Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc
Indian Point Energy Center
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August 21, 2002

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-02-112

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

Subject: Proposed Steam Generator Examination Program – 2002 Refueling Outage (2R15)

Reference:
1) Entergy letter to NRC, "Response to Request for Additional Information Regarding Steam Generator Surveillance Requirements, Indian Point Nuclear Generating Unit 2 (TAC No. MB0770)," dated November 5, 2001
2) Entergy letter to NRC, "Steam Generator Inspection Commitment (TAC No. MB0770)," dated December 7, 2001
3) NRC letter 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

Pursuant to the requirements of Indian Point Unit 2 Technical Specification 4.13.C.1, Entergy Nuclear Operations, Inc. (ENO) hereby submits its proposed steam generator examination program (Attachment 1) to be conducted during the 2002 refueling outage (2R15). This examination program was developed in accordance with industry guidelines defined in Nuclear Energy Institute (NEI) 97-06: "Steam Generator Program Guidelines," Rev. 1, and Electric Power Research Institute (EPRI) Report TR-109569-V1R5: "PWR Steam Generator Examination Guidelines," Rev. 5.

By letter dated November 5, 2001 (Reference 1), ENO provided responses to the staff's request for additional information relative to a proposed amendment regarding revised steam generator primary to secondary leakage limits, and steam generator tube inservice surveillance requirements. By Reference 1, ENO committed to include with its steam generator examination plan submitted in accordance with Technical Specification 4.13.C.1, the plugging criteria, and a description of the methodology used for its development. Subsequently, in a telephone call on November 28, 2001, the NRC further requested that the subject submittal include certain specific items. Reference 2 summarized the specific items to be addressed.

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ENO contracted Westinghouse Electric Corporation to perform a Regulatory Guide 1.121 analysis to determine steam generator tube structural limits. The results from this analysis are provided in WCAP-15909-P, "Regulatory Guide 1.121 Analysis for the Indian Point Unit 2 Steam Generators," dated August 2002 (Proprietary). The structural analysis addresses all anticipated degradation forms except circumferential cracking. With the exception of the locked tube/support plate condition, various tube loading scenarios were evaluated. The locked tube/support plate condition is not expected to develop at IP-2 due to the new steam generator's broached stainless steel support plates. The structural limits established by the analysis substantiate the tube acceptance criteria identified in the Technical Specifications. Based upon the completion of this analysis, ENO is providing a supplemental response (Attachment 2) to Request No. 1 of the NRC's Request for Additional Information Regarding Steam Generator Surveillance Requirements dated September 5, 2001 (Reference 3).

Accordingly, ENO is enclosing the following additional items:

1. One (1) copy of WCAP-15909-P, "Regulatory Guide 1.121 Analysis for the Indian Point Unit 2 Steam Generators," dated August 2002 (Proprietary).
2. One (1) copy of WCAP-15909-NP, "Regulatory Guide 1.121 Analysis for the Indian Point Unit 2 Steam Generators," dated August 2002 (Non-Proprietary).

Also enclosed are a Westinghouse authorization letter dated August 6, 2002 (CAW-02-1544), accompanying affidavit, Proprietary Information Notice, and Copyright Notice. As Item 1 contains information proprietary to Westinghouse Electric Corporation, it is accompanied by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly, it is respectfully requested that the information that is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.790 of the Commission's regulations.

Correspondence with respect to the copyright on proprietary aspects of the items listed above or the supporting affidavit should reference CAW-02-1544 and should be addressed to H. A. Sepp, Regulatory and Licensing Engineering, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

No new regulatory commitments are being made by ENO in this correspondence.

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

Very truly yours,

Fred Dacimo
Vice President – Operations
Indian Point 2

Attachment:

1. Proposed Steam Generator Examination Program 2002 Refueling Outage
2. Supplemental Response to Request No. 1 for NRC Request for Additional Information Regarding Steam Generator Surveillance Requirements, Indian Point Unit No. 2 (TAC No. MB0770), dated September 5, 2001.
3. WCAP-15909-P, "Regulatory Guide 1.121 Analysis for the Indian Point Unit 2 Steam Generators," dated August 2002 (Proprietary).
4. WCAP-15909-NP, "Regulatory Guide 1.121 Analysis for the Indian Point Unit 2 Steam Generators," dated August 2002 (Non-Proprietary).
5. Westinghouse letter dated August 6, 2002 (CAW-02-1544)

C: Mr. Hubert J. Miller (w/att)
Regional Administrator-Region I
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ATTACHMENT 1 TO NL-02-112

Proposed Steam Generator
Examination Program
2002 Refueling Outage

Entergy Nuclear Operations, Inc.
Indian Point Unit No. 2
Docket No. 50-247

Indian Point 2

Proposed Steam Generator Examination Program

2002 Refueling Outage

A comprehensive steam generator (SG) examination program has been developed for implementation at Indian Point 2 (IP-2) for the first inservice inspection of the replacement steam generators during the Fall 2002 refueling outage (2R15). The examination program and methods comply with the IP-2 Technical Specifications and IP-2 Station Administrative Order (SAO)-180: "Administrative Steam Generator Program Plan." The steam generator examination program was developed in accordance with industry guidelines defined by the Nuclear Energy Institute (NEI) 97-06: "Steam Generator Program Guidelines," Rev. 1, and Electric Power Research Institute (EPRI) Report TR-109569-V1R5: "PWR Steam Generator Examination Guidelines," Rev. 5.

The steam generator examination program incorporates both primary and secondary-side inspections. The scope of the inspections to be performed and the methods employed are detailed in Westinghouse Electric Nuclear Services Report No. SG-SGDA-02-029: "Steam Generator Degradation Assessment for Indian Point Unit 2 RFO 15," Rev. 0. (Reference 1). The degradation assessment defines an integrated plan for the detection, quantification and assessment of degradation of both primary and secondary side steam generator components that could affect structural integrity, pressure boundary leak tightness, and operating reliability.

The primary-side examination plan utilizes eddy current test (ECT) methods to detect and assess potential steam generator tube degradation, and visual inspection to assess the condition of the primary channel head, steam generator tube plugs, and tube-to-tubesheet seal welds. The secondary-side examination plan assesses steam generator internals, both in bundle and steam drum, and top of tubesheet regions. Visual examination is utilized to detect loose parts and to assess other secondary-side component conditions that could affect the structural integrity and leak tightness of pressure boundaries.

Elements of the primary and secondary-side inspections described herein address compliance with Technical Specifications, NEI 97-06, and industry guidelines. ENO's compliance with NEI 97-06 is mandated by SAO-180 Administrative Steam Generator Program Plan, which provides for management discretion to define the scope and frequency of certain steam generator examinations that go beyond the requirements of the Technical Specifications. To the extent that the steam generator examination plan for the Fall 2002 exceeds existing Technical Specification requirements, no new licensing commitments are intended or implied in this plan. Specific details of the 2002 refueling outage steam generator examination program are summarized below.

Steam Generator Primary-Side Inspection

Primary-side steam generator examinations are summarized in Table 1. One hundred percent (100%) of active steam generator tubes will be examined from tube end to tube end utilizing eddy current test (ECT) methods as specified in the Degradation Assessment (Ref. 1). A full length bobbin probe inspection will be performed of the tubes in Rows 3 and higher. In Rows 1 and 2, a bobbin probe inspection will be performed of the hot and cold straight leg sections inclusive of the upper support plate, while Ubends in these two rows will be inspected by rotating Plus Point probe. In addition 20% of hot leg tubes in four steam generators will be inspected at the top of the tubesheet +/- 3 inches by rotating Plus Point probe.

Potential and actual indications of degradation by the bobbin probe will be further characterized and confirmed by rotating Plus Point probe. Row 1 and Row 2 Ubends that exhibit unresolved rotating probe signals may also be inspected by high frequency Plus Point rotating probe. The basis used to determine the scope of any selected inspection sample or need to perform an expansion of the inspection scope shall comply with the requirements of the Technical Specification 4.13 and the EPRI PWR Steam Generator Examination Guidelines Rev. 5.

Discretionary ECT testing may include the use of high frequency rotating Plus Point probe in a random selection of Row 1 and 2 Ubends. Other inspection probes and methods may be used at the discretion of ENO. Supplementing steam generator tube ECT inspections, visual inspection will be performed of the primary channel heads and tubesheet, including steam generator tube to tubesheet seal welds and previously plugged tubes (two tubes in SG 24 were plugged at the factory during manufacturing), in accordance with the requirements of the Westinghouse SG Technical Manual and site procedures.

The ECT methods employed to inspect steam generator tubes meet the requirements of the EPRI PWR Steam Generator Examination Guidelines, Rev. 5, and are qualified in accordance with Appendix H. ECT data analysts will be qualified using a site specific training program in accordance with Appendix G of the same EPRI guideline document. Bobbin probe inspection will be performed of all straight leg tube sections and Ubends in Rows 3 and higher with the maximum diameter probe feasible, which is typically a 720 mil diameter probe. While there is no expectation that any significant tube diameter restriction will be encountered, a tube that does not permit passage of the 720 mil diameter probe will be tested through the region of the restriction with progressively smaller qualified probes as required down to a 610 mil diameter probe, and the cause of the restriction will be assessed. Unrestricted portions of the same tube will be tested with the largest practical diameter probe size from either hot and/or cold leg sides in accordance with EPRI Appendix H qualified procedures. The 610 mil bobbin probe will be used only for gauging. If the tube does not permit passage of the 610 mil bobbin probe it will be plugged. Any tube portion whose inspection is limited to use of the 610 mil bobbin probe, shall also be inspected by rotating Plus Point probe. Depending on the cause of the restriction, if this latter inspection data establishes that the tube is not defective, the tube will remain in service.

The results of ECT shall be reviewed, and degraded and defective steam generator tubes shall be identified. The cause of degradation and degradation measurement parameters in degraded steam generator tubes shall be assessed against established structural limits (Ref. 2) and Technical Specification criteria, and the result shall be incorporated in the Operational Assessment. If any defective tubes are detected, a bounding selection of defective tubes shall be pressure tested and the results shall be compared against performance criteria for structural integrity and accident leakage and incorporated into the Condition Monitoring Assessment. For tubes with existing degradation, the Operational Assessment shall suitably account for uncertainties in eddy current measurements and continued tube wall degradation between consecutive inspection periods.

Degraded tubes, as defined by TS 4.13.A.1.d, shall be considered acceptable for continued service only if the degradation 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 including consideration of ECT error and degradation growth. Tubes that contain degradation that exceeds the more limiting requirement shall be either repaired or removed from service by plugging. ENO may administratively plug tubes for other reasons. Prior to leaving a degraded tube in service, Entergy will submit to the NRC the bases of such decision including the method of inspection, the plugging criterion used, and a description of the methodology used to develop this criterion.

Steam Generator Secondary-Side Inspection

The steam generator secondary-side examination plan assesses steam generator internals, both in-bundle and steam drum, and top of tubesheet regions. Visual inspection is utilized to assess the presence of loose parts or other steam generator secondary-side component conditions that could affect the structural integrity of the primary boundary and leak tightness.

The secondary-side inspection will incorporate sludge lancing and foreign object search and retrieval (FOSAR). In-bundle inspection will be performed in approximately every fifth column. The upper bundle inspection will be performed by both looking up from the bottom and also by removing the inspection ports located above the top support plate. This inspection will be performed on all four steam generators. On one steam generator, the secondary side manway will be removed to perform a visual inspection of upper parts of the steam generator. This inspection will include but not be limited to the "J" nozzles, the feedring, and risers. Discrepant and unusual conditions will be noted and dispositioned in accordance with ENO procedure.

References:

1. Westinghouse Electric Nuclear Services Report No. SG-SGDA-02-029: "Steam Generator Degradation Assessment for Indian Point Unit 2 RFO15," Rev. 0
2. Westinghouse Electric Company Report WCAP-15909: "Regulatory Guide 1.121 Analysis for the Indian Point Unit 2 Steam Generators"

Table 1: Steam Generator Primary Side Inspection Plan for 2R15

Inspection	Inspection Scope	Number of Steam Generators
ECT Bobbin Coil	100% of tubes – end to end[1]	Four steam generators
ECT Rotating Probe (Plus Point)	Row 1 and Row 2 U-bends	Four steam generators
ECT Rotating Probe (Plus Point)	20% of hot leg tubes at the top of tubesheet +/- 3 inches	Four steam generators
ECT Rotating Probe (Hi-Freq Plus Point)	Sample Row 1 and Row 2 U-bends	Two steam generators
ECT Rotating Probe (Plus Point)	Indications of degradation by Bobbin probe	To be determined
Visual	Channel head, cladding, plugs, seal welds	Four steam generators

Notes:
1) Bobbin coil is not qualified for Row 1 and Row 2 U-bends.

Table 2: Steam Generator Secondary Side Inspection Plan for 2R15

Task and Inspection Method	Inspection Scope	Number of Steam Generators
Sludge lancing	Top of tubesheet	Four steam generators
In-bundle FOSAR - visual	Top of tubesheet and tube bundle	Four steam generators
Outer annulus visual	Region between wrapper and outer shell	Four steam generators
Upper internals visual	Steam drum, J-tubes, feed ring	One steam generator
Transition cone visual	Girthweld region	One steam generator
Support plates visual	Support plates	One steam generator

ATTACHMENT 2 TO NL-02-112

Supplemental Response to Request No. 1 for NRC Request for Additional Information Regarding Steam Generator Surveillance Requirements, Indian Point Unit No. 2 (TAC No. MB0770), dated September 5, 2001

Entergy Nuclear Operations, Inc.
Indian Point Unit No. 2
Docket No. 50-247

Supplemental Response to Request for Additional Information

Request No. 1

The Basis section of TS 4.13 states that steam generator tube burst and collapse tests have demonstrated that tubes having wall thickness of not less than 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 any 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

A Regulatory Guide 1.121 analysis has been performed to determine tube structural limits for Indian Point 2 steam generator tubing. A summary of this analysis is provided in WCAP-15909: "Regulatory Guide 1.121 Analysis for the Indian Point Unit 2 Steam Generators."

The analysis performed assumes uniform wall thinning in both axial and circumferential directions. This assumption bounds both localized wall loss such as pitting corrosion and single axial throughwall or partial depth cracks. The structural limit criteria developed are not applicable to circumferential cracks. The analysis addresses regions of steam generator tubes corresponding to anti-vibration bar intersections, support plate intersections, and straight leg regions of the tube, including the tubesheet region. The worst case scenario calculated demonstrates that remaining tube wall thickness can be 0.024 inch, and still exhibit adequate margin of safety against steam generator tube burst or collapse. For small pits and wear scars, remaining wall thickness could be as low as 0.016 inch. The Regulatory Guide 1.121 analysis therefore substantiates the adequacy of current Technical Specification Basis.

The evaluation further confirms leak-before-break behavior utilizing test data on leakage rates and burst strength as a function of through wall crack length. Compliance to existing Technical Specification leak limits provides reasonable assurance that plant shutdown can be achieved before a leaking tube would rupture under normal operating or accident conditions.

Any decision to leave degraded tubes in service at Indian Point 2 will be documented in the Condition Monitoring and Operational Assessment (CMOA) Report and justified in accordance with the requirements of the Steam Generator Program Plan, NEI 97-06 and EPRI Steam Generator Examination Guidelines Rev. 5. The basis to leave degraded tubes in service will duly consider ECT inaccuracy and projected degradation growth rate over the next operating cycle.

ATTACHMENT 4 TO NL-02-112

WCAP-15909-NP, "Regulatory Guide 1.121 Analysis for the Indian Point Unit 2 Steam Generators," dated August 2002 (Non-Proprietary)

Entergy Nuclear Operations, Inc.
Indian Point Unit No. 2
Docket No. 50-247