurizer safety valves. Accordingly, with an inoperable OPS and an RCS temperature above 411°F, 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.9.c(3) which requires the presence of a pressurizer bubble to preclude RCS overpressurization during inadvertent mass inputs. Specification 3.1.A.9.c(3) also places restrictions on the number of charging and SI pumps capable of feeding the RCS (see Specifications 3.3.A.9, 3.3.A.10, and 3.3.A.11). Any pump can be

5. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.3 within the time periods specified in 3.3.A.4; then:

   a. If the reactor is critical, it shall be in the hot shutdown condition within four hours and the cold shutdown condition within the following 24 hours.

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.A.3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition using normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

6. The reactor shall not exceed hot shutdown conditions unless a boron concentration of ≥2300 ppm is maintained in safety injection piping from the RWST to the safety injection pumps, and in the Boron Injection Tank (BIT) header from the safety injection pumps up to and including the BIT. If the boron concentration is not within limits, restore the boron concentration to required limits within 8 hours. If the boron concentration is not restored to within limits within 8 hours, be in hot shutdown within 6 hours.

7. When the reactor coolant system T_{avg} is greater than 200°F and less than 350°F, the following decay heat removal requirements shall be met:

   a. Two residual heat removal pumps together with their associated heat exchangers, piping, and valves shall be operable,

   OR

   b. A minimum of one residual heat removal pump and heat exchanger and a minimum of one reactor coolant pump and steam generator together with their associated piping and valves, shall be operable,

   OR

   c. A minimum of two reactor coolant pumps and two steam generators, together with their associated piping and valves, shall be operable,

   OR

   d. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system. Otherwise, if sufficient equipment is available, be in cold shutdown within 20 hours.

8. When the reactor coolant T_{avg} is less than 200°F, but not in the refueling operation condition, two residual heat removal pumps, together with their associated heat exchangers, piping and valves, shall be operable.

a. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system.

b. The above requirements may be suspended during maintenance, modifications, testing, inspection or repair provided that:

1) an alternate means of decay heat removal is available and return of the system within sufficient time to prevent exceeding cold shutdown requirements is assured;

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.

9. When the RCS average cold leg temperature (T_{cold}) is below 319°F, 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.

10. The requirements of 3.3.A.9 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.

11. The requirements of 3.3.A.9 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% and the plant is vented in accordance with Technical Specification 3.1.A.9.c.1. (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 $\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:

   a. Fan cooler unit 32, 34, or 35 or the flow path for fan cooler unit 32, 34, or 35 may be out of service for a period not to exceed 24 hours provided both containment spray pumps are operable.

   OR

   Fan cooler unit 31 or 33, or the flow path for fan cooler unit 31 or 33 may be out of service for a period not to exceed 7 days provided both containment spray pumps are operable.

   b. One containment spray pump may be out of service for a period not to exceed 24 hours, provided the five fan cooler units are operable.

   c. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to an operable status within 24 hours and all valves in the system that provide the duplicate function are operable.

3. If the Containment Cooling and Iodine Removal are not restored to meet the requirements of 3.3.B.1 within the time period specified in 3.3.B.2, then:

   a. If the reactor is critical, it shall be in the hot shutdown condition within four hours and in the cold shutdown condition within the following 24 hours.

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.B.1 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would damage the fuel elements or result in a release in excess of 10 CFR 100 and 10 CFR 50 dose limits.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and, therefore, in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are based on two considerations:

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power, the magnitude of the decay heat decreases after initiating hot shutdown. Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 1 hour of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case the reactor is to be put into the cold shutdown condition.

The limits for the Refueling Water Storage Tank and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.[(9)(13)]

The limiting MSLB accident analysis for containment peak pressure is the 102% power case. This case assumes a boron concentration in the safety injection piping from the RWST to the safety injection pumps and, in the BIT header, from the safety injection pump up to and including the BIT. Surveillance testing is required by TS Table 4.2-1 and, if RHR cooling has been used, RWST recirculation assures no dilution. The containment peak pressure analysis at hot zero power continues to assume no boron concentration in this piping.

Due to the distribution of the five fan cooler units and two containment spray pumps on the 480 volt buses, the closeness to which the combined equipment approaches minimum safeguards varies with which particular component is out of service. Accordingly, the allowable out of service periods vary according to which component is out of service. Under no conditions do the combined equipment degrade below minimum safeguards.

The seven day out of service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is consistent with W Standardized Technical Specifications. This is allowable because no credit has been taken for operation of these systems in the calculation of off-site accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems.[11] The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems for the maximum containment calculated peak accident. A WC & PPS zone is considered that portion of piping downstream of the air receiver discharge check valve up to the last component pressurized by that system portion.

Some portions of the Weld Channel Pressurization System (WCPS) piping would not be practicably accessible for repair if they became inoperable. A section of WCPS piping is considered to be inoperable if it brings the air consumption of the WC & PPS above the required 0.2% of the containment volume per day or if the section can not maintain a pressure above the required 43 psig. If it is determined, by written evaluation, that an inoperable section of piping is not practicably accessable for repair, then that portion of the WCPS may be disconnected from the system. Inoperable sections of WCPS piping which can be considered for disconnection will satisfy one of the following criteria: 1) the piping is covered by concrete and repairs of the piping would involve the removal of some portion of the containment structure; or 2) the piping is located behind plant equipment in the containment building and repairs of the piping would involve the relocation of the equipment. The integrity of the welds associated with any disconnected portions of the WCPS is verified by integrated leak rate testing. The provision that allows for the disconnection of portions of the WCPS piping does not apply to any other WC & PPS piping.

The Component Cooling System is not required during the injection phase of a loss-of-coolant accident. The component cooling pumps are located in the Primary Auxiliary Building and are accessible for repair after a loss-of-coolant accident.[6] During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards.[7]

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%) and the plant is vented in accordance with Technical Specification 3.1.A.8.c.1 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
15) Deleted.
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.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distribution and to limits on control rod operations.

Objectives:

To ensure:

1. Core subcriticality after reactor trip.
2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and
3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

3.10.1 Shutdown Reactivity

3.10.1.1 Whenever $T_{avg} > 200°F$ the shutdown margin shall be $\geq 1.3\%$ $\Delta k/k$. For cycle 11, the shutdown margin shall be $\geq 2.9\%$ $\Delta k/k$ whenever hot shutdown is exceeded.

3.10.1.2 When the conditions of specification 3.10.1.1 are not met, initiate boration to restore shutdown margin within limit.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_Q\ (Z) \leq (F_Q^{RTP}/p)\ \times K(Z)$ for $P > 0.5$

$F_Q\ (Z) \leq (F_Q^{RTP}/0.5)\ \times K(Z)$ for $P \leq 0.5$

$F\Delta_H^N \leq F\Delta_H^{RTP}\ (1 + PF\Delta_H\ (1-P))$

Where P is the fraction of full power at which the core is operating, K(Z) is the fraction specified in the

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each one percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

A sufficient shutdown margin insures that: 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at end of life (EOL), at 102% power, and is associated with a postulated steam line break accident resulting in uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin of 1.3% $\Delta k/k$ is required to control the reactivity transient. For cycle 11, after a burnup of 3000 MWD/MTU, a minimum shutdown margin of $\geq$2.9% $\Delta k/k$ is required to control the reactivity transient whenever hot shutdown is exceeded. In this case, some boron is assumed in the SI piping. Accordingly, the shutdown margin requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

The action to be taken when shutdown margin in out of limit is to borate using the best available source. In the determination of the required combination of boration flow rate and boron concentration, there is no unique Design Basis Event which must be satisfied. It is imperative to raise the boron concentration of the Reactor Coolant System as soon as possible. Therefore, the operator should begin boration with the best possible source available for the plant condition.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequence of a hypothetical rod ejection accident. The available control rod reactivity, or excess beyond needs, decreases with decreasing boron concentration because the negative reactivity required to reduce the core power level from full power to zero is largest when the boron concentration is low.

TABLE 4.1-2 (Sheet 1 of 2)

FREQUENCIES FOR SAMPLING TESTS

Sample	Analysis	Frequency	Maximum Time Between Analysis
1. Reactor Coolant	Gross Activity(1) Tritium Activity Boron Concentration Radiochemical (gamma)(2) Spectral Check Oxygen and Chlorides Concentration Fluorides Concentration Ē Determination (3) Isotopic Analysis for I-131, I-133, I-135	5 days/week(1)(4) Weekly(1) 2 days/week Monthly 3 times per 7 days Weekly Semi-Annually Once per 14 days(5)	3 day(4) 10 days 5 days 45 days 3 days 10 days 30 Weeks 20 days
2. Boric Acid Tank	Boron Concentration, Chlorides	Weekly	10 days
3. Spray Additive Tank	NaOH Concentration	Monthly	45 days
4. Accumulators	Boron Concentration	Monthly	45 days
5. Refueling Water Storage Tank	Boron Concentration pH, Chlorides Gross Activity	Monthly Quarterly	45 days 16 weeks
6. Secondary Coolant	I-131 Equivalent (Isotopic Analysis) Gross Activity	Monthly 3 times per 7 days	45 days 3 days
7. Component Cooling Water	Gross Activity, Corrosion Inhibitor and pH	Monthly	45 days
8. Spent Fuel Pool (when fuel stored)	Gross Activity Boron Concentration, Chlorides	Monthly	45 days
9.a. BIT header(6) b. SI suction(7)	Boron Concentration Boron Concentration	31 days Prior to exceeding hot shutdown	+25 percent N/A

TABLE 4.1-2 (Sheet 2 of 2)

FREQUENCIES FOR SAMPLING TESTS

FOOTNOTES:

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.

(2) The radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 10 minutes making up at least 95% of the total activity of the primary coolant.

(3) E determination will be started when the gross activity analysis indicates $\geq$ 10 μCi/cc and will be redetermined if the primary coolant gross radioactivity changes by more than 10 μCi/cc in accordance with Specification 3.1.D.

(4) Whenever the Gross Failed Fuel Monitor is inoperable, the sampling frequency shall be increased to twice per day, five days per week. The maximum time between analyses shall be sixteen hours for the two samples taken on a given day and three days between daily analysis. This accelerated sampling frequency need only be performed until the Gross Failed Fuel Monitor is declared operable.

(5) Once per 4 hours whenever the DOSE EQUIVALENT I-131 exceeds 1.0 μCi/cc or one sample after two hours but before six hours following a thermal power change exceeding 15 percent of the rated thermal power within a one-hour period.

(6) The boron sample shall be taken from a point downstream of the BIT tank on the BIT header prior to functional testing of the safety injection pumps. The first sample is due prior to exceeding hot shutdown.

Basis

The containment is designed for a pressure of 47 psig.[1] While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and an average maximum temperature of approximately 130°F. The Design Basis Accidents (DBA) that represent a challenge to the containment structure are a Loss of Coolant Accident (LOCA) and a Main Steam Line Break (MSLB). The limiting calculated peak containment pressure is a result of the MSLB [7], which is less than the stated design pressure of 47 psig. In addition, DBA analyses demonstrate that the calculated peak containment temperature will remain less than the Equipment Qualification (EQ) envelope temperature of 290 degrees F.

The containment structure is designed to contain, within established leakage limits, radioactive material that may be released from the reactor core following a DBA. The containment was designed with an allowable leakage rate of 0.1 weight percent of containment air per day. This leakage rate, used to evaluate offsite doses resulting from DBAs is defined in 10CFR 50 Appendix J as L_a; the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing performed in accordance with the program required by Technical Specification 6.14. The minimum test pressure of 42.72 psig used for this program is based upon the test pressure of $1.1P_a$ psig specified for the isolation valve seal water system (IVSWS) in Technical Specification 6.14. FSAR Section 5.2 says the IVSWS is tested at 47 psig which is 1.1 times 42.72 psig. The minimum test pressure is slightly higher than peak pressures previously calculated and is intended as a nominal value that will bound the peak calculated pressure (identified in the FSAR).

Prior to initial operation, the containment was strength-tested at 54 psig and was leak-tested. The acceptance criterion for this pre-operational leakage rate test was established as 0.075 W/o (.75 L_a) per 24 hours at 40.6 psig and 263°F, which were the peak accident pressure and temperature conditions at that time. This leakage rate is consistent with the construction of the containment, [2] which is equipped with a Weld Channel and Penetration Pressurization System for continuously pressurizing the containment penetrations and the channels over certain containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10 W/o per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10CFR100 values in the event of the design basis accident.[3]

Maintaining the containment operable requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in surveillance requirement 4.4.D does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage

The maximum permissible inleakage rate from the containment isolation valves sealed with service water for the full 12-month period of post accident recirculation without flooding the internal recirculation pumps is 0.36 gpm per fan cooler.

REFERENCES

(1) FSAR - Section 5
(2) FSAR - Section 5.1.7
(3) FSAR - 14.3.5
(4) Deleted.
(5) FSAR - Section 6.6
(6) FSAR - Section 6.5
(7) " Indian Point Unit 3, Cycle 11 Reload Safety Evaluation," September 1999.
(8) SECL-96-103, Indian Point Unit 3 Safety Evaluation of 24-Month Fuel Cycle Phase I Instrument Channel Uncertainties, June 1996
(9) Indian Point 3 Safety Evaluation Report, Supplement 2, December 3, 1975.
(10) NRC Safety Evaluation Related to Amendment 129 to Operating License DPR-64.

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, P_a, for the limiting design basis accident is less than 42.72 psig, the minimum test pressure.

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

ATTACHMENT II TO IPN-00-008

SAFETY EVALUATION OF

PROPOSED TECHNICAL SPECIFICATION CHANGE

REGARDING REVISED MAIN STEAM LINE BREAK ACCIDENT

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

I. Description of Changes

This section provides a description of the proposed changes to the Indian Point 3 Technical Specifications (TS). Minor changes in format, such as type font, margins or hyphenation, are not described in this submittal. The proposed TS changes are as follows:

- TS pages 3.3-5 and 3.3-6 - Renumber existing TS 3.3.A.6, 3.3.A.7, 3.3.A.8, 3.3.A.9 and 3.3.A.10 to TS 3.3.A.7, 3.3.A.8, 3.3.A.9, 3.3.A.10 and 3.3.A.11, respectively on TS pages.

- TS page 3.3-5 - Insert new TS 3.3.A.6 which says "The reactor shall not exceed hot shutdown conditions unless a boron concentration of $\geq$2300 ppm is maintained in safety injection piping from the RWST to the safety injection pumps, and in the Boron Injection Tank (BIT) header from the safety injection pumps up to and including the BIT. If the boron concentration is not within limits, restore the boron concentration to required limits within 8 hours. If the boron concentration is not restored to within limits within 8 hours, be in hot shutdown within 6 hours."

- TS page 3.10.1.1 - Insert a second sentence which says "For cycle 11, the shutdown margin shall be $\geq$2.9% $\Delta k/k$ whenever hot shutdown is exceeded."

- TS Table 4.1-2 sheet 1 - Insert two new surveillance requirements under the headings of "Sample," "Analysis," "Frequency," and "Maximum Time Between Analyses," respectively. The first says "9.a BIT [(6)]," "Boron Concentration," "31 days," and "+25 percent." The second says "b. SI suction," "Boron Concentration," "Prior to exceeding hot shutdown," and "N/A."

- TS Table 4.1-2 sheet 2 - Insert a note 6 that says "(6) The boron sample shall be taken from a point downstream of the BIT tank on the BIT header prior to functional testing of the safety injection pumps required by TS 4.5. The first sample is due prior to exceeding hot shutdown." Insert a note 7 that says "(7) A boron sample shall be taken representative of the RWST outlet piping at the connection to the SI pump suction line. The sample is required prior to exceeding hot shutdown and after RHR cooling is exited."

- TS page 6-22 - Change the second paragraph to say "The peak calculated containment internal pressure, P_a, for the limiting design basis accident is less than 42.72 psig, the minimum test pressure."

The associated changes to the bases are as follows:

- Basis page 3.1- 9 - Revise references to TS 3.3.A.8, 3.3.A.9 and 3.3.A.10 to reflect the revised TS numbering.

- Basis page 3.3-15 - Add a paragraph saying "The limiting MSLB accident analysis for containment peak pressure is the 102% power case. This case assumes a boron concentration in the safety injection piping from the RWST to the safety injection pumps and, in the BIT header, from the safety injection pump up to and including the

BIT. Surveillance testing is required by TS Table 4.2-1. The containment peak pressure analysis at hot zero power continues to assume no boron concentration in this piping."

- Bases page 3.3-18 - Delete "of 42.42 psig [(15)]" in the second paragraph.

- Bases page 3.3-21 - In reference 15 delete "WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95° at IP3" and add "Deleted."

- Bases page 3.10-15 - In the third paragraph change "with T_{avg} at no load operating temperature" to "at 102% power" and add, before the last sentence, "For cycle 11, after a burnup of 3000 MWD/MTU, a minimum shutdown margin of $\geq$2.9% $\Delta k/k$ is required to control the reactivity transient whenever hot shutdown is exceeded. In this case, some boron is assumed in the SI piping."

- Bases page 4.4-7 - In the first paragraph delete "of 42.40 psig." In the second paragraph delete the last two sentences and add, at the end of the paragraph, the following "The minimum test pressure of 42.72 psig used for this program is based upon the test pressure of $1.1P_a$ psig specified for the isolation valve seal water system (IVSWS) in Technical Specification 6.14. FSAR Section 5.2 says the IVSWS is tested at 47 psig which is 1.1 times 42.72 psig. The minimum test pressure is slightly higher than peak pressures previously calculated and is intended as a nominal value that will bound the peak calculated pressure (identified in the FSAR). "

- Bases page 4.4-10 - In reference 4 delete "WCAP - 12269 Rev. 1, "Containment Margin Improvement Analysis for IP-3 Unit 3." and add "Deleted." In reference (7) delete "Nuclear Safety Evaluation 98-3-013 MULT, "Integrated Safety Evaluation of 24 Month Cycle Instrument Channel Uncertainties, June 1996." and add "Indian Point Unit 3, Cycle 11 Reload Safety Evaluation," September 1999."

II. Purpose of Proposed Change

The purpose of the proposed changes to TS is to revise non-conservative TS. NRC Administrative Letter 98-10 (Reference 1) states that "the discovery of an improper or inadequate TS value or required action is considered a degraded or non-conforming condition as defined in GL 91-18." IP3 LER-99-008 (Reference 2) reported the plant outside design basis due to a non-conservative value for non-isolatable feedwater following a postulated failure of the feedwater control valve on a faulted steam generator. The LER reported that a cycle 10 main steam line break (MSLB) analysis demonstrated continued plant operability, but changed assumptions that were not conservative with regard to TS. The LER reported that a final MSLB would be performed for cycle 11.

Westinghouse performed the MSLB analysis for cycle 11 (Reference 3). The worst case was the case of a failed feedwater isolation valve on the faulted SG at 102% power. The analysis credited a 2.9% shutdown margin (after 3000 MWD/MTU) and boron in the safety injection (SI) system piping (2300 parts per million boron (ppm) boron in piping upstream of the SI pumps, in the Boron Injection Tank (BIT), and upstream of the BIT) to prevent a substantial return to power. The peak calculated pressure (41.41 psig) occurs at 102%

power and the peak temperature remained below the value used for equipment qualification. The analysis at hot zero power retained the original FSAR assumptions of 0 ppm in the SI piping and 1.3 percent shutdown margin. The TS are non-conservative since there are no requirements for sampling boron in the SI piping and the TS currently indicate a 1.3 % shutdown margin. Administrative controls are in place to sample boron concentration in the BIT and to take the required actions if the boron concentration is not within required limits. The cycle 11 MSLB analysis had the consequential effect of changing the peak calculated pressure inside containment and this has to be corrected in the TS.

III. Safety Implications of the Proposed Changes

The proposed changes to the TS revise the TS to reflect the Cycle 11 MSLB analysis that is currently the basis for plant operations. These assumptions are controlled by TS to ensure that the plant remains within design basis. The following discusses the acceptability of the assumptions used, the acceptability of the surveillance requirements, the basis for the action statements, and the changes to reflect the changed peak calculated pressure.

<u>SI piping contains ≥2300 ppm of boron</u>

The MSLB analysis assumed that when the reactor was critical there would be 2300 ppm of boron in the SI piping from the refueling water storage tank (RWST) to the SI pumps and, on the BIT header, in the piping from the SI pumps to the BIT and in the BIT. In the hot zero power condition (equivalent to the IP3 hot shutdown condition, i.e., "When the reactor is subcritical, by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{avg} is > 200°F but ≤ 555°F"), the boron in this piping was assumed to be 0 ppm. The revised assumption while above hot shutdown is reasonable since the initial boron concentration in the SI lines is established when the SI system is filled and vented with a subsequent full flow test (References 4 and 5) and during quarterly testing of the SI pumps (Reference 6). These actions are performed using RWST borated water with a boron concentration between 2400 and 2600 ppm. An LCO was added in proposed TS 3.3.A.6 to require the assumed boron concentration of ≥2300 ppm to be met prior to exceeding hot shutdown.

Prior to exceeding hot shutdown, a surveillance (sampling) is necessary upstream of the SI pumps to assure that the LCO of TS 3.3.A.6 is met. The potential exists for dilution of borated water in the suction line from the RWST when there is RHR cooling. The dilution potential exists because the discharge pressure of the RHR pumps is higher than the pressure in the RWST so valve leakage could result in some dilution of piping between the RWST and the RHR pumps. A sample of the RWST outlet piping taken upstream of the purification pump is adequate to demonstrate boron concentration of the SI suction since the purification loop connects to the RWST outlet pipe closer to RHR than the SI piping. The purification loop will be purged prior to the sample to assure the concentration is representative of the RWST outlet piping and is adequate. Typically purging is 3 volumes. This is initiated after termination of RHR cooling and prior to exiting hot shutdown.

Prior to exceeding hot shutdown, a surveillance (sampling) is necessary downstream of the BIT to assure that the LCO of TS 3.3.A.6 is met. No potential dilution of the SI header downstream of the BIT has been identified. Leakage is possible through the two check valves in each SI line to a cold leg and through two check valves and a closed motor operated valve to a hot leg. The primary system typically has not been pressurized for long enough for leakage to occur since entry into the hot shutdown condition is usually short term and surveillances during power operation assure adequate boron. Leakage to the SI header from the accumulators (several paths exist) is not considered a dilution mechanism because the accumulators are required to have 2000 ppm of boron and, historically, have been above 2300 ppm boron. Dilution is not postulated from other lines connected to SI pump suction from the RWST because they are to the suction of the RHR, charging, purification and recirculating pumps. Dilution is not postulated from lines connected to the BIT tank (i.e., boric acid transfer pump (BATP) fill of BIT, and discharge of BIT to the boric acid storage tanks (BAST) and the chemical volume and control system (CVCS) holdup tanks) because the BIT has been retired in place (high concentrations of boron are no longer maintained in the tank) and the lines are not in use. Although the above does not indicate dilution is of concern, a surveillance is performed prior to going critical to provide evidence that leakage has not occurred. Operation of the SI pumps prior to such testing would not be of concern because the testing is confirmatory prior to startup and sampling during operation provides assurance that dilution is not occurring.

During operation, a surveillance (sampling) is necessary to assure that the LCO of TS 3.3.A. is met because of the potential for back leakage from the primary system, discussed above. The RWST surveillance requirement (Table 4.1-2, item 5) assures an adequate boron concentration upstream of the SI pumps. A proposed surveillance of the boron concentration downstream of the BIT assures adequate boron upstream of the BIT and in the BIT. The surveillance will be scheduled every 31 days (the 25% allowance for surveillance testing is applied to this time), the same as the current surveillance interval in TS Table 4.2-1 for the RWST (31 to 45 days). The sample, taken downstream of the BIT, is taken prior to the surveillance testing for the SI pumps. This assures an accurate sample because two of the SI pumps recirculate RWST water through the BIT header which would disguise dilution of the water in the BIT header. The proposed TS change to Table 4.2-1 by the addition of item 9 and note 6 will require testing of the boron concentration in the SI lines consistent with this discussion of safety implications.

An allowed outage time (AOT) of 8 hours and a shutdown requirement of 6 hours to hot shutdown is proposed in TS 3.3.A.11. The AOT and shutdown time are based on Standard Technical Specification (STS) 3.5.4 (Reference 7) for the RWST. The bases for STS 3.5.4 note that the emergency core cooling system cannot perform its design function when the boron concentration is not within limits and that the 8 hour AOT was developed considering the time required to return the boron concentration to within limits and the continued availability of the RWST for injection. STS 3.5.4 requires the plant to be in hot standby (the equivalent of IP3 hot shutdown) within 6 hours and cold shutdown within 36 hours, reasonable completion times based on operating experience. The proposed shutdown requirement is based upon an analogous situation when the boron in the SI piping is insufficient. The proposed AOT is the same and the subsequent action will bring the plant to the hot shutdown condition within 6 hours. Cold shutdown is not required because the LCO no longer applies once hot shutdown is entered.

Shutdown margin exceeds 2.9%

The shutdown margin for the control rods is a value calculated based on core design. The shutdown margin for cycle 11 exceeds the assumed 2.9%. This value is added to the TS in the proposed revision to TS 3.10.1.1. There is no associated surveillance requirement or action statement required because this is a reload specific value. Cycle 11 is specifically referenced because this value will have to be reevaluated for cycle 12.

Change to peak calculated pressure

Prior to the revised MSLB evaluation, the MSLB analysis for hot zero power resulted in the peak calculated pressure (i.e., 42.40 psig). The revised MSLB analysis deleted some conservatism in the analysis. As reported in LER 99-008, this had been expected to account for the increased feedwater. It did not account for the increased feedwater but did decrease the MSLB peak calculated pressure so that the peak calculated pressure is 41.41 psig. TS 6.14 currently identifies the peak calculated pressure as 42.42 psig which is higher than both the MSLB calculated pressure and the LOCA calculated pressure. The proposed TS change will identify a minimum peak pressure to be used for containment leakage testing and require the peak calculated pressure to remain less than this value. The minimum test pressure is 42.72 psig. This value was derived by taking the isolation valve seal water system (IVSWS) test specified in the FSAR (i.e., 47 psig) and deriving the associated peak accident pressure from the TS 6.14 required IVSWS test pressure of $1.1P_a$. This proposed TS change provides a small margin over calculated peak pressure and provides a savings by limiting the number of changes required to the TS for changes in peak calculated pressure. The current peak calculated pressure is found in Chapter 14.3 of the FSAR. The actual tests done for the containment isolation valves meet the FSAR requirement of 43 psig and are therefore conservative with respect to proposed change. The last Type A test (Reference 8) was done at more than 44 psig and is therefore conservative with respect to the proposed change. The proposed change will not affect current design or testing.

IV. Evaluation of Significant Hazards Consideration

Operation of the Indian Point 3 plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92 since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes include revised assumptions in the TS to correct non-conservative TS and revised TS with respect to the peak calculated containment pressure for a postulated MSLB. The changes take credit for existing boron in the SI system and existing shutdown margin, perform surveillance to verify the boron concentration, and revise the containment testing program to reflect a minimum test pressure that must bound the peak calculated pressure. These changes cannot increase the probability of an accident previously evaluated since they do not change plant operations and are not related to accident initiators. These changes will not increase the consequences of an accident previously evaluated since they do not change system operation to mitigate any accident and the use of a minimum containment test pressure ensures the TS required testing bounds the calculated peak calculated pressure.

2. create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes include revised assumptions in the TS to correct non-conservative TS and revised TS with respect to the peak calculated pressure. The changes take credit for existing boron in the SI system and existing shutdown margin, perform surveillance to verify the boron, and revise the containment testing program to reflect a minimum test pressure that must bound the peak calculated pressure. These changes do not physically alter the plant since they take credit for existing plant conditions and the physical act of sampling meets system design and Technical Specification requirements. Therefore, these changes do not create the possibility of a new or different type of accident from those previously evaluated.

3. involve a significant reduction in a margin of safety.

   The proposed changes include revised assumptions in the TS to correct non-conservative TS and revised TS with respect to the peak calculated pressure. The changes take credit for existing boron in the SI system and existing shutdown margin, perform surveillance to verify the boron, and revise the containment testing program to reflect a minimum test pressure that must bound the peak calculated pressure. These changes do not involve a significant reduction in the margin of safety since the credited boron is part of the existing system design that has not been credited since the BIT tank retirement. The credited shutdown margin is typical of the excess shutdown margin resulting from cycle specific core design.

V. Implementation of The Proposed Change

The proposed changes do not affect the operation of the SI system, no modifications will be required, and testing is within the design and licensing limits of the SI system. There would be no significant effect on the ALARA Program. Although sampling of boron is in a radiologically controlled area, it is not in a high radiation area or contaminated area. There would be no adverse effect on the Security and Fire Protection Programs since there are no physical changes that could affect security and Appendix R requirements. There would be no adverse effect on the Emergency Plan since equipment allowed outage times are not a consideration. The FSAR was revised as part of the core reload safety evaluation (Reference 9) to reflect the cycle 11 MSLB and the revised assumptions. The Improved Technical Specification submittal will have to be revised to add a new specification for boron concentration but not for the shutdown margin (this was relocated to the COLR). There will be no effect on overall plant operations and the environment since system operation will remain the same and there will be no changes plant effluents or radwaste. This amendment meets the eligibility criteria for categorical exclusion relative to requiring a specific environmental assessment as set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. The amendment involves no significant hazards consideration. This was discussed previously.

2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. There are no design changes or changes to system operations and the only new sample is a boron sample.

3. There is no significant increase in individual or cumulative occupational radiation exposure. There are no design changes or changes to system operations and the only new sample is a boron sample which is not taken in an area of radiological concern.

VI. Conclusion

The Plant Operating Review Committee (PORC) and Safety Review Committee (SRC) have reviewed the proposed TS change and have concluded that it does not involve a significant hazards consideration, will not endanger the public health and safety and meet the criteria for exclusion from a specific environmental assessment.

V. References

1. NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient To Assure Plant Safety," dated 12/29/98.

2. IPN-99-092, "Licensee Event Report # 1999-008-00, Plant outside Design Basis Due to an Error in an Assumption In the Main Steam Line Break Analysis," dated August 23, 1999.

3. Westinghouse Letter INT-99-254, "SLB Inside Containment Sensitivities for FCV Failure," October 8, 1999.

4. System Operating Procedure SOP-SI-01, Safety Injection System Operation, Revision 25.

5. Surveillance Procedure 3PT-R64, "High Head Safety Injection Check Valves", Revision 12.

6. Surveillance Procedure 3PT-Q116A, B, and C, "Safety Injection Pump Functional Test," Revision 0.

7. NUREG 1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1, dated April 1995.

8. IP3-91-016, "Reactor Containment Building Integrated Leak Rate Test," dated March 8, 1991.

9. NSE-98-3-138 RCS, "Core Reload For Cycle 11," Revision 3.

ATTACHMENT III TO IPN-00-008

MARKUP OF EXISTING TECHNICAL SPECIFICATION TO SHOW PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING REVISED MAIN STEAM LINE BREAK ACCIDENT

(FOR INFORMATION ONLY)

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

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 3.1.A-2) 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 and/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 319°F 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 2.00 square inches.

The OPS enable temperature of 319°F permits the performance of an RCS hydrostatic test (see Fig. 4.3-1) without activating the OPS.

Upon OPS inoperability, the RCS may be heated above 411°F. This temperature is that value for which the RCS heatup and cooldown curves (Figures 3.1-1 and 3.1-2) permit pressurization to the setting of the pressurizer safety valves. Accordingly, with an inoperable OPS and an RCS temperature above 411°F, 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

5. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.3 within the time periods specified in 3.3.A.4; then:

   a. If the reactor is critical, it shall be in the hot shutdown condition within four hours and the cold shutdown condition within the following 24 hours.

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.A.3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition using normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

Insert 1

~~6~~ 7. When the reactor coolant system T_{avg} is greater than 200°F and less than 350°F, the following decay heat removal requirements shall be met:

   a. Two residual heat removal pumps together with their associated heat exchangers, piping, and valves shall be operable,

   OR

   b. A minimum of one residual heat removal pump and heat exchanger and a minimum of one reactor coolant pump and steam generator together with their associated piping and valves, shall be operable,

   OR

   c. A minimum of two reactor coolant pumps and two steam generators, together with their associated piping and valves, shall be operable,

   OR

   d. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system. Otherwise, if sufficient equipment is available, be in cold shutdown within 20 hours.

8 ~~7~~. When the reactor coolant T_{avg} is less than 200°F, but not in the refueling operation condition, two residual heat removal pumps, together with their associated heat exchangers, piping and valves, shall be operable.

   a. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system.

   b. The above requirements may be suspended during maintenance, modifications, testing, inspection or repair provided that:

      1) an alternate means of decay heat removal is available and return of the system within sufficient time to prevent exceeding cold shutdown requirements is assured;

Insert 1

6. The reactor shall not exceed hot shutdown conditions unless a boron concentration of ≥2300 ppm is maintained in safety injection piping from the RWST to the safety injection pumps, and in the Boron Injection Tank (BIT) header from the safety injection pumps up to and including the BIT. If the boron concentration is not within limits, restore the boron concentration to required limits within 8 hours. If the boron concentration is not restored to within limits within 8 hours, be in hot shutdown within 6 hours.

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.

9 8. When the RCS average cold leg temperature (T_{cold}) is below 319°F, 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.

10 9. The requirements of 3.3.A.8 9 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.

11 10. The requirements of 3.3.A.8 9 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% and the plant is vented in accordance with Technical Specification 3.1.A.8 9.c.1. (Alternate methods and instrumentation may be used to confirm actual RCS elevation.)

B. <u>Containment Cooling and Iodine Removal Systems</u>

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:

cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would damage the fuel elements or result in a release in excess of 10 CFR 100 and 10 CFR 50 dose limits.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and, therefore, in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are based on two considerations:

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power, the magnitude of the decay heat decreases after initiating hot shutdown. Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 1 hour of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case the reactor is to be put into the cold shutdown condition.

The limits for the Refueling Water Storage Tank and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.[9][11]

Insert 2

Insert 2

The limiting MSLB accident analysis for containment peak pressure is the 102% power case. This case assumes a boron concentration in the safety injection piping from the RWST to the safety injection pumps and, in the BIT header, from the safety injection pump up to and including the BIT. Surveillance testing is required by TS Table 4.2-1. The containment peak pressure analysis at hot zero power continues to assume no boron concentration in this piping.

Due to the distribution of the five fan cooler units and two containment spray pumps on the 480 volt buses, the closeness to which the combined equipment approaches minimum safeguards varies with which particular component is out of service. Accordingly, the allowable out of service periods vary according to which component is out of service. Under no conditions do the combined equipment degrade below minimum safeguards.

The seven day out of service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is consistent with W Standardized Technical Specifications. This is allowable because no credit has been taken for operation of these systems in the calculation of off-site accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems.[11] The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems for the maximum containment calculated peak accident pressure of 42.42 psig.[15] A WC & PPS zone is considered that portion of piping downstream of the air receiver discharge check valve up to the last component pressurized by that system portion.

Some portions of the Weld Channel Pressurization System (WCPS) piping would not be practicably accessible for repair if they became inoperable. A section of WCPS piping is considered to be inoperable if it brings the air consumption of the WC & PPS above the required 0.2% of the containment volume per day or if the section can not maintain a pressure above the required 43 psig. If it is determined, by written evaluation, that an inoperable section of piping is not practicably accessable for repair, then that portion of the WCPS may be disconnected from the system. Inoperable sections of WCPS piping which can be considered for disconnection will satisfy one of the following criteria: 1) the piping is covered by concrete and repairs of the piping would involve the removal of some portion of the containment structure; or 2) the piping is located behind plant equipment in the containment building and repairs of the piping would involve the relocation of the equipment. The integrity of the welds associated with any disconnected portions of the WCPS is verified by integrated leak rate testing. The provision that allows for the disconnection of portions of the WCPS piping does not apply to any other WC & PPS piping.

The Component Cooling System is not required during the injection phase of a loss-of-coolant accident. The component cooling pumps are located in the Primary Auxiliary Building and are accessible for repair after a loss-of-coolant accident.[6] During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards.[7]

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%) and the plant is vented in accordance with Technical Specification 3.1.A.8.c.1 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" Deleted.
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.

Amendment No. 67, 94, 98, 108, 145, 154, ~~179~~,

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distribution and to limits on control rod operations.

Objectives:

To ensure:

1. Core subcriticality after reactor trip.

2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and

3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

Insert 1

3.10.1 Shutdown Reactivity

3.10.1.1 Whenever $T_{avg} > 200°F$ the shutdown margin shall be $\geq 1.3\% \ \Delta k/k$.

3.10.1.2 When the conditions of specification 3.10.1.1 are not met, initiate boration to restore shutdown margin within limit.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$$F_Q(Z) \leq (F_Q^{RTP}/p) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq (F_Q^{RTP}/0.5) \times K(Z) \text{ for } P \leq 0.5$$

$$F_{\Delta H}^N \leq F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$$

Where P is the fraction of full power at which the core is operating, K(Z) is the fraction specified in the

Insert 3

For cycle 11, the shutdown margin shall be $\geq$2.9% $\Delta k/k$ whenever hot shutdown is exceeded.

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each one percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

A sufficient shutdown margin insures that: 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

102% power

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at end of life (EOL), with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident resulting in uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin of 1.3 % Δk/k is required to control the reactivity transient. Accordingly, the shutdown margin requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

Insert 4

The action to be taken when shutdown margin in out of limit is to borate using the best available source. In the determination of the required combination of boration flow rate and boron concentration, there is no unique Design Basis Event which must be satisfied. It is imperative to raise the boron concentration of the Reactor Coolant System as soon as possible. Therefore, the operator should begin boration with the best possible source available for the plant condition.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequence of a hypothetical rod ejection accident. The available control rod reactivity, or excess beyond needs, decreases with decreasing boron concentration because the negative reactivity required to reduce the core power level from full power to zero is largest when the boron concentration is low.

Insert 4

For cycle 11, after a burnup of 3000 MWD/MTU, a minimum shutdown margin of $\geq$2.9% Δk/k is required to control the reactivity transient whenever hot shutdown is exceeded. In this case, some boron is assumed in the SI piping.

TABLE 4.1-2 (Sheet 1 of 2)

FREQUENCIES FOR SAMPLING TESTS			
Sample	Analysis	Frequency	Maximum Time Between Analysis
1. Reactor Coolant	Gross Activity[1] Tritium Activity Boron concentration Radiochemical (gamma)[2] Spectral Check Oxygen and Chlorides Concentration Fluorides Concentration $\bar{E}$ Determination [3] Isotopic Analysis for I-131, I-133, I-135	5 days/week[1][4] Weekly[1] 2 days/week Monthly 3 times per 7 days Weekly Semi-Annually Once per 14 days[5]	3 days[4] 10 days 5 days 45 days 3 days 10 days 30 weeks 20 days
2. Boric Acid Tank	Boron Concentration, Chlorides	Weekly	10 days
3. Spray Additive Tank	NaOH Concentration	Monthly	45 days
4. Accumulators	Boron Concentration	Monthly	45 days
5. Refueling Water Storage Tank	Boron Concentration pH, Chlorides Gross Activity	Monthly Quarterly	45 days 16 weeks
6. Secondary Coolant	I-131 Equivalent (Isotopic Analysis) Gross Activity	Monthly 3 times per 7 days	45 days 3 days
7. Component Cooling Water	Gross Activity, Corrosion Inhibitor and pH	Monthly	45 days
8. Spent Fuel Pool (when fuel stored)	Gross Activity Boron Concentration, Chlorides	Monthly	45 days

1

Insert 5

9.a. BIT header[(6)] b. SI suction[(7)]	Boron Concentration Boron Concentration	31 Days Prior to exceeding hot shutdown	+25 percent N/A

TABLE 4.1-2 *(Sheet 2 of 2)*

FREQUENCIES FOR SAMPLING TESTS

FOOTNOTES:

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 10 minutes making up at least 95% of the total activity of the primary coolant.

(3) $\bar{E}$ determination will be started when the gross activity analysis indicates $\geq$ 10 μCi/cc and will be redetermined if the primary coolant gross radioactivity changes by more than 10 μCi/cc in accordance with Specification 3.1.D.

(4) Whenever the Gross Failed Fuel Monitor is inoperable, the sampling frequency shall be increased to twice per day, five days per week. The maximum time between analyses shall be sixteen hours for the two samples taken on a given day and three days between daily analysis. This accelerated sampling frequency need only be performed until the Gross Failed Fuel Monitor is declared operable.

(5) Once per 4 hours whenever the DOSE EQUIVALENT I-131 exceeds 1.0 μCi/cc or one sample after two hours but before six hours following a thermal power change exceeding 15 percent of the rated thermal power within a one-hour period.

Insert 6

Insert 6

(6) The boron sample shall be taken from a point downstream of the BIT tank on the BIT header prior to functional testing of the safety injection pumps required by TS 4.5. The first sample is due prior to exceeding hot shutdown.

(7) A boron sample shall be taken representative of the RWST outlet piping at the connection to the SI pump suction line. The sample is required prior to exceeding hot shutdown and after RHR cooling is exited.

Basis

The containment is designed for a pressure of 47 psig. [1] While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and an average maximum temperature of approximately 130°F. The Design Basis Accidents (DBA) that represent a challenge to the containment structure are a Loss of Coolant Accident (LOCA) and a Main Steam Line Break (MSLB). The limiting calculated peak containment pressure of 42.40 psig is a result of the MSLB [7], which is less than the stated design pressure of 47 psig. In addition, DBA analyses demonstrate that the calculated peak containment temperature will remain less than the Equipment Qualification (EQ) envelope temperature of 290 degrees F.

The containment structure is designed to contain, within established leakage limits, radioactive material that may be released from the reactor core following a DBA. The containment was designed with an allowable leakage rate of 0.1 weight percent of containment air per day. This leakage rate, used to evaluate offsite doses resulting from DBAs is defined in 10CFR 50 Appendix J as L_a; the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing performed in accordance with the program required by Technical Specification 6.14. The minimum test pressure of 42.42 psig used for this program is based on analyses performed to support an increase of the ultimate heat sink temperature, [4] as incorporated by Technical Specification Amendment 98. The minimum test pressure, 42.42 psig, bounds the current limiting DBA pressure, 42.40 psig.

Insert 7

Prior to initial operation, the containment was strength-tested at 54 psig and was leak-tested. The acceptance criterion for this pre-operational leakage rate test was established as 0.075 W/o (.75 L_a) per 24 hours at 40.6 psig and 263°F, which were the peak accident pressure and temperature conditions at that time. This leakage rate is consistent with the construction of the containment, [2] which is equipped with a Weld Channel and Penetration Pressurization System for continuously pressurizing the containment penetrations and the channels over certain containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10 W/o per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10CFR100 values in the event of the design basis accident. [3]

Maintaining the containment operable requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in surveillance requirement 4.4.D does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage

Insert 7

The minimum test pressure of 42.72 psig used for this program is based upon the test pressure of $1.1P_a$ psig specified for the isolation valve seal water system (IVSWS) in Technical Specification 6.14. FSAR Section 5.2 says the IVSWS is tested at 47 psig which is 1.1 times 42.72 psig. The minimum test pressure is slightly higher than peak pressures previously calculated and is intended as a nominal value that will bound the peak calculated pressure (identified in the FSAR).

The maximum permissible inleakage rate from the containment isolation valves sealed with service water for the full 12-month period of post accident recirculation without flooding the internal recirculation pumps is 0.36 gpm per fan cooler.

REFERENCES

Deleted.

(1) FSAR - Section 5
(2) FSAR - Section 5.1.7
(3) FSAR - 14.3.5
(4) WCAP - 12269 Rev. 1, "Containment Margin Improvement Analysis for IP-3 Unit 3"
(5) FSAR - Section 6.6
(6) FSAR - Section 6.5
(7) Nuclear Safety Evaluation 98-3-013-MULT, "Integrated Safety Evaluation of 24-Month Cycle Instrument Channel Uncertainties," Revision 0, dated March 3, 1998.
(8) SECL-96-103, Indian Point Unit 3 Safety Evaluation of 24-Month Fuel Cycle Phase I Instrument Channel Uncertainties, June 1996
(9) Indian Point 3 Safety Evaluation Report, Supplement 2, December 1975.
(10) NRC Safety Evaluation Related to Amendment 129 to Operating License DPR-64.

"Indian Point Unit 3, Cycle 11 Reload Safety Evaluation," September 1999.

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.

less than 42.72 psig,

The peak calculated containment internal pressure, P_a, for ~~the limiting~~ design basis accident is 42.40 psig. The minimum test pressure is 42.42 psig.

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

ATTACHMENT IV TO IPN-00-008

MARKUP OF IMPROVED TECHNICAL SPECIFICATION TO

SHOW EFFECT OF PROPOSED TECHNICAL SPECIFICATION CHANGES

REGARDING REVISED MAIN STEAM LINE BREAK ACCIDENT

(FOR INFORMATION ONLY)

Revise ITS as follows:

Remove Page	Insert Page
none	3.5.5-1

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

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 ECCS Boron Concentration

LCO 3.5.5 Maintain a boron concentration of $\geq$2300 PPM in safety injection piping from the RWST to the safety injection pumps, and in the BIT header up to and including the BIT.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A Boron concentration not within limits.	A.1 Restore boron concentration to required limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.5.1	Verify BIT boron concentration is $\geq$ 2300 ppm.	------------------NOTE------------------ 1. Prior to safety injection pump functional test. -- 31 days
SR 3.5.5.2	Verify safety injection suction line boron concentration is $\geq$ 2300 ppm.	Prior to entering Mode 2.