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

July 5, 2001

Mr. A. Alan Blind
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - AMENDMENT RE: ADMINISTRATIVE CHANGES TO TECHNICAL SPECIFICATIONS (TAC NO. MA9033)

Dear Mr. Blind:

The Commission has issued the enclosed Amendment No. 216 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated February 14, 2000, as supplemented on May 3, 2001.

The amendment revises the TSs to correct various editorial errors and make other administrative changes. Specifically, the amendment makes administrative changes that revise: (a) Tables 3.6-1 and 4.4-1 to correct listing and editorial errors, (b) TS 3.8.B.10 to reflect the wording in 10 CFR 50.54(m)(2)(iv), (c) Figures 3.10-2 through 3.10-6 to remove these figures, (d) Table 4.1-1 to reflect change in level indication components, (e) TS 4.19.B and 6.14.1.1 to correct editorial errors, (f) TS 6.12.1 to reflect an organizational title change, and (g) TS 6.13.2 to correct a typographical error. In your May 3 letter, you requested that the proposed changes to TS 6.12.1 regarding references to the current sections of 10 CFR Part 20 be withdrawn.

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

Sincerely,

Patrick D. Milano, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures: 1. Amendment No. 216 to DPR-26
2. Safety Evaluation

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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 216
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated February 14, 2000, as supplemented on May 3, 2001, 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-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 216 , 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 5, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 216

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Replace the following pages 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
3.3-13	3.3-13
Table 3.6-1	Table 3.6-1
3.8-3	3.8-3
3.10-2	-----
3.10-3	-----
3.10-4	-----
3.10-5	-----
3.10-6	-----
Table 4.1-1 (Page 2 of 8)	Table 4.1-1 (Page 2 of 8)
Table 4.4-1 (Page 9 of 9)	Table 4.4-1 (Page 9 of 9)
4.19-1	4.19-1
6-12	6-12
6-13	6-13

The containment cooling function is provided by two independent systems: (1) fan-coolers and (2) containment spray. 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 95ºF)[12]. In the event of a Design Basis Accident, sufficient cooling to reduce containment pressure at a rate consistent with limiting offsite doses to acceptable values is provided by three fan-cooler units and one spray pump. These constitute the minimum safeguards and are capable of being operated on emergency power with one diesel generator inoperable.

The iodine removal function is provided by two independent operating trains of the containment spray system. In the event of a Design Basis Accident, one containment spray pump provides sufficient flow to remove air borne elemental and particulate iodine at a rate consistent with limiting offsite doses to acceptable values.

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 offsite power or operation of all emergency diesel generators.

The operability of the recirculation fluid pH control system ensures that there is sufficient trisodium phosphate (TSP) available in containment to guarantee a sump pH $\geq$ 7.0 during the recirculation phase of a postulated LOCA. This pH level is required to reduce the potential for chloride induced stress corrosion of austenitic stainless steel and assure the retention of iodine in the recirculating fluid. The specified amounts of TSP will result in a recirculation fluid pH between 7.0 and 9.5.

One of the five fan cooler units is permitted to be inoperable during power operation. This is an abnormal operating situation, in that the normal plant operating procedures require that an inoperable fan-cooler be repaired as soon as practical.

However, because of the difficulty of gaining access to make repairs, it is important on occasion to be able to operate temporarily without at least one fan-cooler. Compensation for this mode of operation is provided by the high degree of redundancy of containment cooling systems during a Design Basis Accident.

The Component Cooling System is different from the system discussed above in that the pumps are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident[6]. During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards[7]. With two operable component cooling pumps, 100% redundancy will be provided. A total of three operable component cooling pumps will provide 200% redundancy. The 14 day out of service period for the third component cooling pump is allowed since this is the 200% redundant pump.

Table 3.6-1

Non-Automatic Containment Isolation Valves Open Continuously
Or Intermittently for Plant Operation

3418	851A	SWN-44-5-A or B(1)	1814B
3419	850A	SWN-51-5(1)	1814C
		SWN-44-1-A or B(1)	
4136	851B	SWN-51-1(1)	5018
			5019
744	850B	SWN-44-2-A or B(1)	5020
		SWN-51-2(1)	
888A	859A	SWN-44-3-A or B(1)	5021
888B	859C	SWN-51-3(1)	5022
958			5023
959	3416	SWN-44-4-A or B(1)	5024
990D	3417	SWN-51-4(1)	5025
1870	5459	SWN-71-5-A or B(1)	E-2
743	753H	SWN-71-1-A or B(1)	E-1
732	753G	SWN-71-2-A or B(1)	E-3
885A	SWN-41-5-A or B(1)	SWN-71-3-A or B(1)	E-5
885B	SWN-42-5	SWN-71-4-A or B(1)	MW-17
			MW-17-1
205	SWN-43-5	SA-24	85C
226	SWN-41-1-A or B(1)	SA-24-1	85D
227	SWN-42-1	PCV-1111-1	95C
250A	SWN-43-1	PCV-1111-2	95D
4925	SWN-41-2-A or B(1)	580A	IIP-500
250B	SWN-42-2	580B	IIP-501
4926	SWN-43-2	UH-43	IIP-502
250C	SWN-41-3-A or B(1)	UH-44	IIP-503
4927	SWN-42-3		IIP-504
250D	SWN-43-3		IIP-505
4928	SWN-41-4-A or B (1)	1814A	IIP-506
869A	SWN-42-4		IIP-507
878A	SWN-43-4		
869B			

(1) Either A or B valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the B valves(s) in the SWN-44 series requires the codesignation of the SWN-51 valve(s) associated with the penetration(s) as an additional required containment isolation valves(s).

refueling crane for this event must be equal to or greater than the maximum load to be assumed by the refueling crane during the refueling operation. A thorough visual inspection of the refueling crane shall be made after the dead-load test and prior to fuel handling.

6. The fuel storage building charcoal filtration system must be operating whenever spent fuel movement is taking place within the spent fuel storage areas unless the spent fuel has had a continuous 35-day decay period.

7. Radiation levels in the spent fuel storage area shall be monitored continuously whenever spent fuel movement is taking place in that area.

8. The equipment door, or a closure plate that restricts direct air flow from the containment, shall be properly installed. In addition, at least one isolation valve shall be operable or locked closed in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

9. Radiation levels in containment shall be monitored continuously.

10. During alteration of the core (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling shall be present to directly supervise the activity and, during this time, this person shall not be assigned other duties.

11. The minimum water level above the top of the reactor pressure vessel flange shall be at least 23 feet (El. 92'0") whenever movement of spent fuel is taking place inside the containment.

12. If any of the conditions specified above cannot be met, suspend all operations under this specification (3.8.B). Suspension of operations shall not preclude completion of movement of the above components to a safe conservative position.

C. The following conditions are applicable to the spent fuel pit any time it contains irradiated fuel:

1. The spent fuel cask shall not be moved over <u>any</u> region of the spent fuel pit until the cask handling system has been reviewed by the Nuclear Regulatory Commission and found to be acceptable. Furthermore, any load in excess of the nominal weight of a spent fuel storage rack and associated handling tool shall

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

Channel Description	Check	Calibrate	Test	Remarks
8.a 6.9 kV Voltage	N.A.	R##	Q	
8.b 6.9 kV Frequency	N.A.	R##	Q (1) R# (2)	1) Underfrequency relay actuation only. 2) The full test including RCP breaker trip upon underfrequency relay actuation and reactor trip logic relay actuation upon tripping of the RCP breaker.
9. Analog Rod Position	S	R#	M	
10. Rod Position Bank Counters	S	N.A.	N.A.	With analog rod position
11. Steam Generator Level	S	R#	Q	Calibration of transmitters extended on a one time basis to 37 months.
12. Charging Flow	N.A.	R#	N.A.	
13. Residual Heat Removal Pump Flow	N.A.	R##	N.A.	Calibration of transmitters extended on a one time basis to 37 months.
14. Boric Acid Tank Level	W	R#	N.A.	
15. Refueling Water Storage Tank Level	W	Q	N.A.	
16. DELETED				
17. Volume Control Tank Level	N.A.	R##	N.A.	
18a. Containment Pressure	D	R#	Q	Wide Range
18b. Containment Pressure	S	R#	Q	Narrow Range

Table 4.4-1

Containment Isolation Valves

Valve No.	System[1]	Test Fluid[2]	Minimum Test Pressure (PSIG)
4399	Sample Return to Cont. Sump.	Water[4]	52
5132	" " "	Water[4]	52
IIP-500	22 S.G. Level	Gas	47
IIP-501	" " "	Gas	47
IIP-502	21 S.G. Level	Gas	47
IIP-503	" " "	Gas	47
IIP-504	Pressurizer Level	Gas	47
IIP-505	" "	Gas	47
IIP-506	Pressurizer Pressure	Gas	47
IIP-507	" "	Gas	47

Notes:

1. System in which valve is located.
2. Gas test fluid indicates either nitrogen or air as test medium.
3. Testable only when at cold shutdown.
4. Isolation Valve Seal Water System.
5. Sealed by Residual Heat Removal System fluid.
6. Sealed by Service Water System. Either A or B valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the B valve(s) in the SWN-44 series requires the codesignation of the SWN-51 valve(s) associated with the penetration(s) as an additional required containment isolation valve(s).
7. Sealed by Weld Channel and Penetration Pressurization System.

4.19 METEOROLOGICAL MONITORING SYSTEM

Applicability

This specification applies to the surveillance requirements for the meteorological monitoring system.

Objective

To verify operability of the meteorological monitoring system such that adequate measurement and documentation of meteorological conditions at the site can be effected.

Specifications

A. Each meteorological monitoring instrumentation channel shall be demonstrated operable by performance of the surveillance testing required by Table 4.19-1.

B. Meteorological data shall be summarized and reported as required for inclusion in the Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.6.

Basis

This specification assures the operability of the meteorological monitoring instrumentation and the collection of meteorological data at the plant site. This data is used for estimating potential radiation doses to the public resulting from routine or accidental releases of radioactive materials to the atmosphere. A meteorological data collection program, as described in this specification, is necessary to meet the requirements of 10 CFR 50.36a (a) (2), Appendix E to 10 CFR 50 and 10 CFR 51.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 As an acceptable alternative to the "control device" or "alarm signal" required by 10 CFR 20.203(c)(2):

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of Specification 6.12.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry to such areas and the keys shall be maintained under the administrative control of the Radiation Protection Manager and/or the Shift Manager on duty.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines), or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-26 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information,

    b. a determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes, and

    c. documentation of the fact that the change has been reviewed and found acceptable by the SNSC.

2. Shall become effective upon review and acceptance by the SNSC.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 The ODCM shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

    a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluation justifying the change(s),

    b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations, and

    c. documentation of the fact the change has been revised and found acceptable by the SNSC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated February 14, 2000, as supplemented on May 3, 2001, the Consolidated Edison Company of New York, Inc. (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 Technical Specifications (TSs). The requested changes would revise the TSs to correct various editorial errors and make other administrative changes. Specifically, the amendment makes administrative changes that revise: (a) Tables 3.6-1 and 4.4-1 to correct listing and editorial errors, (b) TS 3.8.B.10 to reflect the wording in 10 CFR 50.54(m)(2)(iv), (c) Figures 3.10-2 through 3.10-6 to remove these figures, (d) Table 4.1-1 to reflect change in level indication components, (e) TS 4.19.B and 6.14.1.1 to correct editorial errors, (f) TS 6.12.1 to reflect an organizational title change, and (g) TS 6.13.2 to correct a typographical error. The May 3 letter, requested that the proposed changes to TS 6.12.1 regarding references to the current sections of 10 CFR Part 20 be withdrawn. Further, the May 3, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

In its February 14, 2000, letter, the licensee proposed a number of administrative and editorial changes to correct errors in the facility TSs. These changes were described as:

2.1 Changes To Tables 3.6-1 and 4.4-1 to Correct Listing and Editorial Errors

In Table 3.6-1, "Non-Automatic Isolation Valves Open Continuously or Intermittently for Plant Operation," the licensee proposed to: (1) delete valves 990A and 990B, (2) change the valves grouping designation in Note 1 from SWN-77 series to SWN-71 series, and (3) add valves IIP-500 through IIP-507. In addition, the licensee proposed the addition of valves IIP-500 through IIP-507 to Table 4.4-1, "Containment Isolation Valves."

The licensee indicated that valves 990A and 990B are normally-closed isolation valves for the recirculation pump discharge sample line and are only designed to be opened intermittently after an accident to sample the recirculation fluid. Further, these motor-operated valves receive a close signal from the Phase A Containment Isolation Signal. Thus, valves 990A and 990B do not fit the category of non-automatic isolation valves open continuously or intermittently for plant operation and should not be listed on Table 3.6-1.

The licensee also proposed to correct the valve group series in Note (1) to Table 3.6-1 by changing SWN-77 to SWN-71. SWN-71 series is the correct group designation, and no SWN-77 series is listed in Table 3.6-1. Also, this change is consistent with a similar note in Table 4.4-1.

The licensee proposed the addition of valves IIP-500 through IIP-507 to Tables 3.6-1 and 4.4-1. These are normally closed, but intermittently operated, manually-operated isolation valves in the indication lines for steam generator level, pressurizer level, and pressurizer pressure. The licensee stated that these valves are listed in Updated Final Safety Analysis Report (UFSAR) Table 5.2-1 and the Inservice Testing Program, and thus, TS Tables 3.6-1 and 4.4-1 were being updated to show these valves.

In its letter of May 3, 2001, the licensee responded to the NRC's request for additional information of March 29, 2001, in which it clarified whether these valves were listed in the appropriate plant operating and emergency procedures. In this regard, the licensee stated that Station Operating Procedure SOP-10.6.4, "Operation and Control of Non-Automatic Containment Isolation Valves," provides the direction to ensure that these valves are closed during post-accident period. Further, the licensee stated that the appropriate controls during normal and emergency operations for these valves are included in Administrative Operating Instruction AOI-27.1.9, "Control Room Inaccessibility Safe Shutdown Control," and Check-off Lists COL-1.1, "Reactor Coolant Systems," and COL-10.6.2, "Containment Integrity."

On the basis of the above discussion, the NRC staff finds these changes to be acceptable.

2.2 Change to TS 3.8.B.10 to Reflect the Wording in 10 CFR 50.54(m)(2)(iv)

The current TS 3.8.B.10 specifies that "A licensed senior reactor operator shall be at the site and designated in charge of the operation whenever changes in core geometry are taking place." The licensee has proposed to revise TS 3.8.B.10 to read: "During alteration of the core (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling shall be present to directly supervise the activity and, during this time, this person shall not be assigned other duties."

The licensee noted it is required to comply with 10 CFR 50.54(m)(2)(iv) which states:

> "Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person."

Thus, the licensee proposes this change to remove any ambiguity that may have existed between the current TS 3.8.B.10 requirement and 10 CFR 50.54(m)(2)(iv).

The NRC staff finds that the proposed changes to TS 3.8.B.10 are in accordance with 10 CFR 50.54(m)(2)(iv) and are acceptable.

2.3 Deletion of Figures 3.10-2 through 3.10-6

The licensee proposes to delete TS Figures: (a) 3.10-2, "Hot Channel Factor Normalized Operating Envelop," (b) 3.10-3, "Rod Bank Insertion Limits," 3.10-5, "Target Band on Indicated Flux Difference as a Function of Operating Power Level," and 3.10-6, "Permissible Operating Band on Indicated Flux Difference as a Function of Burnup." The licensee also noted that Figure 3.10.4 was previously deleted by Amendment No. 152, dated June 26, 1990.

The licensee stated that in Amendment No. 194, dated June 27, 1997, TS Section 3.10, "Control Rod and Power Distribution Limits," was revised and references to the Figures 3.10-2 through 3.10-6 were eliminated. In lieu of these figures, the amended section referenced the Core Operating Limits Report (COLR). Thus, TS Figures 3.10-2 through 3.10-6 should have been deleted, but were inadvertently left in, when Amendment No. 194 was issued.

The NRC staff finds that the current TSs for the power distribution and rod insertion limits reference the COLR for the applicable limits. Thus, the staff finds that the change is acceptable since TS Figures 3.10-2 through 3.10-6 should have been deleted in Amendment 194.

2.4 Change to Table 4.1-1 to Reflect Change in Level Indication Components

In the Remarks column for Item 14, "Boric Acid Tank Level," in TS Table 4.1-1, "Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels," it is stated that the "Bubbler tube rodded during calibration." The licensee proposes to delete this statement.

The licensee stated that the original design for Boric Acid Storage Tanks included a nitrogen bubbler system to determine the level in each tank. However, this tank level instrumentation experienced frequent problems because of plugging in the bubbler tube from the crystallization of boric acid. To rectify the problem the bubbler tube had to be rodded out to clear the blockage. Under the original configuration, the statement, "Bubbler tube rodded during calibration," was included since if the bubbler tube was not rodded during the calibration of the level sensor, there could be an inaccurate calibration which could result in erroneous level indication.

Subsequently, the licensee has modified the boric acid storage tanks level instrumentation to a non-intrusive system having no contact with the process fluid. The modification removed the bubbler system and added a microwave transmitter/receiver sensor mounted above the top of each tank with a process seal to completely isolate the instrument from the tank contents.

The NRC staff finds that the remark to rod the bubbler tube during calibration is no longer applicable due to the modification replacing the level instrumentation. This remark should have been deleted when the modification was made. Thus, the NRC staff finds the change acceptable.

2.5 Change To Sections 4.19.B And 6.14.1.1 To Correct Editorial Errors

In Section 4.19.B and in Section 6.14.1.1, the licensee proposes to change "Semiannual Radioactive Effluent Release Report" to "Annual Radioactive Effluent Release Report."

The licensee stated that in Amendment No. 172, dated July 21, 1994, the submittal frequency of the Radioactive Effluent Release Report was changed from semiannual to annual. In Amendment 198, issued in August 12, 1998, TSs 3.9.A.2.c, 3.9.A.5.b, 3.9.B.2.c, 4.11.A.4, 4.11.B.3, 4.11.B.4, and Table 4.11-1 were revised to replace "Semiannual Radioactive Effluent Release Report" with "Annual Radioactive Effluent Release Report." Thus, Technical Specification Sections 4.19.B and 6.14.1.1 should have been included in the prior amendment.

The NRC staff finds that the frequency was previously approved by the NRC and the proposed change to TS 4.19.B and 6.14.1.1 correct items previously overlooked in the earlier amendment.

2.6 <u>Change to Section 6.12.1 to Reference the Current Sections of 10 CFR Part 20</u>

In a letter dated May 3, 2001, the licensee withdrew its request for changes to TS 6.12.1.

2.7 <u>Change to TS 6.12.1 to Reflect an Organizational Title Change</u>

The licensee proposed to change the title for the "Senior Watch Supervisor" to "Shift Manager" in TS Section 6.12.1. The licensee stated that the change was in title only and not in the responsibilities or functions performed by this individual. The licensee also noted that the change in title will be included in the next UFSAR update.

The NRC staff finds that this change acceptable since the functions and responsibilities of this position have not been revised.

2.8 <u>Change to Section 6.13.2 to Correct a Typographical Error</u>

In Section 6.13.2 change "DOR Guidelines of NUREG-0588" to "DOR Guidelines or NUREG-0588."

Technical Specification Section 6.13.1 states:

> "By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors 'Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors' (DOR Guidelines), of NUREG -0588, 'Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,' December, 1979. Copies of these documents are attached to Order of Modification of License No. DPR-26, dated October 24, 1980."

The licensee stated that the DOR Guidelines and NUREG-0588 are two separate documents. Therefore, the statement: "(DOR Guidelines), of NUREG-0588" should actually be "(DOR Guidelines), or NUREG-0588." The licensee believes that the typographical error occurred since the "r" key is just above and to the left of the "f" key on a keyboard.

The NRC staff finds that the proposed change corrects a typographical error and is acceptable.

2.9 Correction of Typographical Error in TS Basis Section 3.3

The licensee stated that the third from the last sentence on page 3.3-13 currently reads: "With two operable component cooling pumps, 100% redundancy will be provide." The word "provide" is grammatically incorrect and should read "provided." The staff does not object to this change since it corrects the tense of this verb and does not change the intent of this statement.

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

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on July 3, 2001 (66 FR 35282).

Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of these amendments will not have a significant effect on the quality of the human environment.

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: R. Clark
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Date: July 5, 2001