Entergy

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

June 13, 2002

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-02-076

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop 0-P1-17
Washington, DC 20555-0001

SUBJECT: License Amendment Request (LAR No. 02-008) – Revision to the Steam Generator Tube Surveillance Requirements for Eddy Current Inspection Probe Diameter in the Indian Point Nuclear Generating Unit No. 2 Technical Specifications

References: 1. NRC letter to Entergy Nuclear Operations, Inc., "Indian Point Nuclear Generating Unit No. 2 – Amendment Re: Technical Specification Changes to Secondary Leakage Limits and Steam Generator Tube Inservice Surveillance Requirements (TAC No. MB0770)," dated April 2, 2002

Pursuant to 10CFR50.90, Entergy Nuclear Operations, Inc. (ENO) hereby requests an amendment to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TS) Section 4.13, "Steam Generator Tube Inservice Surveillance." The purpose of this request is to allow the use of the optimum tube inspection probe diameter when performing the eddy current testing of the steam generator tubes during inservice inspections. The existing prescriptive IP2 TS surveillance requirements do not allow for implementation of improved inspection techniques or equipment without a license amendment. These requirements were imposed as the original steam generators aged and tube degradations and deformations (denting) were documented. Additionally, two grammatical oversights found in Section 4.13 are being corrected.

Therefore, the proposed change will provide for improved steam generator tube inservice inspection capabilities while also representing a significant cost savings and increased operational flexibility to IP2 without any decrease in the level of safety and protection provided to the public.

Attachment 1 to this letter provides the description and evaluation of the proposed change. The revised TS pages are provided in Attachment 2 (strikeout and shaded format).

A001

ENO requests approval of the proposed change by July 26, 2002 with an implementation date within 30 days of approval to allow incorporation into the proposed steam generator examination program that is required to be submitted to the NRC at least 60 days prior to the scheduled examination by TS 4.13.C.1.

The onsite and offsite safety review committees have reviewed the proposed change and both committees concur that the proposed change involves no significant hazards consideration as defined by 10 CFR 50.92(c).

In accordance with 10 CFR 50.91, a copy of this submittal and the associated attachments are being submitted to the designated New York State official.

There are no commitments contained in this letter.

Should you or your staff have any questions regarding this submittal, please contact Mr. John F. McCann, Manager licensing, Indian Point Energy Center at (914) 734-5074.

I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

F Dacimo

Executed on June 13, 2002

Fred Dacimo
Vice President – Operations
Indian Point Energy Center
Unit 2

cc: See page 3

Attachments

cc:

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# ATTACHMENT 1 TO NL-02-076

# LICENSE AMENDMENT REQUEST

## REVISION TO THE STEAM GENERATOR TUBE SURVEILLANCE REQUIREMENTS
## FOR
## EDDY CURRENT INSPECTION PROBE DIAMETER
## IN THE
## INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 TECHNICAL SPECIFICATIONS

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

# LICENSE AMENDMENT REQUEST

## DESCRIPTION OF THE PROPOSED CHANGE

The proposed change involves revising IP2 TS Section 4.13.A.3.f to allow the use of the optimum size eddy current inspection probe diameter during the IP2 steam generator tube inservice surveillances.

## REASONS FOR THE CHANGE

The restrictive specificity of the existing IP2 TS surveillance requirements prohibits the use of improved inspection equipment without processing a license amendment. The current restrictions were imposed during an era when the nature of the problems with the reactor coolant pressure boundary in the steam generator tube region were being discovered and the tools and techniques to prevent and detect degradation of the pressure boundary were still under development. New inspection techniques and more precise instrumentation have been and are continuing to be developed. The proposed change will allow the use of a more optimum eddy current inspection probe diameter during the steam generator tube inservice surveillance.

## EVALUATION OF THE PROPOSED CHANGE

General Design Criteria (GDC) 30 and 32 of Appendix A to 10 CFR Part 50 require that components that are part of the reactor coolant pressure boundary receive periodic inspection and testing of critical areas to assess their structural and leak tight integrity A brief history of the development of the current license basis for the steam generator tube surveillance requirements is necessary to understand the basis for the proposed change.

Before initial operation of the plant, a baseline eddy current inspection of tubing in all four steam generators was performed. During the first cycle of operation, in response to an NRC request dated July 18, 1974, an amendment to the IP2 Technical Specifications, Section 4.2, "Inservice Inspection and Testing," was proposed to provide an acceptable program for in-service inspection of the steam generator tubes. The proposed inspection program was based on the requirements of Section XI of the ASME Code for In-Service Inspection of Nuclear Reactor Coolant Systems dated January 1970. The proposed amendment did not specify any eddy current inspection probe size and, in fact, the proposed associated TS bases for Section 4.2 contained a discussion of the continued development of inspection equipment. Prior to the approval of the requested amendment, the NRC issued a request to revise the initial submittal to be consistent with model technical specifications that had been developed by the NRC. The model technical specifications were based on the guidance contained in Revision 1 of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," dated July 1975.

Following the steam generator tube failure that occurred at Surry Unit No. 2 in 1976, Consolidated Edison Company of New York, Inc. (Con Ed), the licensee for IP2 at that time, provided the NRC with their program to investigate the potential for a similar tube failure at IP2. The steam generator tube inspection results were subsequently provided to the NRC along with a request for the NRC's concurrence to return the reactor to power operation. The concurrence was granted along with the imposition of a license condition specifying the length of time that the plant could be operated with a primary coolant temperature greater than 350°F without further authorization. The license amendment transmittal letter also included a request that the details of steam generator inspection program planned for the next inspection be submitted to the NRC for review and concurrence, no later than 30 days prior to the date the next inspection was expected to commence. The steam generator tube inspection requirements in TS Section 4.13 were originally established by License Amendment No. 31, which was issued on June 28, 1977. The steam generator tube inservice surveillance requirements were based on the model technical specifications provided by the NRC.

The steam generator inspection program for the next planned inspection followed by a report of the inspection results and a request to resume power operation until the next scheduled inspection continued to be submitted to the NRC. The NRC requested, in a May 15, 1981 letter granting concurrence to return to power operation, that an amendment to the license and Technical Specifications for future inspections be proposed. The proposed amendment would remove the need for NRC approval for restart from steam generator inspection outages under certain conditions. In addition to the surveillance requirements specified in the model technical specifications, the proposed amendment also included some of the details from the steam generator tube inspection program such as the use of the "standard" 700-mil diameter probe, the use of a 610-mil diameter probe if the 700 mil diameter probe would not pass through the tube being inspected and the use of a 540-mil diameter probe for certain tube locations. On October 21, 1982, the NRC issued License Amendment No. 81 to redefine the steam generator tube inspection requirements. These requirements have remained essentially unchanged since that time except for allowances added and then removed that approved repair methods as alternatives to removing defective tubes from service.

By letter dated December 11, 2000, Con Ed (still the licensee at the time) submitted an application for an amendment to the IP2 TS requesting 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 that were installed earlier in 2000. On September 6, 2001, Con Ed's operating authority under the license was transferred to Entergy Nuclear Operations, Inc. (ENO). On September 20, 2001, ENO requested the NRC to review and act on all requests that had been submitted prior to the transfer. The NRC reviewed the application for amendment and requested additional information, which was provided by ENO in letters dated November 5, and December 7, 2001.

The requested license amendment was issued as Amendment No. 226 on April 2, 2002, based on the prior replacement of the steam generators. The amendment deleted all references to inspection and repair requirements related to the sleeving repair method, the F* ARC, and the tube denting. Because the NRC has not approved the use of the sleeving repair method or the F* ARC in the replacement steam generators, the references to these requirements are not necessary. The inspection requirements related to denting were added to the TSs for the original steam generators because denting had been identified as a dominant SG degradation mechanism. Because the denting phenomenon has not been identified as a dominant tube degradation mechanism in the replacement SGs, the references to denting in the TSs are no longer necessary.

The NRC staff, in their safety evaluation related to the approved amendment, expressed concern that ENO had elected to continue to use the eddy current inspection probe sizes specified in the existing TSs rather than to propose changes to a more optimum probe size. This would seem justified due to the differences between the design of the original SGs and the replacement SGs. The NRC staff stated that a larger diameter probe size would result in a larger fill factor, which would improve the probe's sensitivity to tube degradation and stabilize the probe during testing. Furthermore, a larger fill factor would improve the signal-to-noise ratio. The 0.700-inch diameter probe has a smaller fill factor than other probe sizes typically used for 7/8" diameter tubing. This smaller fill factor could result in reduced electromagnetic coupling, reduced sensitivity to defects, and increased noise. Although this probe size is not optimal for detecting tube degradation, the staff found that it would provide adequate examination performance.

The IP2 guidelines for steam generator tube inspection and assessment were described to the NRC in the response to Generic Letter 97-05, Steam Generator Tube Inspection Techniques, for Indian Point Unit 2. They are in accordance with industry standards and were found acceptable to satisfy Generic Letter 97-05 by the NRC. The proposed amendment will remove the unnecessarily prescriptive restriction on eddy current probe size from the examination criteria and allow selection of the optimum equipment for performing the steam generator tube inservice surveillance. Regulatory control will not be reduced since the minimum probe size is maintained and current TS 4.13.C.1 requires the licensee to submit the steam generator inspection plan for NRC review at least 60 days prior to the scheduled inspection. Therefore, the proposed change will improve the ability of ENO to assure continued integrity of the steam generator tubes as part of the reactor coolant system pressure boundary.

In addition, two typographical errors are also proposed for correction. The last definition under TS 4.13.A.1 should be numbered with a lower case "i" instead of an upper case "I" and the first "of" should be "or" in TS 4.13.A.4.1.a.

## NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

ENO has determined that the proposed Technical Specification change does not involve a significant hazards consideration as defined by 10CFR50.92(c).

1. **Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.**

The integrity of the steam generator tube portion of the reactor coolant pressure boundary will continue to be monitored, as before, therefore the probability of the failure of the pressure boundary is not affected by the proposed change. Likewise, the probability of the extent of any pressure boundary rupture will not be affected by the proposed changes since the depth and scope of the steam generator tube surveillances are not reduced by the proposed changes. The proposed changes facilitate the application of more advanced diagnostic techniques. The changes involve updating TS Section 4.13.A.3.f. to permit more flexibility in the eddy current inspection probes used in the steam generator tube inspections and to reflect current technological advances in inspection equipment, while still maintaining the minimum 610-mil diameter probe restriction. These changes do not affect possible initiating events for previously evaluated accidents or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical specifications remain unchanged. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

2. **Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed change is to allow improvements to the inspection techniques and equipment used to perform the steam generator tube inservice inspections. These inspections are only performed while the reactor is in the cold shutdown condition and the steam generators are not capable of affecting the operation of any system, structure or component that maintains the protection of a fission product barrier. The proposed changes facilitate the application of current diagnostic techniques. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change does not create a new accident initiator or precursor, or create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.**

The integrity of the steam generator portion of the reactor coolant pressure boundary is not affected by the proposed changes to facilitate the application of current diagnostic techniques. Since the elimination of redundant and restrictive controls over the techniques and equipment used to inspect the steam generator tubes does not result in changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report design basis, accident assumptions, or the Technical Specification Bases are not affected. Therefore, the integrity of the reactor coolant system pressure boundary is not affected and operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

## CONCLUSIONS

Based on the above evaluation, ENO has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed, and does not result in a reduction in any margin of safety. Therefore, operation of IP2 in accordance with the proposed amendment does not involve a significant hazards consideration. In addition, the onsite and offsite safety review committees have reviewed the proposed change to the TS and both committees concur that the proposed change does not involve a significant hazards consideration.

## ENVIRONMENTAL ASSESSMENT

An environmental assessment is not required for the above proposed change because the requested change to the Indian Point Unit No. 2 Technical Specifications conforms to the criteria for "actions eligible for categorical exclusion," as specified in 10CFR51.22(c)(9). The proposed change is to a steam generator tube in-service inspection surveillance requirement that is performed during plant shutdown and will have no impact on the environment. The proposed change does not involve a significant hazards consideration as discussed in the preceding section. The proposed change does not involve a significant change in the types or a significant increase in the amount of any effluents that may be released offsite. In addition, the proposed change does not involve a significant increase in individual or cumulative occupational radiation exposure.

**ATTACHMENT 2 TO NL-02-076**

**INDIAN POINT NUCLEAR GENERATING UNIT NO. 2**

**TECHNICAL SPECIFICATION PAGES IN**

**STRIKEOUT/SHADED FORMAT**

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

g. Plugging Limit is the degradation depth at or beyond which the tube must be plugged or repaired.

h. Hot-Leg Tube Examination is an examination of the hot-leg side tube length. This shall include the length from the point of entry at the hot-leg tube sheet around the U-bend to the top support of the cold leg.

i. 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.

2. Extent and Frequency of Examination

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

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

c. 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.

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

3. Basic Sample Selection and Examination

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

b. At least 25% of the tubes inspected in Specification 4.13.A.3.a above shall be subjected to a cold-leg examination.

c. DELETED

d. 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.

e. DELETED

f. Examination shall be by eddy current techniques as specified by the steam generator examination program submitted to the NRC in accordance with TS 4.13.C.1. ~~A 700-mil diameter probe shall be used unless previous data indicates that a 700-mil diameter probe would not pass through the tube. If the 700-mil diameter probe cannot pass through the tube, the largest size probe that is expected to pass through the tube shall be used.~~ In all cases, a probe with at least a 610-mil diameter shall be used.

4. Additional Examination Criteria

1. Degradation

a. If 5% or more of the tubes examined in a steam generator exhibit degradation or if any of the tubes examined in a steam generator are defective, additional examinations shall be required as specified in Table 4.13-1.

b. Tubes for additional examination shall be selected from the affected area of the tube array and the examination may be limited to that region of the tube where degradation or defective tube(s) were detected.

c. The second and third sample inspections in Table 4.13-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.