ATTACHMENT I TO IPN-93-028

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

BORON INJECTION TANK ELIMINATION

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

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

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

1. Two charging pumps shall be operable.

2. Two boric acid transfer pumps shall be operable.

3. The boric acid storage system shall contain a minimum of 6100 gallons of 11 1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) boric acid solution at a temperature of at least 145°F.

4. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid storage system and one flow path from the refueling water storage tank (RWST) to the Reactor Coolant System.

5. The appropriate boric acid storage tank level indicator(s) shall be operating.

6. Two channels of heat tracing shall be operable for the flow path from the boric acid storage system to the Reactor Coolant System.

7. City water piping and valves shall be operable to the extent required to provide emergency cooling water to the charging pumps and flush water for the concentrated boric acid piping from the outlet of the boric acid storage tanks to the charging pump suction.

C. The requirements of 3.2.B may be modified to allow any one of the following components to be inoperable at any one time:

1. One of the two operable charging pumps may be removed from service provided a second charging pump is restored to an operable status within 24 hours.

2. One boric acid transfer pump may be inoperable for a period not to exceed 48 hours.

3. The boric acid storage system may be inoperable for a period not to exceed 48 hours provided that the RWST is operable.

4. One channel of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided the failed channel is restored to an operable status within 7 days and the redundant channel is demonstrated to be operable daily during that period.

D. If the Chemical and Volume Control System is not restored to meet the requirements of 3.2.B within the time period specified in 3.2.C, then:

1. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

2. 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.

3. In either case, if the requirements of 3.2.B 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.

BASIS

The Chemical and Volume Control System[1] provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with either one of the two boric acid transfer pumps. An alternate method of boration will be to use the charging pumps taking suction directly from the refueling water storage tank. A third method will be to depressurize and use the safety injection pumps.

There are three sources of borated water available for injection through 3 different paths:

1. The boric acid transfer pumps can deliver the contents of the boric acid storage system to the charging pumps.

2. The charging pumps can take suction from the refueling water storage tank.

3. Injection of borated water from the refueling water storage tank with the safety injection pumps[2].

The quantity of boric acid in storage from either the boric acid storage system or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life.

A combined minimum deliverable volume of 6100 gallons with an averaged concentration of the 11 1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) of boric acid are required to meet cold shutdown conditions. An upper concentration limit of 13% (22,735 ppm of boron) boric acid in the boric acid storage system is specified to maintain solution solubility at the specified low temperature limit of 145°F. One channel of heat tracing is sufficient to maintain the specified low temperature limit. The second channel of heat tracing provides backup for continuous plant operation when one channel is inoperable. Should both channels of heat tracing become inoperable, the reactor will be shutdown and can easily be borated before the line temperature is reduced near the boric acid precipitive temperature.

The city water system is used as a source of water for emergency cooling of the charging pumps and as a source of flush water to remove concentrated boric acid from the piping between the outlet of the boric acid storage tanks and the inlet to the charging pumps in the unlikely event of a complete loss of electrical power and/or a complete loss of service water resulting from turbine missiles.

References

1) FSAR - Section 9.2
2) FSAR - Section 6.2
3) "Revised Feasibility Report For BIT Elimination For Indian Point Unit 3," July 1988 (Westinghouse report).

c. One residual heat removal pump and heat exchanger together with the associated piping and valves operable.

d. One recirculation pump together with its associated piping and valves operable.

2. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.1 within 1 hour the reactor shall be in the cold shutdown condition within the next 20 hours.

3. The reactor coolant system T_{avg} shall not exceed 350°F unless the following requirements are met:

a. The refueling water storage tank contains a minimum of 346,870 gallons of water at a boron concentration $\geq$2400 ppm and $\leq$2600 ppm.

b. DELETED

c. The four accumulators are pressurized between 600 and 700 psig and each contains a minimum of 775 ft^3 and a maximum of 815 ft^3 of water at a boron concentration $\geq$2000 ppm and $\leq$2600 ppm. Accumulator isolation valves 894A, B, C, and D shall be open and their power supplies deenergized whenever the reactor coolant system pressure is above 1000 psig.

a. The accumulators may be isolated during the performance of the reactor coolant system hydrostatic tests.

For the purpose of accumulator check valve leakage testing, one accumulator may be isolated at a time, for up to 8 hours, provided the reactor is in the hot shutdown condition.

b. One safety injection pump may be out of service, provided the pump is restored to an operable status within 24 hours and the remaining two pumps are demonstrated to be operable.

c. One residual heat pump may be out of service, provided the pump is restored to an operable status within 24 hours and the other residual heat removal pump is demonstrated to be operable.

d. One residual heat exchanger may be out of service provided that it is restored to an operable status within 48 hours.

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

f. DELETED

g. One refueling water storage tank low level alarm may be inoperable for up to 7 days provided the other low level alarm is operable.

cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

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]

TABLE 4.1-1 (Sheet 2 of 5)

Channel Description	Check	Calibrate	Test	Remarks
10. Steam Generator Level	S	18M	Q	
11. Residual Heat Removal Pump Flow	N.A.	18M	N.A.	
12. Boric Acid Tank Level	S	18M	N.A.	Bubbler tube rodded during calibration
13. Refueling Water Storage Tank Level	W	18M	N.A.	Low level alarms
14. Containment Pressure	S	18M	Q	High and High-High
15. Process and Area Radiation Monitoring Systems	D	18M	Q	
16. Containment Water Level Monitoring System:				
a. Containment Sump	N.A.	18M	N.A.	Narrow Range, Analog
b. Recirculation Sump	N.A.	18M	N.A.	Narrow Range, Analog
c. Containment Water Level	N.A.	18M	N.A.	Wide Range
17. Accumulator Level and Pressure	S***	18M	N.A.	
18. Steam Line Pressure	S	18M	Q	
19. Turbine First Stage Pressure	S	18M	Q	
20. Reactor Protection Relay Logic	N.A.	N.A.	TM	
21. Turbine Trip Low Auto Stop Oil Pressure	N.A.	18M	N.A.	
22. DELETED	DELETED	DELETED	DELETED	

Amendment No. 8, 38, 63, 68, 74, 93, 107, 123,

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)	5 days/week(1)(4)	3 days(4)
	Tritium Activity	Weekly(1)	10 days
	Boron concentration	2 days/week	5 days
	Radiochemical (gamma)(2) Spectral Check	Monthly	45 days
	Oxygen and Chlorides Concentration	3 times per 7 days	3 days
	Fluorides Concentration	Weekly	10 days
	$\overline{E}$ Determination (3)	Semi-Annually	30 weeks
	Isotopic Analysis for I-131, I-133, I-135	Once per 14 days(5)	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	Monthly	45 days
	Gross Activity	Quarterly	16 weeks
6. Secondary Coolant	I-131 Equivalent (Isotopic Analysis)	Monthly	45 days
	Gross Activity	3 times per 7 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

d. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

2. Acceptance Criteria

The measured leakage rate shall be less than 0.75 L_a where L_a is equal to 0.1 w/o per day of containment steam air atmosphere at 42.42 psig.

3. Frequency

A set of three leakage rate tests shall be performed (during plant shutdown), at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant in service inspection.

B. DELETED

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 limiting peak containment temperature, based on LOCA containment response, is 261.5°F. [7] The peak containment pressure, also based on LOCA containment response, is 42.29 psig. [7] The acceptance criteria of specification 4.4.A.2. was changed by amendment 98 to reflect analysis [4] done for the ultimate heat sink temperature increase. As stated, the current peak pressure, calculated for high head safety injection flow balancing, is 42.29 psig. The acceptance criteria of 42.42 psig is conservative with respect to the current calculated peak pressure of 42.29.

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 both the penetrations and the channels over all 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]

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, the containment isolation valves are to be closed in the normal manner and without preliminary exercising or adjustments.

These specifications have been developed using Appendix J (issue effective date March 16, 1973) of 10CFR50 (with the surveillance frequency exception noted previously) and ANSI N45.4-1972 "Leakage Rate Testing of Containment structures for Nuclear Reactors" (March 16, 1972) for guidance.

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) WCAP - 12269 Rev. 1, "Containment Margin Improvement Analysis for IP-3 Unit 3"
(5) FSAR - Section 6.6
(6) FSAR - Section 6.5
(7) SECL-92-131, Indian Point Unit 3 High Head Safety Injection Flow Changes Safety Evaluation, June 1992

ATTACHMENT II TO IPN-93-028

SAFETY EVALUATION

RELATED TO

BORON INJECTION TANK ELIMINATION

TECHNICAL SPECIFICATION CHANGES

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

Section I - Description of Changes

This application for amendment to the Indian Point Unit 3 (IP3) Technical Specifications proposes to revise sections 3.2, 3.3, and 4.1 of Appendix A of the Operating License. The proposed changes to these sections eliminate references to the Boron Injection Tank (BIT). Westinghouse has developed improved analytical techniques to allow removal of the BIT. These analytical techniques were applied to demonstrate the feasibility of elimination of the BIT at Indian Point 3. The IP3 specific analysis (included as Attachment III) concludes that the BIT may be bypassed, eliminated, or the boric acid concentration reduced.

The BIT is a component of the safety injection system whose sole function is to provide concentrated boric acid to the reactor coolant system to mitigate the consequences of postulated steam line break accidents. The only postulated accident analyses affected by BIT removal are the steamline break and the associated mass and energy release/containment pressure analyses. These transients are affected with respect to both core integrity and mass and energy release to containment. For hypothetical steamline break analysis the existing FSAR criteria were applied to the BIT elimination analysis. For the credible steamline break analysis (i.e. a failed secondary safety or relief valve, with offsite power available), a revised criterion (allowing re-criticality) is used that is in compliance with the NRC and ANS criteria. These analyses, as presented in Attachment III, demonstrate the acceptability of BIT removal for the Indian Point 3 Nuclear Power Plant, while continuing to meet applicable safety criteria.

This application also seeks to amend section 4.4 to remove a reference to containment temperature in the acceptance criteria for the containment leak test. The leak test specification and basis are amended to indicate the postulated peak containment temperature and pressure determined by current analysis.

Section II - Evaluation of Changes

Analyses have been performed for Indian Point 3 and are contained in the attached report, "Revised Feasibility Report for BIT Elimination for Indian Point Unit 3." In the IP3 steamline break analysis, the system transient parameters were calculated using the LOFTRAN (WCAP-7907) computer code. The changes in safety injection system volumes, initial concentrations, and temperatures corresponding to elimination of the BIT are introduced into the analyses in the LOFTRAN code.

For the transient behavior of the hypothetical breaks, the differences from the old FSAR cases and the new BIT removal cases show only small changes in reactor coolant system parameters, except for core power. Departure from Nucleate Boiling (DNB) analyses show that the DNB design basis, for all cases, is met and that no consequential fuel failures are expected.

For the transient behavior for the credible steamline break, the DNB design basis must still be met, in order to meet the 10 CFR 100 dose requirements. DNB analyses for this case show that the DNB design basis is met and no fuel failures are predicted.

The impact of the BIT elimination on the Mass and Energy/Containment Pressure analysis was addressed to assure the containment pressure and temperature remain below design limits. This analysis resulted in new postulated peak accident pressure and temperature below the containment design limits.

Subsequent to these analyses, changes to plant operations required amendments to the plant technical specifications. These changes included operation with an increased ultimate heat sink temperature (95°F) and the addition of a six (6) second time delay to the safety injection (SI) and steam line isolation (SLI) actuation signals generated by high steam flow coincident with either low T_{avg} or low steamline pressure condition (hereafter referred to as high steam flow coincidence logic). To support operation with 95°F ultimate heat sink temperature, analyses were performed and the resulting reports (References 4 and 5) were docketed with the amendment submittals. These analyses concluded that for the main steamline break, operation with a 95°F ultimate heat sink temperature and zero (0) boron concentration in the BIT would result in a peak containment postulated pressure of 42.42 psig. The analysis to support operation with an additional six (6) second time delay assumed in the high steam flow coincidence logic (Reference 6) considered both core response and containment pressure response assuming zero (0) ppm boron concentration in the BIT. This analysis also concluded that the applicable DNB limits were met and the margin to containment design limits was maintained. The analyses performed demonstrate that it is acceptable to bypass, eliminate, or reduce the boric acid concentration of the Indian Point 3 Boron Injection Tank.

This amendment also seeks to remove the reference to peak containment temperature from the acceptance criteria for the containment leak test. This is consistent with Westinghouse Standard Technical Specifications. The pressure test was never intended to be performed at the temperature indicated in the acceptance criteria and only indicates the postulated peak accident temperature determined by the containment response analyses. An analysis (Reference 7) to support the safety injection system flow balance test, resulted in a peak containment temperature of 261.5°F as a result of the design basis loss of coolant accident (LOCA). Therefore, 261.5°F will be incorporated in the bases section. Specification 4.4.A.2 and the associated basis is also being changed to reflect the fact that the analysis of Reference 7 resulted in a new peak containment pressure of 42.29 psig. The acceptance criteria of specification 4.4.A.2 (42.42 psig) is conservative with respect to the current peak calculated pressure of 42.29 psig.

The results of the Main Steam Line Break (MSLB) analysis using current design basis techniques resulted in a peak calculated temperature of 269.9°F, which is below the present environmental qualification limit of 290°F. However, based upon NUREG-0458 and detailed analyses performed on a similar design, Westinghouse Electric Corporation concluded that LOCAs produce the most severe environmental conditions for equipment because of the time at temperature. Therefore, the 261.5°F resulting from the LOCA analysis is deemed to be the limiting peak containment temperature.

<u>Section III - No Significant Hazards Evaluation</u>

Consistent with the requirements 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 changes do not involve a significant increase in the probability of a previously-analyzed accident. These changes involve systems relied on to mitigate the consequences of an accident. In one accident case, analyses performed by Westinghouse show a higher core power. This arises because the safety injection system must purge more water (conservatively assumed to be at 0 ppm boron) before injecting boron into the cold leg and the boron source (refueling water storage tank (RWST)) is both colder ($T_{RWST} < 40°F$ assumed) and contains a lower boron concentration than the BIT. This causes the power to initially rise to a higher peak owing to the delay and to subsequent decay at a slower rate after the boron reaches the core. However, the consequences of this increase in core power during a postulated accident clearly show adequate margins of safety within acceptable NRC criteria. The analyses performed demonstrate that it is acceptable to bypass, eliminate, or reduce the boric acid concentration of the Indian Point 3 Boron Injection Tank. The changes to the leak rate acceptance criteria and associated basis do not involve a significant increase in the probability or consequences of any accident. The leak rate test is not performed at the peak calculated temperature. The current postulated peak accident Containment pressure and temperature will be 42.29 psig and 261.5°F, which are clearly within design values. The new peak values are below the Containment design pressure and temperature of 47 psig and 271°F and the equipment qualification temperature of 290°F, and the leak rate acceptance criteria of 42.42 psig is conservative with respect to the current peak calculated pressure of 42.29 psig.

(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 changes, as analyzed, do not involve new or different kinds of accidents from those previously evaluated. The proposed changes involve existing systems and do not have un-analyzed affects on the ability to mitigate the consequences of postulated accidents. The consequences of postulated accidents involving these systems are presently described in the FSAR. The Westinghouse analyses only present the affects of the proposed changes on these consequences and the intended modifications of the systems to allow continued safe operation and increased plant availability and reliability. The changes to the leak rate acceptance criteria and associated basis will not change the overall system operation or testing.

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

Response:

The elimination of the Boron Injection Tank (BIT) does not involve a significant reduction in a margin of safety. The analyses performed to increase the ultimate heat sink allowed temperature (Reference 4) assumed a 0 ppm boron concentration in the BIT. For the Mass and Energy/Containment Pressure analysis, the impact of the BIT elimination was addressed to assure the containment pressure and temperature remain below design limits. For cases including increases in core power, Departure from Nucleate Boiling (DNB) analyses show that the DNB design basis is met and that no consequential fuel failures are expected. For the changes to the leak rate acceptance criteria and associated basis, both the peak temperature and pressure of the latest analysis remain within design values, and the leak rate acceptance criteria of 42.42 psig is conservative with respect to the current peak calculated pressure of 42.29 psig.

In the April 6, 1983 Federal Register, Vol. 048, No. 67, Page 14870, the NRC published a list of examples of amendments that are not likely to involve a significant hazards concern. Example (vi) of that list applies to the elimination of the BIT and states:

> (vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

Section IV - Impact of Changes

These changes will not adversely impact 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 increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not significantly reduce the margin of safety as defined in the bases for any Technical Specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

1) IP3 FSAR.
2) IP3 SER.
3) "Revised Feasibility Report for BIT Elimination for Indian Point Unit 3," July 1988.
4) "Containment Margin Improvement Analysis For Indian Point Unit 3," WCAP - 12269, Revision 1, May 1989.
5) "Safety Evaluation for An Ultimate Heat Sink Temperature Increase To 95°F at Indian Point 3," WCAP - 12313, July 1989.
6) Safety Analysis Of The Hypothetical Steam Line Rupture Accidents For Indian Point Unit 3 with an Additional Time Delay on the High Steam Flow Coincidence Logic, January 1989, Revised March 1990.
7) SECL-92-131, "Indian Point Unit 3 High Head Safety Injection Flow Changes Safety Evaluation," June 1992.
8) NRC Generic Letter 85-16, "High Boron Concentrations," dated August 23, 1985.

ATTACHMENT III TO IPN-93-028

FEASIBILITY REPORT

RELATED TO

BORON INJECTION TANK ELIMINATION

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