Mr. James Knubel
Chief Nuclear Officer
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

February 7, 2000

Template NRR-058

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 RE: REMOVAL OF CHEMICAL AND VOLUME CONTROL TECHNICAL SPECIFICATIONS (TAC NO. MA3943)

Dear Mr. Knubel:

The Commission has issued the enclosed Amendment No. 200 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated October 16, 1998, as supplemented January 28, 1999.

The amendment relocates Technical Specification 3.2, "Chemical and Volume Control System" and associated surveillance requirements and Bases to a licensee-controlled document.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly <u>Federal Register</u> notice.

Sincerely,

/RA/

George F. Wunder, Project Manager, Section 1
Project Directorate 1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No200 to DPR-64
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 7, 2000

Mr. James Knubel
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George F. Wunder, Project Manager, Section 1
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Docket No. 50-286

Enclosures: 1. Amendment No200 to DPR-64
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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 200
License No. DPR-64

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Power Authority of the State of New York (the licensee) dated October 16, 1998, as supplemented January 28, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 200 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Marsha K. Gamberoni, Acting Chief, Section 1
Project Directorate 1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 7, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 200

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Replace the following page of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
ii	ii
3.2-1	3.2-1
Table 4.1-1 (Sheet 2 of 6)	Table 4.1-1 (Sheet 2 of 6)
Table 4.1-1 (Sheet 4 of 6)	Table 4.1-1 (Sheet 4 of 6)
Table 4-1-2 (Sheet 1 of 2)	Table 4.1-2 (Sheet 1 of 2)
Table 4.1-3 (Sheet 1 of 2)	Table 4.1-2 (Sheet 1 of 2)

3.2 Deleted

TABLE 4.1-1 (Sheet 2 of 6)

Channel Description	Check	Calibrate	Test	Remarks
8. 6.9 KV Voltage	N.A.	18M	Q	Reactor protection circuits only
6.9 KV Frequency	N.A.	24M	Q	Reactor protection circuits only
9. Analog Rod Position	S	24M	M	
10. Steam Generator Level	S	24M	Q	
11. Residual Heat Removal Pump Flow	N.A.	24M	N.A.	
12. Deleted				
13. Refueling Water Storage Tank Level				
a. Transmitter	W	18M	N.A.	Low level alarm
b. Indicating Switch	W	6M	N.A.	Low level alarm
14a. Containment Pressure - narrow range	S	24M	Q	High and High-High
14b. Containment Pressure - wide range	M	18M	N.A.	
15. Process and Area Radiation Monitoring:				
a. Fuel Storage Building Area Radiation Monitor (R-5)	D	24M	Q	
b. Vapor Containment Process Radiation Monitors (R-11 and R-12)	D	24M	Q	
c. Vapor Containment High Radiation Monitors (R-25 and R-26)	D	24M	Q	
d. Wide Range Plant Vent Gas Process Radiation Monitor (R-27)	D	24M	Q	

TABLE 4.1-1 (Sheet 4 of 6)

Channel Description	Check	Calibrate	Test	Remarks
25. Level Sensors in Turbine Building	N.A.	N.A.	24M	
26. Deleted				
27. Deleted				
28. Auxiliary Feedwater:				Low-Low
a. Steam Generator Level	S	24M	Q	
b. Undervoltage	N.A.	24M	24M	
c. Main Feedwater Pump Trip	N.A.	N.A.	24M	
29. Reactor Coolant System Subcooling Margin Monitor	D	24M	N.A.	
30. PORV Position Indicator	N.A.	N.A.	24M	Limit Switch
31. PORV Position Indicator	D	24M	24M	Acoustic Monitor
32. Safety Valve Position Indicator	D	24M	24M	Acoustic Monitor
33. Auxiliary Feedwater Flow Rate	N.A.	18M	N.A.	
34. Plant Effluent Radioiodine/ Particulate Sampling	N.A.	N.A.	18M	Sample line common with monitor R-13
35. Loss of Power				
a. 480v Bus Undervoltage Relay	N.A.	24M	M	
b. 480v Bus Degraded Voltage Relay	N.A.	18M	M	
c. 480v Safeguards Bus Undervoltage Alarm	N.A.	24M	M	
36. Containment Hydrogen Monitors	D	Q	M	

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 day(4) 10 days 5 days 45 days 3 days 10 days 30 Weeks 20 days
2. Deleted			
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

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS

	Check	Frequency
1. Control Rods	Rod drop times of all control rods	24M
2. Control Rods	Movement of at least 10 steps in any one direction of all control rods	Every 31 days during reactor critical operations
3. Pressurizer Safety Valves	Set Point	24M
4. Main Steam Safety Valves	Set Point	24M
5. Containment Isolation System	Automatic actuation	24M
6. Refueling System Interlocks	Functioning	Each refueling, prior to movement of core components
7. Primary System Leakage	Evaluate	5 days/week
8. Diesel Generators Nos. 31, 32 & 33 Fuel Supply	Fuel Inventory	Weekly
9. Turbine Steam Stop Control Valves	Closure	Not to exceed 6 months**
10. L.P. Steam Dump System (6 lines)	Closure	Monthly
11. Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Quarterly
12. Deleted		

** The turbine steam stop and control valves shall be tested at a frequency determined by the methodology presented in WCAP-11525, "Probablistic Evaluation of Reduction in Turbine Valve Test Frequency," as updated by Westinghouse Report, WOG-TVTF-93-17, "Update of BB-95/96 Turbine Valve Failure Rates and Effect on Destructive Overspeed Probabilities." The maximum test interval for these valves shall not exceed six months. Surveillance interval extension as per Technical Specification 1.12 is not applicable to the maximum test interval.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 200 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated October 16, 1998, as supplemented by letter dated January 28, 1999, the Power Authority of the State of New York (the licensee) requested an amendment to the Technical Specifications (TSs) for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendment would relocate the TSs for the Chemical Volume and Control System (CVCS) to a licensee-controlled document. The CVCS TS contains the requirements for the charging pumps, the boric acid transfer pumps and storage tanks, the boration flow paths, the heat tracing for the boration systems and the city water piping with the valves used for back-up cooling to the charging pumps. The licensee intends to relocate these requirements to a licensee-controlled document subject to the controls contained in the 10 CFR 50.59 change process. The licensee has justified the change by stating that the CVCS does not meet any of the four criteria contained in 10 CFR 50.36(c)(2)(ii), Limiting Conditions for Operation (LCOs), which establishes the components that require TS LCOs.

2.0 EVALUATION

At 10 CFR 50.36, the regulations describe what needs to be contained in the plant TSs; specifically, in 10 CFR 50.36(c)(2)(ii), the regulation describes the four items or criteria that establish when LCOs are needed. The staff, in the statements of consideration for the rulemaking on 50.36 stated that, "LCOs that do not meet any of the criteria, and their associated actions and surveillance requirements, may be proposed for relocation from the technical specifications to licensee-controlled documents, such as the FSAR." The submittal states that TSs related to the CVCS system are not required to be in the TS because they do not meet the criteria contained in 10 CFR 50.36(c)(2)(ii). The four criteria are discussed below.

> Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The CVCS is not used to detect or indicate a degradation in the reactor coolant pressure boundary.

> Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The CVCS is not credited with mitigating any design-basis accident (DBA) and the CVCS TS does not establish the initial conditions of a DBA.

> Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The CVCS is not credited with mitigating any DBA.

> Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The licensee has used a risk assessment to demonstrate that the CVCS is not significant to public health and safety. The staff's detailed analysis of the licensee's risk assessment is provided in the following section.

2.1 Evaluation of Risk Assessment

Criterion 4 of 10 CFR 50.36(c)(2)(ii) states that TS would be required for "A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety." In demonstrating that CVCS does not meet this criterion, the licensee stated that the core damage frequency (CDF) contribution from system-related CVCS failures is small at about 6.5E-7/year. This risk contribution represents about 1.4 percent of the total plant CDF, 4.4E-5/year, as described in the plant Individual Plant Examination. The risk contribution consists of accident sequences involving the CVCS boration function for anticipated transient without scram (ATWS) and the reactor coolant pump (RCP) seal cooling function for station blackout, RCP seal loss-of-coolant accident (LOCA), and Appendix R sequences. The licensee further stated that the TS relocation does not change system design, operation, operational set points or failure rate of system components and no credit has been taken for the existence of relocated requirements for system operation and testing.

The staff agrees that the CDF contribution from sequences which involve CVCS components is low. With respect to the assumption that excluding CVCS from the plant TS would have no impact on the reliability of system components, we agree that the likelihood of CVCS component failure rates changing due to relocation is low; however, it is noted that this assumption is yet to be supported by real data. We also note that CVCS is covered in the IP3 Maintenance Rule scope and that monitoring and trending are performed to meet the performance criteria set by the licensee.

Along with the CDF contribution, the staff reviewed the IP3 Individual Plant Evaluation (IPE) to examine the importance of CVCS components with respect to the total plant risk. The IPE ranked 315 most important basic events according to their risk-increase importance measure, whose range included a Risk Achievement Worth (RAW) value of about 7600 (highest ranking event) to a RAW value of nearly 1 (lowest ranking event). Of these 315 most important basic events, only three CVCS components appeared on this list and were ranked near the middle of the pack. These are check valves CH-210B, CH-210D, and CH-374 and the risk-increase measures for all three valves were the same and corresponded to a Risk Achievement Worth (RAW) of about 2. The failure of these valves (to open) would prohibit charging injection into the reactor coolant system for small-small break loss-of-coolant accident and boric acid injection for post ATWS event. The reactor coolant pump seal injection is not dependent on these valves because seal injection uses a separate flow path that does not pass through these check valves. We consider the relatively low RAW values of these check valves (with no other CVCS components appearing on the risk-increase importance rank list) and the use of these valves in accident scenarios not involving seal injection to support the licensee's conclusion that risk contribution from CVCS is not significant.

The staff guidance in SECY 95-128, "Final Rulemaking Package for 10 CFR 50.36, Technical Specifications," dated May 19, 1995, states that PRA insights should be used to indicate "...whether the provisions to be relocated contain constraints of importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk." Based on the above discussion, we agree that components in CVCS do not dominate risk and support the licensee's conclusion that the probabilistic safety assessment does not show CVCS to be significant to public health and safety.

With respect to the quality of PRA, we examined the quality of the risk-related information contained in the submittals as well as the staff's original evaluation of the IP3 IPE which concluded that the licensee's IPE was complete with regards to Generic Letter 88-20 requirements and that its results were reasonable.

The CDF contribution from sequences which involve CVCS components was determined to be low at about 6.5E-7/year which is about 1.4 percent of the total plant CDF. The RAW values of the three check valves listed on the most important risk-increase measure list (with no other CVCS components appearing on the IPE risk-increase importance rank list) are low and these valves are relied on in accident scenarios not requiring seal injection. Based on these findings, the staff concludes that components in CVCS meets the guidance provided in SECY 95-128, "Final Rulemaking Package for 10 CFR 50.36, Technical Specifications" and do not dominate risk and that the risk contribution from CVCS is not significant to public health and safety.

The NRC staff has reviewed the amendment request and the justification provided in the licensee's submittals and determined that TS 3.2 for the CVCS does not meet the criteria in 10 CFR 50.36. As a result, the staff finds it acceptable to remove the specification from the TS and relocate it to a licensee-controlled document subject to 10 CFR 50.59 and finds the revised TS, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (64 FR 9200). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: C. Jackson
S. Lee

Date: February 7, 2000

Template NRR-058

DATED: February 7, 2000

AMENDMENT NO. 200 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT UNIT 3

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