Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601
Tel 914 272 3200
Fax 914 272 3205

Michael R. Kansler
President

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

SUBJECT: Indian Point Nuclear Generating Unit No. 2
Docket No. 50-247
Proposed Change to Technical Specifications:
One-Time Change to the Indian Point 2 Steam
Generator Tube Inspection Requirements

Dear Sir:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc, (ENO) hereby requests the following amendment to the Operating License for Indian Point Nuclear Generating Unit No.2.

This amendment request seeks a one-time change to Indian Point 2 Technical Specification 5.5.7.b.1. Specifically, a footnote would be added to modify the Steam Generator (SG) inspection frequency for Indian Point 2. This proposed one-time change would extend the SG inspection interval from 24 months to a maximum of 43 months, based on one SG inspection that results in a C-1 classification. Although Indian Point 2 does not have a Technical Specification that pertains to two consecutive C-1 classifications, the analyses and evaluations presented in Attachment I address two consecutive C-1 classifications because other plants that have had similar changes approved for 40 months, have had TSs with wording similar to: "...two consecutive inspections resulting in C-1 classification". ENO is also cognizant of the industry efforts underway to develop revised technical specifications regarding SG inspections. While these industry efforts have not been completed at this time, ENO believes that these efforts, along with the plant specific information and the proposed compensatory measure, both addressed in Attachment I, provide justification to extend the period from 24 to a maximum of 43 months. It should be noted that the 43 months are calendar months when the actual operating months would be less than 40 months. This change is being proposed to eliminate unnecessary steam generator inspections, which will result in significant dose and cost savings. ENO requests approval of this proposed License Amendment prior to April 30, 2004 to support planning efforts for the Indian Point 2 fall 2004 refueling outage.

Attachment I provides the analysis of the proposed change to the Technical Specifications including information supporting a finding of no significant hazard consideration and provides a technical evaluation of the proposed change, which includes the most recent SG inspection results. In addition, Attachment II provides a markup page in the proposed pending improved technical specification format.

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There are no significant changes in the types or significant increase in the amounts of effluents that may be released offsite and there is no significant increase in individual or cumulative occupational radiation exposure. Therefore, this proposed one-time change satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment and the human environment is not affected by this amendment.

The commitment described in Attachment I is listed in Attachment III. If you have any questions, please contact Ms. Charlene Faison at 914-272-3378.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 10/21/03

Very truly yours,

Michael R. Kansler
President
Entergy Nuclear Operations, Inc.

Attachments:
I. Analysis of the Proposed Technical Specification Change.
II. Proposed Technical Specification Change – Markup Pages.
III. Commitments.

cc: Mr. Patrick D. Milano, Project Manager
Project Directorate I,
Division of Reactor Projects I/II
U.S. Nuclear Regulatory Commission
Mail Stop O-8-C2
Washington, DC 20555

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

Resident Inspector's Office
Indian Point Unit 2
U.S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, NY 10511

Mr. Peter R. Smith, Acting President
New York State Energy, Research and
Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Mr. Paul Eddy
New York State Dept. of Public Service
3 Empire Plaza
Albany, NY 12223

ATTACHMENT I

PROPOSED CHANGE TO TECHNICAL SPECIFICATIONS:

ONE-TIME CHANGE TO THE INDIAN POINT 2 STEAM

GENERATOR TUBE INSPECTION REQUIREMENTS

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

1.0 DESCRIPTION

This submittal is a request to amend Operating License DPR-26, Docket No. 50-247 for Indian Point Nuclear Generating Unit No. 2 on a one-time basis.

This amendment request seeks to revise Indian Point 2 Technical Specification 5.5.7.b.1. Specifically, a footnote is being added to modify the Steam Generator (SG) inspection frequency for Indian Point 2. This proposed one-time change would extend the SG inspection interval from 24 months to a maximum of 43 months, after one SG inspection resulting in a C-1 classification. While Indian Point 2 does not have a Technical Specification that pertains to two consecutive C-1 classifications, the analyses and evaluations presented in this Attachment address two consecutive C-1 classifications because other plants that have had similar changes approved for 40 months, have had TSs with wording similar to: "...two consecutive inspections resulting in C-1 classification". Entergy Nuclear Operations, Inc. (ENO) is also cognizant of industry efforts underway to develop revised technical specifications regarding SG inspections. While these efforts have not been completed at this time, ENO believes that these industry efforts along with the plant specific information and the compensatory measure, both addressed in Section 4.0 below, provide justification to extend the inspection interval from 24 months to a maximum of 43 months. It should be noted that the 43 months are calendar months, the actual operating months would be less than 40 months.

The reason for this one-time change is to eliminate unnecessary steam generator inspections, resulting in significant dose and cost savings.

2.0 PROPOSED CHANGES

Indian Point Technical Specification 5.5.7.b.1 would be modified by adding a footnote (asterisk) that would allow for a maximum 43-month inspection interval after one SG inspection resulting in a C-1 classification. The proposed footnote will read:

" * Except that the surveillance related to the steam generator tube inspection due no later than November 17, 2004, may be deferred until June 17, 2006."

3.0 BACKGROUND

The inspection of the SG tubes ensures that the structural integrity of this portion of the reactor coolant system pressure boundary will be maintained. Inservice inspection of SG tubes is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of progressive degradation of mechanical damage due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of SG tubes also provides a means of characterizing the nature and cause of any degradation so that timely corrective measures can be taken.

Technical Specification 5.5.7.b.1 requires that inservice inspections of steam generator tubes be conducted not less than twelve nor later than twenty-four months after the

previous examination. Several plants (see precedent section in this Attachment) have technical specifications that have a provision that if two consecutive SG inspections result in C-1 classifications, then the inspection interval may be extended from 24 to 40 months. In addition, there is an industry effort to extend the SG inspection interval beyond 40 months, for thermally treated Alloy 600TT and 690TT.

Indian Point 2 has thermally treated Alloy 600TT steam generator tubes, which exhibited minimal wear during the last inspection (see Section 4 below). This inspection was the first inspection of the Indian Point 2 replacement steam generators conducted during the first refueling outage subsequent to their installation. Section 4.0 below provides analyses of the results of the aforementioned inspection showing that the data support a one-time extension of the next Indian Point 2 SG inspection to a maximum of 43 months from the last inspection.

4.0 TECHNICAL ANALYSIS

The justification to extend the Indian Point 2 replacement steam generator (SG) inspection interval from 24 to a maximum of 43 calendar months on a one-time basis, is based on several factors. First, the SGs incorporated significant design improvements over the original steam generators (OSG). Second, industry experience with these design improvements has demonstrated that SGs can operate many years with minimal degradation. Third, Indian Point 2 is operated and the SGs maintained in accordance with NEI 97-06 "Steam Generator Program Guidelines" and its referenced EPRI documents. Lastly, results from a very thorough first in-service inspection (ISI) identified very few tubes with measurable indications and all of those were removed from service. An operational assessment for an assumed operation duration of 4.0 effective full power years (EFPY) confirmed that the SG tube structural and leakage integrity will be maintained until the proposed SG inspection in 2006. The following evaluation utilizes previous accepted data and information to address extending the inspection interval from 24 to 43 months.

SG Design Improvements

Industry experience with recirculating SGs using mill annealed Alloy 600 tubing, including the Indian Point 2 and Indian Point 3 original SGs, has led to significant design improvements in replacement SG design and fabrication. Problems associated with tube degradation (e.g., stress corrosion cracking (SCC), intergranular attack (IGA), pitting and wastage) have been addressed through changes in tube materials and stress relief. Problems associated with secondary system fouling and flow-induced vibration and wear have been addressed with changes to the tube bundle support system. These design improvements, along with others, have been incorporated into the Westinghouse replacement SG design and are discussed below.

- Thermally treated Alloy 600 (600TT) tubing provides much improved resistance to stress corrosion cracking.

  Each of the four Indian Point 2 SGs contains 3214 thermally treated Alloy 600 U-tubes that have a nominal outer diameter of 0.875 inches and a nominal thickness of 0.050 inches. The replacement SGs have thermally treated alloy 600

because that was the preferred alloy at the time of fabrication in 1986. Laboratory testing as well as extensive operating experience with the Alloy 600TT has demonstrated its superior corrosion resistance over mill annealed Alloy 600 tubing.

- Secondary stress relieving of the first eight rows improves resistance to stress corrosion cracking.

  In addition to the thermal treatment process that was performed on all tubing, the first eight rows of tubing (to a radius of 10.8 inches) received an additional heat treatment after bending to reduce residual stresses from the bending process.

- Tighter fitup of anti-vibration bars (AVB) provides for a more stable tube bundle, and limits potential for both wear and high cycle fatigue of tubes.

  There are two sets of AVBs installed in the U-bend region to stiffen the tube bundle, maintain proper tube spacing and alignment, and reduce tube vibration. The AVBs consist of V-shaped, square cross section bars of chrome plated Alloy 600 material.

  The combined tube-to-AVB fitup gap is limited by use of tubes having tightened U-bend outside diameter ovality control at AVB crossover points, close tolerance AVB widths, and special AVB assembly procedures. These controls were used in previous Westinghouse Model replacement steam generators where the AVB wear experience has ranged from none to limited. During the first ISI of the Indian Point 2 replacement SGs, all AVB/tube intersections were inspected with a bobbin probe. Some AVB wear was detected and those tubes were removed from service.

  The largest AVB wear, at Indian Point 2, that developed over the 21 Effective Full Power Months (EFPM) operating cycle was 20% through wall (TW). AVB wear typically declines over time as the gap between the AVB and the tubes increases. If the largest linear growth rate is assumed over the proposed 37 EFPM inspection interval, a new indication would be expected to grow 35.5% TW. The sensitivity of the inspection technique for AVB wear is about 5% TW or less. If it is assumed that a 5% TW AVB indication existed at the start of the current operating period, the maximum expected depth for that indication at the end of the proposed inspection period would be 40.5% TW. This is just above the repair limit of 40% and well below the tube burst structural limit of 67.8% TW.

- Corrosion resistant tube support plate material limits potential for crevice corrosion product buildup.

  The original Indian Point 2 SGs experienced crevice corrosion product buildup and tube denting. Improved material selection has reduced the potential for this problem.

  The tubes are intermittently supported on the secondary side by six tube support plates. The support plate's material is Type 405 stainless steel, a material with

improved corrosion resistance over the carbon steel used in the original Indian Point 2 SGs.

- Structural broach hole cutouts in the tube support plates improve axial flow within the tube bundle and minimize the tube-to-tube support contact area.

  The six tube support plates contain structurally arranged, quatrefoil shaped tube support holes. This design reduces tube dryout and chemical concentration where the tubes pass through the tube support plates.

- Narrower flow slots in support plates resist deformation and maintain better tube alignment.

  The denting that occurred in the original SGs deformed the tube support plates. There was visible hour glassing of the flow slots and in some cases ligament breaks. In addition to the more corrosion resistant stainless steel material and quatrefoil tube support holes, the width of the flow slots in the bottom five support plates was reduced from 2.75 to 1.75 inches. This increases the resistance to support plate deformation and maintains alignment of the tubing.

- Elimination of flow slots in the top support plate reduces the potential for deformation and addition of stresses to tubing.

  The replacement SGs have two rows of flow holes in the top support plate instead of flow slots used in the original SGs. Flow holes make the support plate more resistant to deformation and protect the U-bend portion of the tubing from additional stresses resulting from any deformation.

- Addition of a flow distribution baffle (FDB) plate produces flow conditions on the secondary side of the tubesheet to minimize the size of the zone where sludge deposition can occur.

  The FDB is fabricated of Type 405 stainless steel and has circular shaped holes (instead of the quatrefoil holes in the tube support plates). The central region of the FDB is open. The result of this design is that the flow velocity across the tubesheet surface is increased and the low flow velocity region (i.e., sludge deposition zone) is in the center of the tube bundle, near the blowdown intake.

- Full depth hydraulic tube expansions minimize the depth of the crevice between the tubes and the top of the tubesheet, thus minimizing the accumulation of contaminants in the tubesheet crevice and minimizing the residual stresses.

  The tube-to-tubesheet joint provides axial load resistance and the physical fastening of the tubing to the vessel. Original SG designs encountered severe corrosion problems with the SG tube-to-tubesheet joint region associated with open (i.e., unexpanded) crevices, and/or SCC at the high residual stress cold worked locations on the surface of the transition zone between the roller expanded and unexpanded tube. The Westinghouse replacement SG design incorporates the following features to address tube-to-tubesheet joint configuration concerns:

- Full depth expansion to eliminate a tubesheet crevice that could result in accumulation of contaminants against the transition zone and minimize the risk of stress corrosion cracking.

- Hydraulic expansion that leaves minimal residual stresses and cold work as compared to mechanical roller expansion techniques.

Overall, the improved design features incorporated in the Indian Point 2 replacement SGs provide assurance that SG tube integrity will be maintained over the proposed operating period. Figure 1 provides a Model 44F steam generator schematic.

Industry Data

The Indian Point 2 SGs are Westinghouse Model 44F and are nearly identical to replacement units at Turkey Point Units 3 & 4, Point Beach Unit 1 and H.B. Robinson Unit 2. Those 4 units have a total of 11 steam generators that have been in service from 13.25 to 14.70 EFPY and operate with higher hot leg temperatures than the Indian Point 2 SGs (597 – 604°F versus 590°F).

To date there has been minimal degradation exhibited in the Westinghouse Model 44F SGs. Some minor AVB wear has been found in some of those units. Additionally, other Westinghouse steam generators with Alloy 600TT tubing and similar AVB designs have seen very few corrosion indications and minor AVB wear with declining growth rates over time.

The Indian Point 3 SGs are Westinghouse Model 44F that have operated for 9 EFPY but have Alloy 690TT tubing and additional anti-vibration bar support. Therefore, comparing the Indian Point 2 and Indian Point 3 SGs would not be relevant due to the differences in SG tube materials and the differences in AVB designs.

First Outage Inspection Results

During the first refueling outage after SG replacement, Entergy conducted a thorough ISI examination of the SGs that exceeded the requirements of both technical specifications and revision 5 of the EPRI PWR SG Examination Guidelines. Technical specifications required that 12% of the tubes in each SG be examined from point of entry on the hot leg to the top support plate on the cold leg. In addition, the cold leg straight sections of 25% of those tubes were required to be examined as well. The EPRI guidelines require a full-length inspection of 100% of the tubes with a general purpose eddy current probe.

During the first ISI for Indian Point 2, Entergy inspected the full-length of 100% of the tubes with a bobbin probe except for the row one and two U-bend sections that were examined with rotating pancake probe (RPC). The bobbin probe does not provide an adequate signal in U-bends with a tight radius. Additionally, 20% of the tubes on the hot leg side of the SGs were examined with RPC from 3 inches above to 3 inches below the secondary face of the tubesheet. This examination included all tubes on the periphery of the annulus and tube lane of each SG to provide increased sensitivity for any potential

loose part wear. The peripheral tubes of the annulus and tube lane on the cold leg side were also examined with RPC from 3 inches above to 3 inches below the secondary face of the tubesheet. A sampling of dents and dings were examined with RPC as well.

The results of the first inservice inspection were transmitted to the NRC by letter dated December 19, 2002 (NL-02-161). Only two degraded tubes were identified and no active degradation mechanisms per the EPRI guidelines' definition exists. The threshold for that definition is the occurrence of 10 or more indications of degradation. The following is a summary of the indications found during the outage and the tube plugging that was performed.

The most significant indications were small volumetric indications in 13 tubes at AVB (anti-vibration bar) locations. Only two of those tubes had indications that met the technical specification and EPRI definition of degradation with depths of 20% through wall (TW). Entergy preventatively plugged all 13 tubes with mechanical plugs to support an extended inspection interval. Stabilization was not performed at the time but will be evaluated for two tubes with wear at all four AVB contact points prior to the next scheduled inspection.

There were three freespan volumetric indications detected. These indications were less than 20% TW, but, Entergy elected to preventatively plug these tubes as well.

Although loose parts were found on the secondary side in all four steam generators, no loose part wear was found by eddy current examination.

In December 2001 an extremely low level of secondary system activity was detected. Based on a review of fabrication records, a very small tube end weld imperfection in SG 22 is considered the most likely source of this activity. Approximately 200 tube ends in 22 SG had weld repairs during fabrication and while they were successfully tested for leak tightness, the possibility exists that a minor flaw was missed or that the thermal stress of operation could have opened a subsurface flaw. Although the activity only has an associated flow rate which currently ranges from approximately 0.01 - 0.03 gallons per day (well below industry guidelines, 5 gpd), it continues to be monitored, and has not changed since first detected. This is consistent with the conclusion that a tube end weld imperfection is the source of the activity. A primary side visual inspection of SG 22 tubesheet to look for possible leaks or surface indications, which might indicate cracking at the tube-to-tubesheet seal welds, found no anomalies. It has not been determined which of the aforementioned tube ends could be the source of this activity.

Laboratory analysis is routinely conducted of samples from the steam jet air ejector. This method initially detected the radioactivity on the secondary side of the plant. The lower limit of detection is in the range of 1E-8 uCi/cc for noble gases such as Xenon 133 and 135 when 4 liter samples are obtained from the air ejector. This method can reliably measure primary to secondary leakage to less than one gpd. The laboratory analysis is performed at a frequency defined by procedure. The current sampling frequency is once per week. As part of the ENO compensatory measures, from the beginning of the 41st month until the end of cycle 17, but, no later than the end of the 43rd month, steam jet air ejector sampling frequency will be increased from once to twice a week.

Condition Monitoring Assessment

After completion of the Indian Point 2 fall 2002 refueling outage SG inspections, a Condition Monitoring Assessment was performed in accordance with EPRI "Steam Generator Integrity Assessment Guidelines," Revision 1, March 2000. This proprietary document provides guidelines for evaluating the condition of SG tubes based on inspection results. Based on the Indian Point 2 fall 2002 refueling outage inspection results of the "as found" condition of all four SGs, all performance criteria were met, as recommended in NEI 97-06 "Steam Generator Program Guidelines".

Operational Assessment

An Operational Assessment was performed in accordance with EPRI Steam Generator Integrity Assessment Guidelines to evaluate the predicted condition of the SGs after two cycles of operation. The Operational Assessment is summarized below.

Indian Point 2 operated for 1.72 EFPY during Cycle 15 (i.e., the first post-SG replacement cycle). The operational assessment evaluated Indian Point 2 for a 4.0 EFPY operating period. This exceeds the combined Cycle 16 and Cycle 17 length estimate of 3.1 EFPY.

The only observed degradation with potential growth is AVB wear. Operating condition changes for a power uprate of 1.4% would have negligible impact on degradation since AVB wear is not temperature dependent and flow changes should be minimal. As a result of the 1.4% Indian Point 2 uprate, the amount of wear expected over 40 years of operation is approximately 2 mil, which is less than the allowable 3 mil. Since the tubes with wear indications were plugged, and the wear rate is expected to show a significant decrease with operating time, it is reasonable to expect that the Indian Point 2 steam generator tube structural and leakage integrity requirements will be maintained through the next two operating cycles. In addition, recent calculations indicate that operation at the projected 4.7% uprate conditions anticipated in the future, will not result in rapid rates of tube wear or high levels of tube vibration.

The operational assessment also concluded that the two tubes with wear indications at all four AVB contact points recommended for stabilization in the future are not expected to wear to the point of potential tube-to-tube contact during the next four cycles.

A possible damage mechanism that could affect replacement SG tube integrity is wear from secondary side foreign objects. During the Fall 2002 refueling outage, sludge lancing was performed on the secondary side tubesheet region of all four SGs. Upon completion of sludge lancing, a video inspection was performed in this region to identify any foreign objects.

Various foreign objects were observed during the Foreign Object Search And Retrieval (FOSAR) of the tubesheet region of the steam generators. Many of the small wires and sludge rocks in the bundle were left due to size, time and personnel exposure to retrieve. The objects that remain inside the SGs can be classified into three general types of objects as indicated below:

- Sludge rocks/sludge pebbles with diameters ranging from 1/8 to 3/8 inch.

- Small wires/bristles with diameters of 1/64 inch and a maximum length of 1.5 inches.
- Oval shaped metallic mass that could not be retrieved about 1/2 by 1/4 inch.

An analysis has determined that with each type of object the time required to wear a tube down to 40% through wall (assuming 20% initial tube wear) is greater than 2 operating cycles. From the eddy current inspection program it has been determined that these foreign objects did not produce any indications of tube wear. Table 1 provides a detailed list of the foreign objects that were observed. Figure 2 provides the SG Tubesheet Map (one leg).

This inspection along with improved SG design and Indian Point 2's Foreign Material Exclusion (FME) Program provides confidence that foreign object wear will not occur over the proposed operating period. In addition, Indian Point 2 has an online acoustic monitoring system to detect loose parts should they enter the steam generator.

Three freespan volumetric indications were also detected during the first inservice inspection. All three freespan indications are well below the condition monitoring limits for structural and leakage integrity established in the degradation assessment. Experience with other plants has shown that tubes with Alloy 600TT material have a tendency to develop freespan indication during the first cycle of operation. At Indian Point 2 many signals were detected by bobbin that did not exist in the preservice inspection. All of these indications were tested with plus point RPC and only three were reported as volumetric indications by plus point. There is no basis for suspecting that these indications are due to a corrosive mechanism. It is presumed that these volumetric indications were deep buff marks that became indications after the sustained heating of the operation cycle, or perhaps caused by a transient loose part.

Therefore, all structural and accident leakage performance criteria in Reference 2 are predicted to be met through the end of Cycle 17.

Some indications (such as pre-existing dings identified in the pre-service inspection) were not attributed to service induced degradation and the indications were left in service. The decision to leave such indications in service is primarily based on either previously identifying the signal during preservice inspections or the signal does not exhibit flaw like characteristics. Historical review of these bobbin probe signals can confirm their existence in the as-built preservice inspection records. If such a signal is unchanged from its baseline appearance, the possible flaw is re-categorized as a non-flaw. If the signal appears to have changed or not to have been present in the baseline records, the signal is +Point tested. If no degradation is found, the +Point result is reported as "NDF" (no degradation found) and the bobbin probe result re-stated as "NQS" (non-quantifiable signal). Tubes with signals classified in this fashion are not designated to be plugged.

NRC Information Notice (IN)-2002-21, dated June 25, 2002, and supplemented on April 1, 2003, reported indications of axially oriented ODSCC, after approximately 10 EFPY operation, in thermally treated Alloy 600 tubing in Seabrook to result from high residual stresses caused by nonoptimal tube processing. The unique eddy current signal offset seen at Seabrook was not seen during the first Inservice Inspection at Indian Point 2. Seabrook indicated that according to vendor records, the three affected heats were used

for steam generators in other PWRs as well. A review has shown the Indian Point 2 steam generator tubes had a different manufacturer and heats when compared to Seabrook. In addition, one of the criteria used to investigate the anomaly discussed in the "First Outage Inspection Results" section, above, was to review the fabrication history of the replacement SGs to identify fabrication anomalies. There were no other issues identified.

Dose and Cost Impact

If the proposed change is not approved for the Indian Point 2 Fall 2004 refueling outage, the plan would be to perform 20% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) bobbin probe inspection of the four SGs. Assuming this scope, a radiation dose of 8 person-REM is predicted based on the first ISI . This person-REM estimate includes SG platform setup, manway removal, eddy current inspection, and tube plugging. In addition, it is estimated that the cost of the inspection would be approximately $ 3,000,000.

Based on improved SG materials, design, first inservice inspection results, and industry experience, Entergy has concluded that SG inspections are not necessary to meet the SG inspection objectives.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

There is no direct increase in SG leakage because the proposed change does not alter the plant design. The scope of the inspection performed during the first refueling outage subsequent to SG replacement (last outage), exceeded the technical specification requirements for the first two refueling outages combined, after replacement. More tubes were inspected than were required by the technical specifications. Indian Point 2 does not have an active SG damage mechanism and will meet the current industry examination guidelines without performing inspections during the next refueling outage. The results of the Condition Monitoring Assessment subsequent to the last outage, demonstrated that all performance criteria were met during the last operating period. The results of the aforementioned Operational Assessment show that all performance criteria will be met over the proposed operating period.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of the inspections performed during the last (first after SG replacement) refueling outage significantly exceeds the Technical Specification requirements for the scope of the first two refueling outages combined subsequent to SG replacement.

The proposed change does not affect the SG design, the method of operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension of the SG tube inservice inspection frequency, and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant system or components.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

SG tube integrity is a function of design, environmental, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SGs. Inspections conducted prior to placing the SGs into service (pre-service inspection) and inspection during the first refueling outage following SG replacement, demonstrate that the SGs do not have fabrication damage or an active damage mechanism. The scope of those inspections significantly exceeds those required by the technical specifications. These inspection results were comparable to similar inspection results for the same model SG installed at other plants, and subsequent inspections at those plants provided results that support the extension request. The improved design of the replacement SGs also provides assurance that significant tube degradation is not likely to occur over the proposed operating period.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy Nuclear Operations, Inc. concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements / Criteria

The proposed change has been evaluated to determine whether applicable regulations, requirements, and criteria are met.

General Design Criterion (GDC) 32 (Inspection of Reactor Coolant Pressure Boundary) states: "Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel". The proposed change only impacts the frequency of inspection of the SG pressure boundary and does not affect the ability of the boundary to be inspected.

GDCs 14 (Reactor Coolant Pressure Boundary) and 31(Fracture Prevention of Reactor Coolant Pressure Boundary) are similarly unaffected by this proposed change.

Regulatory Guide (RG) 1.83 (Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes) paragraph states: "If two consecutive inspections, not including the preservice inspection, results in less than 10% of the tubes with detectable wall penetration (20%) and no significant (10%) further penetration of tubes with previous indications, the inspection frequency should be extended to 40 months intervals." This regulatory position is being impacted because ENO is proposing that after replacement of the Indian Point 2 SGs only one C-1 inspection after the initial inservice inspection is necessary for extending the inspection interval. The impact to the discussion of the RG is the same as that being requested for the Indian Point 2 technical specification change.

R.G. 1.121 (Bases for Plugging Degraded PWR Steam Generator Tubes) describes a method acceptable to the NRC Staff for establishing the limiting safe conditions of the tube degradation of steam generator tubing, beyond which defective tubes as established by inservice inspection should be removed from service by weld plugs at each end of the tube. This R.G. is unaffected by the proposed change.

10 CFR 50.65, the Maintenance Rule, classifies SGs as risk significant components because they are relied upon to remain functional during design basis events. SGs are monitored under (a)(2) of the Maintenance Rule against industry established performance criteria. If the performance criteria are not met, a cause determination shall be done and the results evaluated to determine if goals should be established per (a)(1) of the Maintenance Rule.

10 CFR 100, Reactor Site Criteria, establishes the reactor-siting criteria with respect to the risk of public exposure to the release of radioactive fission products. SG tubing and associated repairs techniques and components, such as tube plugs, must be capable of maintaining reactor coolant inventory and pressure in order to prevent excessive leakage.

NEI 97-06, "Steam Generator Program Guidelines, Rev 1" is not affected by the proposed change.

ENO has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the technical specification change, and does not affect conformance with any GDC differently than described in the FSAR. R.G. 1.83 compliance is affected the same way, as that of the proposed technical specification change and no additional action is considered necessary.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

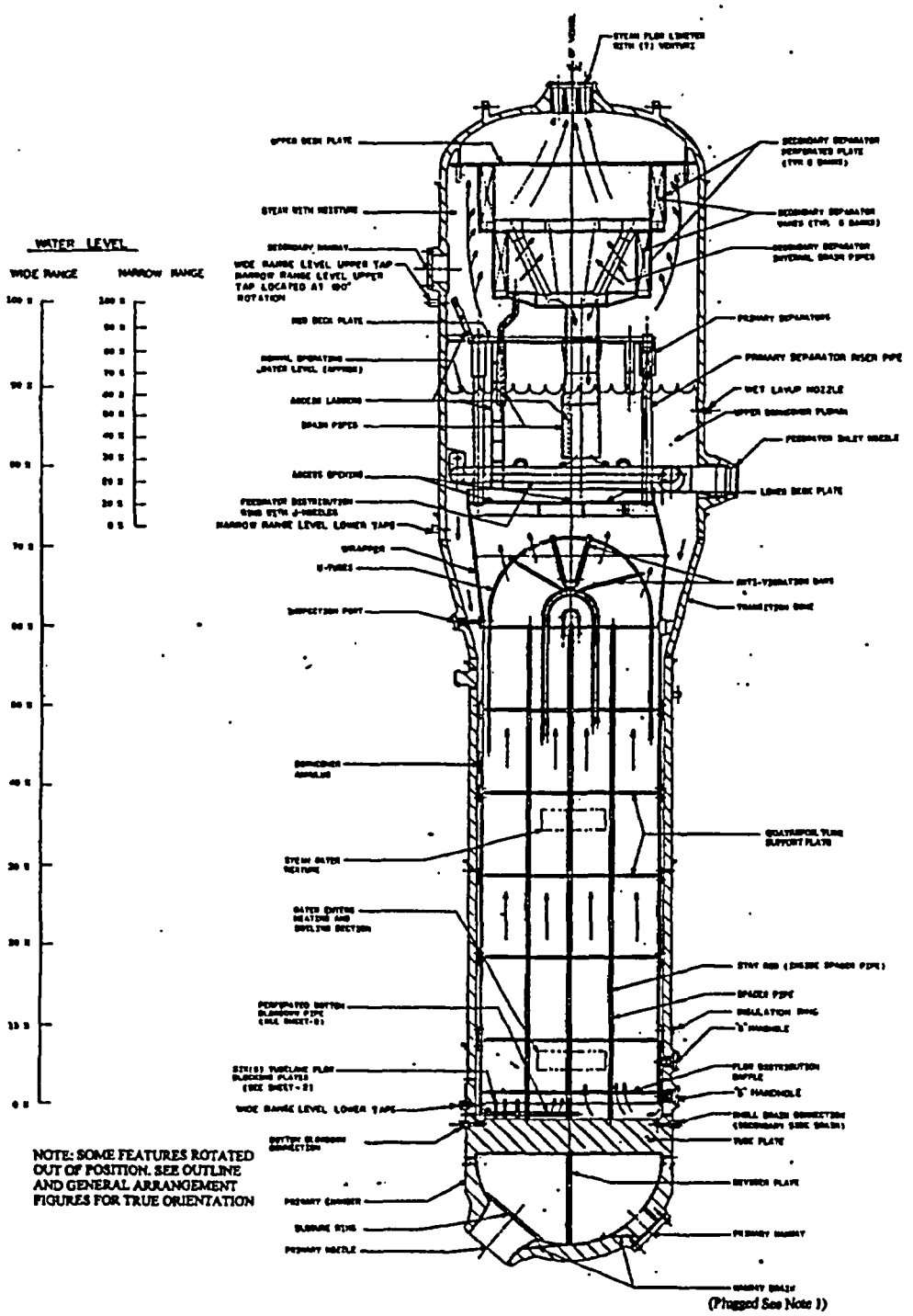
6.0 PRECEDENCE

Several nuclear plants that have replaced steam generators recently, have requested extensions of their SG inspections intervals from 24 months to 40 months, after one C-1 classification. The plants that have requested this type of extension and have received NRC approval include Braidwood 1, Farley 1 and 2, South Texas Project, and ANO-2.

In addition, the plant specific information, the proposed compensatory measure along with the industry effort to revise SG technical specification, provide the justification to extend the Indian Point 2 SG inspection interval from 24 months to a maximum of 43 months after one C-1 classification, on a one-time basis.

7.0 FIGURES AND TABLES

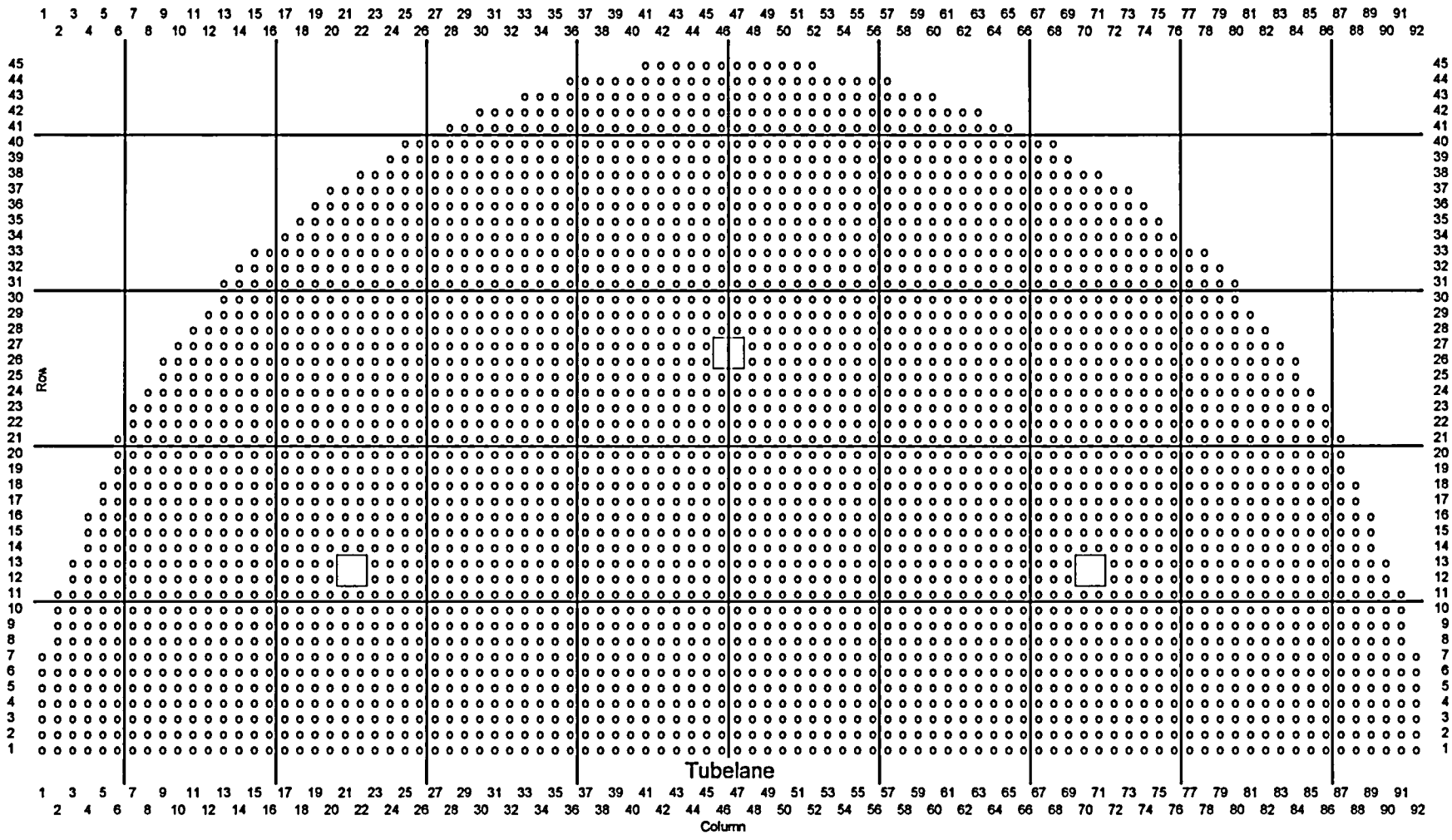
The aforementioned Figures 1 and 2, and Table 1 are provided following this page.



NOTES:

1) Internal Drains have been plugged with welded Inconel plugs per separate modification

Figure 1 – Model 44F Steam Generator Schematic



Steam Generator Tubesheet Map (one leg)
Westinghouse Model 44F (Indian Point 2)
Column 1 is the nozzle side of SG

Figure 2

Indian Point 2 R15 Secondary Side Inventory

Item #	SG	LOCATION	DESCRIPTION	DIMENSIONS	RETRIEVED YES / NO
1	21	CL R30C25	Wire Bristle	3/8" L X 1/64" Dia.	No
2	21	CL R28C30	Wire Bristle	1/4" L X 1/64" Dia.	No
3	21	CL R43C40	Sludge Rock	1/4" X 1/4"	No
4	21	CL R43C40	Sludge Rock	1/4" X 1/4"	No
5	21	CL R42C40	Wire Bristle	3/8" L X 1/64" Dia.	No
6	21	CL R41C40	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
7	21	CL R45C43	Weld Slag	1/4" W X 3/16" H X 1" L	Yes
8	21	CL R40C43	Wire Bristle	1/16" W X 1/64" Thick X 1/4" L	No
9	21	CL R40C43	Wire Bristle	3/8" L X 1/64" Dia.	No
10	21	CL R38C43	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
11	21	CL R36C43	Wire Bristle	3/8" L X 1/64" Dia.	No
12	21	CL R34C43	Wire Bristle	3/8" L X 1/64" Dia.	No
13	21	CL R32C43	Wire Bristle	1/4" L X 1/64" Dia.	No
14	21	HL R27C20	Wire Bristle	1/4" L X 1/64" Dia.	No
15	21	HL R43C35	Wire Bristle	3/16" L X 1/64" Dia.	No
16	21	HL R41C40	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
17	21	HL R31C40	Wire Bristle	1/4" L X 1/64" Dia.	No
18	21	HL R40C43	Wire Bristle	1/4" L X 1/64" Dia.	No
19	21	HL R40C43	Sludge Rock	1/8" X 1/8"	No
20	21	HL R38C43	Wire Bristle	1/4" L X 1/64" Dia.	No
21	21	HL R38C43	Sludge Rock	1/8" X 1/8"	No
22	21	HL R37C43	Wire Bristle	3/8" L X 1/64" Dia	No
23	21	HL R35C43	Wire Bristle	3/8" L X 1/64" Dia	No
24	21	HL R10C43	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
25	21	HL R10C43	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
26	21	CL R34C62	Wire Bristle	1/4" L X 1/64" Dia.	No
27	21	CL R33C62	Wire Bristle	1/4" L X 1/64" Dia.	No
28	21	CL R32C62	Wire Bristle	1/4" L X 1/64" Dia.	No
29	21	CL R28C62	Wire Bristle	1/8" L X 1/64" Dia.	No
30	21	CL R43C57	Sludge Rock	1/8" X 1/8"	No
31	21	CL R40C57	Wire Bristle	1/4" L X 1/64" Dia.	No
32	21	CL R34C57	Wire Bristle	1/4" L X 1/64" Dia.	No
33	21	CL R31C57	Wire Bristle	1/4" L X 1/64" Dia.	No
34	21	CL R30C57	Wire Bristle	1/4" L X 1/64" Dia.	No
35	21	CL R23C57	Wire Bristle	1/4" L X 1/64" Dia.	No
36	21	CL R18C57	Wire Bristle	1/8" L X 1/64" Dia.	No
37	21	CL R37C52	Wire Bristle	1/8" L X 1/64" Dia.	No
38	21	CL R27C52	Wire Bristle	1/8" L X 1/64" Dia.	No
39	21	HL R14C87	Wire Bristle	1/8" L X 1/64" Dia.	No
40	21	HL R41C62	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
41	21	HL R40C62	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
42	21	HL R39C62	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
43	21	HL R43C57	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
44	21	HL R42C57	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
45	21	HL R40C57	Assorted Wire Bristle	1/8" to 3/8" L X 1/64" Dia.	No
46	21	HL R41C56	Wire Bristle	3/8" L X 1/64" Dia	No

Item #	SG	LOCATION	DESCRIPTION	DIMENSIONS	RETRIEVED YES / NO
47	21	HL R37C56	Wire Bristle	1/8" L X 1/64" Dia.	No
48	21	HL R32C56	Wire Bristle	1/8" L X 1/64" Dia.	No
49	21	HL R41C54	Wire Bristle	3/8" L X 1/64" Dia	No
50	21	HL R37C54	Wire Bristle	3/8" L X 1/64" Dia	No
51	21	HL R42 C59/60	Metallic Mass Rock-Like, Oval Shape	1/2" x 1/4"	No
52	22	**HL R29C15**	**Flexitalic Gasket**	**2"L X 3/16"**	**Yes**
53	22	HL R19C15	Wire Bristle	3/8"L X 1/64"	No
54	22	HL R23C20	Wire Curl	1/4" high X 1/64"	No
55	22	HL R20C20	Wire Bristle	3/8"L X 1/64"	No
56	22	HL R32C25	Wire Bristle	1/2"L X 1/64"	No
57	22	HL R27C25	Wire Bristle	3/8"L X 1/64"	No
58	22	HL R40C30	Wire Bristle Pile	3/8"L X 1/64"	No
59	22	HL R38C30	Wire Bristle	3/8"L X 1/64"	No
60	22	HL R38C35	Wire Bristle	1/4" L X 1/64"	No
61	22	HL R37C35	Assorted Wire Bristle	1/4" L X 1/64" to 3/8"L x 1/64"	No
62	22	HL R12C40	Assorted Wire Bristle / Collaring	1/4" L X 1/64" to 3/8"L x 1/64"	No
63	22	HL R11C40	Wire Bristle	5/8" L X 1/64" Dia.	No
64	22	HL R42C43	Wire Bristle	1/2" L X 1/64" Dia.	No
65	22	CL R18C5	Anomaly In Tube Sheet (Treated As Sludge)	1/4" W X 3/4"L X 1/32" Deep	N/A
66	22	CL R26C10	Sludge Rock	1/4" X 3/8"	No
67	22	CL R23C10	Assorted Wire Bristle	1/4" L X 1/64" to 3/8"L X 1/64"	No
68	22	CL R22C10	Assorted Wire Bristle	1/4" L X 1/64" to 3/8"L x /64"	No
69	22	CL R18C10	Wire Bristle	3/8"L X 1/64" Dia.	No
70	22	CL R13C10	Wire Bristle	1/2"L X 1/64" Dia.	No
71	22	CL R8C15	Wire Bristle	1/4" L X 1/64" Dia.	No
72	22	CL R29C15	Wire Bristle	3/8"L X 1/64" Dia.	No
73	22	CL R23C15	Assorted Wire Bristle	1/4" L X 1/64" to 3/8"L x 1/64"	No
74	22	CL R15C15	Wire Bristle	3/8"L X 1/64" Dia.	No
75	22	CL R35C20	Assorted Wire Bristle	1/4" L X 1/64" to 1/2"L x 1/64"	No
76	22	CL R15C20	Wire Bristle	3/8"L X 1/64" Dia.	No
77	22	CL R7C20	Wire Bristle	3/8"L X 1/64" Dia.	No
78	22	CL R39C40	Wire Bristle	3/8"L X 1/64" Dia.	No
79	22	CL R28C72	Wire Curl	1/4" H X 1/4"L X 1/64" Dia.	No
80	22	HL R38C57	Wire Bristle	3/8"L X 1/64" Dia.	No
81	22	HL R34C52	Wire Bristle	3/8"L X 1/64" Dia.	No
82	23	CL R29C30	Wire Bristle	3/4"L x 1/64" Dia.	No
83	23	CL R32C35	Wire Bristle	1/4"L x 1/64" Dia.	No
84	23	CL R32C35	Wire Bristle	1/4"L x 1/64" Dia.	No
85	23	CL R38C40	Wire Bristle	1/4"L x 1/64" Dia.	No
86	23	CL R32C40	Wire Bristle	1/4"L x 1/64" Dia.	No
87	23	CL R30C40	Wire Bristle	1/4"L x 1/64" Dia.	No
88	23	CL R40C43	Wire Bristle	1/8"L x 1/64 Dia.	No
89	23	CL R35C43	Wire Bristle	3/4"L x 1/64 Dia.	No
90	23	CL R28C43	Wire Bristle	3/4"L x 1/64 Dia.	No
91	23	HL R27C20	Wire Bristle	1/2"L x 1/64 Dia.	No

Item #	SG	LOCATION	DESCRIPTION	DIMENSIONS	RETRIEVED YES / NO
92	23	HL R25C20	Wire Bristle	1/2"L x 1/64 Dia.	No
93	23	HL R23C20	Wire Bristle	1/2"L x 1/64 Dia.	No
94	23	HL R35C25	Wire Bristle	3/4"L x 1/64 Dia.	No
95	23	HL R32C25	Sludge Rock	1/8" Dia.	No
96	23	HL R35C30	Wire Bristle	1/3"L x 1/64"	No
97	23	HL R32C30	Wire Bristle	1/2" x 1/64"	No
98	23	HL R41C35	Wire Bristle	1 1/2"L x 1/64" Dia.	No
99	23	HL R37C35	Wire Bristle	1"L x 1/64" Dia.	No
100	23	HL R35C35	Sludge Rock	1/3" Dia.	No
101	23	HL R31C35	Wire Bristle	1/3"L x 1/64" Dia.	No
102	23	HL R42C40	Weld Slag	3/4" x 1/3"	Yes
103	23	HL R42C40	Weld Slag	3/4" x 1/3"	Yes
104	23	HL R39C40	Wire Bristle	1"L x 1/64" Dia.	No
105	23	CL R32C67	Wire Bristle	1/2"L x 1/64 Dia.	No
106	23	CL R35C62	Wire Bristle	1/4"L x 1/64 Dia.	No
107	23	CL R35C57	Wire Bristle	1/4"L x 1/64 Dia.	No
108	23	CL R35C52	Wire Bristle	1/2'L x 1/64 Dia.	No
109	23	HL R42C67	Wire Bristle	1/2"L x 1/64 Dia.	No
110	23	HL R43C57	Wire Bristle	1/2" x 1/2" x 1/2" x 1/8" W	Yes
111	24	HL annulus	Steel Block w/ Bolt	1" X 1/64"Dia.	Yes
112	24	HL annulus	Steel Block	1" X 1/64"Dia.	Yes
113	24	HL annulus	Block Magnet	1" X 2" X 3/8"	Yes
114	24	HL annulus	Block Magnet	1" X 2" X 3/8"	Yes
115	24	HL annulus	Block Magnet	1" X 2" X 3/8"	Yes
116	24	HL annulus	Block Magnet	1" X 2" X 3/8"	Yes
117	24	HL annulus	Block	1" X 2" X 1/4"	Yes
118	24	HL annulus	Block	1" X 2" X 1/4"	Yes
119	24	HL annulus	Block	1" X 2" X 1/4"	Yes
120	24	HL R42C35	Sludge Pebble	1/8" X 1/16"	No
121	24	HL R39C35	Wire Bristle	1/2" X 1/64" Dia.	Yes
122	24	HL R32C35	Wire Bristle	5/8" X 1/64" Dia.	Yes
123	24	HL R15C35	Wire Bristle	1/2" X 1/64" Dia.	Yes
124	24	HL R43C40	Rivet	3/16" Dia. X 1/2"lg.	Yes
125	24	HL R42C40	Wire Bristle	5/8" X 1/64" Dia.	No
126	24	HL R31C40	Wire Bristle	3/8" X 1/64"Dia.	No
127	24	HL R20C40	Wire Bristle	3/8"X 1/64"Dia.	Yes
128	24	HL R41C43	Sludge Rock	1/4" X 3/16"	No
129	24	HL R12C43	Wire Bristle	1/2" X 1/64"Dia.	No
130	24	HL R11C43	Wire Bristle	X 1/64"Dia.	No
131	24	HL R10C43	Sludge Pebble	1/4" X 1/4"	No
132	24	CL R31C35	Wire Bristle	3/8" X 1/64"Dia.	No
133	24	CL R42C43	Sludge Rock	1/4" X 3/16"	No
134	24	HL R40C52	Wire Bristle	3/8" X 1/64"Dia.	No
135	24	HL R40C52	Wire Bristle	3/8" X 1/64"Dia.	No
136	24	HL R39C52	Sludge Rock	3/16" X 3/16"	No
137	24	HL R35C52	Wire Bristle	1/2" X 1/64"Dia.	No
138	24	HL R37-38C22-24	Metal Lever	2 5/8 " x 1/4" x 1/4"	Yes

ATTACHMENT II

MARKUP OF TECHNICAL SPECIFICATION PAGES

REGARDING A PROPOSED

ONE-TIME CHANGE TO THE INDIAN POINT 2

STEAM GENERATOR INSPECTION REQUIREMENTS

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

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Tube Surveillance Program (continued)

9. Cold-Leg Tube Examination is an examination of the cold-leg side tube length. This shall include the tube length between the top support of the cold leg and the face of the cold-leg tube sheet.

b. Extent and Frequency of Examination

1. Steam generator examinations shall be conducted not less than 12 months nor later than twenty four calendar months after the previous examination. *

2. Scheduled examinations shall include each of the four steam generators in service.

3. Unscheduled steam generator examinations shall be required in the event there is a primary to secondary leak exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steamline or feedwater line break.

4. Unscheduled examinations may include only the steam generator(s) affected by the leak or other occurrence.

c. Basic Sample Selection and Examination

1. At least 12% of the tubes in each steam generator to be examined shall be subjected to a hot-leg examination.

2. At least 25% of the tubes inspected in Technical Specification 5.5.7.c.1 above shall be subjected to a cold-leg examination.

3. Tubes selected for examination shall include, but not be limited to, tubes in areas of the tube bundle in which degradation has been reported, either at Indian Point 2 in prior examinations, or at other utilities with similar steam generators.

4. Examination shall be by eddy current techniques as specified by the steam generator examination program submitted to the NRC in accordance with Technical Specification 5.5.7. In all cases, a probe with at least a 610-mil diameter shall be used.

*** Except that the surveillance related to the steam generator tube inspection due no later than November 17, 2004, may be deferred until June 17, 2006.**

ATTACHMENT III

COMMITMENTS REGARDING PROPOSED CHANGE TO TECHNICAL SPECIFICATIONS: ONE-TIME CHANGE TO THE INDIAN POINT 2 STEAM GENERATOR TUBE INSPECTIONS REQUIREMENTS

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

COMMITMENTS REGARDING PROPOSED CHANGE TO TECHNICAL SPECIFICATIONS: ONE-TIME CHANGE TO THE INDIAN POINT 2 STEAM GENERATOR TUBE INSPECTION REQUIREMENTS

Number	Commitment	Due Date
NL-03-165-01	Increase steam jet air ejector sampling to twice a week while the plant is operating, for the balance of cycle 17, beginning 40 calendar months after completion of the last steam generator C-1 inspection.	Beginning March 17, 2006, until June 17, 2006.