UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 5, 2014

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: H* ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBE INSPECTION AND REPAIR (TAC NO. MF3369)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 277 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 16, 2014, as supplemented by letters dated April 2 and April 15, 2014.

The amendment revises TS 3.4.13, "RCS Operational Leakage," and TS 5.5.7, "Steam Generator (SG) Program," to exclude portions of the SG tube below the top of the SG tubesheet from periodic inspections and plugging by implementing the H* alternate repair criteria on a permanent basis. In addition, TS 5.6.7, "Steam Generator Tube Inspection Report," is also being revised to include additional reporting requirements.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures:
1. Amendment No. 277 to DPR-26
2. Safety Evaluation

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NUCLEAR REGULATORY COMMISSION
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ENTERGY NUCLEAR INDIAN POINT 2, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

AND TECHNICAL SPECIFICATIONS

Amendment No. 277
License No. DPR-26

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated January 16, 2014, as supplemented on April 2 and April 15, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A, B and C, as revised through Amendment No. 277, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley

Benjamin G. Beasley, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: September 5, 2014

ATTACHMENT TO

LICENSE AMENDMENT NO. 277

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO 50-247

Replace the following page of the License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.4.13-1	3.4.13-1
3.4.13-2	3.4.13-2
5.5-7	5.5-7
5.5-8	5.5-8
5.6-5	5.6-5

instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; — Amdt. 42 10-17-78

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility and Indian Point Nuclear Generating Unit No. 3 (IP3). — Amdt. 220 09-06-01

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal. — Amdt. 241 10-27-2004

(2) Technical Specifications

The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 277, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

(3) The following conditions relate to the amendment approving the conversion to Improved Standard Technical Specifications:

1. This amendment authorizes the relocation of certain Technical Specification requirements and detailed information to licensee-controlled documents as described in Table R, "Relocated Technical Specifications from the CTS," and Table LA, "Removed Details and Less Restrictive Administrative Changes to the CTS" attached to the NRC staff's Safety Evaluation enclosed with this amendment. The relocation of requirements and detailed information shall be completed on or before the implementation of this amendment.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE, and

d. 85 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	**- NOTES -** 1. Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	**- NOTE -** Not required to be performed until 12 hours after establishment of steady state operation. Verify primary to secondary LEAKAGE is ≤ 85 gallons per day through any one SG.	72 hours

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program (continued)

contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gpd per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following SG tube alternate plugging criteria shall be applied as an alternative to the preceding criteria.

Tubes with service-induced flaws located greater than 18.9 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 18.9 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 18.9 inches below the top of the tubesheet on the hot leg side to 18.9 inches below the top of the tubesheet on the cold leg side, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program (continued)

location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-line indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing,

h. The effective plugging percentage for all plugging in each SG,

i. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

j. The calculated accident leakage rate from the portion of the tubes below 18.9 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident leakage rate from the most limiting accident is less than 1.75 times the maximum primary to secondary leakage rate, the report should describe how it was determined, and

k. The results of monitoring for tube displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 277 TO FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR INDIAN POINT 2, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated January 16, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14027A514), as supplemented by letters dated April 2, 2014 (ADAMS Accession No. ML14107A042), and April 15, 2014 (ADAMS Accession No. ML14112A460), Entergy Nuclear Operations, Inc. (Entergy, the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TSs). The amendment revises TS 3.4.13, "RCS [Reactor Coolant system] Operational Leakage," TS 5.5.7, "Steam Generator (SG) Program," and TS 5.6.7, "Steam Generator Tube Inspection Report," in order to implement the H* alternate repair criteria on a permanent basis.

The supplemental letters dated April 2 and April 15, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration.

2.0 BACKGROUND

IP2 has four Model 44F replacement steam generators (SGs), which were designed by Westinghouse Electric Company, LLC (Westinghouse), and installed in 2000. There are 3,214 thermally treated Alloy 600 (Alloy 600TT) tubes with a nominal outside diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. The thermally treated tubes are hydraulically expanded for the full depth of the 21-inch thick tubesheet and are welded to the tubesheet at each tube end.

Prior to the fall of 2004, no instances of stress corrosion cracking (SCC) affecting the tubesheet region of Alloy 600TT tubing had been reported at any nuclear power plant in the United States. In the fall of 2004, crack-like indications were found in tubes in the tubesheet region of SGs at Catawba Nuclear Station, Unit 2 (Catawba Unit 2). These crack-like indications were found in a tube overexpansion (OXP) that was approximately seven inches below the top of the tubesheet (TTS) on the hot-leg side in one tube, and just above the tube-to-tubesheet (T/TS) weld, in a region of the tube known as the tack expansion region, in several other tubes. Indications were

also reported near the T/TS welds, which join the tube to the tubesheet. An OXP is created when the tube is expanded into a tubesheet borehole that is not perfectly round. These out-of-round conditions were created during the tubesheet drilling process by conditions such as drill bit wandering or chip gouging. The tack expansion is an approximately 1-inch long expansion at each tube end. The purpose of the tack expansion is to facilitate performing the T/TS weld, which is made prior to the hydraulic expansion of the tube over the full tubesheet depth.

Since the initial findings at Catawba Unit 2, in the fall of 2004, other nuclear plants with Alloy 600TT tubing have found crack-like indications in tubes within the tubesheet as well. Most of the indications were found in the tack expansion region near the tube-end welds and were a mixture of axial and circumferential primary water SCC. Over time, these cracks can be expected to become more and more extensive, necessitating more extensive inspections of the lower tubesheet region and more extensive tube plugging or repairs, with attendant increased cost and the potential for shortening the useful lifetime of the SGs. To avoid these impacts, the affected licensees and their contractor, Westinghouse, developed proposed alternative inspection and repair criteria applicable to the tubes in the lowermost region of the tubesheets. These criteria are referred to as the H*(H-star) criteria. The H* distance is the minimum engagement distance between the tube and tubesheet, measured downward from the TTS, that is proposed as needed to ensure the structural and leakage integrity of the T/TS joints. The proposed H* alternate repair criteria would exclude the portions of tubing below the H* distance from inspection and plugging requirements, on the basis that flaws below the H* distance are not detrimental to the structural and leakage integrity of the T/TS joints.

Requests for permanent H* amendments were proposed for a number of plants as early as 2005. The U.S. Nuclear Regulatory Commission (NRC, or the Commission) staff identified a number of issues with these early proposals and in subsequent proposals made in 2009, and was unable to approve H* amendments on a permanent basis pending resolution of these issues. The staff found it did have a sufficient basis to approve H* amendments on an interim (temporary) basis, based on the relatively limited extent of cracking existing in the lower tubesheet region at the time the interim amendments were approved. The technical basis for approving the interim amendments is provided in detail in the staff's safety evaluations (SEs) accompanying issuance of these amendments. The staff recently approved similar permanent H* amendments for other plants with Model 44F SGs, such as Turkey Point Nuclear Generating Station, Units 3 and 4 (Reference 4).

3.0 REGULATORY EVALUATION

The following explains the applicability of general design criteria (GDC) for IP2. The construction permit for IP2 was issued by the Atomic Energy Commission (AEC) on October 14, 1966, and the operating license was issued on September 28, 1973. The plant GDC are discussed in the Updated Final Safety Analysis Report (UFSAR) Chapter 1.3, "General Design Criteria," with more details given in the applicable UFSAR sections. The AEC published the final rule that added Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971, with the rule effective on May 21, 1971.

In accordance with a staff requirements memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated

September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the Appendix A GDC to plants with construction permits issued prior to May 21, 1971. Therefore, the GDC which constitute the licensing bases for IP2 are those in the UFSAR.

As discussed in the UFSAR, the licensee for IP2 has made some changes to the facility over the life of the unit that committed to some of the GDCs from 10 CFR Part 50, Appendix A. The extent to which the Appendix A GDC have been invoked can be found in specific sections of the UFSAR and in other IP2 licensing basis documentation, such as license amendments.

The SG tubes are part of the reactor coolant pressure boundary (RCPB) and isolate fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis. The GDC in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 provide regulatory requirements that state the RCPB shall have "an extremely low probability of abnormal leakage ... and of gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDCs 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing...to assess...structural and leaktight integrity" (GDC 32). IP2 received a construction permit prior to May 21, 1971, which is the date the GDC in Appendix A of 10 CFR Part 50 became effective. Although the plant is exempt from the current GDC, the licensee states it is in compliance with the 1967 GDC that were in effect when IP2 was licensed and discusses how IP2 meets each of these GDC in Section 4.1.3 of the Updated Final Safety Analysis Report. A review of the 1967 GDC shows that the GDC applicable to the RCPB and SG are comparable to the requirements of the current GDC.

10 CFR, Part 50, Section 50.55a(c) specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), except as provided in 10 CFR 50.55a(c)(2), (3), and (4). Section 50.55a further requires that throughout the service life of pressurized water reactor (PWR) facilities like IP2, ASME Code Class 1 components must meet the Section XI requirements of the ASME Code to the extent practical, except for design and access provisions and pre-service examination requirements. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to in-service inspection of SG tubing are augmented by additional requirements in the TSs.

10 CFR 50.36, "Technical specifications," includes requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

10 CFR 50.36(c)(5), "Administrative controls," includes "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." Programs established by the licensee to operate the facility in a safe manner including the SG program, are listed in the administrative controls section of the TSs. For IP2, the requirements for performing SG tube inspections and repair are in TS 5.5.7, while the requirements for reporting the SG tube inspections and repair are in TS 5.6.7.

The TS for all PWR plants require that an SG program be established and implemented to ensure that SG tube integrity is maintained. For IP2, SG tube integrity is maintained by meeting the performance criteria specified in TS 5.5.7.b for structural and leakage integrity, consistent with the plant design and licensing basis. TS 5.5.7.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. TS 5.5.7.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube and that may satisfy the applicable tube repair criteria. The applicable tube repair criteria, specified in TS 5.5.7.c, are that tubes found during inservice inspection that contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, unless the tubes are permitted to remain in service through application of alternate repair criteria provided in TS 5.5.7.c, such as is being proposed for IP2. The use of the proposed alternate repair criteria does not impact the integrity of the SG tubes and, therefore, the SG tubes still meet the design requirements and the requirements for Class 1 components in Section III of the ASME Code.

The plant TS 3.4.13 also includes a limit on operational primary-to-secondary leakage, beyond which the plant must be promptly shutdown. Should a flaw exceeding the tube repair limit not be detected during the periodic tube surveillance required by the plant TSs, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired, consistent with the design and licensing bases. The proposed amendment would reduce the LCO limit on primary to secondary leakage from the current value of 150 gallons per day (gpd) to a more restrictive value of 85 gpd.

As part of a plant's licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBAs), such as a SG tube rupture and a main steam line break (MSLB). These analyses consider primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of 10 CFR Part 50.67 or 10 CFR Part 100.11 for offsite doses; GDC 19 for control room operator doses (or some fraction thereof as appropriate to the accident); or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analyses for IP2 are being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The proposed changes maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated DBAs for SG tubes.

4.0 TECHNICAL EVALUATION

Proposed Changes to the TS

The current TSs are shown below with the proposed changes. The proposed changes are shown in markup form for clarity, with additions shown in bold text and deletions shown with strikethrough text.

The TS 3.4.13 is to be revised to implement the SG tube alternate repair criteria. The reactor coolant system operational leakage limit is being changed from 150 gallons per day to 85 gallons per day.

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE, and

d. ~~150~~ **85** gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: [No change/not shown]

ACTIONS: [No change/not shown]

SURVEILLENCE REQUIREMENTS

SR 3.4.13.1 [No change/not shown]

SR 3.4.13.2 Verify primary to secondary LEAKAGE is ≤~~150~~ **85** gallons per day through any one SG.

[SR 3.4.13.2 notes and frequency requirements are unchanged/not shown]

The TS 5.5.7.c is to be revised to implement the SG tube alternate repair criteria. Wording is being added to specify that the portion of the tube below 18.9 inches (H* value) from the TTS is excluded from plugging.

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following SG tube alternate plugging criteria shall be applied as an alternative to the preceding criteria.

Tubes with service-induced flaws located greater than 18.9 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 18.9 inches below the top of the tubesheet shall be plugged upon detection.

The TS 5.5.7.d is being revised to implement the SG tube alternate repair criteria. Wording is being added to specify that the portion of the tube below 18.9 inches (H* value) from the TTS is excluded from inspection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from **18.9 inches below the top of the tubesheet on the hot leg side to 18.9 inches below the top of the tubesheet on the cold leg side,** ~~the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet,~~ and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

Additional reporting requirements are being included as TS 5.6.7.i, j and k, which will establish the appropriate reporting criteria for the tubes that require plugging under the proposed alternate repair criteria and a quantification of the operational and accident-induced leakage that could potentially be attributable to the uninspected region of the SG tubes.

5.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.7, SG Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in situ testing,

h. The effective plugging percentage for all plugging in each SG,

i. The primary to secondary leakage rate observed in each SG (if it is not

practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

j. The calculated accident leakage rate from the portion of the tubes below 18.9 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident leakage rate from the most limiting accident is less than 1.75 times the maximum primary to secondary leakage rate, the report should describe how it was determined, and

k. The results of monitoring for tube displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

4.1 Technical Evaluation of H* Alternate Repair Criteria

The T/TS joints are part of the pressure boundary between the primary and secondary systems. Each T/TS joint consists of the tube, which is hydraulically expanded against the bore of the tubesheet, the T/TS weld located at the tube end, and the tubesheet. The joints were designed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, as welded joints, not as friction joints. The T/TS welds were designed to transmit the tube end-cap pressure loads, during normal operating and design-basis-accident (DBA) conditions, from the tubes to the tubesheet with no credit taken for the friction developed between the hydraulically expanded tube and the tubesheet. In addition, the welds serve to make the joints leak tight.

This design basis is a conservative representation of how the T/TS joints actually work, since it conservatively ignores the role of friction between the tube and tubesheet in reducing the tube end-cap load that is transmitted to the T/TS weld. The initial hydraulic expansion of the tubes against the tubesheet produces an "interference fit" between the tubes and the tubesheet; thus, producing a residual contact pressure between the tubes and tubesheet, which acts normally to the outer surface of the tubes and the inner surface of the tubesheet boreholes. Additional contact pressure between the tubes and tubesheet is induced by operational conditions as discussed in detail below. The amount of friction force that can be developed between the outer tube surface and the inner surface of the tubesheet bore is a direct function of the contact pressure between the tube and tubesheet, multiplied by the applicable coefficient of friction.

To support the proposed TS changes, the licensee's contractor, Westinghouse, has defined a parameter called H*. The parameter H* is the distance below the TTS over which sufficient frictional force, with acceptable safety margins, can be developed between each tube and the tubesheet to prevent significant slippage or pullout of the tube from the tubesheet, assuming the tube is fully severed at the H* distance below the TTS. The analyses used to define the H* parameter assumed tube end-cap pressure loads associated with normal operating and DBA conditions. For IP2, the proposed H* distance is 18.9 inches. Given that the frictional force developed in the T/TS joint over the H* distance is sufficient to resist the tube end-cap pressure loads, it is the licensee's and Westinghouse's position that the length of tubing between the H* distance and the T/TS weld is not needed to resist any portion of the tube end-cap pressure loads. Thus, the licensee is proposing to change the TS to not require inspection of the tubes

below the H* distance and to exclude tube flaws located below the H* distance (including flaws in the T/TS weld) from the application of the TS tube repair criteria. Under these changes, the T/TS joint would now be treated as a friction joint extending from the TTS to the H* distance below the TTS for purposes of evaluating the structural and leakage integrity of the joint. The regulatory standard by which the NRC staff has evaluated the subject license amendment is that the amended TSs should continue to ensure that tube integrity will be maintained, consistent with the current design and licensing basis. This includes maintaining structural safety margins consistent with the structural performance criteria in TS 5.5.7.b.1 and the design basis, as is discussed in Section 4.2.1.1 below. In addition, this includes limiting the potential for accident-induced primary-to-secondary leakage to values not exceeding the accident-induced leakage performance criteria in TS 5.5.7.b.2, which are consistent with values assumed in the licensing basis accident analyses. Maintaining tube integrity in this manner ensures that the amended TS comply with all applicable regulations. The staff's evaluation of joint structural integrity and accident-induced leakage integrity is discussed in Sections 4.2.1 and 4.2.2 of this safety evaluation, respectively.

4.2.1 Joint Structural Integrity

4.2.1.1 Acceptance Criteria

Westinghouse has conducted extensive analyses to establish the necessary H* distance to resist pullout under normal operating and DBA conditions. Based on the physical geometry of the SG tubesheet, the NRC staff finds that pullout is the structural failure mode of interest, since the tubes are radially constrained against axial rupture by the presence of the tubesheet. The axial force that could produce pullout derives from the end-cap pressure loads, due to the primary-to-secondary pressure differentials associated with normal operating and DBA conditions. Westinghouse determined the needed H* distance on the basis of maintaining a factor of three against pullout under normal operating conditions and a factor of 1.4 against pullout under DBA conditions. The NRC staff finds that these are the appropriate safety factors to apply to demonstrate structural integrity, because they are consistent with the safety factors embodied in the structural integrity performance criteria in TS 5.5.7.b.1 and with the design basis; namely, the stress limit criteria in the ASME Code, Section III.

The above approach equates tube pullout to gross structural failure, which is conservative. Should the pullout load be exceeded, tube slippage would generally be limited by the presence of adjacent tubes and support structures, such that the tube would not be expected to pull out of the tubesheet.

In Reference1, the licensee committed to monitor for tube slippage as part of the SG inspection program. Under the proposed license amendment, TS 5.6.7.k will require that the results of slippage monitoring be included as part of the SG tube inspection report required by TS 5.6.7. In addition, TS 5.6.7.j will require that should slippage be discovered, the implications of the discovery and corrective action shall be included in the report. The NRC staff finds that slippage is not expected to occur for the reasons discussed in this SE. However, in the unexpected event it should occur, it will be important to understand why it occurred so that the need for corrective action can be evaluated. Therefore, the staff concludes the commitment to monitor for slippage and the accompanying reporting requirements are acceptable.

4.2.1.2 3-D Finite Element Analysis

A detailed 3-D finite element analysis (FEA) of the lower SG assembly (consisting of the lower portion of the SG shell, the tubesheet, the channel head, and the divider plate separating the hot- and cold-leg inlet plenums inside the channel head) was performed to calculate tubesheet displacements due to primary pressure acting on the primary face of the tubesheet and SG channel head; secondary pressure acting on the secondary face of the tubesheet and SG shell; and the temperature distribution throughout the entire lower SG assembly. The calculated tubesheet displacements were used as input to the T/TS interaction analysis evaluated in Section 4.2.1.3 below.

The tubesheet boreholes were not explicitly modeled. Instead, the tubesheet was modeled as a solid structure with equivalent material property values selected such that the solid model exhibited the same stiffness properties as the actual perforated tubesheet. This is a classical approach for analyzing perforated plates that the NRC staff finds acceptable.

Two versions of the 3-D FEA model were used to support the subject request for a license amendment: A "reference model" (Reference 5), which was submitted to support a previous request for a permanent H* amendment for Turkey Point Nuclear Generating Station, Units 3 and 4, and a "revised model" (Reference 6), which was submitted in support of the subject IP2 permanent H* amendment. The reference 3-D FEA model was used to provide displacement input to the thick shell T/TS interaction model described in Section 4.2.1.3.1 below. The revised 3-D FEA model was used to provide displacement input to the square cell T/TS interaction model described in Section 4.2.1.3.2 below.

The revised 3-D FEA model employs a revised mesh near the plane of symmetry (perpendicular to the divider plate) to be consistent with the geometry of the square cell model, such that the displacement output from the 3-D FEA model can be applied directly to the edges of the square cell model. In addition, the mesh near the TTS was enhanced to accommodate high temperature gradients in this area during normal operating conditions. This allowed the temperature distributions throughout the lower SG assembly, including the tubesheet region, to be calculated directly in the 3-D FEA from the assumed plant temperature conditions (e.g., from the assumed primary and secondary water temperatures), for each operating condition. The NRC staff finds this is a more realistic approach (relative to the reference analysis where a linear temperature distribution was assumed to exist through the tubesheet thickness and an adjustment factor was applied to the H* calculations for normal operating conditions) to account for the actual temperature distribution in the tubesheet, based on sensitivity analyses.

Some non-U.S. reactor units have experienced cracks in the weld between the divider plate and the stub runner attachment on the bottom of the tubesheet. If cracks in the divider plate weld ultimately caused the divider plate to become disconnected from the tubesheet, vertical and radial tubesheet displacements under operational conditions could be significantly increased, relative to those for an intact divider plate weld. Although the industry believes that there is little likelihood that cracks such as those seen in non-U.S. reactor units could cause a failure of the divider plate weld, the 3-D FEA conservatively considered both the case of an intact divider plate weld and a detached divider plate weld, to ensure a conservative analysis. The case of a detached divider plate weld was found to produce the most limiting H* values. In the reference analyses (Reference 5), a factor was applied to the 3-D FEA results to account for a non-functional divider plate, based on earlier sensitivity studies. The revised 3-D FEA model

assumes the upper 5 inches of the divider plate to be non-existent. The NRC staff finds this further improves the accuracy of the 3-D FEA for the assumed condition of a non-functional divider plate.

4.2.1.3 T/TS Interaction Model

4.2.1.3.1 Thick Shell Model

The resistance to tube pullout is the axial friction force developed between the expanded tube and the tubesheet over the H* distance. The friction force is a function of the radial contact pressure between the expanded tube and the tubesheet. In the reference analysis (Reference 5), Westinghouse used classical thick-shell equations to model the interaction effects between the tubes and tubesheet under various pressure and temperature conditions for purposes of calculating contact pressure (T/TS interaction model). Calculated displacements from the 3-D FEA of the lower tubesheet assembly (see section 4.2.1.2 above), were applied to the thick shell model as input to account for 1) the increment of tubesheet bore diameter change caused by the primary pressure acting on the primary face of the tubesheet and SG channel head, 2) the secondary pressure acting on the secondary face of the tubesheet and SG shell, and 3) the temperature distribution throughout the entire lower SG assembly. However, the tubesheet bore diameter change from the 3-D FEA tended to be non-uniform (eccentric) around the bore circumference. The thick shell equations used in the T/TS interaction model are axisymmetric. Thus, the non-uniform diameter change from the 3-D finite element analyses had to be adjusted to an equivalent uniform value before it could be used as input to the T/TS interaction analysis.

A 2-D plane stress finite element model was used to define a relationship for determining a uniform diameter change that would produce the same change to the average T/TS contact pressure as the actual non-uniform diameter changes from the 3-D finite element analyses.

In Reference 7, Westinghouse identified a difficultly in applying this relationship to Model D5 SGs under Main Steam Line Break (MSLB) conditions. In reviewing the reasons for this difficulty, the NRC staff developed questions related to the conservatism of the relationship and whether the tubesheet bore displacement eccentricities were sufficiently limited to ensure that T/TS contact was maintained around the entire tube circumference. This concern was applicable to all SG models with alloy 600 TT tubing. In Reference 8, the NRC staff documented a list of questions that would need to be addressed satisfactorily before the staff would be able to approve a permanent H* amendment. These questions related to the technical justification for the eccentricity adjustment, the distribution of contact pressure around the tube circumference, and a new FEA model under development by Westinghouse, to address the aforementioned issue encountered with the Model D5 SGs.

On June 14 and 15, 2010, the NRC staff conducted an audit at the Westinghouse Waltz Mill Site (Reference 9). The purpose of the audit was to gain a better understanding of the H* analysis pertaining to eccentricity, to review draft responses to the NRC staff's questions in Reference 8, and to determine which documents would need to be provided on the docket to support any future requests for a permanent H* amendment. Based on the audit, including review of pertinent draft responses to Reference 8, the staff concluded that eccentricity did not appear to be a significant variable affecting either average T/TS contact pressure at a given elevation or calculated values of H*. The staff found that average contact pressure at a given elevation is

primarily a function of average bore diameter change at that elevation associated with the pressure and temperature loading of the tubesheet. Accordingly, the staff concluded that no adjustment of computed average bore diameter change considered in the thick shell model is needed to account for eccentricities computed by the 3-D FEA. The material reviewed during the audit revealed that computed H* values from the reference analyses continued to be conservative when the eccentricity adjustment factor is not applied.

4.2.1.3.2 Square Cell Model

The square cell model is a 2-D plane stress FEA model of a single square cell of the tubesheet with a borehole in the middle and each side of the cell measuring one tube pitch in length. Displacement boundary conditions are applied at the edges of the cell, based on the displacement data from the revised 3-D FEA model. The model includes the tube cross-section inside the bore. Displacement compatibility between the tube outer surface and bore inner surface is enforced except at locations where a gap between the tube and bore tries to occur.

The square cell model was originally developed in response to the above-mentioned difficulty encountered when applying the eccentricity adjustment to Model D5 SGs T/TS interaction analysis under MSLB conditions using the thick shell model. Early results with this model indicated significant differences compared to the thick shell model, irrespective of whether the eccentricity adjustment was applied to the thick shell model. The square cell model revealed a fundamental problem with how the results of the 3-D FEA model of the lower SG assembly were being applied to the tubesheet bore surfaces in the thick shell model. As discussed in Section 4.2.1.2 above, the perforated tubesheet is modeled in the 3-D FEA model as a solid plate whose material properties were selected such that the gross stiffness of the solid plate is equivalent to that of a perforated plate under the primary-to-secondary pressure acting across the thickness of the plate. This approach tends to smooth out the distribution of tubesheet displacements as a function of radial and circumferential location in the tubesheet, and ignores local variations of the displacements at the actual bore locations. These smoothed out displacements from the 3-D FEA results were the displacements applied to the bore surface locations in the thick shell model. The square cell model provides a means for post-processing the 3-D FEA results to account for localized variations of tubesheet displacement at the bore locations, as part of the T/TS interaction analysis. Based on these findings, square cell models were developed for all SG model types, including the Model 44F SGs at IP 2.

The square cell model is applied to nine different elevations, from the top to the bottom of the tubesheet, for each tube and loading case analyzed. The square cell slices at each elevation are assumed to act independently of one another. T/TS contact pressure results from each of the nine slices are used to define the contact pressure distribution from the top to the bottom of the tubesheet.

The resisting force to the applied end-cap load, which is developed over each incremental axial distance from the TTS, is the average contact pressure over that incremental axial distance multiplied by the tubesheet bore surface area (equal to the tube outer diameter surface area), and then multiplied by the coefficient of friction. The NRC staff reviewed the coefficient of friction used in the analysis and found it a reasonable lower bound (conservative) estimate. The H* distance for each tube was determined by integrating the incremental friction forces from the TTS to the distance below the TTS where the friction force integral equaled the applied end-cap load multiplied by the appropriate safety factor as discussed in Section 4.2.1.1. The square

cell model assumes as an initial condition that each tube is fully expanded against the tubesheet bore such that the outer tube surface is in contact with the inner surface of the tubesheet bore under room temperature, atmospheric pressure conditions, with zero residual contact pressure from the hydraulic expansion process. The staff finds the assumption of zero residual contact pressure in all tubes to be a conservative assumption.

The limiting tube locations, in terms of H*, were determined by the reference analysis to lie along the plane of symmetry perpendicular to the divider plate. The outer edges of the square cell model conform to the revised mesh pattern along this plane of symmetry in the 3-D FEA model of the lower SG assembly, as discussed in Section 4.2.1.2. Because the tubesheet boreholes were not explicitly modeled in the 3-D FEA, only the average displacements along each side of the square cell are known from the 3-D FEA. Three different assumptions for applying displacement boundary conditions to the edges of the square cell model were considered, to allow for a range of possibilities about how local displacements might vary along the length of each side. The most conservative assumption, in terms of maximizing the calculated H* distance, was to apply the average transverse displacement uniformly over the length of each edge of the square cell.

Primary pressure acting on the inside tube surface and crevice pressure[1] acting on both the tube outside surface and tubesheet bore surface are not modeled directly, as in the case of the thick shell model. Instead, the primary side (inside) of the tube is assumed to have a pressure equal to the primary pressure minus the crevice pressure. Note the crevice pressure varies as a function of the elevation being analyzed, as discussed in Section 4.2.1.4.

The NRC staff concludes that the square cell model provides for improved compatibility between the 3-D FEA model of the lower SG assembly and the T/TS interaction model, more realistic and accurate treatment of the T/TS joint geometry, and added conservatism relative to the thick shell model used in the reference analyses.

4.2.1.4 Crevice Pressure Evaluation

The H* analyses postulate that interstitial spaces exist between the hydraulically expanded tubes and tubesheet bore surfaces. These interstitial spaces are assumed to act as crevices between the tubes and the tubesheet bore surfaces. The NRC staff finds that the assumption of crevices is conservative since the pressure inside the crevices acts to push against both the tube and the tubesheet bore surfaces, thus reducing contact pressure between the tubes and tubesheet.

For tubes that do not contain through-wall flaws within the thickness of the tubesheet, the pressure inside the crevice is assumed equal to the secondary system pressure. For tubes that contain through-wall flaws within the thickness of the tubesheet, a leak path is assumed to exist from the primary coolant inside the tube, through the flaw, and up the crevice to the secondary system. Hydraulic tests were performed on several tube specimens that were hydraulically expanded against tubesheet collars, to evaluate the distribution of the crevice pressure from a location where through-wall holes had been drilled in the tubes, to the top of the crevice

[1] Although the tubes are in tight contact with the tubesheet bore surfaces, surface roughness effects are conservatively assumed to create interstitial spaces, which are effectively crevices, between these surfaces. See Section 4.2.1.4 for more information.

location. The T/TS collar specimens were instrumented at several axial locations to permit direct measurement of the crevice pressures. Tests were run for both normal operating and MSLB pressure and temperature conditions.

The NRC staff finds that the use of the drilled holes, rather than through-wall cracks, is conservative since it eliminates any pressure drop between the inside of the tube and the crevice at the hole location. This maximizes the pressure in the crevice at all elevations, thus reducing contact pressure between the tubes and tubesheet.

The crevice pressure data from these tests were used to develop a crevice pressure distribution as a function of normalized distance between the TTS and the H* distance below the TTS where the tube is assumed to be severed. These distributions were used to determine the appropriate crevice pressure at each axial location of the T/TS interaction model and are concluded to be acceptable for this purpose by the NRC staff.

Because the crevice pressure distribution is assumed to extend from the H* location, where crevice pressure is assumed to equal primary pressure, to the TTS, where crevice pressure equals secondary pressure, an initial estimate as to the H* location must be made before iteratively solving for H* using the T/TS interaction model and 3-D finite element model. The resulting new H* estimate becomes the initial estimate for the next H* iteration.

4.2.1.5 <u>H* Calculation Process</u>

The calculation of H* consists of the following steps for each loading case considered:

1. Perform initial H* estimate (mean H* estimate) using the T/TS interaction model and 3-D FEA models, assuming nominal geometric and material properties, and that the tube is severed at the bottom of the tubesheet for purposes of defining the contact pressure distribution over the length of the T/TS crevice. Two sets of mean H* estimates are pertinent to the proposed H* value, mean H* estimates calculated with the reference T/TS interaction and 3-D FEA models, and mean H* estimates calculated with the square cell T/TS interaction and revised 3-D FEA models. The mean H* estimate from the revised analysis (for the worst sector of tubes) is approximately 4 inches longer than the mean H* distance from the reference analysis, for the most limiting case of normal operating conditions with the associated safety factor of 3 (as evaluated in section 4.2.1.1). The NRC staff finds that the difference in mean H* estimates between the reference analysis and the revised analysis is predominantly due to the improved post-processing of the 3-D FEA model displacements for application to the T/TS interaction model.

2. In the reference analysis (Reference 5), a 0.3-inch adjustment was added to the initial H* estimate to account for uncertainty in the bottom of the tube expansion transition (BET) location relative to the TTS, based on an uncertainty analysis on the BET for Model F SGs conducted by Westinghouse. This adjustment is not included in the revised H* analysis accompanying the subject amendment request, as discussed and evaluated in Section 4.2.1.5.1 of this SE. In the reference analysis (Reference 5) for normal operating conditions only, an additional adjustment was added to the initial H* estimate to correct for the actual temperature distribution in the tubesheet compared to the linear distribution assumed in the reference 3-D FEA analysis. This adjustment is no

longer necessary, as discussed in Section 4.2.1.2, since the temperature distributions throughout the tubesheet were calculated directly in the revised 3-D FEA supporting the current request for an H* amendment.

3. Steps 1 through 3 yield a so-called "mean" estimate of H*, which is deterministically based. Step 4 involves a probabilistic analysis of the potential variability of H*, relative to the mean estimate, associated with the potential variability of key input parameters for the H* analyses. This leads to a "probabilistic" estimate of H*, which includes the mean estimate. The NRC staff's evaluation of the probabilistic analysis is provided in Section 4.2.1.6 and 4.2.1.7 of this SE.

4. Add a crevice pressure adjustment to the probabilistic estimate of H* to account for the crevice pressure distribution that results from the tube being severed at the final H* value, rather than at the bottom of the tubesheet. This step is discussed and evaluated in Section 4.2.1.5.2 of this SE.

5. Add an additional adjustment to the probabilistic estimate of H*, to account for the Poisson contraction of the tube radius due to the axial end-cap load acting on each tube. This step has been added to the H* calculation process since the reference analysis, to support the subject amendment request. This step is discussed and evaluated in Section 4.2.1.5.3 of this SE.

4.2.1.5.1 BET Considerations

The diameter of each tube transitions from its fully expanded value to its unexpanded value near the TTS. The BET region is located a short distance below the TTS, to avoid any potential for over-expanding the tube above the TTS. In the reference H* analysis (Reference 5), a 0.3-inch adjustment was added to the mean H* estimate to account for the BET location being below the TTS, based on an earlier survey of BET distances conducted by Westinghouse. This adjustment was necessary since the reference analysis did not explicitly account for the lack of contact between the tube and tubesheet over the BET distance.

Measurements of BET locations, based on eddy current testing, have subsequently been performed for all tubes at IP2. These measurements confirm that the original 0.3 inch BET assumption is bounding on a 95-percentile basis for SGs 1, 2, and 3, and the 95-percentile value for SG 4 is 0.31 inches. The maximum BET value at IP2 is 0.74 inches (Reference 10).

In considering the acceptability of the 0.3 inch BET assumption, the NRC staff determined that the most recent H* analyses using the square cell T/TS interaction model (Reference 6) has made the need for a BET adjustment unnecessary, as the square cell Model shows a loss of contact pressure over a distance from the TTS that is greater than the possible variation in the BET location. This observation applies to all radial locations with local mean H* values within one inch of the maximum, mean H* value. The loss of contact pressure at the TTS shown in the square cell model (which is unrelated to BET location) is compensated for by a steeper contact pressure gradient than previously shown in the thick shell model H* analysis. The NRC staff concludes that the proposed H* value adequately accounts for the range of BET values at IP2.

4.2.1.5.2 Crevice Pressure Adjustment

As discussed in Section 4.2.1.5, steps 1 through 3 of the H* calculation process leading to a probabilistic H* estimate are performed with the assumption that the tube is severed at the bottom of the tubesheet for purposes of calculating the distribution of crevice pressure as a function of elevation. If the tube is assumed to be severed at the initially computed H* distance and steps 1 through 3 repeated, a new H* may be calculated which will be incrementally larger than the first estimate. This process may be repeated until the change in H* becomes small (convergence). Sensitivity analyses conducted with the thick shell model showed that the delta between the initial H* estimate and final (converged) estimate is a function of the initial estimate for the tube in question. This delta (i.e., the crevice pressure adjustment referred to in step 5 of Section 4.2.1.5) was plotted as a function of the initial H* estimate for the limiting loading case and tube radial location. Although the sensitivity study was conducted with the thick shell model, the deltas from this study were used in the Reference 6 (square cell model) analysis to make the crevice pressure adjustment to H*, as updating this sensitivity study would have been very resource intensive, requiring many new 2-D FEA square cell runs.

In response to an NRC staff question as to whether it is conservative to rely on the existing sensitivity study as opposed to updating it to reflect the square cell model, Westinghouse submitted an analysis (Reference 11) demonstrating that if the sensitivity study were updated, it would show that the crevice pressure adjustment for H* is negative, not positive, as is shown by the existing study. This is because the square cell model predicts a much longer zone (approximately 6 inches) of reduced T/TS contact pressure below the TTS than does the thick shell model. Therefore, the crevice pressure must reduce from primary side pressure (at the iterative H* location) to secondary side pressure about six inches below the TTS. This leads to higher predicted pressure differentials across the tube wall over the iterative H* distance than exists during the initial iteration when crevice pressure is assumed to vary from primary pressure at the bottom of the tubesheet to secondary pressure at the TTS. Based on its review of the Westinghouse analysis, the NRC staff concludes that the positive crevice pressure adjustment to H* in the Reference 10 analysis, which is based on the existing sensitivity study, is conservative and that an updated sensitivity analysis based on use of the square cell model would show that a negative adjustment can actually be justified. Thus, the staff concludes the crevice pressure adjustment performed in support of the proposed H* amendment is conservative and acceptable.

4.2.1.5.3 Poisson Contraction Effect

The axial end-cap load acting on each tube is equal to the primary-to-secondary pressure differential multiplied by the tube cross-sectional area. For purposes of resisting tube pullout under normal and accident conditions, the end-cap loads used in the H* analyses are based on the tubesheet bore diameter, which the NRC staff finds to be a conservative assumption. The axial end-cap load tends to stretch the tube in the axial direction, but causes a slight contraction in the tube radius due to the Poisson's Ratio effect. This effect, by itself, tends to reduce the T/TS contact pressure and, thus, to increase the H* distance. The axial end-cap force is resisted by the axial friction force developed at the T/TS joint. Thus, the axial end-cap force begins to decrease with increasing distance into the tubesheet, reaching zero at a location before the H* distance is reached. This is because the H* distances are intended to resist pullout under the end-cap loads with the appropriate factors of safety applied as discussed in Section 4.2.1.1.

A simplified approach was taken to account for the Poisson radial contraction effect. First, thick shell equations were used to estimate the reduction in contact pressure associated with application of the full end-cap load, assuming none of this end-cap load had been reduced by the tubesheet. The T/TS contact pressure distributions determined in the H* calculation process (in Section 4.2.1.5) were reduced by this amount. Second, the friction force associated with these reduced T/TS contact pressures were integrated with distance into the tubesheet, and the length of engagement necessary to reduce one times the end-cap loading (i.e., no safety factor applied) was determined. At this distance (termed attenuation distance by Westinghouse), the entire end-cap loading was assumed to have been reduced by the tubesheet, and the axial load in the tube below the attenuation distance was assumed to be zero. Thus, the T/TS contact pressures below the attenuation distance were assumed to be unaffected by the Poisson radial contraction effect. Finally, a revised H* distance was calculated, where the T/TS contact pressures from Section 4.2.1.5 were reduced only over the attenuation distance.

The NRC staff finds the simplified approach for calculating the H* adjustment for the Poisson contraction effect to contain significant conservatism relative to a more detailed approach. Regarding the safety factor of unity assumption, Westinghouse states that it is unrealistic to apply a safety factor to a physical effect such as Poisson's ratio. The staff has not reached a conclusion on this point. However, irrespective of whether a safety factor is applied to the Poisson's contraction effect (consistent with Section 4.2.1.1 above), the staff concludes there is ample conservatism embodied in the proposed H* distance to accommodate the difference.

4.2.1.6 Acceptance Standard - Probabilistic Analysis

The purpose of the probabilistic analysis is to develop an H* distance that ensures with a probability of 0.95 that the population of tubes will retain margins against pullout consistent with criteria evaluated in Section 4.2.1.1 of this SE, assuming all tubes to be completely severed at their H* distance. The NRC finds this probabilistic acceptance standard is consistent with what the staff has approved previously and is acceptable. For example, the upper voltage limit for the voltage based tube repair criteria in NRC Generic Letter 95-05 (Reference 12) employs a consistent criterion. The staff also notes that use of the 0.95 probability criterion ensures that the probability of pullout of one or more tubes under normal operating conditions and conditional probability of pullout under accident conditions is well within tube rupture probabilities that have been considered in probabilistic risk assessments (References 13 and 14).

In terms of the confidence level that should be attached to the 0.95 probability acceptance standard, it is industry practice for SG tube integrity evaluations, as embodied in industry guidelines, to calculate such probabilities at a 50 percent confidence level. The NRC staff has been encouraging the industry to revise its guidelines to call for calculating such probabilities at a 95 percent confidence level when performing operational assessments and a 50 percent confidence level when performing condition monitoring (Reference 15). In the meantime, the calculated H* distances supporting the amendment currently being requested have been evaluated at the 95 percent confidence level, as recommended by the staff.

Another issue relating to the acceptance standard for the probabilistic analysis is determining what population of tubes needs to be analyzed. For accidents such as MSLB or feed line break, the NRC staff and licensee find that the tube population in the faulted SG is of interest, since only that SG experiences a large increase in the primary-to-secondary pressure differential.

However, normal operating conditions were found to be the most limiting in terms of meeting the tube pullout margins in Section 4.2.1.1. For normal operating conditions, tubes in all SGs at the plant are subject to the same pressures and temperatures. Although there is not a consensus between the staff and industry on which population needs to be considered in the probabilistic analysis for normal operating conditions, the calculated H* distances for normal operating conditions supporting the requested amendment are 0.95 probability/95 percent confidence estimates based on the entire tube population for the plant, consistent with the staff's recommendation.

Based on the above, the NRC staff concludes that the proposed H* distance in the subject license amendment request is based on acceptable probabilistic acceptance standards evaluated at acceptable confidence levels.

4.2.1.7 Probabilistic Analyses

4.2.1.7.1 Reference Analyses

Sensitivity studies were conducted during the reference analyses (Reference 5) and demonstrated that H* was highly sensitive to the potential variability of the coefficients of thermal expansion (CTE) for the Alloy 600 tubing material and the SA-508 Class 2a tubesheet material. Given that no credit was taken in the reference H* analyses (Reference 5) for residual contact pressure associated with the tube hydraulic expansion process[2], the sensitivity of H* to other geometry and material input parameters was judged by Westinghouse to be inconsequential and were ignored, with the exception of Young's modulus of elasticity for the tube and tubesheet materials. Although the Young's modulus parameters were included in the reference H* analyses sensitivity studies, these parameters were found to have a weak effect on the computed H*. Based on its review of the analysis models and its engineering judgment, the NRC staff finds that the sensitivity studies adequately capture the input parameters which may significantly affect the value of H*. This conclusion is based, in part, on no credit being taken for residual contact pressure during the reference H* analyses.

These sensitivity studies were used to develop influence curves describing the change in H*, relative to the mean H* value estimate (see Section 4.2.1.5), as a function of the variability of each CTE parameter and Young's Modulus parameter, relative to the mean values of CTE and Young's Modulus. Separate influence curves were developed for each of the four input parameters. The sensitivity studies showed that of the four input parameters, only the CTE parameters for the tube and tubesheet material interacted with one another. A combined set of influence curves containing this interaction effect was also created.

Two types of probabilistic analyses were performed independently in the reference analyses. One was a simplified statistical approach utilizing a "square root of the sum of the squares" method and the other was a detailed Monte Carlo sampling approach. The NRC staff's review of the reference analysis relied on the Monte Carlo analysis, which provides the most realistic treatment of uncertainties. The staff reviewed the implementation of probabilistic analyses in the reference analyses and questioned whether the H* influence curves had been conservatively treated. To address this concern, new H* analyses were performed as

2 Residual contact pressures are sensitive to variability of other input parameters.

documented in References 16 and 17. These analyses made direct use of the H* influence curves in a manner the staff finds to be acceptable.

The revised reference analyses in Reference 16 divided the tubes by sector location within the tube bundle and all tubes were assumed to be at the location in their respective sectors where the initial value of H* (based on nominal values of material and geometric input parameters) was at its maximum value for that sector. The H* influence curves discussed above, developed for the most limiting tube location in the tube bundle, were conservatively used for all sectors. The revised reference analyses also addressed a question posed by the NRC staff concerning the appropriate way to sample material properties for the tubesheet, whose properties are unknown but do not vary significantly for a given SG, in contrast to the tubes whose properties tend to vary much more randomly from tube to tube in a given SG. This issue was addressed by a staged sampling process where the tubesheet properties were sampled once and then held fixed, while the tube properties were sampled a number of times equal to the SG tube population. This process was repeated 10,000 times, and the maximum H* value from each repetition was rank ordered. The final H* value was selected from the rank ordering to reflect a 0.95 probability value at the desired level of confidence for a single SG tube population or all SG populations, as appropriate. The staff concludes that this approach addresses the NRC staff's question in a realistic fashion and is acceptable.

4.2.1.7.2 Revised Analyses to Reflect Square Cell and Revised 3-D FEA Models

New Monte Carlo analyses using the square cell model to evaluate the statistical variability of H* due to the CTE variability for the tube and tubesheet materials were not performed. This was because such an approach would have been extremely resource intensive and because a simpler approach involving good approximation was available. The simplified approach involved using the results of the Monte Carlo analyses from the reference analysis, which are based on the thick shell T/TS interaction model, to identify CTE values for the tube and tubesheet associated with the probabilistic H* values near the desired rank ordering. Tube CTE values associated with the upper 10 percent rank order estimates are generally negative variations from the mean value whereas tubesheet CTE values associated with the higher ranking order estimates are generally positive variations from the mean value. For the upper 10 percent of the Monte Carlo results ranking order, a combined uncertainty parameter, "alpha," was defined as the square root of the sum of the squares of the associated tube and tubesheet CTE values for each Monte Carlo sample. Alpha was plotted as a function of the corresponding H* estimate and separately as a function of rank order. Each of these plots exhibited well defined "break lines," representing the locus of maximum H* estimates and maximum rank orders associated with a given value of alpha. From these plots, three paired sets of tube and tubesheet CTE values, located near the break line, were selected. These CTE values were then input to the lower SG assembly 3-D FEA model and the square cell model to yield probabilistic H* estimates which approximate the H* values for these same rank orderings, had a full Monte Carlo been performed with the square cell and revised 3-D FEA models. These H* estimates were then plotted as a function of rank ordering, allowing the interpolation of H* values at the other rank orders. The resulting 95/95 upper bound H* estimate is approximately 7 inches longer than the mean estimate discussed in Section 4.2.1.5. With adjustments for Poisson's contraction (see Section 4.2.1.5.3) and crevice pressure (Section 4.2.1.5.2), the final 95/95 upper bound H* estimate is 18.9 inches, and that is the value selected in the subject amendment request.

The NRC staff considers that the above break line approach to be a very good approximation of what an actual Monte Carlo would show. A perfect approximation would require performing a square cell analysis for each paired set of tube and tubesheet CTE values associated with the top 10 percent of rank orders, and plotting the resulting H* values versus the original rank ordering associated with the CTE couple, the calculated H* values should monotonically increase from rank order to rank order. Westinghouse performed additional square cell analyses with CTE pairs for five consecutive rank orders for both Model D5 and Model F SGs. The results showed deviations from monotonically increasing values of H* with rank order to be on the order of only 0.3 inches for the Model D5 SGs and 0.1 inches for the Model F SGs. The staff considers these deviations to be representative for the Model 44F SGs at IP2. The staff concludes that use of the break line approach adds little imprecision to the probabilistic H* estimates and is acceptable.

4.2.1.8 Coefficient of Thermal Expansion

During operation, a large part of contact pressure in a SG T/TS joint is derived from the difference in CTE between the tube and tubesheet. As discussed in Section 4.2.1.7, the calculated value of H* is highly sensitive to the assumed values of these CTE parameters. However, CTE test data acquired by an NRC contractor, Argonne National Laboratory (ANL), suggested that CTE values might vary substantially from values listed in the ASME Code for design purposes. In Reference 18, the NRC staff highlighted the need for a rigorous technical basis of the CTE values, and their potential variability, to be employed in future H* analyses.

In response, Westinghouse had a subcontractor review the CTE data in question, determine the cause of the variance from the ASME Code CTE values, and provide a summary report (Reference 19). The analysis of the CTE data revealed that the CTE variation with temperature had been developed using a polynomial fit to the raw data, over the full temperature range from 75 °F to 1300 °F. The polynomial fit chosen resulted in mean CTE values that were significantly different from the ASME Code values from 75 °F to about 300 °F. When the raw data was reanalyzed using the locally weighted least squares regression method, the mean CTE values determined were in good agreement with established ASME Code values.

Westinghouse also formed a panel of licensee experts to review the available CTE data in open literature, review the ANL provided CTE data, and perform an extensive CTE testing program on Alloy 600 and SA-508 steel material to supplement the existing database. Two additional sets of CTE test data (different from those addressed in the previous paragraph) had CTE offsets at low temperatures that were not expected. Review of the test data showed that the first test, conducted in a vacuum, had proceeded to a maximum temperature of 1300 °F, which changed the microstructure and the CTE of the steel during decreasing temperature conditions. Because of the altered microstructure, the CTE test data generated in the second test, conducted in air, was also invalidated. Because of the large "dead band" region and the altered microstructure, both data sets were excluded from the final CTE values obtained from the CTE testing program. The test program included multiple material heats to analyze chemistry influence on CTE values and repeat tests on the same samples were performed to analyze for test apparatus influence. Because the tubes are strain hardened when they are expanded into the tubesheet, strain hardened samples were also measured to check for strain hardening influence on CTE values.

The data from the test program was combined with the Argonne National Laboratory data that were found to be acceptable and the data obtained from the open literature search. A statistical analysis of the data uncertainties was performed by comparing deviations to the mean values obtained at the applicable temperatures. The correlation coefficients obtained indicated a good fit to a normal distribution, as expected. Finally, an evaluation of within-heat variability was performed due to increased data scatter at low temperatures. The within-heat variability assessment determined that the increase in data scatter was a testing accuracy limitation that was only present at low temperatures. The CTE report is included as Appendix A to Reference 5.

The testing showed that the nominal ASME Code values for Alloy 600 and SA-508 steel were both conservative relative to the mean values from all the available data. Specifically, the CTE mean value for Alloy 600 was greater than the ASME Code value and the CTE mean value for SA-508 steel was smaller than the ASME Code value. Thus, the H* analyses utilized the ASME Code values as mean values in the H* analyses. The NRC staff finds this to be conservative because it tends to lead to an over-prediction of the expansion of the tubesheet bore and an under-prediction of the expansion of the tube, thereby resulting in an increase in the calculated H* distance. The statistical variances of the CTE parameters from the combined database were utilized in the H* probabilistic analysis.

Based on its review of the Westinghouse CTE program, the NRC staff concludes that the CTE values used in the H* analyses are fully responsive to the concerns stated in Reference 18 and are acceptable.

4.2.2 Leakage Considerations

Operational leakage integrity is assured by monitoring primary-to-secondary leakage relative to the applicable TS Limiting Conditions for Operation in TS 3.4.13, "RCS Operational LEAKAGE." However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during a DBA to exceed the accident leakage performance criteria in TS 5.5.7.b.2, which are based on the leakage values assumed in the plant licensing basis accident analyses.

If a tube is assumed to contain a 100 percent through-wall flaw some distance into the tubesheet, a potential leak path between the primary and secondary systems is introduced between the hydraulically expanded tubing and the tubesheet. The leakage path between the tube and tubesheet has been modeled by the licensee's contractor, Westinghouse, as a crevice consisting of a porous media. Using Darcy's model for flow through a porous media, leak rate is proportional to differential pressure and inversely proportional to flow resistance. Flow resistance is a direct function of viscosity, loss coefficient, and crevice length.

Westinghouse performed leak tests of T/TS joint mockups to establish loss coefficient as a function of contact pressure. A large amount of data scatter, however, precluded quantification of such a correlation. In the absence of such a correlation, Westinghouse has developed a leakage factor relationship between accident induced leak rate and operational leakage rate, where the source of leakage is from flaws located at or below the H* distance.

Using the Darcy model, the leakage factor for a given type accident is the product of four quantities. The first quantity is ratio of the maximum primary-to-secondary pressure difference

during the accident divided by that for normal operating conditions; for IP2; this quantity is 1.75. The second quantity is the ratio of viscosity under normal operating primary water temperature divided by viscosity under the accident condition primary water temperature; for IP2; this quantity is conservatively assumed to be 1. The third quantity is the ratio of crevice length under normal operating conditions to crevice length under accident conditions. This ratio equals 1, if it can be shown that positive contact pressure is maintained along the entire H* distance for both conditions, which is the case for IP2.

The fourth quantity is the ratio of loss coefficient under normal operating conditions to loss coefficient under the accident condition. Although the absolute value of these loss coefficients is not known, Westinghouse has previously assumed that the loss coefficient is constant with contact pressure such that the ratio is equal to 1. The NRC staff considers this a conservative assumption, provided there is a positive contact pressure for both conditions along the entire H* distance and provided that contact pressure increases at each axial location along the H* distance when going from normal operating to accident conditions. The square cell analyses confirmed positive contact pressure along the entire H* distance, but at the tube bundle radius of 2.655 inches (from approximately 1 to 4 inches below the TTS), the average contact pressure during MSLB conditions was calculated to be less than normal operating conditions (by up to almost 200 pounds per square inch). Westinghouse states, in Reference 6, that these contact pressure results are essentially the same, within the accuracy of the analysis, and it is, therefore, acceptable to conclude that the loss coefficient sub-factor is equal to 1. Because the delta between the normal operating and MSLB condition is small and because areas above and below this region in the tubesheet have contact pressures during MSLB conditions that are greater than during normal operating conditions, the NRC staff finds it acceptable to assume the loss coefficient sub-factor is equal to 1.

Leakage factors were calculated for DBAs exhibiting a significant increase in primary-to-secondary pressure differential, including MSLB, locked rotor, and control rod ejection. The design basis MSLB transient was found to exhibit the highest leakage factor, 1.75, meaning that the MSLB transient is expected to result in the largest increase in leakage relative to normal operating conditions.

In Reference 2, the licensee provided a commitment describing how the leakage factor will be used to satisfy TS 5.5.7.a for condition monitoring and TS 5.5.7.b.2 regarding performance criteria for accident induced leakage. All plants receiving NRC approval for a permanent license amendment for H* have committed to using their plant-specific leakage factor developed in the appropriate WCAP similar to what the licensee provided below:

> *Indian Point Unit 2 will apply a factor of 1.75 to the normal operating leakage associated with the tubesheet expansion region in the condition monitoring and operational assessment. Specifically, for the condition monitoring assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 1.75 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment, the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 1.75 and compared to the observed operational leakage. An administrative limit will be established to not exceed the calculated value.*

The licensee's commitment is not imposed as a condition of the plant license, which is consistent with the fact that details of how condition monitoring and operational assessments are performed are generally not included as part of an operating license, including the TSs. Extensive industry guidance on conducting condition monitoring and operational assessments is available as part of the industry NEI 97-06 initiative (Reference 20). The above commitments ensure that plant procedures address the above leakage factor issue consistent with industry guidelines and will be administratively controlled under the licensee's existing commitment management program.

Per the above commitment, the administrative limit for operational primary to secondary leakage can never exceed the current TS 3.4.13 LCO limit for primary to secondary leakage of 150 gpd divided by the 1.75 leakage factor, equaling 85 gpd. Accordingly, the licensee is proposing to revise the LCO limit from 150 gpd to 85 gpd. This change provides added assurance that during a hypothetical DBA, the resulting primary to secondary leakage will be within the accident leakage performance criteria of TS 5.5.7.b.2 (i.e., 150 gpd). The NRC staff finds this proposed change to be acceptable.

Finally, the subject amendment request includes reporting requirements (TS 5.6.7.i and 5.6.7.j) relating to operational leakage existing during the cycle preceding each SG inspection and condition monitoring assessment, and the associated potential for accident induced leakage from the lower portion of the tubesheet below the H* distance. These reporting requirements will allow the NRC staff to monitor how the leakage factor is actually being used, and are acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (79 FR 15147). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the

amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Entergy Nuclear Northeast, LLC, "Proposed License Amendment for Alternate Repair Criteria for Steam Generator Tube Inspection and Repair, Indian Point, Unit Number 2," January 16, 2014 (NRC ADAMS Accession No. ML14027A514).

2. Entergy Nuclear Northeast, LLC, "Indian Point, Unit 2 – Response to Request for Additional Information Regarding the Steam Generator License Amendment Request to Revise Technical Specification for Permanent Alternate Repair Criteria," April 2, 2014 (NRC ADAMS Accession No. ML14107A042).

3. Entergy Nuclear Northeast, LLC, "Correction to Entergy Letter NL-14-045, Indian Point Unit Number 2," April 15, 2014 (NRC ADAMS Accession No. ML14112A460).

4. NRC letter to Florida Power and Light Company, "Turkey Point Nuclear Generating Station Unit Nos. 3 and 4 – Issuance of Amendments Regarding Permanent Alternate Repair Criteria for Steam Generator Tubes," November 5, 2012 (NRC ADAMS Accession No. ML12292A342).

5. Westinghouse Electric Company LLC, WCAP-17091-P (Proprietary) and WCAP-17091-NP (Non-Proprietary), Rev. 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)," June 2009, NRC ADAMS Accession Nos. ML092300061 (Proprietary) and ML092300060 (Nonproprietary).

6. Westinghouse Electric Company LLC, WCAP-17828-P (Proprietary) and WCAP-17828-NP (Non- Proprietary), Rev. 0, "Indian Point Unit 2 H* Alternate Repair Criteria for the Tubesheet Hydraulic Expansion Region (Model 44F – 4-Loop)," January 2014; NRC ADAMS Accession Nos. ML14027A516 (Proprietary) and ML14027A515 (Nonproprietary).

7. Westinghouse Electric Company LLC report, WCAP-17072-P (Proprietary) and WCAP-17072-NP (Nonproprietary), Rev. 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," May 2009, NRC ADAMS Accession Nos. ML101730386/ ML101730393 (Proprietary) and ML101730389 (Nonproprietary).

8. NRC letter to Southern Nuclear Operating Company, "Vogtle Electric Generating Plant, Units 1 and 2, Transmittal of Unresolved Issues Regarding Permanent Alternate Repair Criteria for Steam Generators," November 23, 2009, NRC ADAMS Accession No. ML093030490.

9. NRC memorandum, R. Taylor to G. Kulesa, "Vogtle Electric Generating Plant – Audit of Steam Generator H* Amendment Reference Documents," July 9, 2010, NRC ADAMS Accession No. ML101900227.

10. Westinghouse Electric Company LLC letter, LTR-SGMP-14-22-P (Proprietary) and LTR-SGMP-14-22-NP (Nonproprietary), Rev. 0, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H* at Indian Point Unit 2," March 2014, NRC ADAMS Accession No. ML14107A044 (Proprietary) and ML14107A043 (Nonproprietary).

11. Virginia Electric and Power Company, "Surry Power Station Units 1 and 2 – Response to Request for Additional Information Related to License Amendment Request for Permanent Alternate Repair Criteria for Steam Generator Tube Inspection and Repair," February 14, 2012, NRC ADAMS Accession No. ML12048A676.

12. NRC Generic Letter 95-05, "Voltage Based Alternate Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995, NRC ADAMS Accession No. ML031070113.

13. NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," September 1988.

14. NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," March 1998.

15. NRC Meeting minutes, "Summary of the January 8, 2009, Category 2 Public Meeting with the Nuclear Energy Institute (NEI) and Industry to Discuss Steam Generator Issues," February 6, 2009, NRC ADAMS Accession No. ML090370782.

16. Westinghouse Electric Company LLC letter, LTR-SGMP-09-100-P (Proprietary) and LTR-SGMP-09-100-NP (Nonproprietary) "Response to NRC Request for Additional Information on H*; Model F and D5 Steam Generators," August 12, 2009, NRC ADAMS Accession No. ML101730397 (Proprietary) and ML101730391 (Nonproprietary).

17. SNC letter NL-09-1317, August 28, 2009, transmitting WEC letter LTR-SGMP-09-104-P Attachment "White Paper on Probabilistic Assessment of H*" dated August 13, 2009, NRC ADAMS Accession Nos. ML092450030 (Proprietary) and ML092450029 (Non- Proprietary).

18. NRC letter to Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station – Withdrawal of License Amendment Request on Steam Generator Tube Inspections," February 28, 2008, NRC ADAMS Accession No. ML080450185.

19. Nuclear Energy Institute (NEI) letter dated July 7, 2008, NRC ADAMS Accession No. ML082100086, transmitting Babcock and Wilcox Limited Canada letter 2008-06-PK-001, "Re-assessment of PMIC measurements for the determination of CTE of SA 508 Steel," dated June 6, 2008, NRC ADAMS Accession No. ML082100097.

20. NEI 97-06, Revision 3, "Steam Generator Program Guidelines," January 2011, NRC ADAMS Accession No. ML111310708.

Principal Contributor: Andrew B. Johnson, NRR/DE/ESGB

Date: September 5, 2014

September 5, 2014

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBE INSPECTION AND REPAIR (TAC NO. MF3369)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 277 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 16, 2014, as supplemented by letters dated April 2 and April 15, 2014.

The amendment revises TS 3.4.13, "RCS Operational Leakage," and TS 5.5.7, "Steam Generator (SG) Program," to exclude portions of the SG tube below the top of the SG tubesheet from periodic inspections and plugging by implementing the H* alternate repair criteria on a permanent basis. In addition, TS 5.6.7, "Steam Generator Tube Inspection Report," is also being revised to include additional reporting requirements.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,
/RA/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures:
1. Amendment No. 277 to DPR-26
2. Safety Evaluation

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