Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

November 5, 2001
Re: Indian Point Unit No. 2
Docket No. 50-247
NL01-129

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Response to Request for Additional Information Regarding Steam Generator Surveillance Requirements, Indian Point Nuclear Generating Unit No. 2 (TAC No. MB0770)

References:
1. Consolidated Edison letter (NL00-147) to NRC, "Proposed Technical Specification Amendment – Changes to Primary to Secondary Leakage Limits and Steam Generator Tube Inservice Surveillance Requirements," dated December 11, 2000
2. NRC letter (RA01-219) to Consolidated Edison, "Request for Additional Information Regarding Steam Generator Surveillance Requirements, Indian Point Nuclear Generating Unit No. 2 (TAC No. MB0770)," dated September 5, 2001
3. Consolidated Edison letter (NL00-135) to NRC, "Response to Request for Additional Information – Generic Letter 97-05, Steam Generator Tube Inspection Techniques, for Indian Point Unit 2. (TAC No. MA0468)," dated November 7, 2000
4. NRC letter (RA01-011) to Consolidated Edison, "Completion of Licensing Action for Generic Letter 97-05 – Indian Point Nuclear Generating Unit No. 2 (TAC No. MA0468)," dated January 8, 2001

By letter dated December 11, 2000 (Ref. 1), Consolidated Edison (the former licensee) submitted an application for an amendment to the Technical Specifications (TS) for Indian Point Unit No. 2 (IP2). The proposed amendment requested revised steam generator primary to secondary leakage limits and steam generator tube inservice surveillance requirements. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed this submittal, determined that additional information was required to complete its review, and requested that additional information in its letter of September 5, 2001 (Ref. 2). This letter submits the Entergy Nuclear Operations, Inc. (ENO – the current licensee) response to the NRC's request for additional information. The specific responses to the three questions are in Attachment 1.

The purpose of the license amendment request was to delete provisions (e.g., criteria for sleeved tubes) in the existing TS that were no longer applicable to the replacement steam generators installed

A079

in 2000. In Ref. 1, Consolidated Edison indicated its intention to request a license amendment request to adopt the industry standard regulatory framework for steam generators, when approved by the NRC, modeled on the Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines." This would have updated the programmatic aspects of the steam generator TS. At the time, early approval of the industry standard regulatory framework was expected. ENO also intends to request a license amendment request to adopt the industry standard regulatory framework for steam generators, when approved by the NRC.

Regulatory control of steam generator inspection requirements is maintained even if the industry standard TS are not issued by the next scheduled IP2 steam generator inspection in the Fall of 2002. Current TS 4.13.C.1 requires ENO to submit the steam generator inspection plan for NRC review at least 60 days prior to the inspection. In addition, the IP2 guidelines for steam generator inspection and assessment were described to the NRC in Ref. 3. They are in accordance with industry (NEI and EPRI) standards and were found acceptable to satisfy Generic Letter 97-05 by the NRC in Ref. 4. Thus, degraded tubes (as defined in TS 4.13.A.1.d) are considered acceptable for continued service only if they meet the more restrictive of the prescriptive requirements of TS 4.13.B.1 or the required industry standard operational assessment for the next period that conservatively demonstrates continued structural integrity. Additionally, ENO reaffirms the commitment made in Ref. 3 to submit to the NRC the applicable plugging criteria, the details of the inspection method, and the technical bases for the decision prior to leaving tubes with degradation in service.

This letter contains two new commitments and a reaffirmed commitment, both identified in Attachment 2.

Should you or your staff have any questions regarding this submittal, please contact Mr. John F. McCann, Manager, Nuclear Safety and Licensing at (914) 734-5074.

Very truly yours,

Fred Dacimo
Vice President – Operations
Indian Point 2

Attachments

cc:

Mr. Hubert J. Miller
Regional Administrator-Region I
US Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Patrick D. Milano, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
US Nuclear Regulatory Commission
Mail Stop O-8-2C
Washington, DC 20555

NRC Senior Resident Inspector
US Nuclear Regulatory Commission
PO Box 38
Buchanan, NY 10511

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. Paul Eddy
NYS Department of Public Service
3 Empire Plaza
Albany, NY 12223

Mr. William F. Valentino, President
NYS ERDA
Corporate Plaza West
286 Washington Ave. Extension
Albany, NY 12223-6399

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of ENTERGY NUCLEAR OPERATIONS, INC. Indian Point Nuclear Generating Unit No. 2	)))	Docket No. 50-247

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission (NRC), Entergy Nuclear Operations, Inc., as holder of Facility Operating License No. DPR-26, hereby submits additional information in support of the December 11, 2000 application for amendment of the Technical Specifications contained in Appendix A of this license. The specific additional information is set forth in Attachment 1.

As required by 10CFR50.91(b)(1), a copy of this submittal has been provided to the appropriate New York State official designated to receive such amendments.

BY: ______________________
Fred Dacimo
Vice President – Operations
Indian Point 2

Subscribed and sworn to
before me this 5 day
November, 2001.

Ersilia A. Amanna
Notary Public

ERSILIA A. AMANNA
Notary Public, State of New York
No. 01AM6039669
Qualified in Westchester County
Commission Expires March 20, 2002

ATTACHMENT 1 TO NL 01-129

Response to Request for Additional Information Regarding Proposed Steam Generator Surveillance Requirements Technical Specifications

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

Response to Request for Additional Information
Steam Generator Surveillance Requirements Techncial Specification Changes
Indian Point Unit 2

Request No. 1

The Bases section of TS 4.13 states that the steam generator tube burst and collapse tests demonstrated that tubes having wall thickness of not less that 0.025 inch have adequate margins of safety. In the December 11 letter, the licensee did not indicate what guidelines, if any, were used to obtain these results. Provide a summary of the analysis, including the loads considered, tube support plate conditions (locked or unlocked), and a list of guidelines (e.g., NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes"), used to support the proposed minimum wall thickness and plugging criteria.

Response to Request No. 1

The plugging limit of 40% in TS 4.13.B.1 and the discussion in its Bases have been unchanged since they were prescriptively added by License Amendment 31.

Both the original Model 44 steam generators and the replacement Model 44F steam generators have tubes with the same nominal OD (0.875 in.) and wall thickness (0.050 in.) In addition, the materials have similar strength properties. Both are Alloy 600 (SB-163) but the original steam generator tubes were mill annealed, while the replacement steam generator tubes are thermally treated for increased margin with respect to stress corrosion performance. Prior to startup from the steam generator replacement outage, a vendor structural evaluation of Model 44F steam generators tubes was performed. Based on that evaluation and the similarity between the original and the replacement steam generators, it is not expected that the plugging limit and the discussion in the TS Bases will change when the detailed calculation of plugging criteria for the IP2 Model 44F steam generators is performed in accordance with current methodology prior to the first inservice inspection of the replacement steam generators during the next refueling outage.

The proposed amendment requested revised steam generator primary to secondary leakage limits and steam generator tube inservice surveillance requirements. The purpose of the license amendment request was to delete provisions (e.g., criteria for sleeved tubes) in the existing TS that were no longer applicable to the replacement steam generators installed in 2000. In Ref. 1, Consolidated Edison indicated its intention to request a license amendment to adopt the industry standard regulatory framework for steam generators, when approved by the NRC, modeled on the Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines." This would have updated the programmatic aspects of the steam generator TS. At the time, early approval of the industry standard regulatory framework was expected. ENO also intends to request a license amendment to adopt the industry standard regulatory framework for steam generators, when approved by the NRC.

Regulatory control of steam generator inspection requirements is maintained even if the industry standard TS are not issued by the next scheduled IP2 steam generator inspection in the Fall of 2002. Current TS 4.13.C.1 requires ENO to submit the steam generator inspection plan for NRC review at least 60 days prior to the inspection. The steam generator inspection plan will contain the plugging criteria developed with the latest methodology that can be used for the outage. In addition, the IP2 guidelines for steam generator inspection and assessment were described to the NRC in Ref. 3. They are in accordance with industry (NEI and EPRI) standards and were found acceptable to satisfy Generic Letter 97-05 by the NRC in Ref. 4. Thus, degraded tubes (as defined in TS 4.13.A.1.d) will be considered acceptable for continued service only if they meet the more restrictive of the prescriptive requirements of TS 4.13.B.1 or the required industry standard operational assessment for the next period that conservatively demonstrates continued structural integrity. Additionally, ENO reaffirms the commitment made in Ref. 4 to submit to the NRC the applicable plugging criteria, the details of the inspection method, and the technical bases for the decision prior to leaving degraded tubes in service at IP2.

Request No. 2

The Basis section of TS 4.13 states that an essentially 100% tube examination was performed on each tube in each steam generator prior to service. Although the licensee indicated in the December 11 letter that no baseline imperfections were identified, it did not state what guidelines, if any, were used to conclude that the tubes were free of imperfections. In addition, the licensee did not indicate what guidelines would be used for the first inservice inspection (ISI). Provide a reference to the guidelines used for the preservice inspection and the first ISI. If the licensee intends to depart from any part of the guidelines, provide a summary of those differences.

Response to Request No. 2

Baseline Inspection Guidelines

The proposed TS marked-up pages contained in Attachment 2 to the December 11 letter (Ref. 1) included an existing and unchanged TS Bases statement that "No significant baseline imperfections were identified." Although anomalous NDE signals were recorded during the preservice inspection (PSI) of the replacement steam generators, the number and type of these signals were typical for new steam generators. The analysis of each of these signals showed that they were benign. The identification and tracking of such anomalous signals is common practice so that the resolution of future reporting and evaluation of these signals is based solely on changes from the PSI. In NRC's Indian Point 2 Inspection Report 05000247/2000-13, dated December 18, 2000, the adequacy of the eddy current testing of the replacement steam generators was evaluated and were noted to be "well planned and conducted." No findings were identified during the NRC's inspection of the eddy current testing.

Analysis of the inspection data was performed in accordance with the "IP2 Analyst Reference Book for the Replacement Steam Generators," dated Summer 2000, which was prepared in accordance with the EPRI PWR Steam Generator Examination Guidelines, Revision 5. The "Analyst Reference

Book," also known in the industry as "Steam Generator NDE Data Analyst Guidelines," was developed prior to the SG inspection and provides detailed and site specific instructions pertaining to the analysis of SG eddy current data.

First Inspection Guidelines

During the first inspection (ISI) after the first full cycle of operation with the replacement steam generators, IP2 will perform a 100% full length tube examination using general purpose eddy current probes in all four steam generators as required by the EPRI Steam Generator Examination Guidelines (TR-107569). Prior to the inspection, a Degradation Assessment will be prepared in accordance with the IP2's Administrative Steam Generator Program Plan, SAO-180, which implements the industry guidelines for a Steam Generator Program as described in NEI 97-06. Furthermore, all inspections will be performed in accordance with the EPRI Steam Generator Examination Guidelines. If any deviations from NEI 97-06 or the EPRI Guidelines referenced are considered appropriate, they will be evaluated and documented in accordance with the EPRI Steam Generator Management Program Administrative Procedures, Revision 0, dated April 2001 (EPRI Report 1000776).

Current TS 4.13.C.1 requires ENO to submit the IP2 steam generator inspection plan for NRC review at least 60 days prior to the inspection. The first inspection is scheduled for the refueling outage in the Fall of 2002. The above does not constitute the submittal required by the TS. Any changes and deviations from the above referenced guidelines will be documented in the required submittal when the outage plans are finalized.

Request No. 3

Provide a summary of the assessment completed to confirm that the issues covered by NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," are not applicable to the replacement steam generators.

Response to Request No. 3

NRC Bulletin 88-02 was reviewed while preparing the Consolidated Edison Modification Package for the Westinghouse Model 44F replacement steam generators (Modification Number FMX-00-52429-D). The bulletin identified that three conditions were necessary to produce a rapidly propagating crack such as occurred at North Anna. The result of review of these conditions as related to the replacement steam generators is summarized below:

1) Denting at the upper support plate

   The replacement steam generators, Westinghouse Model 44F, are designed and fabricated with stainless steel support plates to resolve the tube denting problems that were associated with earlier Westinghouse steam generators that had been fabricated with carbon steel support plates. The Westinghouse Model 44F replacement steam generators installed at Indian Point 2 were

inspected by visual and eddy current methods during fabrication and just prior to installation. No such denting was noted, as would be expected for new steam generators. While future denting is not anticipated to occur, planned eddy current tube inspection and secondary side inspection programs will detect this degradation were it to occur. In such event, an appropriate analysis would be conducted and the issue would be handled in accordance with the IP2 Steam Generator Program.

2) A fluid-elastic stability ratio approaching that for the tube that ruptured at North Anna.

   Bulletin 88-02 identifies a fluid-elastic stability ratio of equal to or exceeding 1.0, beyond which the displacement response of a tube increases rapidly. Westinghouse performed a Model 44F steam generator replacement unit tube analysis (Westinghouse WTP-EM-79-089, Proprietary). Specifically evaluated was flow induced tube vibration. It was concluded that the Westinghouse Model 44F steam generator fluid-elastic stability ratio for 1% damping is 0.7. It was further concluded that the contribution to tube fatigue caused by flow-induced tube vibration is negligible.

3) Absence of effective anti-vibration bar support

   The Westinghouse Model 44F four peripheral anti-vibration bars were relocated (as compared to the original Westinghouse Model 44 locations), and two additional anti-vibration bars were added to the interior of the tube pattern. Thus, the tubes are considered to be adequately supported to prevent flow induced vibration of the type responsible for the North Anna tube rupture event.

In summary none of conditions identified in NRC Bulletin No. 88-02 that would indicate susceptibility to tube rupture due to a rapidly propagating fatigue crack are present in the replacement steam generators at IP2. Thus, NRC Bulletin 88-02 is applicable 'for information only' since IP2 possesses Westinghouse Model 44F replacement steam generators.

ATTACHMENT 2 TO NL 01-129

COMMITMENTS

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

No.	Commitment	Implementation
1.	A degraded tube (as defined by TS 4.13.A.1.d) shall be considered acceptable for continued service only if it meets the more restrictive of the requirements of TS 4.13.B.1 or the required industry standard (NEI 97-06) operational assessment for the next period that conservatively demonstrates continued structural integrity.	Whenever a degraded tube is found.
2.	The steam generator inspection plan submitted in accordance with Current TS 4.13.C.1 at least 60 days prior to the Fall 2002 inspection will contain the plugging criteria and a description of the methodology used for its development.	At least 60 days prior to the Fall 2002 inspection.
3.	Prior to leaving a degraded tube (as defined by TS 4.13.A.1.d) in service, the applicable plugging criteria, the details of the inspection method, and the technical bases for the decision will be submitted to the NRC.	Reaffirmed.