April 4, 2007

Mr. Dennis L. Koehl
Site Vice President
Point Beach Nuclear Plant
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE REPAIR IN THE TUBESHEET (TAC NO. MD2583)

Dear Mr. Koehl:

The Commission has issued the enclosed Amendment No. 226 to Renewed Facility Operating License No. DPR-24 for Point Beach Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 11, 2006, and supplemented on January 19, March 9 and 26, 2007.

This amendment revises TS 5.5.8, "Steam Generator Program," to change the inspection and repair criteria for the portion of the tubes within the hot-leg region of the tubesheet for the single operating cycle following Refueling Outage 30. The amendment defines a distance downward into the hot-leg tubesheet, below which flaws may remain in service regardless of size. As a result, tube inspection within the hot-leg region would be required only within 17 inches of the top of the tubesheet. An administrative change is made to correct a page number in the TS table of contents and to delete two blank pages in TS Section 5.0.

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/

Patrick D. Milano, 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. 226 to DPR-24
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

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Package Accession Number: ML070800705 Amendment Accession Number: ML070800625
TS Accession Number: ML070950265

OFFICE	NRR/LPL2-2	LPL3-1/PM	LPL3-1/LA	DCI/CSG	OGC	LPL3-1/BC
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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 226
License No. DPR-24

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated July 11, 2006, as supplemented on January 19, March 9 and 26, 2007 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 3.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. 226, are hereby incorporated in the renewed operating license. NMC 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 within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, 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: April 4, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 226

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-24

DOCKET NO. 50-266

Replace the following pages of the Facility Operating License 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
License Page 3	License Page 3
iii	iii
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.5-11	5.5-11
5.5-12	5.5-12
5.5-13	5.5-13
5.5-14	5.5-14
5.5-15	5.5-15
5.5-16	5.5-16
5.5-17	- - -
5.5-18	- - -

D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NMC 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, NMC 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. <u>Maximum Power Levels</u>

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

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 226, are hereby incorporated in the renewed operating license. NMC shall operate the facility in accordance with Technical Specifications.

C. <u>Spent Fuel Pool Modification</u>

The licensee[2] 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.

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[2] Reference to the licensee in License Conditions 4.C, 4.E and 4.H refers to Wisconsin Electric Power Company and is maintained for historical purposes.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 226 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-24

NUCLEAR MANAGEMENT COMPANY, LLC

POINT BEACH NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-266

1.0 INTRODUCTION

By application dated July 11, 2006, as supplemented on January 19, March 9 and 26, 2007 (Agencywide Documents Access and Management System Accession Nos. ML062050338, ML070220084, ML070680413, and ML070860501, respectively), Nuclear Management Company (the licensee) requested changes to the Technical Specifications (TSs) for Point Beach Nuclear Plant, Unit 1. The amendment would revise TS 5.5.8, "Steam Generator (SG) Program." The proposed change would modify the inspection and plugging requirements for portions of the SG tubing within the hot leg tubesheet region to make these requirements applicable only to the portion of tubing within the upper 17 inches of the tubesheet thickness, for Refueling Outage 30 (U1R30) and the subsequent operating cycle until the next scheduled inspection.

The licensee's supplements dated January 19, March 9 and 26, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 29, 2006.

2.0 REGULATORY EVALUATION

2.1 Background

Point Beach Unit 1 has two Westinghouse Model 44F steam generators. Each SG has 3,214 thermally-treated Alloy 600 tubes with an outside diameter of 0.875 inches and a 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.

The licensee has used bobbin probes for most ultrasonic inspections of the length of tubing within the tubesheet. Since the bobbin probe is not capable of reliably detecting stress corrosion cracks (SCC) in the tubesheet region, these inspections have been supplemented with rotating probe inspections in the hot-leg expansion transition region, typically to a depth of 3 inches below the top of the tubesheet. The expansion transition contains significant residual stress and, combined with the relatively high temperature, is considered one of the more likely areas for SCC to develop. Until the fall of 2004, there had not been any reported instances of SCC affecting the tubesheet region of thermally-treated Alloy 600 tubing, either at Point Beach Unit 1 or elsewhere in the U.S.

In the fall of 2004, crack-like indications were found in tubes in the tubesheet region of Catawba Unit 2, which has Westinghouse model D5 SGs. Like Point Beach 1, the Catawba SGs employ thermally-treated Alloy 600 tubing that is hydraulically-expanded against the tubesheet. Catawba had accumulated 14.7 effective full-power years (EFPY) of service, slightly less than Point Beach 1 (17.7 EFPY as of U1R29), with a higher (by 28 °F) hot-leg operating temperature. The crack-like indications at Catawba were found in bulges (or over-expansions) in the tubesheet region, in the tack roll region, and in the tube-to-tubesheet weld. (The tack expansion is an initial 0.7-inch long expansion at each tube end and is formed prior to the hydraulic expansion over the full tubesheet depth. Its purpose was to facilitate performing the tube to tubesheet weld.) Crack-like indications were found in a bulge in one tube and in the tack expansion in nine tubes. Approximately 6 of the 196 tube-to-tubesheet weld indications extended into the parent tube.

During the Fall 2005 refueling outage (U1R29), as a result of the Catawba findings, the licensee expanded the scope of previous rotating probe inspections at Point Beach 1 by inspecting the full length of 20 percent of the tubes in the hot-leg portion of the tubesheet. These inspections detected no crack-like indications. The licensee believes that any flaws located at elevations more than 17 inches below the top of the tubesheet (i.e., in the bottom five inches of the tubesheet region, including the tack expansion region and the tubing in the vicinity of the welds) have no potential to impair tube structural or leakage integrity and, thus, do not pose a safety concern. According to the proposed license amendment, the portion of tubing below 17 inches from the top of the tubesheet would be excluded from the tube inspection and plugging requirements for Refueling Outage 30 (U1R30) and the subsequent operating cycle.

2.2 Regulatory Evaluation

Steam generator 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 coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, tube integrity means that the tubes are capable of performing these functions in accordance with the plant design and licensing basis.

Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the fundamental regulatory requirements with respect to the integrity of the SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 state that the RCPB shall have “an extremely low probability of abnormal leakage...and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards possible" (GDC 30), and shall be designed to permit "periodic inspection and testing ... to assess ... structural and leak tight 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 (ASME) *Boiler and Pressure Vessel Code* (Code). Section 50.55a further requires, in part, that throughout the service life of a PWR facility, 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 [ISI] 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 ISI of SG tubing are augmented by additional SG tube surveillance requirements in the TS.

As part of the plant licensing basis as described in the Final safety Analysis Report, applicants for pressurized water reactor (PWR) licenses are required to analyze the consequences of postulated design-basis accidents (DBAs) such as an SG tube rupture (SGTR) and main steamline break (MSLB). These analyses consider the primary-to-secondary leakage through the tubing 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).

TS 5.5.8 for Point Beach 1 requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Tube integrity is maintained by meeting specified performance criteria (in TS 5.5.8b) for structural and leakage integrity, consistent with the plant design and licensing bases. TS 5.5.8 requires a condition monitoring assessment be performed during each outage during 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 inspection 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 applicable tube repair criteria. The applicable tube repair criterion specified in TS 5.5.8c is that 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 proposed license amendment would limit the required inspections and any resulting plugging on the hot-leg side of the 22-inch thick tubesheet region to the upper 17 inches of the tubesheet region only during Refueling Outage 30 and the subsequent operating cycle. This amendment is similar to amendments approved for Byron Unit 2, Braidwood Unit 2, Vogtle Unit 2, and Wolf Creek.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes

a. TS 5.5.8.c currently states: “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 licensee proposed an alternative to this 40-percent criterion that would be added as follows:

   > The following alternate tube repair criteria may be applied as an alternative to the 40% depth-based criteria:
   >
   > 1. For 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 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.

b. TS 5.5.8.d currently states:

"Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and the method 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 requirements 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 licensee proposed to revise TS 5.5.8.d as follows:

> Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and the method 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 requirements 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.

c. In addition to the TS changes proposed above, the licensee committed in its January 19, 2007, letter to provide the following information in the 180-day Steam Generator Tube Inspection Report for Point Beach Unit 1 Refueling Outage 30:

- A listing of indications detected in the upper 17 inches of the hot-leg region of the tubesheet with respect to location, orientation, and size.

- The operational primary-to-secondary leakage rate observed in each steam generator during the cycle preceding the inspection.

- The calculated accident induced leakage rate for the most limiting accident for each steam generator from the hot-leg region of the tube sheet below 17 inches from the top of the tubesheet.

3.2 Staff 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 in accordance with the ASME Code, Section III, not as a friction or expansion joint. The weld itself was designed as a pressure boundary element in accordance with the ASME Code, Section III. It was designed to transmit the entire end cap pressure load during normal and design basis accident conditions from the tube to the tubesheet with no credit taken for the friction developed between the tubesheet and the hydraulically-expanded portion of the tube. In addition, the weld makes the joint leak-tight.

In effect, the licensee is proposing to exclude, during Refueling Outage 30 and the subsequent operating cycle, the lower 5 inches of the 22-inch deep tubesheet region from a tube inspection and to exclude tubes with flaw indications in the lower 5-inch zone from the need to plug. The latter part of this proposal (i.e., to exclude tubes from plugging) is needed as a practical matter since, although rotating coil probe inspections will not be performed in this region, the bobbin probe will necessarily be recording any signals produced in this zone. This proposal, in effect, redefines 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 region. Under this proposal, no credit is taken for the lower 5 inches of the tube or the tube-to-tubesheet weld in contributing to the structural or leakage integrity of the joint. In effect, the lower 5 inches of the tube and weld are assumed not to exist.

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. This includes maintaining structural safety margins consistent with the structural performance criteria in TS 5.5.8.b.1 and the design basis as is discussed in Section 3.2.1 below. In addition, this includes limiting the potential for accident-induced primary to secondary leakage to values not exceeding the accident leakage performance criteria in TS 5.5.8.b.2 consistent with the licensing basis accident analyses. Maintaining tube integrity in this manner ensures that the amended TS are in compliance with all applicable regulations. The NRC staff's evaluation of joint structural integrity and leakage integrity is discussed in Sections 3.2.1 and 3.2.2, respectively, of this safety evaluation.

The licensee is also proposing to plug, during Refueling Outage 30 and the subsequent operating cycle, all tubes found with flaws in the upper 17-inch region of the tubesheet on the hot-leg side. The staff finds this proposed requirement acceptable because (a) it is more conservative than the current TS 40-percent plugging limit and (b) will provide added assurance that the length of tubing along the entire proposed 17-inch inspection zone will be effective in resisting tube pull out under tube end cap pressure loads and in resisting primary-to-secondary leakage between the tube and tubesheet.

According to the proposed TS 5.5.8.d, the portion of the tube below 17 inches from the top of the hot-leg tubesheet region can only be excluded from inspection when the alternate repair criteria proposed in TS 5.5.8.c are implemented. This provision is meant to ensure that the lower 5 inches are excluded from inspection only when flaws in the upper 17 inches are plugged on detection (and evaluated and reported according to TS 5.6.8). The alternate repair criteria do not apply to the tube positioned at row 38 column 69 in the A steam generator. This tube is not fully expanded and cannot be assumed to have the same structural and leakage integrity as the fully expanded condition used in the licensee's technical justification for the amendment. This tube will be inspected over the full length of the tubesheet and the 40-percent throughwall repair criteria will apply.

3.2.1 Joint Structural Integrity

The licensee's contractor, Westinghouse, calculated the engagement (embedment) length of hydraulically-expanded tubing inside the tubesheet that is necessary to resist pullout under normal operating and design basis accident conditions. Pullout is the structural failure mode of interest because the tubes are radially constrained against axial fishmouth rupture by the presence of the tubesheet. The axial force, which could produce pullout, derives from the pressure end-cap loads due to the primary to secondary pressure differentials associated with normal operating and design basis accident conditions. Westinghouse determined the required engagement distance on the basis of maintaining a factor of 3 against pullout under normal operating conditions and a factor of 1.4 against pullout under accident conditions. The NRC staff concurs that these are the appropriate safety factors to apply to demonstrate structural integrity. These safety factors are consistent with the safety factors embodied in the structural integrity performance criteria in TS 5.5.8.b.1 and with the design basis; namely the stress limit criteria in the ASME Code, Section III.

The resistance to pullout is the axial friction force developed between the expanded tube and the tubesheet over the engagement distance. The friction force is a function of the radial contact pressure between the expanded tube and the tubesheet. The radial contact pressure derives from several sources including: (1) the contact pressure associated directly with the hydraulic expansion process, (2) additional contact pressure due to differential radial thermal expansion between the tube and tubesheet under hot operating conditions, (3) additional contact pressure caused by the primary pressure inside the tube, (4) reduced contact pressure due to pressure inside the crevice between the tube and tubesheet, and (5) additional or reduced contact pressure associated with tubesheet bore dilation (distortion) caused by tubesheet bow (deflection) as a result of the primary to secondary pressure load acting on the tubesheet. The licensee's application conservatively assumed there was no contribution from the hydraulic expansion process. Westinghouse performed conventional and finite element analyses to evaluate the contributions of the remaining three contributors to the radial contact pressure.

Tubesheet bore distortion caused by tubesheet bow under primary to secondary pressure can increase or decrease contact pressure depending on the tube location within the bundle and the location along the length of the tube in the tubesheet region. The tubesheet acts essentially as a flat, circular plate under an upward acting net pressure load. The tubesheet is supported axially around its periphery with a partial restraint against tubesheet rotation provided by the SG shell and channel head. Over most of the tubesheet away from the periphery, the bending moment resulting from the applied primary to secondary pressure load can be expected to put the tubesheet into tension at the top and compression at the bottom. Thus, the resulting distortion of the tubesheet bore (tubesheet bore dilation) tends to be such as to loosen the tube to tubesheet joint at the top of the tubesheet and to tighten the joint at the bottom of the tubesheet.

Based on these analyses, Westinghouse concluded that the required engagement distances to ensure the safety factor criteria against pullout are achieved vary in the hot-leg region from 3.3 to 11.0 inches. The value depends on the radial location of the tube within the tube bundle, with the largest engagement distances needed toward the center. The most limiting value, which corresponds to the cold-leg region at normal operating conditions, is 12.1 inches; however, this amendment request does not apply to the cold-leg region.

The March 9 and 26, 2007, letters provided revised analyses which, in part, addressed recent test results indicating that a fundamental assumption in the original analyses (i.e., the analyses provided with the July 11, 2006, letter) was not justified. Specifically, the original analysis assumed that, for a throughwall flaw located 17 inches or more below the top of the tubesheet, primary water inside the tube flashes to steam at secondary side pressure when it leaks through the flaw into the tube to tubesheet crevice. The recent tests were performed with several small, throughwall round holes representing the flaw under hot conditions. These tests indicated that the leakage through the holes remains in liquid state. Pressure inside the crevice ranges from primary pressure at the hole location to saturation pressure (based on primary water temperature) near the top of the crevice. The net effect relative to the original analyses is to reduce the pressure drop across the tube wall and, thus, to reduce the contact pressure between the tube and tubesheet.

The revised analyses included a revised finite element model of the tubesheet. The revised model is described by Westinghouse as a more detailed finite element model than that used in the original analyses. Westinghouse states that the original model was overly conservative because it did not account for features in the lower steam generator region that act to increase the resistance of the tubesheet to vertical deflections. For example, the finite element model did not include the tube lane and the channel head to divider plate weld.

The revised analyses also considered a case where the divider plate is assumed to provide no restraint to vertical deflection of the tubesheet, which is subject to the primary to secondary pressure differential. This case was analyzed in response to a staff request for additional information (NRC letter dated December 13, 2006, ADAMS Accession No. ML063400204) concerning the implications of cracks being found by inspection in the welds connecting the tubesheet to divider plate at certain foreign reactors.

In its March 9, 2007, letter, the licensee stated that the required engagement distances in the original proposal submitted for Point Beach Unit 1 were not affected by the revised Westinghouse analyses because the methodology used for Point Beach was not based on the tube pullout tests. In the March 26, 2007, letter, the licensee provided another revised analysis, which used the tube pullout data and the methodology employed by another plant with the same steam generator design and materials. The analysis also accounted for differences in operating parameters for the two plants (i.e., temperature and pressure). Based on the revised analyses, including the assumption of no divider plate restraint against tubesheet deflection, the licensee concluded that the maximum tube to tubesheet engagement distance needed in the hot-leg region to provide the required margins against pullout is 11.8 inches. This is for tube locations near the center of the bundle and compares to the value of 11.0 inches calculated in the original analysis. The revised engagement distance is well within the proposed 17-inch inspection zone.

The NRC staff has not reviewed the Westinghouse analyses in sufficient detail to conclude whether 11.8 inches of engagement (termed H* criterion by Westinghouse) is adequate to ensure that the necessary safety margins against pullout are maintained in the hot-leg region. The licensee is therefore proposing to inspect the tubes in the hot-leg tubesheet region such as to ensure a minimum of 17 inches of effective engagement, well in excess of the length that the Westinghouse analyses indicate is needed. Based on the following considerations, the NRC staff concludes the proposed 17-inch engagement length is acceptable to ensure the structural integrity of the tubesheet joint:

- Pullout tests of 9 samples performed for a comparable plant with the same model of steam generators indicate that the radial contact pressure between the tube and tubesheet produced by the tube hydraulic expansion, coupled with the contact pressure due to differential thermal expansion between the tube and tubesheet (due to a higher thermal expansion coefficient for the Alloy 600 TT tubing as compared to the A508 steel tubesheet) for joint temperatures ranging from room temperature to 600 °F, is such as to require an engagement distance of about 1 to 5 inches to ensure the appropriate safety margins against pullout (for both plants). This 1- to 5-inch range reflects considerable scatter in the pullout data but is well within the proposed 17-inch inspection zone.

- The primary coolant system pressure inside the tube exceeds the average pressure outside the tube over the length of the tube to tubesheet crevice, thus acting to tighten the joint relative to unpressurized conditions under which the pullout tests were performed for the comparable plant. (The pressure differential across the tube wall is reduced in the revised analyses (discussed above) relative to the original analysis, but remains positive when averaged over the 17-inch inspection zone.)

- The amount of tubesheet bore dilation and resulting change in joint contact pressure would be expected to vary in a linear fashion from top to bottom of the tubesheet. Given the neutral axis to be at approximately the axial mid-point of the tubesheet thickness (i.e., approximately 11 inches below the top of the tubesheet), tubesheet bore dilation effects would be expected to further tighten the joint from 11 inches below the top of the tubesheet to 17 inches below the top of the tubesheet which would be the lower limit of the proposed tubesheet region inspection zone. Combined with the effects of the joint tightening associated with differential pressure across the tube wall, contact pressure over at least a 6-inch distance will be higher than the contact pressure simulated in the pullout tests.

- At the periphery of the tubesheet, the magnitude of the bending moment decreases to between zero and a value opposite in sign with respect to the moment acting inside the peripheral region. The amount of the decrease depends on the degree of structural restraint against tubesheet rotation at the peripheral edge of the tubesheet. If the bending moment is zero, tubesheet bore dilation effects from tubesheet bow will have no influence on contact pressure between the tube and tubesheet. A moment of the opposite sign would tend to increase contact pressures for the 11 inches above the neutral plane, thus increasing the contact pressure in this zone relative to that simulated in the pullout tests.

3.2.2 Joint Leakage Integrity

If no credit is to be taken for the presence of the tube-to-tubesheet weld, a potential leak path between the primary and secondary sides is introduced between the hydraulically-expanded tubing and the tubesheet. In addition, not inspecting the tubing in the lower 5 inches of the tubesheet region may lead to an increased potential for 100 percent throughwall flaws in this zone and the potential for leakage of primary coolant through the crack and up between the hydraulically-expanded tubes and tubesheet to the secondary system. Operational leakage integrity is assured by monitoring primary to secondary leakage relative to the applicable TS LCO limits. However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during design basis accidents to exceed the accident leakage performance criteria 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 by limiting primary-to-secondary leakage to 1 gallon per minute (gpm) in the faulted SG during MSLB.

To support its H* criterion (discussed above), Westinghouse has developed a detailed leakage prediction model, which considers the resistance to leakage from cracks located within the thickness of the tubesheet. The NRC staff has neither reviewed nor accepted this model. For the proposed 17-inch inspection zone, Westinghouse cited a number of qualitative arguments supporting a conclusion that a minimum 17-inch engagement length ensures that leakage during MSLB will not exceed two times the observed leakage during normal operation. Westinghouse refers to this as the "bellwether approach." Thus, for an SG leaking at the TS LCO limit (i.e., 150 gallons per day (gpd) or 0.104 gpm) under normal operating conditions, Westinghouse estimates that leakage would not be expected to exceed 0.208 gpm, which is less than the 0.347 gpm assumed in the licensing basis accident analyses for MSLB.

The factor of 2 upper bound is based on the Darcy equation for flow through a porous media where leakage rate would be proportional to differential pressure. Westinghouse considered normal operating pressure differentials between 1200 and 1400 psi and accident differential pressures on the order of 2560 to 2650 psi, essentially a factor of 2 difference. The factor of 2 as an upper bound is based on a premise that the flow resistance between the tube and tubesheet remains unchanged. Westinghouse states that the flow resistance varies as a log normal linear function of joint contact pressure. The NRC staff finds that the factor of 2 upper bound is reasonable given the stated premise. The staff notes that the assumed linear relationship between leak rate and differential pressure is conservative relative to alternative models such as the Bernoulli or orifice models, which assume leak rate to be proportional to the square root of differential pressure.

The NRC staff reviewed the qualitative arguments developed by Westinghouse regarding the conservatism of assuming that flow resistance between the expanded tubing and the tubesheet does not decrease under the most limiting accident relative to normal operating conditions. Most of the Westinghouse observations are based on insights derived from the finite element analyses performed to assess joint contact pressures and from test data relating leak flow resistance to joint contact pressure, neither of which has been reviewed by the staff in detail. The Westinghouse data indicate that for all tubes there is a zone of at least 6 inches in the upper 17 inches of the tubesheet (from 11 to 17 inches below the top of the tubesheet) where there is an increase in joint contact pressure, and, thus, leak flow resistance, due to higher primary pressure inside the tube and changes in tubesheet bore dilation along the length of the tubes.

The revised analyses described in the licensee's March 9, 2007 and March 26, 2007, letters (discussed in Section 5.1 of this safety evaluation) do not affect this result, although they do revise the amount of flow resistance expected. The original analysis concluded that at 17 inches below the top of the tubesheet, the flow resistance would increase by about 60 percent in going from normal operating conditions to steam line break conditions. Based on the elevated temperature leak rate testing used in the revised analysis, the licensee concluded there would still be an increase in flow resistance associated with steam line break conditions, but it would be at least 20 percent rather than 60 percent.

Although joint contact pressures and leak flow resistance decrease over other portions of the tube length, Westinghouse expects a net increase in total leak flow resistance on the basis of its insights from leakage test data that leak flow resistance is more sensitive to changes in joint contact pressure as contact pressure increases due to the linear log normal nature of the relationship. The NRC staff's depth of review did not permit it to credit this aspect of the Westinghouse assessment. However, it is clear from the above discussion that there should be no significant reduction in leakage flow resistance when going from normal operating to accident conditions.

Finally, the NRC staff has considered that undetected cracks in the lower 5 inches are unlikely to produce leakage rates during normal operation that would approach the TS LCO operational leakage limits during normal operation, thus providing additional confidence that such cracks will not result in leakage in excess of the values assumed in the accident analyses. Any axial cracks will be tightly clamped by the tubesheet against opening of the crack faces. In addition, little of the end cap pressure load should remain in the tube below 17 inches and therefore any circumferential cracks would be expected to remain tight. Thus, irrespective of the flow resistance in the upper 17 inches of the tubesheet between the tube and tubesheet, the tightness of the cracks themselves should limit leakage to very small values.

Based on the above, the NRC staff concludes that there is reasonable assurance that the proposed one time exclusion of the lower 5 inches of the tubes in the tubesheet region from the tube inspection and plugging and repair requirements will not impair the leakage integrity of the tube-to-tubesheet joint.

3.3 Administrative TS Changes

The licensee proposed two administrative changes: (1) to correct a page number in the TS Table of Contents and (2) to delete two blank pages in TS Section 5.0. The NRC staff has no objection to either of these changes.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change a surveillance requirement. The staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (71 FR 51230). Accordingly, these amendments meet 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 these amendments.

6.0 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.

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