BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

7LA00L

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

APPLICATION FOR AMENDMENT TO
OPERATING LICENSE

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

The proposed changes to the Indian Point 3 Technical Specifications serve to amend certain Sections of Appendix A to the Operation license in accordance with the requirements of Generic Letter No. 82-16.

The proposed changes to the Technical Specifications are presented in Attachment I to this Application. The Safety Evaluation is included in Attachment II.

POWER AUTHORITY OF THE STATE
OF NEW YORK

BY *C. M. Wilverding* for
J. P. Bayne
Executive Vice President
Nuclear Generation

Subscribed and Sworn to before
me this 1st day of June 1983

Linda Della Donna
Notary Public

LINDA DELLA DONNA
NOTARY PUBLIC, State of New York
No. 60-6669935
Qualified in Westchester County
Term Expires March 30, 1984

Attachment A

Response to Generic Letter No. 82-16

NUREG-0737 Technical Specifications

IPN-83-49

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

The information below is previded in response to Generic Letter No. 82-16 regarding implementation of NUREG-0737 Technical Specifications. Each of the Items contained in the Generic Letter is addressed below. The revised Technical Specification pages are contained in Attachment No. 2 of this submittal.

Item (1): STA Training (I.A.1.1.3)

As stated in the Generic Letter there is currently no requirement for submittal of Technical Specifications regarding Shift Technical Advisor qualifications, training and on-shift duties.

Item (2): Limit Overtime (I.A.1.3)

As stated in the Authority's letter dated February 15, 1983 (IPN-83-13), current Administrative Procedures contain restrictions which meet the requirements of the NRC policy statement attached to Generic Letter No. 82-12 as they apply to persons on shift coverage. The Authority believes that Section 6.8, Procedures, of the Indian Point 3 Technical Specifications adequately addresses this concern and therefore proposes no Specifications with this letter.

Items (3) and (4) Auxiliary Feedwater System (II.E.1.1 and II.E.1.2)

The Limiting Conditions for Operation (LCO) and the Surveillance requirements for the Auxiliary Feedwater (AFWS) System are defined in Technical Specification sections 3.4.1 and 4.8, respectively. Both the LCO and the Surveillance Requirements for the AFW System are similar to those for other safety-related systems and the requirements of the Generic Letter are thereby fulfilled. NRC acceptance of the current Indian Point 3 AFWS Technical Specifications has been previously documented in Mr. S. A. Varga's letter dated August 10, 1982 and no additional Technical Specification changes are required.

Item (5) Dedicated Hydrogen Penetrations (II.E.4.1)

Indian Point 3 utilizes hydrogen recombiners located inside reactor containment. Hence, this Item is not applicable and no Technical Specifications changes are required.

Item (6): Containment Pressure Setpoint (II.E.4.2.5)

NRC acceptance of the current Indian Point 3 containment pressure Technical Specification setpoint has been previously documented in Mr. S. A. Varga's letter dated November 24, 1981 and no additional Technical Specification changes are required.

Item (7): Containment Purge Valves (II.E.4.2.6)

A proposed Technical Specification 3.6.1 is included in Attachment No. 2 of this submittal reflecting the NRC position summarized in the Generic Letter. The proposed Specification requires that the containment purge and exhaust valves be de-energized in the closed position when the reactor is not in the cold shutdown mode, except when used for safety-related purposes. In addition, the valve positions will be verified at least once per month. The valves will be leak rate tested in accordance with the requirements of 10 CFR 50, Appendix J.

Item (8): Radiation Signal on Purge Valves (II.E.4.2.7)

The proposed Technical Specification 3.6.1 discussed in Item 7 requires that one process radiation monitor capable of initiating automatic isolation of the purge supply and exhaust valves be operable whenever the purge supply and exhaust valves are open. The setpoint for automatic isolation of the purge supply and exhaust valves will be set in accordance with the Radiological Environmental Technical Specifications (RETS) and the Offsite Dose Calculation Manual (ODCM). The Authority feels that no additional Technical Specification changes are required for this item. NRC acceptance of the Authority's compliance with Items II.E.4.2.6 and II.E.4.2.7 has been previously documented in Mr. S. A. Varga's letter dated November 9, 1982.

Item (9): Upgrade B&W AFW System (Item II.K.2.8)

This Item is not applicable to the Indian Point 3 Nuclear Power Plant.

Item (10): No Item (10) was included in the Attachment to Generic Letter No. 82-16.

Item (11): B&W Thermal-Mechanical Report (Item II.K.2.13)

This Item is not applicable to the Indian Point 3 Nuclear Power Plant.

Item (12): Reporting SV and RV Failures and Challenges (Item II.K.3.3)

Technical Specification Section 6.9 has been revised to reflect the requirements of NUREG-0660 regarding SV and RV challenges and failures. Specifically, Specification 6.9.1.7 has been revised to require prompt notification of SV and RV failures and Specification 6.9.1.9, Annual Reports, has been proposed to require annual reporting of SV and RV challenges. The appropriate revised Technical Specification pages are included in Attachment No. 2.

Item (13): Anticipatory Trip on Turbine Trip (Item II.K.3.12)

The Limiting Conditions for Operation and the Surveillance Requirements for the anticipatory trip instrumentation (turbine electrical overspeed protection) are contained in Technical Specification Table 3.52 and Table 4.1-1, respectively. NRC acceptance of the current Indian Point 3 "Anticipatory Trip" Technical Specifications has been previously documented in Mr. J. O. Thoma's letter dated October 29, 1981 and no additional Technical Specification changes are required.

ATTACHMENT NO. 1

RESPONSE TO GENERIC LETTER NO. 82-16

NUREG-0737 TECHNICAL SPECIFICATIONS

IPN-83- 49

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

TABLE OF CONTENTS

3.6.1 Containment Purge System

Applicability

Applies to the containment purge system whenever the plant is above the cold shutdown condition.

Objective

To define the operating status of the containment purge system.

Specification

1. Operation of the containment purge supply and exhaust valves shall be limited to safety-related activities.

2. At least one process radiation monitor capable of automatically initiating closure of the purge supply and exhaust valves must be operable whenever the purge supply and exhaust valves are open.

3. Except as allowed in Item 1. above, the purge supply and exhaust valves shall be maintained in the closed position with their associated actuation systems de-energized. The purge supply and exhaust valves shall be verified to be in the closed position at least once per month.

REPORTABLE OCCURRENCES

6.9.1.6 The REPORTABLE OCCURRENCES of Specifications 6.9.1.7 and 6.9.1.8 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.7 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected system when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment. 5/

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any planned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Failure of a pressurizer power-operated relief valve or safety valve to function when called upon to operate during plant operations.

5/ Leakage of packing, gaskets, mechanical joints and seal welds within the limits for identified leakage set forth in technical specifications need not be reported under this item, Steam Generator tube degradation need not be reported under this item except where leakage exceeds the limits of specification 3.1.f.

d. Abnormal degradation of systems other than those specified in 6.9.1.7.c above designed to contain radioactive material resulting from the fission process. 7/

ANNUAL REPORTS

6.9.1.9 A summary of any challenges to the pressurizer power-operated relief valves and (or) safety valves shall be submitted to the Regional Administrator of Region I on an annual basis.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the acitivities identified below pursuant to the requirements of the applicable reference specification:

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

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Primary coolant activity in excess of limits (Specification 3.1.D)

d. Seismic event analysis (Specification 4.10)

e. Inoperable fire Protection and detection equipment (Specification 3.14)

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

6.10 RECORD RETENTION

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

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

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

c. ALL REPORTABLE OCCURRENCES submitted to the Commission.

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

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

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

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

i. Records of reactor tests and experiments.

7/ Sealed sources or calibration sources are not included under this item. Leakage of packing, gaskets, mechanical joints and seal welds within the limits for identified leakage set forth in technical specifications need not be reported under this item.

ATTACHMENT NO. 2

RESPONSE TO GENERIC LETTER NO. 82-16

NUREG-0737 TECHNICAL SPECIFICATIONS

IPN-83-49

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

Section I - Description of Modification

Certain Sections of those Technical Specifications are being amended in accordance with the requirements of Generic Letter No. 82-16.

Section II- Purpose of Modifications

The purpose of these modifications is to implement those Technical Specifications related to NUREG-0737 identified in Generic Letter No. 82-16.

Section III - Impact of Change

These modifications will not alter the conclusions reached in the FSAR and SER accident analyses nor will they impact the ALARA or Fire Protection Program at IP-3.

Section IV - Conclusion

The incorporation of these modifications: 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 reduce the margin of safety as defined in the basis 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 V - References

(a) IP-3 FSAR
(b) IP-3 SER
(c) Generic Letter No. 82-16