March 25, 2003

Mr. Michael Kansler
Senior Vice President and
Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: PRESSURIZER LEVEL LIMIT IN MODE 3 (TAC NO. MB5296)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 216 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 June 3, 2002, as supplemented on January 23, 2003.

The amendment revises TS 3.4.9, "Pressurizer," to increase the pressurizer water level limit when the plant is in MODE 3 (Hot Standby). The current pressurizer water level limit for Modes 1 and 2 remains unchanged. The amendment also revises TS 3.8.4, "DC Sources - Operating," to remove the notes that refer to the one-time amendment allowing the online replacement of station batteries 31 and 32. The notes were no longer applicable since the batteries have been replaced.

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,

/RA/

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

Docket No. 50-286

Enclosures: 1. Amendment No. 216 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

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Chief Operating Officer
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Package Number: ML
TSs: ML
Accession Number: ML

OFFICE	PDI-1\PM	PDI-1\LA	SRXB\SC	OGC	PDI-1\SC
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DATE	03/6/03	03/6/03	3/7/03	03/24/03	03/25/03

Official Record Copy

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Indian Point Nuclear Generating Unit No. 3

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DATED: March 25, 2003

AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE NO. DPR-64 INDIAN POINT UNIT 3

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ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 216
License No. DPR-64

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated June 3, 2002, as supplemented on January 23, 2003, 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. 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. This amendment shall be implemented only after completion of the required procedural changes as described in the licensee's letter dated January 23, 2003, and the NRC safety evaluation dated March 25, 2003.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 25, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 216

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

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
3.4.9-1	3.4.9-1
3.4.9-2	3.4.9-2
3.8.4-1	3.8.4-1
3.8.4-1.a	-----
3.8.4-2	3.8.4-2
3.8.4-3	3.8.4-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE NO. DPR-64

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated June 3, 2002, as supplemented on January 23, 2003, Entergy Nuclear Operations, Inc. (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs). The requested changes would revise TS 3.4.9, "Pressurizer," to increase the pressurizer water level limit when the plant is in Mode 3 (Hot Standby). The current pressurizer water level limit is applicable for Modes 1, 2, and 3, and will remain unchanged for Modes 1 and 2 (Power Operation and Startup, respectively). The proposed amendment would also revise TS 3.8.4, "DC Sources - Operating," to remove the notes that refer to the one-time amendment allowing the online replacement of station batteries 31 and 32. The notes are no longer applicable since the batteries have been replaced. The January 23 letter provided clarifying information that did not enlarge the scope of the original *Federal Register* notice or change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) staff finds that the licensee in its June 3, 2002, application identified the applicable regulatory requirements. The regulatory requirements and guidance which the staff considered in its review of the requested action are as follows:

1. Criterion 2 of 10 CFR 50.36(c)(2)(ii) requires, in part, that a TS limiting condition for operation (LCO) be established for a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that presents a challenge to the integrity of a fission product barrier. In this regard, pressurizer level is an initial condition for these analyses. Limiting the LCO maximum operating water level preserves the steam space for pressure control and ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients.

2. IP3 Final Safety Analysis Report (FSAR) Section 4.2.2, "Components," states that the pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and

contraction during normal load transients, and prevents the pressure in the reactor coolant system (RCS) from exceeding the design pressure.

3.0 TECHNICAL EVALUATION

3.1 MODE 3 Pressurizer Water Level Limit

3.1.1 Background

The pressurizer with a steam vapor space provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes. The pressurizer water level is maintained by a control system that varies level as a function of reactor coolant average temperature. The temperature-dependent water level provides sufficient water in the pressurizer to prevent the pressurizer from emptying on a reactor trip from 100% power, while maintaining a sufficient steam space to prevent overfilling the pressurizer with water following an overpressure event, such as loss of load at 100% power.

IP3 TS LCO 3.4.9 specifies the maximum pressurizer water level limit during MODES 1, 2, and 3 operations to ensure that the pressurizer is capable to establish and maintain pressure control for steady-state operation and to minimize the consequences of potential overpressure transients. This LCO pressurizer water level limit is assumed as the initial condition in the safety analyses performed from a critical reactor condition.

The RCS relies on the pressurizer safety valves (PSVs) for overpressure protection during MODES 1, 2, and 3 operations. Three Mile Island (TMI) Action Item II.D.1 in NUREG-0737, “Clarification of TMI Action Plan Requirements,” called for licensees to conduct testing to qualify the RCS relief and safety valves under expected operating conditions for design-basis transients and accidents. Because the IP3 PSVs are not qualified for water relief, the PSV overpressure protection operation should be limited to steam relief. Water relief through PSVs could result in a failure of the PSVs to re-close, causing a small-break loss-of-coolant accident due to the unisolable PSV opening. This would not comply with the acceptance criteria, stated in Section 15.5.1 - 15.5.2 of NUREG-0800, “Standard Review Plan,” that accidents of moderate frequency should not generate a more serious plant condition without other faults occurring independently. Therefore, the maximum pressurizer water level limit should be such that pressurizer overfill and PSV water relief are avoided during anticipated operational occurrences.

3.1.2 Proposed TS Change

The licensee proposed increasing the pressurizer water level limit from 58.3% to 90% for MODE 3 operation. The licensee stated that this higher water level limit in MODE 3 provides additional operational flexibility and efficiency with expected time savings of 1 to 2 hours for performing plant cooldown at or near the maximum allowable rate. This is because the higher water level compensates for reactor coolant contraction and allows greater flexibility for establishing boron concentration required for shutdown margin.

TS LCO 3.4.9 requires that, during MODES 1, 2, and 3 operations, the pressurizer shall be OPERABLE by complying with the specified limits for the pressurizer water level (i.e., pressurizer water level $\leq$ 58.3%) and heaters power supply. The licensee proposed to amend

TS 3.4.9 to allow for a higher pressurizer water level limit for MODE 3 compared to MODES 1 and 2. Specifically, the proposed amendment covers the following LCO, surveillance requirement (SR) and Basis.

- Revise LCO 3.4.9, item a, from “Pressurizer water level ≤ 58.3%” to “Actual pressurizer water level ≤ 58.3% in MODES 1 and 2 or ≤ 90% in MODE 3.”

- Revise SR 3.4.9.1 from “Verify pressurizer water level is ≤ 58.3%” to “Verify actual pressurizer water level is ≤ 58.3% in MODES 1 and 2 <u>or</u> ≤ 90% in MODE 3.”

- Revise Basis 3.4.9 to reflect the proposed changes.

3.1.3 Staff Evaluation

The LCO pressurizer water level limit is the initial condition in the safety analyses for overpressure events, such as loss of load and loss of normal feedwater. The limiting scenario for these accident analyses is with the reactor at full power. The proposed TS change to increase the maximum pressurizer water level limit from 58.3% to 90% applies to MODE 3 hot standby only. Therefore, this TS change does not affect the existing requirement for MODES 1 and 2; nor does it affect the validity of the initial condition assumption and the result of the design-basis safety analyses of transients and accidents initiated at power operating conditions.

The staff’s evaluation of the revised pressurizer water level limit is based on prevention of pressurizer overfill to avoid PSV water relief for events initiated from MODE 3 operation. The licensee contended that in MODE 3, a higher initial pressurizer level is acceptable because the potential magnitude of a pressurizer insurge due to thermal expansion of the reactor coolant is much smaller than that which would occur in MODE 1 with the plant at full power. However, the staff does not agree with the licensee’s argument of MODE 3 being bounded by MODE 1 since the proposed TS change would result in a smaller steam space in the pressurizer in MODE 3 than MODE 1. Since the overpressure events of loss of load or loss of normal feedwater are not applicable for MODE 3 operation, the staff requested the licensee to evaluate the impact of higher water level in MODE 3 on such events as inadvertent safety injection (SI) and malfunction of the chemical and volume control system (CVCS) on the overfill of pressurizer.

In its supplemental letter of January 23, 2003, the licensee provided an assessment of the time that the operator has to respond to a CVCS malfunction to avoid overfilling the pressurizer when the pressurizer level is set at 90 percent in MODE 3. In the event of a charging pump operating without letdown, the operator would have more than 20 minutes to respond to the condition. In the unlikely event that all three charging pumps are operating without letdown, the operator would have nearly 8 minutes to respond to the condition.

Since the purpose of the license amendment request is to support a specific and limited plant evolution (e.g., plant cooldown from MODE 3 to MODE 4), the licensee committed to implement administrative controls by requiring that a dedicated operator be assigned for operating and controlling the CVCS, including monitoring pressurizer level, whenever pressurizer level in MODE 3 is above the current TS limit of 58.3 percent. Specifically, the licensee made the following commitments to revise the TS BASES and the operating procedure for plant cooldown to implement this dedicated operator requirement.

a. Commitment No. NL-03-019-01:

> Revise Technical Specification Bases to specify a requirement that a dedicated operator is assigned for operating and controlling the chemical and volume control system, including monitoring pressurizer level, whenever pressurizer level in Mode 3 is above the existing Technical Specification limit of 58.3%.

The licensee committed to revise the TS Bases on or before the implementation date established when the License amendment is issued.

b. Commitment No. NL-03-019-02:

> Revise the operating procedure for plant cooldown from Mode 3 to Mode 4 to implement the requirement for a dedicated operator as stated in the revised Technical Specification Bases.

The licensee committed to revise the procedure prior to first use of the relaxed limit on pressurizer water level in MODE 3.

The licensee's administrative processes under its Commitment Management Program will ensure proper and timely implementation of these commitments. Therefore, the staff finds these commitments satisfactory.

The licensee also stated that the effect of an inadvertent SI on pressurizer water level is limited because IP3 is designed with low-head centrifugal SI pumps. The pressure-temperature limits for operating the plant in MODE 3 are established, in part, by the operating curves which ensure that the reactor pressure boundary fracture toughness requirement of Appendix G to 10 CFR Part 50 are satisfied. The SI pump nominal shutoff head of 1500 psig is bounded by the upper pressure limit curve. Therefore, in the event of an inadvertent SI actuation in MODE 3 with pressure above the SI pump shutoff head, no mass injection would occur and the pressurizer level would not be affected. In the event of an SI with RCS pressure below the pump shutoff head, the resulting mass injection would result in the RCS pressure increasing until it reaches the pump shutoff head. The licensee states that even assuming an "enhanced pump" (e.g., an SI pump with a higher shutoff head of 1600 psig), there is only a very narrow temperature range in MODE 3 (350 °F to approximately 352 °F) where the upper pressure limit for the Appendix G curve could be slightly exceeded by approximately 25 psi overpressure at 350 °F. However, because the centrifugal SI pump flow rate decreases with increasing back pressure, the inadvertent SI scenario is bounded by the CVCS malfunction, three-charging-pump maximum charging condition described above. Therefore, the dedicated operator monitoring pressurizer level would also be able to take appropriate action to limit the potential for exceeding the Appendix G pressure-temperature limits.

Based on the above evaluation and the licensee's commitment to revise the TS BASES and the operating procedure to require a dedicated operator be assigned for operating and controlling the CVCS, and monitoring pressurizer level whenever the pressurizer level in MODE 3 is above the existing TS limit of 58.3%, the staff concludes that there is reasonable assurance that the pressurizer overfill and water relief through the PSVs can be avoided in the events of CVCS malfunction or inadvertent SI actuation during MODE 3 operation with the pressurizer water level at 90% limit. In addition, even if the water relief occurs, the safety function of PSVs for overpressure protection of the RCS pressure boundary may still be met without failure to re-close. Therefore, the staff concludes that the proposed TS change to increase the pressurizer water level limit for MODE 3 from 58.3% to 90% is acceptable. It should be noted that the proposed change does not affect the limit of 58.3% for MODES 1 and 2 operation.

The licensee also proposed adding the word "actual" before the "pressurizer water level" in the current LCO and SR. This is because the analytical limit specified in the LCO includes an allowance of 7% for instrument uncertainty. The TS Bases states that "The required pressurizer level of ≤ 58.3% is the analytical limit used as initial condition in the accident analysis. An additional margin of approximately 7% must be allowed for instrument error (i.e., the indicated level should not exceed 51.3% for MODES 1 and 2, or 83% for MODE 3)." The NRC staff concludes that the addition of the word "actual" is editorial with the BASIS providing clarification of its meaning. Therefore, the staff finds the change acceptable.

3.2 Removal of One-Time Notes in LCO 3.8.4 and SRs 3.8.4.3 and 3.8.4.4

In IP3 TS LCO 3.8.4, Required Action B requires that when one DC electrical power subsystem inoperable, restore DC electrical power subsystem to OPERABLE status within 2 hours. The 2-hour completion time contains a footnote for an additional one-time allowed completion time. The footnote states that:

> On a one-time (per battery) only basis for Station batteries 31 and 32, the batteries may be inoperable for up to 10 days each, as necessary, to allow on-line replacement of the batteries. The time period during which this allowance may be exercised will end on May 31, 2002. The following additional requirements shall also be met to invoke this extended one-time allowed outage time: No risk significant planned maintenance or testing activities, which may impact AC or DC normal or emergency distribution sources or ESF systems, shall be performed during this replacement period.

SRs 3.8.4.3 and 3.8.4.4 also contain a footnote, which states:

> This battery surveillance may be performed on a one-time only basis during replacement of Station batteries 31 and 32 when the unit is in Mode 1, 2, 3, or 4 in order to support the one-time allowed outage time change of 10 days, as indicated in Section 3.8.4.B. This testing shall be done when the battery is disconnected from the DC bus.

These footnotes were added by License Amendment No. 208 dated September 19, 2001, to support the one-time replacement of station batteries 31 and 32 with the plant on-line. The licensee states that station battery replacement was successfully performed in February and March 2002, and the notes are no longer needed. Therefore, the licensee requested removal of these footnotes to restore the TS requirements to a condition previously approved by the NRC. Since the batteries have been replaced and the allowable period has passed, the staff finds that the change is administrative and is acceptable.

4.0 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.

5.0 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 and changes surveillance requirements. 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 (67 FR 45566). 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.

6.0 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 Contributor: Y. Hsii

Date: March 25, 2003