July 17, 1990

Docket No. 50-286

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Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
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Dear Mr. Brons:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT 3 (TAC NO. 76468)

The Commission has issued the enclosed Amendment No. 101 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 in response to your application transmitted by letter dated March 28, 1990.

The amendment revises Technical Specifications Sections 2.1, 2.3, 3.3, 4.2, 4.3, 4.5, 5.3 and 6.13, Table of Contents, List of Tables, List of Figures and Figures 3.1.A-5 and 3.1.A-6 of Appendix A. Proposed changes to the Table of Contents and Section 5.3 of Appendix B, Part II, are also included. The proposed amendment is administrative in nature and consists of various changes to achieve consistency between Technical Specification sections, clarify Bases, correct inadvertent errors made by previous amendments, and delete and/or update superseded text.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Joseph D. Neighbors, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.101 to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

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CVogan	DNeighbors:rsc		RACapra
7-11-90	7/11/90	7-12-90	7/12/90

DOCUMENT NAME: AMENDMENT 76468

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P PDC

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Power Authority of the State
of New York

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

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. 101
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated March 28, 1990, 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:

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P PDC

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 101 , 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 17, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

	Remove Pages	Insert Pages
Appendix A:		
	i	i
	ii	ii
	v	v
	List of Figures	List of Figures
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	2.1-3	2.1-3
	2.3-6	2.3-6
	Figure 3.1.A-5	Figure 3.1.A-5
	Figure 3.1.A-6	Figure 3.1.A-6
	3.3-11	3.3-11
	3.3-12	3.3-12
	3.3-17	3.3-17
	4.2-2	4.2-2
	Table 4.2-1	---
	4.3-1	4.3-1
	4.5-2	4.5-2
	4.5-11	4.5-11
	5.3-2	5.3-2
	6.19	6.19
Appendix B:		
	Table of Contents, Section 5.0	Table of Contents, Section 5.0
	5-1	5-1
	5-4	5-4

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INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

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In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting rod is greater than or equal to the applicable DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. The DNBR uncertainty combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin is maintained by performing DNB design evaluations to a higher DNBR value, called the Safety Limit DNBR.

The curves of Figure 2.1-1 show the loci of points of thermal power, Reactor Coolant System pressure and vessel inlet temperature for which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The calculation of these limits includes:

1. $F^N_{\Delta H}$ of 1.56

2. an equivalent steam generator tube plugging level of up to 30% in any steam generator provided the equivalent average plugging level in all steam generators is less than or equal to 24%, (2)

3. a reactor coolant system total flow rate of greater than or equal to 332,240 gpm as measured at the plant,

4. a reference cosine with a peak of 1.55 for axial power shape.

Figure 2.1-1 includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$$F^N_{\Delta H} \leq 1.56\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of Rated Thermal Power.

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.6% measurement uncertainty on Flow and a 4% measurement uncertainty of $F_{\Delta H}$ have already been included in the above limits.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (Figure 3.10-4) assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

References

1. FSAR Section 3.2.2.

2. "Safety Evaluation for Indian Point Unit 3 With Asymmetric Tube Plugging Among Steam Generators", WCAP-10705 (Westinghouse Non-Proprietary), October 1984.

The constants delta-T_o and T' for each overtemperature and overpower protection channel are set in accordance with the measured delta-T and T_{avg} at rated power existing in the loop from which the process inputs for a particular protection channel are supplied. This is done to account for loop to loop differences in delta-T and T_{avg} which may exist as a result of asymmetric steam generator tube plugging.

The low flow reactor trip protects the core against DNB in the event of a loss of one or two reactor coolant pumps. The undervoltage reactor trip protects the core against DNB in the event of a loss of two or more reactor coolant pumps. The set points specified are consistent with the values used in the accident analysis. (8) The low frequency reactor coolant pump trip also protects against a decrease in flow. The specified set point assures a reactor trip signal by opening the reactor coolant pump breaker before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1600 ft^3 of water (39.75 ft. above the lower instrument tap) corresponds to 92% of span. The specified set point allows margin for instrument error and transient level overshoot beyond their trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against postulated loss of feedwater accidents. This specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the Auxiliary Feedwater System (9).

Specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set points at which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above the P-8 setpoint for four-loop operation, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below the applicable safety limit during normal operational transients.

A turbine trip causes a direct reactor trip, when operating at or above 10% power. This anticipatory trip will operate in advance of the pressurizer high pressure reactor trip to reduce the peak Reactor Coolant System pressure. No credit was taken in the accident analyses for operation of this trip. (10)

The steam-feedwater flow mismatch trip does not appear in the specification as this setting is not used in the transient and accident analysis (FSAR Section 14).

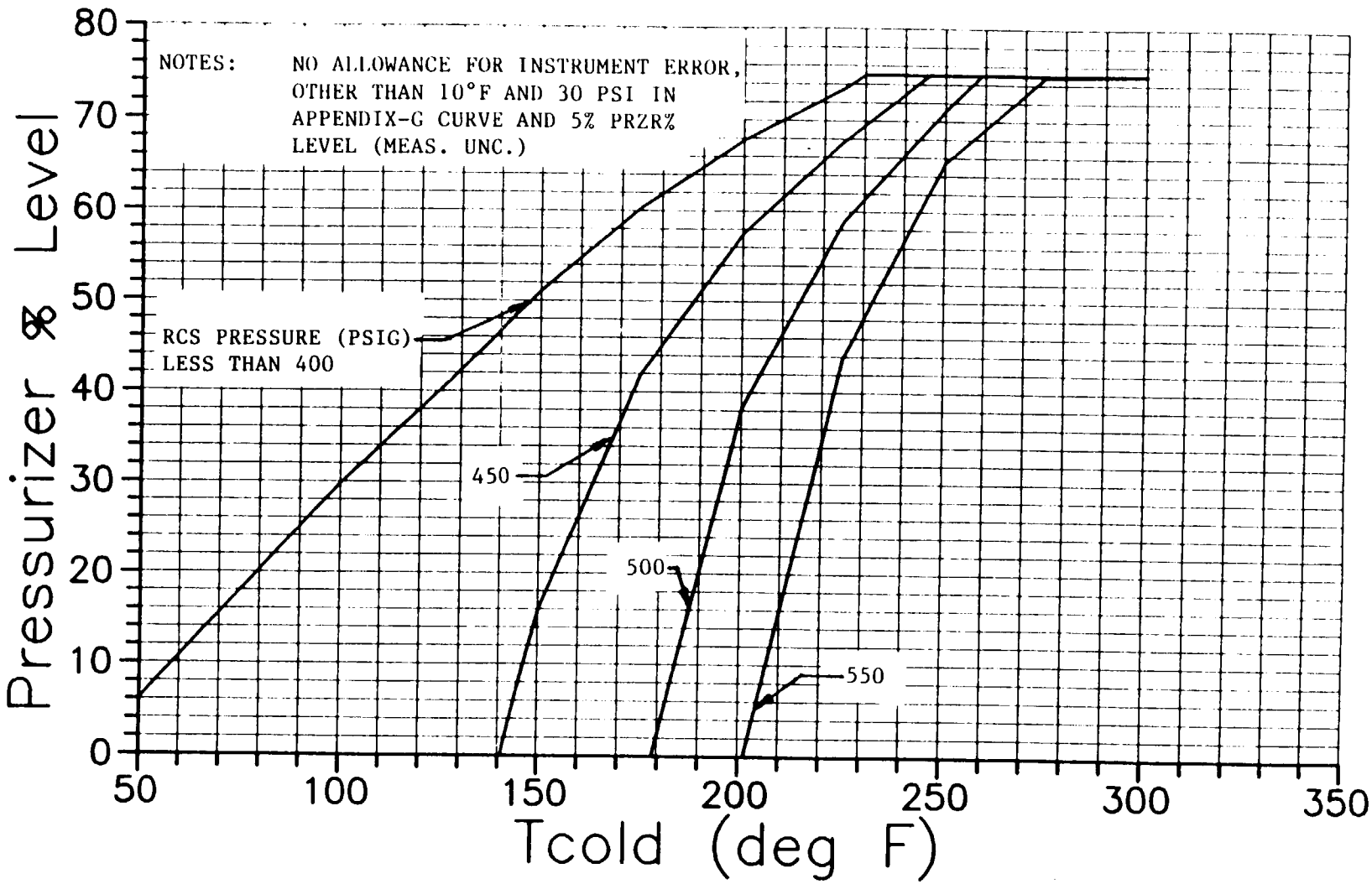

Figure 3.1.A–5

Figure applicable to 9.26 EFPY

MAXIMUM PRESSURIZER LEVEL WITH OPS INOPERABLE AND ONE (1) CHARGING PUMP ENERGIZED

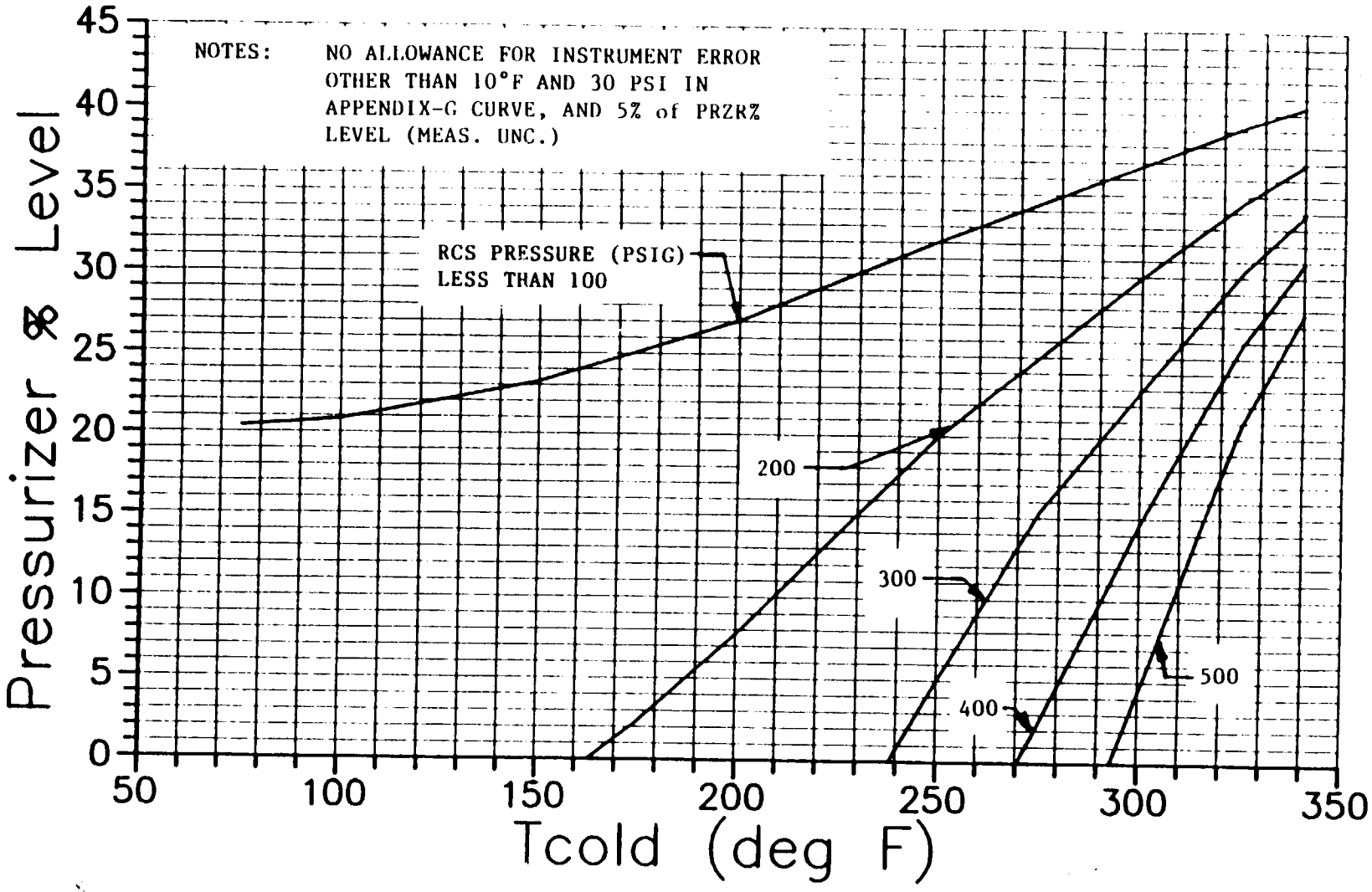

Figure 3.1.A–6

Figure applicable to 9.26 EFPY

MAXIMUM PRESSURIZER LEVEL WITH OPS INOPERABLE AND ONE (1) SAFETY INJECTION PUMP AND/OR THREE (3) CHARGING PUMPS ENERGIZED

G. Hydrogen Recombiner and Containment Hydrogen Monitoring Systems

1. The reactor T_{avg} shall not exceed 350° unless the following requirements are met:

   a) Both hydrogen recombiner units together with their associated piping, valves, oxygen supply system and control system are operable.

   b) One hydrogen monitor including a flow path and associated containment fan cooler unit shall be operable.

   c) Hydrogen and oxygen supplies shall not be connected to the hydrogen recombiner units except under conditions of an accident or those specified in 4.5.A.3.

2. The requirements of 3.3.G.1 may be modified to allow any one of the following components to be inoperable at any one time:

a. One hydrogen recombiner unit or its associated flow path, or oxygen supply system or control system may be inoperable for a period not to exceed 15 days, provided the other recombiner unit is demonstrated to be operable.

b. Both containment hydrogen monitoring systems may be inoperable for a period not to exceed 7 days.

3. If the Hydrogen Recombiner System is not restored to meet the requirements of 3.2.G.1 within the time period specified in 3.3.G.2, then:

a. 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.

b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more then 25°F and 100 psi, respectively, over existing values.

c. In either case, if the requirements of 3.3.G.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.

H. Control Room Ventilation System

1. The control room ventilation system shall be operable at all times when containment integrity is required as per specification 3.6.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95°F). (4) In the event of a Design Basis Accident, any one of the following configurations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also in the event of a Design Basis Accident, any one of three configurations of fan-cooler units (with charcoal filters) and/or containment spray pumps (with sodium hydroxide addition) will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values. (5) Any one of these three configurations constitutes the minimum safeguards for iodine removal.

The combination of three fan-coolers and one containment spray pump is capable of being operated on emergency power with one diesel generator failing to start. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

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.

4.2.1.4 BASES

This specification ensures that inservice inspection of ASME Code Class 1,2 and 3 components will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

Under the terms of this Specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda.

4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

Specification

a) When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig and in accordance with NDT requirements for temperature.

b) When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components, the new welds will meet the requirements of ASME Section XI.

c) The reactor coolant system leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first 9.26 EFPY's of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure 3.1-2.

Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (Operating pressure + 100 psi: ± 100 psi is normal system pressure fluctuation), it will be leak tight during normal operation.

2. Containment Spray System

a. System tests shall be performed at each reactor refueling interval. The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. The spray nozzles shall be checked for proper functioning at least every five years.

c. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

3. Hydrogen Recombiner and Containment Hydrogen Monitoring Systems

a. A complete recombiner system test shall be performed at each normal reactor refueling on each unit. The test shall include verification of ignition and attainment of normal operating temperature.

b. A complete control system test shall be performed at intervals not greater than six months on each unit. The test shall consist of a complete dry-run startup using artificially generated signals to simulate light off.

c. Containment hydrogen monitoring system tests shall be performed at intervals no greater than six months. The test shall include drawing a sample from the fan cooler units.

d. The above tests will be considered satisfactory if visual observations and control panel indication indicate that all components have operated satisfactorily.

e. Each recombiner air-supply blower shall be started at intervals not greater than two months. Acceptable levels of performance shall be that the blowers start, deliver flow, and operate for at least 15 minutes.

With the efficiencies of the HEPA filters and charcoal adsorbers as specified, further assurance is provided that the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10CFR Part 50.

A pressure drop across the combined HEPA filters and charcoal adsorbers of less than or equal to 6.0 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability. Proper operation of the system fans should also be verified at least every refueling by either direct or indirect measurements.

If results of charcoal tests are unsatisfactory, two additional samples may be tested. If both of these tests are acceptable, the charcoal may be considered satisfactory for use in the plant. Should the charcoal of any of these air filtration systems fail to satisfy the test criteria outlined in this specification, the charcoal beds will be replaced with new charcoal which satisfies the requirements for new charcoal outlined in Regulatory Guide 1.52 (Revision June, 1973).

The hydrogen recombiner system is an engineered safety feature which would be used only following a loss-of-coolant accident to control the hydrogen evolved in the containment. The system is not expected to be started until approximately 12 days have elapsed following the accident. At this time, the hydrogen concentration in the containment will have reached 2% by volume, which is the design concentration for starting the recombiner system. (3) Actual starting of the system will be based upon containment atmosphere sample analysis. The complete functional tests of each unit at refueling shutdown will demonstrate the proper operation of the recombiner system. More frequent tests of the recombiner control system and air-supply blowers will assure operability of the system. The bi-annual testing of the containment hydrogen monitoring system will demonstrate the availability of this system.

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with the stainless steel. (5)

B. Reactor Coolant System

1. The design of the reactor coolant system complies with the code requirements. (6)

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no loss of function.

3. The total liquid volume of the reactor coolant system, at rated operating conditions, is 11,522 cubic feet.

References

(1) FSAR Section 3.2.2

(2) FSAR Section 3.2.1

(3) FSAR Section 3.2.1

(4) FSAR Section 3.2.3

(5) FSAR Sections 3.2.1 & 3.2.3

(6) FSAR Table 4.1-9

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 Environmental qualification of electric equipment important to safety shall be in accordance with the provisions of 10CFR 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.

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5.0 ADMINISTRATIVE CONTROLS

5.1 RESPONSIBILITIES

The responsibiliites of the Plant Operating Review Committee and the Safety Review Committee associated with the implementation of the Radiological Environmental Technical Specifications are delineated in the appropriate sections of Appendix A Specification 6.5

5.2 PROCEDURES AND PROGRAMS

Reference to implementation of the procedures and programs necessary for the implementation of the Radiological Environmental Technical Specifications are delineated in Appendix A Specification 6.8.1

5.3 REPORTING REQUIREMENTS

5.3.1 SPECIAL REPORTS

The activities requiring the submittal of special reports are delineated in Appendix A Specification 6.9.2

5.3.2 ROUTINE REPORTS

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT *

5.3.2.1 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plant," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof or as modified in the R.E.T.S.

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* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

5.3.2.2 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison, as appropriate, with preoperational studies, with operational controls, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 2.8.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 2.9; discussion of all deviations from the sampling schedule of Table 2.7-1; and discussion of all analyses in which the LLD required by Table 3.7-1 was not achievable.

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* A single submittal may be made for a multiple unit station.

** One map shall cover stations near the site boundary; a second shall include the more distant stations.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 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

INTRODUCTION

By letter dated March 28, 1990, the Power Authority of the State of New York (the licensee) requested an amendment to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The proposed amendment would revise the Technical Specifications (TS) to achieve consistency between TS sections, clarify Bases, correct inadvertent errors made by previous amendments, and delete and/or update superseded text.

DESCRIPTION AND EVALUATION

(1) Table of Contents of Appendix A:

The proposed changes reflect the change to the title of Technical Specifications 3.3.G and 4.5.A.3.

The changes are purely administrative in nature, and update the listing to incorporate changes being made elsewhere in the Technical Specifications.

(2) List of Tables of Appendix A:

The proposed changes to the List update the list and ensure consistency between the titles listed and those appearing on the tables contained in the Technical Specifications.

One of the changes to the List of Tables corrects the title listed for Table 3.6-1 to match with that appearing on the table in the Technical Specifications. Additionally, Table 3.10-1, which was previously incorporated into the Technical Specifications by Amendment 34, and Tables 3.14-2 and 3.14-3, incorporated by Amendment No. 45, but which were not included in the List, are being added. Moreover, reference to Table 4.2-1 is being deleted since this table is no longer applicable. The changes update the listing and are acceptable.

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(3) List of Figures of Appendix A:

The proposed changes update the List and ensure consistency between the titles listed and those appearing on the figures contained in the Technical Specifications.

Two of the changes to the List of Figures correct the titles listed for Figures 3.1.A-5 and 3.1.A-6 to match with those appearing on the figures in the Technical Specifications. Figure 4.2-1 was deleted from the Technical Specifications by Amendment No. 57. The title for this figure is being deleted from the list. The changes update the listing and are acceptable.

(4) Section 2.1 of Appendix A:

The proposed change adds the title of a reference made in the Basis.

The change reinserts Reference 2 which was inadvertently deleted by Amendment No. 86 to the Technical Specifications.

(5) Section 2.3 of Appendix A:

The proposed change to this section's Basis removes reference to the turbine trip exclusion from the Technical Specifications.

On April 30, 1986 (IPN-86-20) the licensee submitted a proposed amendment concerning Anticipatory Reactor Trip on Turbine Trip. Amendment No. 68 to Indian Point 3's operating license incorporated Technical Specifications for an Anticipatory Reactor Trip on Turbine Trip. The consequent change to page 2.3-6 added the basis for including this trip in the list of the protective instrumentation settings for reactor trip interlocks. However, the last paragraph of page 2.3-6 addresses text which has been invalidated by the addition of the turbine trip to the Technical Specifications. Removal of this text is acceptable.

(6) Figures 3.1.A-5 and 3.1.A-6 of Appendix A:

The proposed changes correct the titles appearing on the figures.

Specification 3.1.A.8.b.(3) specifies use of the figures under the following conditions: 1) for one PORV and/or its series MOV inoperable for greater than seven consecutive days, or 2) for both PORVs and/or their series MOVs inoperable. The titles of the figures imply use for only PORVs inoperable. The figure is to be used under other conditions as well. The titles are being changed to indicate use with OPS inoperable; the exact cause of OPS inoperability not being specified.

(7) Section 3.3 of Appendix A:

a. The proposed change to the title of Specification 3.3.G emphasizes that requirements for the Containment Hydrogen Monitoring System are also contained in this section.

The change to the title of Specification 3.3.G is administrative in nature. The change emphasizes that Containment Hydrogen Monitoring System requirements are contained in this Specification, in addition to those for the Hydrogen Recombiner System.

b. Other proposed changes to this section clarify the Technical Specification Basis regarding post-LOCA iodine removal and containment pressure reduction using the fan cooler units and the containment spray pumps.

The Basis for Technical Specification 3.3.B does not clearly state the following fact. In the event of a Design Basis Accident, any one of the following combinations will both reduce airborne organic and molecular iodine from the containment atmosphere, and reduce containment pressure, at a rate consistent with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, and (3) three fan-cooler units and one spray pump. The change clarifies the Basis and is acceptable.

(8) Section 4.2 of Appendix A:

The proposed change deletes reference to the augmented inspection of steam generator girth weld number 6.

This section contains information regarding the surveillance and inspection of steam generator girth weld number 6. With replacement of the steam generators at Indian Point 3 having been performed during the Cycle 6/7 Refueling Outage, this information is no longer applicable and deletion of the information is acceptable.

(9) Section 4.3 of Appendix A:

The proposed change deletes reference to articles IS400 and IS500 of Section XI of the ASME Boiler and Pressure Vessel Code.

The change deleting reference to ASME Section XI Articles IS400 and IS500 on page 4.3-1 reflects conformance with 10 CFR Part 50, Section 50.55a(g) which prescribes the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code. The Technical Specifications currently reference requirements derived from the 1971 edition of the Code which are no longer applicable. Deletion of reference to superseded Articles IS400 and IS500 is acceptable.

(10) Section 4.5 Appendix A:

The proposed change to the title of Specification 4.5.A.3 emphasizes that requirements for the Containment Hydrogen Monitoring System are also contained in this section.

The proposed changes to Specification 4.5.A.3.c and to the corresponding Basis deletes reference to the containment atmosphere sampling system which was replaced by the containment hydrogen monitoring system.

Specifications for the containment atmosphere sampling system, which was replaced by the containment hydrogen monitoring system, were deleted by Amendment No. 80. Correction of current Technical Specification 4.5.A.3.c was missed at that time. One of the changes deletes reference to the superseded system. The other change revises the title of Specifications 4.5.A.3 to include reference to the Containment Hydrogen Monitoring System. These changes are acceptable.

(11) Section 5.3 Appendix A:

The proposed change corrects the units given for the reactor coolant system liquid volume.

The change corrects an inadvertent deletion of the word "cubic" which is part of the units of measure describing the liquid volume of the reactor coolant system contained in Specification 5.3.B.3.

(12) Section 6.13 of Appendix A:

The proposed changes update Equipment Qualification requirements pursuant to 10 CFR 50.49.

On November 19, 1984, (Federal Register Vol. 49, No. 224, p. 45571), the Commission issued a final rule deleting the June 30, 1982, deadline for the environmental qualification of electrical equipment from individual licenses. The rule required, instead, that licensees meet the schedule set forth in 10 CFR 50.49, which mandates the environmental qualification of electrical equipment important to safety for nuclear power plants. Its provisions include the preparation of a list identifying the electrical equipment requiring environmental qualification, and the maintenance of complete and auditable records detailing the degree of compliance with the Equipment Qualifications (EQ) Program for all equipment identified on the list. The changes to page 6-19 reflect conformance with 10 CFR 50.49.

(13) Table of Contents of Appendix B, Part II:

The proposed changes to this Table incorporate proposed changes to Section 5.3 of Appendix B, Part II.

The changes are purely administrative in nature, and update the listing to incorporate changes being made elsewhere in the Technical Specifications. Therefore, these changes are acceptable.

(14) Section 5.3 of Appendix B, Part II:

The proposed change deletes reference to a previously deleted section of Appendix A.

Section 5.3.1 of Appendix B references Appendix A Specification 6.9.1.8 which was deleted by Amendment No. 59. Specification 5.3.1 is no longer applicable and deletion is proposed. Consequently, renumbering of subsections of Specification 5.3 is also acceptable.

SUMMARY

Based on our review, the changes are administrative in nature. The changes provide consistency between Technical Specification sections, clarify Bases, correct inadvertent errors made by previous amendments, and delete or update superseded text. We find the changes to be acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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 this amendment.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 17, 1990

PRINCIPAL CONTRIBUTION:

J. D. Neighbors, PDI-1