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

October 7, 2008

Mr. Larry Meyer
Site Vice President
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION 5.5.8 AND 5.6.8 (TAC NO. MD8800)

Dear Mr. Meyer:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 234 to Renewed Facility Operating License No. DPR-24 for the Point Beach Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 28, 2008, as supplemented by letter dated July 18, 2008.

The amendment changes the repair requirements of TS Section 5.5.8, "Steam Generator (SG) Program," and to the reporting requirements of TS Section 5.6.8, "Steam Generator (SG) Tube Inspection Report."

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Jack Cushing, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-266

Enclosures:
1. Amendment No. 234 to DPR-24
2. Safety Evaluation

cc w/encls: See next page

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

FPL ENERGY POINT BEACH, LLC

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 234
License No. DPR-24

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

   A. The application for amendment by FPL Energy Point Beach, LLC (the licensee), dated May 28, 2008, as supplemented by letter dated July 18, 2008, 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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 4.B of Renewed Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 234, are hereby incorporated in the renewed operating license. FPLE Point Beach shall operate the facility in accordance with Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented prior to entering MODE 4 during startup of PBNP Unit 1 from the refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Lois M. James, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications
and Facility Operating License

Date of issuance: October 7, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 234

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-24

DOCKET NOS. 50-266

Replace the following pages of the Facility Operating Licenses and 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	INSERT
Unit 1 License Page 3	Unit 1 License Page 3
TS page 5.5-8	TS page 5.5-8
----------	TS page 5.5-8a
TS page 5.6-6	TS page 5.6-6
---------	TS Page 5.6-7

D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, FPLE Point Beach 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; and

E. Pursuant to the Act and 10 CFR Parts 30 and 70, FPLE Point Beach to possess such byproduct and special nuclear materials as may be produced by the operation of the facility, but not to separate such materials retained within the fuel cladding.

4. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and 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 below:

A. Maximum Power Levels

FPLE Point Beach is authorized to operate the facility at reactor core power levels not in excess of 1540 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 234, are hereby incorporated in the renewed operating license. FPLE Point Beach shall operate the facility in accordance with Technical Specifications.

C. Spent Fuel Pool Modification

The licensee is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

5.5.8 Steam Generator (SG) Program (continued)

for all SGs and leakage rate for an individual SG.
Leakage is not to exceed 500 gallons per day 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 alternate tube repair criteria shall be applied as an alternative to the 40% depth-based criteria:

1. For Unit 1 Refueling Outage 31 and the subsequent operating cycle, tubes with flaws having a circumferential component less than or equal to 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet do not require plugging. Tubes with flaws having a circumferential component greater than 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet shall be removed from service.

   Tubes with service-induced flaws located within the region from the top of the tubesheet to 17 inches below the top of the tubesheet shall be removed from service. Tubes with service-induced axial cracks found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging.

When more than one flaw with circumferential components is found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet with the total of the circumferential components greater than 203 degrees and an axial separation distance of less than 1 inch, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceeds 94 degrees, then the tube shall be removed from service. When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceed 94 degrees, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

This alternate tube repair criteria is not applicable to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.

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 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

5.6 Reporting Requirements

5.6.7 Tendon Surveillance Report (continued)

Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

5.6.8 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 5.5.8, Steam Generator (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, and

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

i. Following completion of an inspection performed in Unit 1 Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, whether initiated on primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 5.5.8,

5.6 Reporting Requirements

j. Following completion of an inspection performed in Unit 1 Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the primary to secondary LEAKAGE rate observed in each steam generator (if it is not practical to assign leakage to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report, and

k. Following completion of an inspection performed in Unit 1 Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the portion of the tube below 17 inches from the top of the tubesheet for the most limiting accident in the most limiting steam generator.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 234 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-24

FPL ENERGY POINT BEACH, LLC

POINT BEACH NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-266

1.0 INTRODUCTION

By letters dated May 28, 2008 (Agencywide Document Access and Management System (ADAMS) Accession No. ML081560178), and July 18, 2008 (ADAMS Accession No. ML082040226), FPL Energy Point Beach, LLC, (the licensee) submitted a license amendment request to change the technical specifications (TS) for Point Beach Nuclear Plant (PBNP), Unit 1. The request proposed changes to the repair requirements of TS Section 5.5.8, "Steam Generator (SG) Program," and to the reporting requirements of TS Section 5.6.8, "Steam Generator (SG) Tube Inspection Report." The proposed changes would establish alternate repair criteria for portions of the SG tubes within the tubesheet, and would be applicable to Unit 1 during Refueling Outage 31 (1R31) and the subsequent operating cycle.

The supplemental letter dated July 18, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination published in the *Federal Register* on August 5, 2008 (73 FR 45481).

In its letter dated May 28, 2008, the licensee submitted Westinghouse Electric Company topical reports, LTR-CDME-08-11, Rev. 1, P-Attachment, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Re[g]ion of the Tubesheet Expansion Zone," dated April 29, 2008, and LTR-CDME-08-43, Rev. 1, P-Attachment, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11, Rev. 1, P-Attachment," dated April 29, 2008. The topical reports contained proprietary information and the required affidavits, signed by the licensee, requesting that NRC withhold the proprietary information from the public, were also submitted in the two letters.

The NRC issued a letter approving the withholding of the information from the public, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 2.390(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended (ADAMS Accession No. ML082540858). There is no proprietary information in this safety evaluation (SE).

2.0 BACKGROUND

PBNP Unit 1 has two Westinghouse Model 44F SGs. There are 3,214 thermally-treated Alloy 600 tubes in each SG, each with an outside diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. The tubes are hydraulically-expanded for the full depth of the tubesheet at each end and are welded to the tubesheet at the bottom of each expansion.

Until the fall of 2004, no instances of stress corrosion cracking (SCC) affecting the tubesheet region of thermally-treated alloy 600 tubing had been reported at PBNP, or other nuclear power plants in the United States. As a result, most plants, including PBNP, had been using bobbin probes for inspecting the length of tubing within the tubesheet. Since bobbin probes are not capable of reliably detecting SCC in the tubesheet region, supplementary rotating coil probe inspections were used in a region extending from 3 inches above the top of the tubesheet (TTS) to 3 inches below the TTS. This zone includes the tube-expansion transition, which contains significant residual stress, and was considered a likely location for SCC to develop.

In the fall of 2004, crack-like indications were found in tubes in the tubesheet region of Catawba Nuclear Station, Unit 2 (Catawba), which has Westinghouse Model D5 SGs. Like PBNP, the Catawba SGs employ thermally-treated alloy 600 tubing that is hydraulically-expanded against the tubesheet. At the time of cracking, Catawba had accumulated 14.7 effective full power years (EFPY) of service, which is similar to the service experience that the SGs at PBNP have accumulated, although the hot-leg operating temperature of PBNP is lower than that of Catawba. The crack-like indications at Catawba were found in bulges (also called over-expansions) in the tubesheet region, in the tack expansion region, and near the tube-to-tubesheet weld. 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 tube-to-tubesheet weld, which is made prior to the hydraulic expansion of the tube over the full tubesheet depth.

As a result of the 2004 Catawba findings, the PBNP licensee expanded the scope of rotating coil inspections to include a 20 percent sample of the hot-leg tubes, from 3 inches above the top of the TTS to the end of the tube, during 1R29 (fall 2005) and reported that they found no degradation meeting the TS plugging requirements. During 1R30 (spring 2007), PBNP performed rotating coil inspections of 50 percent of the hot-leg tubes, from 3 inches above to 17 inches below the TTS. The inspections focused on the upper 17 inches of the tube within the tubesheet, since the licensee concluded that flaws located below 17 inches from the TTS (i.e., in the bottom 4 inches of the tube within the tubesheet) had no potential to impair tube integrity. The NRC approved restricting the inspection and repair of flawed tubes to the upper 17 inches of the tube within the hot-leg tubesheet, in Amendment No. 226 for PBNP Unit 1, on April 4, 2007. Amendment No. 226 applied to 1R30 and the subsequent operating cycle.

By letter dated November 29, 2007 (ADAMS Accession No. ML073380100), Southern Nuclear Operating Company submitted a license amendment request for Vogtle Electric Generating Plant (VEGP), Units 1 and 2, which would make the inspection and repair modifications of Amendment No. 141 (which was similar to Amendment No. 226 for PBNP Unit 1) permanent and would add some additional reporting requirements under TS Section 5.6.8, "Steam Generator Tube Inspection Report." The permanent amendment request was based on a technical analysis approach, identified as H*/B*, that was also used as a basis for a permanent amendment request submitted by Wolf Creek Nuclear Operating Corporation (WCNOC) for the

Wolf Creek Generating Station on February 21, 2006. After three requests for additional information (RAIs) and several meetings with WCNOC, the staff informed WCNOC during a phone call on January 3, 2008, that it had not provided sufficient information to allow the staff to review and approve the permanent license amendment request.

Since the lack of information in the technical analysis mentioned above prevented the NRC from approving a permanent amendment to the TS inspection and reporting criteria, both VEGP and WCNOC submitted revised applications with a more conservative interim alternate repair criteria (IARC) approach. After WCNOC and VEGP responded to NRC RAIs regarding the IARC, the NRC approved the IARC amendments in letters dated April 4, 2008 (ADAMS Accession No. ML080840004), and April 9, 2008 (ADAMS Accession No. ML080950247).

By letter dated May 28, 2008 (ADAMS Accession No. ML081560178), the licensee submitted a license amendment request (LAR) to change the TSs for PBNP Unit 1. The request proposed changes to the repair requirements of TS Section 5.5.8, "Steam Generator (SG) Program," and to the reporting requirements of TS Section 5.6.8, "Steam Generator Tube Inspection Report." The proposed changes would establish IARC for portions of the SG tubes within the tubesheet, and would be applicable during 1R31 (planned for fall 2008) and the subsequent operating cycle.

By letter dated July 18, 2008 (ADAMS Accession No. ML082040226), the licensee submitted a supplement to the LAR that formalized, as a regulatory commitment, the licensee's intention of using a ratio of 2.5 in the completion of both the condition monitoring and operational integrity assessments, upon implementation of the IARC. The supplement also clarified that the additions to TS 5.6.8 were applicable to PBNP Unit 1 only.

3.0 REGULATORY EVALUATION

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs. In 10 CFR 50.36(d)(5), administrative controls are stated to be "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." This also includes the programs established by the licensee and listed in the administrative controls section of the TSs for the licensee to operate the facility in a safe manner. The requirements for (1) SG tube inspections and repair, and (2) reporting on these inspections and repair for PBNP are in TS 3.4.17 "Steam Generator (SG) Tube Integrity," and TS 5.5.8 and 5.6.8, respectively.

The TSs for all pressurized-water reactor (PWR) plants require that a SG program be established and implemented to ensure that SG tube integrity is maintained.
For PBNP, SG tube integrity is maintained by meeting specified performance criteria (in TS 5.5.8.b) for structural and leakage integrity, consistent with the plant design and licensing basis. TS 5.5.8.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 5.5.8 also includes provisions regarding the scope, frequency, and methods of SG tube inspections. Of relevance to the subject amendment request, 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, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the tube repair criteria (except as indicated above regarding the one-cycle application of a limited scope of inspection in the tubesheet region). The tube repair criteria, specified in TS 5.5.8.c, are that tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged except if permitted to remain in service through application of the alternate repair criteria provided in TS 5.5.8.c.1.

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, serve to isolate radiological fission products in the primary reactor coolant from the secondary coolant and the environment. For the purposes of this SE, SG tube integrity means that the tubes are capable of performing these safety functions in accordance with the plant design and licensing basis.

PBNP was licensed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50. As such, PBNP is not licensed to the Appendix A GDC. The PBNP Final Safety Analysis Report (FSAR) lists the plant-specific GDC to which the plant was licensed. The PBNP GDC are similar in content to the draft GDC proposed for public comment in 1967. The PBNP GDC addressing the reactor coolant pressure boundary are PBNP GDC 9 (Reactor Coolant Pressure Boundary), GDC 33 (Reactor Coolant Pressure Boundary Capability), GDC 34 (Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention), and GDC 36 (Reactor Coolant Pressure Boundary Surveillance). The applicable criteria for this system are discussed in FSAR Section 4.1, "Reactor Coolant System - Design Basis. PBNP GDC 9, 33, 31, 34, and 36 are similar to Appendix A GDC 14, 15, 31, and 32.

The General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 provide regulatory requirements which state that the RCPB shall have "an extremely low probability of abnormal leakage . . . and 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). To this end, 10 CFR 50.55a 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). Section 50.55a further requires, in part, that throughout the service life of a PWR facility like PBNP, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements, in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. Section XI requirements pertaining to inservice inspection (ISI) of SG tubing are augmented by additional requirements in the TSs.

As part of the plant 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 main steamline break (MSLB). These analyses consider primary-to-secondary leakage which may occur during these events and must show that the offsite radiological consequences do not

exceed the applicable limits of the 10 CFR Part 100 guidelines for offsite doses, GDC 19 criteria 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 analysis for PBNP is being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed.

The licensee-proposed changes to TS 5.5.8 stay within the GDC requirements for the SG tubes and maintain the accident analysis and consequences that NRC has reviewed and approved for the postulated DBAs for SG tubes. Amendment No. 226 modified the TS wording at PBNP Unit 1, to restrict the required inspection and plugging in the hot-leg tubesheet region to the uppermost 17 inches of the tubesheet region for 1R30 and the subsequent operating cycle. This excluded the lowermost 4 inches of tubing in the hot-leg tubesheet from the TS inspection and plugging requirements. License Amendment No. 226 also added a requirement that all tubes found with flaws in the upper 17 inches of the hot-leg tubesheet region be plugged, to provide added assurance that tube-to-tubesheet joint integrity would be maintained.

The proposed amendment is applicable to 1R31 and the subsequent operating cycle. This license amendment differs from Amendment No. 226 in a number of ways. First, the lowermost 4 inches of tubing in the tubesheet would no longer be excluded from the TS inspection requirements in TS 5.5.8.d. The lowermost 4 inches of tubing would be subject to the same inspection requirements as the rest of the tubing. Second, any flaws in the lowermost 4 inches of tubing in the tubesheet would not be excluded from requirements to plug. Under the proposed amendment, flaws found in the lowermost 4 inches of tubing would be subject to the IARC in lieu of the aforementioned 40 percent depth-based criterion; the latter criterion would continue to be applicable outside of the tubesheet region. Third, the proposed amendment applies to both the hot- and cold-leg sides of the tubesheet. Fourth, the proposed amendment would include new reporting requirements to allow the NRC staff to monitor the implementation of the amendment. As with Amendment No. 226 for the hot-leg side, the proposed amendment would require the plugging of all tubes found with flaws in the upper 17 inches of the tubesheet region on both the hot- and cold-leg sides.

4.0 TECHNICAL EVALUATION

4.1 Proposed Changes to the TSs

TS 5.5.8 - Steam Generator (SG) Program

TS 5.5.8.c currently states:

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

The following alternate tube repair criteria may be applied as an alternative to the 40 percent depth-based criteria:

1. For Unit 1 Refueling Outage 30 and the subsequent operating cycle, flaws found in the portion of the tube below 17 inches from the top of the hot-leg tubesheet do

not require plugging. All tubes with flaws identified in the portion of the tube within the region from the top of the hot-leg tubesheet to 17 inches below the top of the tubesheet shall be plugged. This alternate tube repair criteria is not applicable to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.

The criterion would be revised as follows, as noted in strikeout and italic type:

The following alternate tube repair criteria *shall* be applied as an alternative to the 40 percent depth-based criteria:

1. For Unit 1 Refueling Outage ~~30~~ 31 and the subsequent operating cycle, *tubes with* flaws *having a circumferential component less than or equal to 203 degrees* found in the portion of the tube below 17 inches from the top of the ~~hot leg~~ tubesheet *and above 1 inch from the bottom of the tubesheet* do not require plugging. ~~All~~ Tubes with flaws ~~identified in the portion of the tube within the region from the top of the hot leg tubesheet to~~ *having a circumferential component greater than 203 degrees found in the portion of the tube below* 17 inches *from* ~~below~~ the top of the tubesheet *and above 1 inch from the bottom of the tubesheet* shall be *removed from service* ~~plugged~~. ~~This alternate tube repair criteria is not applicable to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.~~

   Tubes with service-induced flaws located within the region from the top of the tubesheet to 17 inches below the top of the tubesheet shall be removed from service. Tubes with service-induced axial cracks found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging.

   When more than one flaw with circumferential components is found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet with the total of the circumferential components greater than 203 degrees and an axial separation distance of less than 1 inch, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

   When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceeds 94 degrees, then the tube shall be removed from service. When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceed 94 degrees, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

This alternate tube repair criteria is not applicable to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.

TS 5.5.8.d currently states:

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 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. For Unit 1 Refueling Outage 30 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded when the alternate repair criteria in TS 5.5.8.c are implemented. This exclusion does not apply to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet. 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.

The provisions would be revised as follows, as noted in strikeout type:

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 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. ~~For Unit 1 Refueling Outage 30 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded when the alternate repair criteria in TS 5.5.8.c are implemented. This exclusion does not apply to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.~~ The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d1, 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.

TS 5.6.8 - Steam Generator Tube Inspection Report

TS 5.6.8 currently states:

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with TS 5.5.8, Steam Generator (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, and

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

TS 5.6.8 would be revised to add the following three additional reporting criteria, as noted in italic type:

i. Following completion of an inspection performed in Unit 1 Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, whether initiated on primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 5.5.8,

j. Following completion of an inspection performed in Unit 1 Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the primary to secondary LEAKAGE rate observed in each steam generator (if it is not practical to assign leakage to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report, and

k. Following completion of an inspection performed in Unit 1 Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the portion of the tube below I7 inches from the top of the tubesheet for the most limiting accident in the most limiting steam generator.

4.2 Technical Evaluation

The tube-to-tubesheet joint consists of the tube, which is hydraulically-expanded against the bore of the tubesheet; the tube-to-tubesheet weld located at the tube end; and the tubesheet. The joint was designed as a welded joint and not as a friction or expansion joint. The weld itself was designed as a pressure boundary element. It was designed to transmit the entire end-cap pressure load during normal and DBA conditions from the tube to the tubesheet with no credit taken for the friction developed between the hydraulically-expanded tube and the tubesheet. In addition, the weld serves to make the joint leak tight.

The one-cycle amendments approved for PBNP Unit 1 (Amendment No. 226) and other plants (such as Vogtle and Braidwood) prior to 2008, exempted the lower 4-inch portion of the tube within the 21-inch-deep tubesheet from inspection and exempted tubes with flaw indications in this region from being removed from service (i.e., plugged). These one-cycle amendments, in effect, redefined the pressure boundary at the tube-to-tubesheet joint as consisting of a friction or expansion joint with the tube hydraulically-expanded against the tubesheet over the top 17 inches of the tubesheet. These amendments took no credit for the lower portion of the tube or the tube-to-tubesheet weld as contributing to the structural or leakage integrity of the joint.

The proposed amendment that is the subject of this SE (and similar amendments approved in 2008 for Wolf Creek, Vogtle, Braidwood, and Surry) differs fundamentally from the one-cycle amendments approved prior to 2008, and is a more conservative approach. The proposed amendment treats the tube-to-tubesheet joint as a welded joint in a manner consistent with the original design basis, with no credit taken for the friction developed between the hydraulically-expanded tube and the tubesheet. The proposed amendment is intended to ensure that the aforementioned end-cap loads can be transmitted down the tube, through the tube-to-tubesheet weld, and into the tubesheet.

4.2.1 Proposed Change to TS 5.5.8.c, "Provisions for SG tube repair criteria"

The 40 percent depth-based tube repair criterion in TS 5.5.8.c is intended to ensure, in conjunction with other elements of TS 5.5.8, that tubes accepted for continued service (i.e., not plugged) satisfy the performance criteria for structural integrity in TS 5.5.8.b.1 and the performance criteria for accident leakage integrity in TS 5.5.8.b.2. The criterion includes an allowance for eddy current measurement error and incremental flaw growth prior to the next inspection of the tube. The alternate tube repair criteria in the existing TSs and the proposed IARC in this amendment are alternatives to this 40 percent depth-based criterion.

4.2.1.1.1 Structural Integrity Considerations

The 40 percent depth-based criterion was developed to be conservative for flaws located anywhere in the SG, including free span regions. In the tubesheet, however, the tubes are constrained against radial expansion by the tubesheet and, therefore, are constrained against an axial (fish-mouth) rupture failure mode. The only potential structural failure mode within the tubesheet is a circumferential failure mode, leading to tube severance.

The proposed IARC would permit tubes with up to 100 percent through-wall flaws in the portion of the tube from 17 inches below the TTS to 1 inch above the bottom of the tubesheet to remain

in service provided the circumferential component of these flaws does not exceed 203 degrees. The 203-degree criterion was determined on the basis of the remaining cross-sectional area of the tube needed to resist the limiting axial end-cap load on the tube and the pressure load on the flaw cross-section, using limit-load analysis, with safety factors consistent with those required by the performance criteria for structural integrity in TS. Because the 203-degree criterion was determined on this basis, the NRC staff finds this approach acceptable.

For the portion of the tube from the bottom of the tubesheet to 1 inch above the bottom of the tubesheet, the proposed ARC would permit tubes with up to 100 percent through-wall flaws to remain in service provided the circumferential component of these flaws does not exceed 94 degrees. This criterion is based on the minimum tube-to-tubesheet weld cross-sectional area needed to resist the limiting axial end-cap load on the tube and the pressure load on the flaw cross-section, using limit load analysis, with safety factors consistent with those required by the performance criteria for structural integrity in the TS. A 203-degree crack in the tube wall immediately above the weld could potentially concentrate the entire end cap load to a 157-degree segment of the weld, whereas a minimum 266 degree segment (i.e., 360 minus 94 degrees) of weld is needed to resist the end-cap load with adequate safety margin. Thus, the 94-degree criterion for the tube in the lowermost 1-inch region is intended to ensure that the weld is not overstressed. Although the NRC staff did not complete its review of the specific limit-load methodology used to calculate the 94-degree criterion, it reviewed the results of the stress analysis of the weld, which was performed to demonstrate that the weld complied with the stress limits of the ASME Code, Section III. The TS performance criteria for tube structural integrity are intended to ensure safety margins consistent with the ASME Code, Section III stress limits. Based on a comparison of the calculated maximum design stress to the ASME Code-allowable stress, the NRC staff concludes that the proposed 94-degree criterion ensures that the weld can carry the end-cap loads with margins to failure consistent with the margins ensured by the ASME stress limits and is, therefore, acceptable.

The 203- and 94-degree criteria include an allowance for incremental flaw growth in the circumferential direction prior to the next inspection. The licensee states that no significant growth rate data exists for the specific case of circumferential cracking in the tubesheet expansion region. The licensee's growth rate estimate is based on a 95 percent upper bound value of available primary water stress corrosion crack (PWSCC) growth rate data for other tube locations. Given the lack of actual growth rate data for cracks that may potentially initiate in the lowermost 4 inches of the tube, the staff attaches only a low level of confidence in the conservatism of the licensee's growth rate estimate. However, the staff notes that the effect of any lack of conservatism in the licensee's estimate is mitigated somewhat by the fact that all of the SGs at PBNP will be inspected at 1R32, should any crack indications be found during 1R31. In addition, the 203- and 94-degree criterion conservatively take no credit for the effects of friction between the tube and tubesheet in any portion of the tube-to-tubesheet joint, for reacting a portion of the axial end cap load before it reaches the cracked cross-section. Thus, the staff concludes that the 203- and 94-degree criteria are conservative, irrespective of growth rate uncertainties.

The 203- and 94-degree criteria do not include an explicit allowance for eddy current measurement error. The licensee will be utilizing an inspection technique that has been qualified for the detection of circumferential PWSCC in tube expansion transitions and in the tack expansion region just above the tube to tubesheet weld. The tack expansion is an

approximately 1-inch long expansion of the tube in the tubesheet that is performed before the tube is hydraulically-expanded for the entire depth of the tubesheet. A fundamental assumption behind the proposed 203- and 94-degree repair criteria is that all detected circumferential flaws in the lowermost 4 inches of the tube are fully 100 percent through wall, irrespective of the actual depth of the flaw. With this assumption, the licensee referenced an Electric Power Research Institute (EPRI) sponsored study that indicated the eddy current measurement of the crack arc length was conservative (i.e., larger than the actual crack size), and resulted in an estimate of the remaining cross-sectional area that was always smaller than values obtained through direct measurement of cracks. Although the NRC staff has not reviewed the EPRI study in detail, it finds, based on the results of the study, that any uncertainties relating to measured arc length of the flaw are not expected to impair the conservatism of the 203- and 94-degree criteria.

The proposed IARC also includes criteria to account for interaction effects for multiple circumferential flaws that are in close proximity. The proposed criteria treat the multiple circumferential flaws located within 1 inch of one another as all occurring at the same axial location. The total arc length of the combined flaw is the sum of the individual flaw arc lengths with overlapping arc lengths counted only once. The licensee stated that the summation of cracks with both located more than 17 inches from the TTS and more than 1 inch from the bottom of the tube will be compared to the 203-degree criterion. The summation of cracks with one flaw located less than 1 inch from the bottom of the tubesheet and the other within 1 inch of the first (or both flaws within 1 inch of the bottom of the tubesheet) would be compared to the 94-degree criterion. Cracks located more than 1 inch apart are assumed to act independently of each other. This 1-inch criterion was determined using a fracture mechanics approach to determine the axial distance from an individual crack tip at which the stress distribution reverts to a nominal stress distribution for an uncracked section. The 1-inch criterion is twice the calculated distance since twice this distance is the necessary separation between two cracks for the cracks to act independently of each other. The NRC staff reviewed the basis for the 1-inch criterion and the fracture mechanics approach to determining the criterion. Because the criterion is based on a valid fracture mechanics approach, the NRC staff finds it acceptable.

The proposed ARC would permit tubes with axial cracks in the lower most 4 inches of the tube to remain in service, irrespective of crack depth. The NRC staff finds this acceptable because axial cracks do not impair the ability of the tube or the weld to resist axial load and because the tube is fully constrained by the tubesheet against an axial failure mode.

Finally, the proposed IARC includes a requirement to plug all tubes in which flaws are detected in the upper 17-inch portion of the tube within the tubesheet. This adds to the conservatism of the 203- and 94-degree criteria since it mitigates any loss of tightness and, thus, any loss of friction between the tube and tubesheet due to flaws in the upper 17-inch region of the joint.

4.2.1.2 Accident Leakage Integrity Considerations

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. Operational leakage integrity is assured by monitoring primary-to-secondary leakage relative to the applicable TS LCO limits 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 DBAs to exceed the

accident-induced leakage performance criterion in TS 5.5.8.b.2, including the leakage values assumed in the plant licensing basis accident analyses. The licensee states that this is ensured for PBNP by limiting primary-to-secondary leakage to 0.35 gallon per minute in the faulted SG during an MSLB accident.

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 tube-to-tubesheet joint mockups to establish loss coefficient as a function of contact pressure. Westinghouse states that the flow resistance varies as a log normal linear function of joint contact pressure, but due to the large scatter of the flow resistance test data, has been assumed to be constant with joint contact pressure at a value which conservatively lower bounds the data.

Using the above model, a "modified B*" approach for calculating accident leakage was initially proposed in the amendment request. The proposed modified B* approach relies to some extent on an assumed, constant value of loss coefficient, based on a lower bound of the data. This contrasts with the "nominal B*" approach which, in its latest form, is not directly impacted by the assumed value of loss coefficient since this value is assumed to be constant with increasing contact pressure between the tube and tubesheet. The NRC staff is not able to make a conclusion as to whether the assumed value of loss coefficient in the "modified B*" approach is conservative at this time. However, the NRC staff has performed some evaluations regarding the potential for the normal operating leak rate to increase under steam-line break conditions. Making the conservative assumption that loss coefficient and viscosity are constant under both normal operating and steam-line break conditions, the ratio of steam-line break leakage rate to normal operating leak rate is equal to the ratio of steam-line break differential pressure to normal operating differential pressure times the ratio of effective crevice length under normal operating conditions (l_{NOP}) to effective crevice length under steam-line break conditions (l_{SLB}). Effective crevice length is the crevice length over which there is contact between the tube and tubesheet. Using various values of (l_{NOP}/ l_{SLB}) determined from the "nominal B*" approach (which does not rely on an assumed value of loss coefficient) and recognizing the issues associated with some of these previous H*/B* analyses, the NRC staff concludes that a factor of 2.5 reasonably bounds the potential increase in leakage from the lowermost 4 inches of tubing that would be realized in going from normal operating to steam-line break conditions.

4.2.1.3 Regulatory Commitment

The licensee provided a regulatory commitment in its July 18, 2008, supplemental letter, stating that it would apply the 2.5 factor in its condition monitoring (CM) and operational assessment (OA) upon implementation of the subject license amendment. Specifically, for the CM assessment, the licensee states that the component of leakage from the lowermost 4 inches for the most limiting SG during the prior cycle of operation will be multiplied by a factor of 2.5 and added to the total leakage from any other source and compared to allowable accident leakage limit. For the OA, the licensee stated that the difference in leakage from the allowable accident leakage limit and the accident leakage from other sources will be divided by 2.5 and compared to the observed (operational) leakage and that an administrative limit (for operational leakage)

will be established to not exceed the calculated value. Since this properly addresses the factor of 2.5 that bounds the potential increase in leakage in the lowermost 4 inches of tubing, the NRC staff finds this acceptable.

The NRC staff finds that reasonable controls for the licensee's implementation and subsequent evaluation of any changes to the regulatory commitment are provided by the licensee's administrative processes, including its commitment management program. The NRC staff has determined that the commitment does not warrant the creation of regulatory requirements, which would require prior NRC approval of subsequent changes. The NRC has agreed that NEI 99-04, Revision 0, provides reasonable guidance for the control of regulatory commitments made to the NRC staff (Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff," dated September 21, 2000). These commitments will be controlled in accordance with the licensee's commitment management program in accordance with NEI 99-04. Any change to the regulatory commitments is subject to licensee management approval and subject to the procedural controls established at the plant for commitment management in accordance with NEI 99-04, which include notification of the NRC. Also, the NRC staff may choose to verify the implementation and maintenance of these commitments in a future inspection or audit.

Based on this, the NRC staff concludes that the regulatory commitment addressed above for this amendment is acceptable.

4.2.2 Proposed Change to TS 5.5.8.d, "Provisions for SG tube inspections"

With the plant entry into 1R31, the sentence added to TS 5.5.8.d in Amendment No. 226 is no longer applicable and the licensee has proposed to delete the sentence. The sentence to be deleted states, "For Unit 1 Refueling Outage 30 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot-leg tubesheet is excluded when the alternate repair criteria in TS 5.5.8.c are implemented. This exclusion does not apply to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet." Therefore, in 1R31, the inspection requirements of TS 5.5.8.d apply to the entire length of tubing from the tube-to-tubesheet weld location at the tube inlet to the tube-to-tubesheet weld location at the tube outlet. TS 5.5.8.d further states that the tube-to-tubesheet weld itself is not considered part of the tube. No changes relative to this wording are being proposed as part of the subject amendment request.

4.2.3 Proposed Change to TS 5.6.8, "Steam Generator Tube Inspection Report"

The NRC staff has reviewed the proposed new reporting requirements and finds that they are sufficient to allow the staff to monitor the implementation of the proposed amendment. Based on this conclusion, the NRC staff finds that the proposed new reporting requirements are acceptable.

4.2.4 Considerations Relating to Tube-to-Tubesheet Welds

The standard technical specifications (STS) and the PBNP TSs state specifically that the tube to tubesheet welds are not part of the tube. Therefore, the requirements of TS 5.5.8 do not apply

to these welds. However, licensees typically visually inspect the tube ends (including the welds) for evidence of leakage while the SG primary manways are open to permit eddy current inspection of the tubes.

Eddy-current inspection of the SG tubes at Catawba Unit 2 in 2007 revealed indications interpreted as cracks at or near the tube-to-tubesheet weld, suggesting the potential for such cracks in similar SGs, such as those at PBNP. An industry peer review was recently conducted for the Catawba Unit 2 cold-leg tube-end indications, to establish whether the reported indications are in the tube material or the welds. A consensus was reached that the indications most likely exist within the tube material. However, some of the indications extend close enough to the tube end that the possibility that the flaws extend into the weld could not be ruled out. An NRC staff member and an expert consultant from Argonne National Laboratory also reviewed these indications and concluded that the industry's position was reasonable. The peer review group and the NRC consultant also reviewed eddy-current signals from a tube-to-tubesheet mockup, which included a circumferential notch in one of the welds, and they concluded that this notch did not produce a detectable signal.

Based on the above evaluation, the NRC staff finds that the proposed license amendment, which is applicable only to 1R31 and the subsequent operating cycle, ensures that SG tube structural and leakage integrity will be maintained during this period with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, and will have no adverse impact on the ability of the tube-to-tubesheet welds to perform their safety-related function. Based on this finding, the NRC staff further concludes that the proposed amendment meets 10 CFR 50.36 and, thus, the proposed amendment is acceptable.

The current TSs and the proposed amendment do not address inspection requirements for the tube-to-tubesheet welds. There are no safety issues with respect to hypothetical cracks in the weld if it can be demonstrated, such as with the H*/B* strategies discussed in Section 2 of this safety evaluation, that the axial end-cap loads in the tube are reacted by frictional forces developed between the tube and tubesheet before any portion of the end-cap load is transmitted to the weld. Currently, all industry requests for a permanent H*/B* amendment have been withdrawn (see Section 2); however, the industry is still pursuing development of the information needed by the NRC staff to support future amendment requests for H*/B*.

The licensee has concluded that cracking exclusively in the weld is not a potential damage mechanism on the basis of the peer review findings. Should it not be possible for the NRC staff to approve an acceptable H*/B* amendment within a reasonable time period, it is the NRC staff's position that the industry will need to develop inspection techniques (e.g., visual, eddy-current) capable of detecting weld cracks to ensure that the welds are capable of performing their safety related function. It should be noted that the NRC staff observed a demonstration of an available visual inspection technique for inspecting the welds, but raised questions on whether this technique was sufficiently reliable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment relates to changes in repair criteria and reporting requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (NSHCD)

The Commission has made a final determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below.

> 1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?
>
> Response: No
>
> Of the various accidents previously evaluated, the proposed changes only affect steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB), locked rotor, and control rod ejection accident evaluations.
>
> Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Model F steam generators has shown that axial loading of the tubes is negligible during an SSE.
>
> At normal operating pressures, leakage from [primary water stress cracking corrosion] PWSCC below 17 inches from the [top of the tubesheet] TTS is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. For the SGTR event, the required structural margins of the steam generator tubes are maintained by limiting the allowable crack size to 203 degrees for circumferential cracks below 17 inches from the TTS for the subsequent operating cycle. Tube rupture is precluded for cracks in the hydraulic-expansion region due to the constraint provided by the tubesheet. The potential for tube pullout is mitigated by limiting the allowable crack size to 203 degrees during the subsequent operating cycle. These allowable crack sizes use a bounding analysis that takes into account eddy current uncertainty and crack growth rate.

It has been shown that a circumferential crack with an azimuthal extent of 203 degrees for the 18-month SG tubing eddy current inspection interval meets the performance criteria of NEI 97-06, Rev. 2, "Steam Generator Program Guidelines" and Draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [pressurized water reactors] Steam Generator Tubes." Therefore, the margin against tube burst pullout is maintained during normal and postulated accident conditions and the proposed change does not result in a significant increase in the probability or consequence of a SGTR.

The probability of a SLB is unaffected by the potential failure of a SG tube as the failure of a tube is not an initiator for a SLB event. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube-to-tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for all axial or circumferentially oriented cracks occurring 17 inches below the top of the tubesheet. Since normal operating leakage is limited to 150 [gallons per day] gpd (approximately 0.10 [gallons per minute] gpm), the attendant accident condition leak rate, assuming all leakage to be from indications below 17 inches from the top of the tubesheet would be bounded by 500 gpd (approximately 0.35 gpm). This value is within the accident analysis assumptions for the design basis accident for PBNP.

Based on the above, the performance criteria of NEI-97-06, Rev. 2 and Draft Regulatory Guide (RG) 1 .I21 continue to be met and the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed change does not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the interim alternate repair criteria. The proposed change does not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

Therefore, based on the above evaluation, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The proposed change maintains the required structural margins of the steam generator tubes for both normal and accident conditions. NEI 97-06, Revision 2 and RG I .I21 are used as the basis in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.I21 describes a method acceptable to the NRC staff for meeting [general design criteria] GDC 14, 15, 231, and 32 by reducing the probability and consequences of an SGTR. PBNP GDC 9, 33, 31, 34, and 36 are similar to Appendix A GDC 14, 15, 31, and 32. RG 1. I21 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME [American Society of Mechanical Engineers] Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking in a tube or the tube-to-tubesheet weld, References 2 and 4 [of the application] define a length of remaining tube ligament that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). Additionally, it is shown that application of the limited tubesheet inspection depth criteria will not result in unacceptable primary-to-secondary leakage during all plant conditions.

Based on the above, it is concluded that the proposed changes do not result in any reduction of margin with respect to plant safety as defined in the Final Safety Analysis Report or Bases of the plant Technical Specifications.

The NRC staff has reviewed the licensee's analysis and based on this review, determined that the three standards of 10 CFR 50.92 are satisfied. Therefore, the NRC staff has determined that the amendment involves no significant hazards consideration.

8.0 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Andrew Johnson, NRR

Date: October 7, 2008

October 7, 2008

Mr. Larry Meyer
Site Vice President
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION 5.5.8 AND 5.6.8 (TAC NO. MD8800)

Dear Mr. Meyer:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 234 to Renewed Facility Operating License No. DPR-24 for the Point Beach Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 28, 2008, as supplemented by letter dated July 18, 2008.

The amendment changes the repair requirements of TS Section 5.5.8, "Steam Generator (SG) Program," and to the reporting requirements of TS Section 5.6.8, "Steam Generator (SG) Tube Inspection Report."

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
/RA/
Jack Cushing, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-266

Enclosures:
1. Amendment No. 234 to DPR-24
2. Safety Evaluation

cc w/encls: See next page

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