April 17, 2001

Mr. Michael Kansler
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: FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING (TAC NO. MB0178)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 206 to Facility Operating License No DPR-64 for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendment is issued in response to the Power Authority of the State of New York application transmitted by letter dated September 6, 2000, as supplemented on January 18, and April 2, 2001. On November 21, 2000, the operating license for IP3 was transferred to Entergy Nuclear Operations, Inc. (ENO). By letter dated January 26, 2001, ENO adopted requests associated with the operating license that were pending at the time of the license transfer.

The amendment revises Technical Specification 5.5.15 to allow a one time change in the 10 CFR Part 50, Appendix J, Type A test interval from the required 10 years to a test interval of 15 years. A co 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/R Laufer for

George F. Wunder, 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. 206 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

ACCESSION NO. ml011020315

*Safety Evaluation dated 03/28/01 was provided and no major changes were made.
** See previous concurrence

OFFICE	PM:PDI-1	LA:PDI-1	SC:SPLB*	SC:SPSB*	OGC**	(A)SC:PDI-1
NAME	RLaufer for GWunder	SLittle	GHubbard	MRubin	CMarco	MBanerjee
DATE	4/12/01	4/12/01	03/28/01	03/28/01	04/06/01	4/12/01

Official Record Copy

Indian Point Nuclear Generating Unit No. 3

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DATED: April 17, 2001

AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. DPR-64 INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

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G. Wunder
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M. Rubin
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cc: Plant Service list

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 206
License No. DPR-64

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

   A. The application for amendment by the Power Authority of the State of New York dated September 6, 2000, as supplemented on January 18, and April 2, 2001, as adopted by Entergy Nuclear Operations, Inc. (the licensee) pursuant to a letter dated January 26, 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-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. 206 , 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

/RA/

Maitri Banerjee, 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: April 17, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 206

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Replace the following pages of 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 Page	Insert Page
5.0-30	5.0-30
5.0-31	5.0-31

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 206 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 September 6, 2000, as supplemented on January 18, and April 2, 2001, the Power Authority of the State of New York (the licensee before November 21, 2000) and Entergy Nuclear Operations, Inc. (ENO, the licensee after November 21, 2000) proposed to amend the Technical Specifications (TSs) for the Indian Point Nuclear Generating Unit No. 3 (IP3). The proposed amendment would allow a one time change in the 10 CFR Part 50, Appendix J, Type A test interval from the required 10 years to a test interval of 15 years.

The January 18, and April 2, 2001, supplements provided clarifying information that did not expand the application beyond the scope of the initial *Federal Register* notice, nor change the staff's initial proposed no significant hazards consideration determination.

On November 21, 2000, the operating license for IP3 was transferred to ENO. By letter dated January 26, 2001, ENO adopted requests associated with the operating license that were pending at the time of the license transfer.

2.0 BACKGROUND

10 CFR Part 50, Appendix J, Option B requires that a Type A test (containment ILRT) be conducted at a periodic interval based on historical performance of the overall containment system. IP3 TS 5.5.15 requires that a program be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. TS 5.5.15 further requires that this program be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. RG 1.163 references Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months under certain circumstances.

The two most recent Type A tests at IP3 have been successful, so their current interval requirement is 10 years.

The licensee has requested a revision to TS 5.5.15 which would allow an exception to the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS change would allow the first Type A test performed after the December 2, 1990, Type A test to be performed no later than December 1, 2005. This would make the interval 15 years since the last test.

3.0 EVALUATION

The licensee's January 18, 2001, supplemental submittal, provided a risk impact assessment of extending the Type A test interval to 15 years. In performing the risk assessment, the licensee followed the guidelines of NEI 94-01, the methodology used in EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and the guidelines of RG 1.174,"An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test interval from 3 in 10 years to 1 in 10 years, would increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of leakage, in percent of person-rem/year, for the pressurized-water reactor representative plant was estimated to increase from .032 percent to .035 percent. This confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 per 10 years to 1 per 10 years leads to an imperceptible increase in risk.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The staff considers the licensee's assessment an improvement of the EPRI study because the leakage from sequences that have the potential to result in large releases if a pre-existing leak were present were quantified. Since the Option B rulemaking in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF the relevant criterion is the change in LERF which the licensee estimated. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the increase in the conditional containment failure probability which helps to ensure that the defense-in-depth philosophy is maintained.

The licensee examined plant-specific accident sequences from their Individual Plant Examination. The following sequences were considered in the assessment:

- Core damage sequences in which the containment remains intact initially and in the long term.
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach, or steam generator manway leakage.
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left 'opened' following a plant post-maintenance test. For example, valve failing to close following a valve stroke test.

Accident sequences involving containment failure induced by severe accident phenomena, containment bypassed, large containment isolation failures, and small containment isolation 'failure-to-seal' events were not accounted for in this evaluation. These sequences are impacted by changes in Type B and C test intervals, not changes in the Type A test interval.

The steps taken by the licensee to perform the risk assessment are as follows:

- Quantified the base-line risk in terms of frequency per reactor year for each of the eight accident classes evaluated in EPRI TR-104285.
- Developed plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes.
- Evaluated the risk impact of extending the Type A test interval from 10 to 15 years.
- Determined the change in risk in terms of LERF in accordance with RG 1.174.

Determining the change in risk in terms of LERF involves the potential that a core damage event that normally would result in only a small radioactive release from containment could in fact result in a large release due to failure to detect a pre-existing leak during the extension period. The licensee designated these sequences as Class 3B sequences and estimated a frequency of 1.02×10^{-6}/year, based on a 10-year test interval. The licensee then used the EPRI methodology to estimate the impact of the Type A test interval on the leakage probability. Extending the Type A test interval from 10 to 15 years increases the average time that a leak detectable only by a Type A test goes undetected from 60 to 90 months. For a 15-year interval there is a 15 percent increase in the overall probability of leakage versus 10 percent for a 10-year interval. Thus, increasing the Type A test interval from 10 years to 15 years results in a 5 percent increase in the overall probability of leakage. Multiplying the above LERF frequency (1.02×10^{-6}/year) by the increase in overall probability of leakage (0.05) gives an increase in LERF of 5.1×10^{-8}/year. If the risk increase is measured from the original 3 in 10-year interval, the increase in LERF is 1.1×10^{-7}/year.

The following conclusions can be drawn from the licensee's risk assessment associated with extending the Type A test frequency:

1. The risk assessment predicted a slight increase in risk when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk (person-rem/year within 50 miles) was found to be 0.048 percent. The increase in the total integrated plant risk, given the change from a 3 in 10-year test interval to a 15-year test interval, was found to be 0.43 percent. This is just slightly greater than the range of risk increase, 0.02 to 0.14 percent, estimated in NUREG-1493 when going from a 3 in 10-year test interval to a 10-year interval. NUREG-1493 concluded that a reduction in the frequency of tests from 3 per 10 years to 1 per 10 years leads to an imperceptible increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small.

2. RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in CDF less than 10^{-6} per reactor year and increases in LERF less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from 1 in 10 years to 1 in 15 years is estimated to be 5.1×10^{-8}/year. The increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 1.1×10^{-7}/year. Increasing the Type A interval to 15 years is considered to be a very small change in LERF.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in the conditional containment failure probability was estimated to be 0.1 percent for the proposed change and 0.4 percent for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed change.

Based on these conclusions, the staff finds that the increase in predicted risk due to the proposed change is within the acceptance criteria while maintaining the defense-in-depth philosophy of RG 1.174 and is, therefore, acceptable.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations, 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment 3 times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

Based on the foregoing evaluation, the staff finds that the interval until the next Type A test at IP3 may be extended to 15 years, and that the proposed changes to TS 5.5.15 are acceptable.

4.0 PUBLIC COMMENT

On January 24, 2001, the NRC published a “Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing,” in the *Federal Register* (66 FR 7665). On February 26, 2001, an individual e-mailed comments on the proposed amendment to the NRC’s Office of Public Affairs (OPA). These comments are addressed below.

4.1 Comments

The following e-mail was received by OPA:

> Good morning. I read the Public Comment notice on Indian Point Unit 3's request for a "one-time extension" to 15 years between integrated leakage rate tests of the primary reactor containment (isn't this the plant right next door to the one that just shutdown because of tube leaks in the S/G?). I'm now told that several other utilities are planning on jumping on the bandwagon (Crystal River, TMI, the rest of the Entergy plants...).
>
> Is this a form of rule-making by exemption request? Didn't the Commission and the industry go through a revision to 10CFR50 Appendix J in 1995? From my understanding, the same "science" was available then as now, and intervals longer than 10 years (including 15, 20 years) were considered and rejected. Less thought and attention has been put into this area since. In fact since that rulemaking, a relatively new Commonwealth Edison plant "failed" the ILRT because of a cracked liner weld, the North Anna and Brunswick plants experienced incidents of through-wall corrosion of their primary containment liners.
>
> I know there are new inspection programs in place under sections IWE/IWL of the boiler-pressure vessel code, but those are visual inspection programs. In the face of increasing evidence that the reactor containments are NOT holding up as well as originally thought, do you think it wise to replace actual tests of the containments' integrity with simple visual inspections, and over-turn the logic behind the rulemaking of 1995? Some may say the recent failures indicate the new, less costly inspection programs are working. I believe the pronouncement pre-mature. Do you think the North Anna problem would have been discovered had it been up in the dome region?
> From my understanding the initial inspection revealed a "coating" problem, and it was the coating expert's inspection that revealed the through-wall corrosion problem.
>
> It seems the NRC is asking the public to accept the "warranty" of these plant's safety when there [their] originally scheduled "oil change" was every 3-4 years, then changed to once every ten years, and before the first round of ten-years maintenance can be completed (let alone evaluated - the Commission doesn't even receive the reports or observe these tests anymore), we're asked to accept a 15 year service interval. Why?

4.2 NRC Response

The commenter asked if the proposed license amendment amounted to rulemaking by exemption. As part of the 1995 change to 10 CFR Part 50 Appendix J, the ILRT interval was removed from the rule under Option B. When IP3 adopted Appendix J, Option B, the ILRT interval was incorporated by reference in its TS and must be changed through a license amendment. Because the interval is not specified in the rule under Appendix J, Option B, and since IP3 has adopted Option B, no exemption is required. License amendments are granted only after review of individual applications; any other licensee requiring a TS change to extend their ILRT interval will have to apply for an amendment and that amendment request will be evaluated by the staff.

The commenter noted that in 1995 the Commission was not willing to approve an ILRT interval longer than 10 years. This is true; however, since 1995 the staff has gained additional insight in using probabilistic risk assessment (PRA) to support decisions to modify an individual plant's licensing basis. In 1998, the Commission issued RG 1.174 to provide guidance on an approach that is acceptable to the staff. In using PRA to evaluate changes to a plant's licensing basis, the staff reviews how the proposed change affects both the CDF and the LERF. In the case of a change to the ILRT interval, there is no effect on the CDF. In the case of IP3, extending the ILRT interval from 10 years to 15 years increases the LERF by an estimated 5.1E-8 (5.1×10^{-8}) per year. This estimate is based on an analysis of data from 144 ILRTs performed at U.S. reactors from 1987 to 1993. RG 1.174 defines very small changes in the risk-acceptance guidelines as those changes that result in increases in CDF less than 1E-6 per reactor year and changes in LERF of less than 1E-7 per reactor year. The extension of the ILRT frequency at IP3 from once in 10 years to once in 15 years meets the definition of a change involving a very small increase in risk. In the context of the IP3 amendment request, the staff considers this small increase in risk to be acceptable.

The commenter cited three examples to demonstrate that reactor containments are not holding up as well as it was originally thought they would. In one case, the degraded containment was detected through the ILRT. The staff was aware of this instance and considered it when making the 1995 rule change. As the commenter noted in the e-mail, licensees must perform inspections required by Sections IWE and IWL of the ASME Code. The commenter also noted in the e-mail that two of the instances cited were detected by visual inspection, not by an ILRT. The NRC regulations, 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment, including the dome, 3 times every 10 years. This requirement will not be changed as a result of the extended ILRT interval. Furthermore, Appendix J, Type B tests performed on containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

Because historical data shows that the probability of an ILRT failure is low, because the PRA for IP3 shows a very small increase in LERF, and because the licensee is required to perform visual inspections in accordance with the ASME Code, the staff has concluded that the one time ILRT extension for IP3 is acceptable. The staff believes that a 15-year interval for the ILRT at IP3 is adequate to protect public health and safety.

5.0 STATE CONSULTATION

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

6.0 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 (66 FR 7665). 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.

7.0 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: J. Pulsipher
M. Snodderly

Date: April 17, 2001