April 18, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer
and Senior Vice President
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO TECHNICAL SPECIFICATIONS FOR THE STEAM GENERATOR PROGRAM (TAC NOS. MD8158 AND MD8159)

Dear Mr. Pardee:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 150 to Facility Operating License No. NPF-72 and Amendment No. 150 to Facility Operating License No. NPF-77 for the Braidwood Station (Braidwood), Units 1 and 2, respectively. The amendments are in response to your application dated February 25, 2008, as supplemented by letters dated March 27, 2008, and April 9, 2008.

The amendments revise Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," and TS 5.6.9, "Steam Generator (SG) Tube Inspection Report." For TS 5.5.9, the amendment replaces the existing alternate repair criteria in the provisions for SG tube repair criteria during Braidwood, Unit 2, Refueling Outage 13 and the subsequent operating cycle. For TS 5.6.9, three new reporting requirements are added to the existing seven requirements for Braidwood, Unit 2. These changes only affect Braidwood, Unit 2; however, this action is docketed for Braidwood, Units 1 and 2, because the TS are common to both units.

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

Sincerely,

/RA/

Marshall J. David, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 150 to NPF-72
2. Amendment No. 150 to NPF-77
3. Safety Evaluation

cc w/encls: See next page

Mr. Charles G. Pardee April 18, 2008
Chief Nuclear Officer
and Senior Vice President
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO TECHNICAL SPECIFICATIONS FOR THE STEAM GENERATOR PROGRAM (TAC NOS. MD8158 AND MD8159)

Dear Mr. Pardee:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 150 to Facility Operating License No. NPF-72 and Amendment No. 150 to Facility Operating License No. NPF-77 for the Braidwood Station (Braidwood), Units 1 and 2, respectively. The amendments are in response to your application dated February 25, 2008, as supplemented by letters dated March 27, 2008, and April 9, 2008.

The amendments revise Technical Specification (TS) 5.5.9, “Steam Generator (SG) Program,” and TS 5.6.9, “Steam Generator (SG) Tube Inspection Report." For TS 5.5.9, the amendment replaces the existing alternate repair criteria in the provisions for SG tube repair criteria during Braidwood, Unit 2, Refueling Outage 13 and the subsequent operating cycle. For TS 5.6.9, three new reporting requirements are added to the existing seven requirements for Braidwood, Unit 2. These changes only affect Braidwood, Unit 2; however, this action is docketed for Braidwood, Units 1 and 2, because the TS are common to both units

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission’s biweekly *Federal Register* notice.

Sincerely,
/RA/
Marshall J. David, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456 and STN 50-457
Enclosures:
1. Amendment No. 150 to NPF-72
2. Amendment No. 150 to NPF-77
3. Safety Evaluation
cc w/encls: See next page

DISTRIBUTION:
PUBLIC
LPL3-2 R/F
RidsNrrDorlLpl3-2
RidsNrrDorlDpr
RidsNrrLAEWhitt
RidsNrrPMMDavid
RidsAcrsAcnw&mMailCenter
RidsRgn3MailCenter
RidsNrrDciCsgb
RidsNrrDirsItsb
RidsOgcRp
GHill (4)
AJohnson, NRR
EMurphy, NRR

ADAMS Accession Nos. Pkg ML080920889 Ltr. Accession No. ML080920879 *SE date NRR-058

OFFICE	LPL3-2/PM	LPL3-2/LA	DCI/CSGB/BC	DIRS/ITSB/BC	OGC	LPL3-2/BC
NAME	MDavid:sp	EWhitt	AHiser*	RElliott	MSmith	RGibbs
DATE	4/14/08	4/14/08	04/10/08	4/16/08	4/16/08	4/18/08

OFFICIAL RECORD COPY

Braidwood Station, Units 1 and 2

cc:

Corporate Distribution
Exelon Generation Company, LLC
Via e-mail

Braidwood Distribution
Exelon Generation Company, LLC
Via e-mail

Mr. Dwain W. Alexander, Project Manager
Westinghouse Electric Corporation
Via e-mail

Ms. Bridget Little Rorem
Appleseed Coordinator
Via e-mail

Howard A. Learner
Environmental Law and Policy
  Center of the Midwest
Via e-mail

Braidwood Resident Inspector
U.S. Nuclear Regulatory Commission
Via e-mail

Ms. Lorraine Creek
RR 1, Box 182
Manteno, IL   60950

Illinois Emergency Management Agency
Division of Nuclear Safety
Via e-mail

Will County Executive
Via e-mail

Attorney General
Springfield, IL 62701
Via e-mail

Chairman, Ogle County Board
Post Office Box 357
Oregon, IL 61061

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
License No. NPF-72

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated February 25, 2008, as supplemented by letters dated March 27, 2008, and April 9, 2008, and 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 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 150 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to the return to service from the Braidwood Station, Unit 2 spring 2008 Refueling Outage 13.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance:
April 18, 2008

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
License No. NPF-77

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated February 25, 2008, as supplemented by letters dated March 27, 2008, and April 9, 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 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 150 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to the return to service from the Braidwood Station, Unit 2 spring 2008 Refueling Outage 13.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: April 18, 2008

ATTACHMENT TO LICENSE AMENDMENT NOS. 150 AND 150

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Facility Operating Licenses and the Appendix A Technical Specifications (TSs) 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 NPF-72	License NPF-72
License Page 3	License Page 3
License NPF-77	License NPF-77
License Page 3	License Page 3
TSs	TSs
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
5.6-17	5.6-17
5.5-18	5.5-18
5.5-19	5.5-19
5.5-20	5.5-20
5.5-21	5.5-21
5.5-22	5.5-22
----	5.5-23
5.6-6	5.6-6
----	5.6-7

(3) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I 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 or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels is not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 150, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Exelon Generation Company, LLC pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts are 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

(5) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I 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 or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at reactor core power levels is not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 150, and the Environmental Protection Plan contained in Appendix B, both of which are attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Emergency Planning</u>

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. NPF-72

AND AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated February 25, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080560512), as supplemented by letters dated March 27, 2008 (ADAMS Accession No. ML080880055), and April 9, 2008 (ADAMS Accession No. ML081000582), Exelon Generation Company, LLC (EGC, the licensee) submitted a license amendment request to change the technical specifications (TSs) for Braidwood Station (Braidwood), Unit 2. The request proposed changes to the repair requirements of TS 5.5.9, “Steam Generator (SG) Program,” and to the reporting requirements of TS 5.6.9, “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 Braidwood, Unit 2, during Refueling Outage 13 and the subsequent operating cycle of Braidwood, Unit 2.

The supplemental letters dated March 27, 2008, and April 9, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change NRC staff’s original proposed no significant hazards consideration determination published in an individual notice in the *Federal Register* on March 11, 2008 (73 FR 13029).

In its letters dated February 25, 2008, and March 27, 2008, the licensee submitted Westinghouse Electric Company (Westinghouse) documents, LTR-CDME-08-11-P, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated January 31, 2008, and LTR-CDME-08-43-P, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11-P – Attachment," dated March 18, 2008. The documents contained proprietary information and the affidavits, executed by Westinghouse requesting that NRC withhold the proprietary information from the public, were also submitted in the two letters. The NRC letters 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, were issued on March 11, 2008, (ADAMS Accession No. ML080660473, for LTR-CDME-08-11-P) and April 4, 2008 (ADAMS Accession No. ML080920688, for LTR-CDME-08-43-P). There is no proprietary information in this safety evaluation (SE).

2.0 BACKGROUND

Braidwood, Unit 2, has four Westinghouse Model D5 SGs. There are 4570 thermally treated Alloy 600 tubes in each SG, each with an outside diameter of 0.750 inches and a nominal wall thickness of 0.043 inches. The tubes are hydraulically expanded for the full depth of the tubesheet (21.2 inches) 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 Braidwood or other nuclear power plants in the United States. As a result, most plants, including Braidwood, had been using bobbin probes for inspecting the length of tubing within the tubesheet. Because 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 (Catawba), Unit 2, which has Westinghouse Model D5 SGs. Like Braidwood, 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 of service, which is similar to the service experience that the SGs at Braidwood have accumulated, with a comparable hot-leg operating temperature. The crack-like indications at Catawba were found in bulges (or over-expansions) in the tubesheet region, in the tack expansion region, and near 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. The purpose of the tack expansion is to facilitate performing the tube to tubesheet weld.

As a result of the Catawba findings, the Braidwood licensee expanded the scope of rotating coil inspections to include overexpansions (OXPs) during the spring 2005 Unit 2 refueling outage and reported that they found no degradation. During the fall 2006 Unit 2 refueling outage, Braidwood again performed rotating coil inspections of OXPs within the tubesheet. The inspections focused on the upper 17 inches of the tube within the tubesheet, because 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 staff approved restricting the inspection and repair of tubes that were found to have flaws to the upper 17 inches of the tube within the hot-leg tubesheet, in Amendment No. 141 for Braidwood, Unit 2, on October 24, 2006. Amendment No. 141 applied to Refueling Outage 12 and the subsequent operating cycle.

By letter dated November 29, 2007 (ADAMS Accession No. ML073450563), the licensee requested an amendment for Byron and Braidwood, which would make the inspection and repair modifications of Amendment 141 permanent and would add some additional reporting requirements under TS 5.6.9, "Steam Generator (SG) 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 NRC staff informed WCNOC during a phone call on January 3, 2008, that it had not provided sufficient information to allow the NRC staff to review and approve the permanent license amendment request.

Because the lack of information in the technical analysis mentioned above also affected Braidwood, the licensee submitted a revised application on February 25, 2008. The revised application proposed to modify TS 5.5.9 and 5.6.9 with a more conservative interim alternate repair criteria (IARC) approach that was only applicable to Braidwood, Unit 2, for Refueling Outage 13 and the subsequent operating cycle.

On March 11, 2008 (ADAMS Accession No. ML080980500), the NRC staff provided the licensee with fourteen questions, by electronic mail, regarding the February 25, 2008, license amendment application for Braidwood. By letters dated March 27, 2008, and April 9, 2008, the licensee submitted revisions to the license amendment request. The revisions clarified the application of the more conservative IARC during Refueling Outage 13 inspections and any inspections performed during the subsequent operating cycle for Braidwood, Unit 2.

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 Braidwood are in TS 3.4.19, "Steam Generator (SG) Tube Integrity," and TS 5.5.9, and in TS 5.6.9, respectively.

In the improved standard technical specifications (STS) in NUREG-1431 for Westinghouse plants like Braidwood, TS 5.5.9 requires that an SG tube program be established and implemented to ensure that SG tube integrity is maintained. SG tube integrity is maintained by meeting specified performance criteria (in TS 5.5.9.b) for structural and leakage integrity, consistent with the plant design and licensing basis. TS 5.5.9 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 5.5.9 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 number and portions of tubes inspected and methods of 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 (except as indicated above regarding the one-cycle application of a limited scope of inspection in the tubesheet region). The applicable tube repair criteria, specified in TS 5.5.9.c, are that tubes found by an inservice inspection (ISI) to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube-wall thickness shall be plugged.

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and 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.

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 that 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 of 10 CFR further requires, in part, that throughout the service life of a pressurized-water reactor (PWR) facility like Braidwood, 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 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 that 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 Braidwood is being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed.

The licensee's proposed changes to TS 5.5.9 are to stay within the GDC requirements for the SG tubes and to maintain the accident analysis and consequences that the NRC staff has reviewed and approved for the postulated DBAs for SG tubes. Braidwood Amendment No. 141 modified the STS wording at Braidwood to restrict the required inspection and plugging in the hot-leg tubesheet region to the uppermost 17 inches of the tubesheet region for Refueling Outage 12 and the subsequent operating cycle of Braidwood, Unit 2. This excluded the lowermost 4 inches of the tubesheet on the hot-leg side from the TS inspection and plugging requirements. This license amendment also added a requirement that all tubes found with flaws in the upper 17 inches of the tubesheet region on the hot-leg side be plugged to provide added assurance that tube-to-tubesheet joint integrity would be maintained.

The requested amendment is applicable to Braidwood, Unit 2 for Refueling Outage 13 and the subsequent operating cycle of Braidwood, Unit 2. The license amendment request differs from Amendment No. 141 in a number of ways. First, the lowermost 4 inches of the tubesheet would no longer be excluded from the TS inspection requirements in TS 5.5.9.d. The lowermost 4 inches would be subject to the same inspection requirements as the rest of the tubing. Second, any flaws in the lowermost 4 inches of the tubesheet would not be excluded from requirements to plug. Under the requested amendment, flaws found in the lowermost 4 inches of tubing would be subject to a specified ARC 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 requested amendment would include new reporting requirements to allow the NRC staff to monitor the implementation of the amendment. As with Amendment

No. 141 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 5.5.9.c currently states:

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate repair criteria discussed in TS 5.5.9.c.4. For Unit 2 only, during Refueling Outage 12 and the subsequent operating cycle, flaws identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection.

2. Sleeves found by inservice inspection to contain flaws with a depth equal to or exceeding the following percentages of the nominal sleeve wall thickness shall be plugged:

    i. TIG welded sleeves (per TS 5.5.9.f.2.i): 32%

3. Tubes with a flaw in a sleeve to tube joint that occurs in the sleeve or in the original tube wall of the joint shall be plugged.

4. The following tube repair criteria may be applied as an alternate to the 40% depth-based criteria of Technical Specification 5.5.9.c.1:

    i. For Unit 2 only, during Refueling Outage 12 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 or repair.

The criteria would be revised as follows, as noted in strikeout and bold type:

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate repair criteria discussed in TS 5.5.9.c.4. For Unit 2 only, during Refueling Outage 1~~2~~**3** and the subsequent operating cycle, flaws identified in the portion of the tube from the top of the ~~hot leg~~ tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection.

2. Sleeves found by inservice inspection to contain flaws with a depth equal to or exceeding the following percentages of the nominal sleeve wall thickness shall be plugged:

    i. **For Unit 2 only**, TIG welded sleeves (per TS 5.5.9.f.2.i): 32%

3. Tubes with a flaw in a sleeve to tube joint that occurs in the sleeve or in the original tube wall of the joint shall be plugged.

4. The following tube repair criteria ~~may~~ **shall** be applied as an alternate to the 40% depth-based criteria of Technical Specification 5.5.9.c.1:

    i. For Unit 2 only, ~~during Refueling Outage 12 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 or repair.~~ **during Refueling Outage 13 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 or repair. 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 axial indications found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging or repair.**

    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 exceeds 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.

TS 5.6.9 currently states:

5.6.9 Steam Generator (SG) 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 Specification 5.5.9, 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 or repaired during the inspection outage for each active degradation mechanism,

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

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

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

i. Repair method utilized and the number of tubes repaired by each repair method.

TS 5.6.9 would be revised as follows to include the addition of the following 3 new reporting requirements:

h. The effective plugging percentage for all plugging and tube repairs in each SG, ~~and~~

i. Repair method utilized and the number of tubes repaired by each repair method~~.~~,

j. For Unit 2, following completion of an inspection performed in Refueling Outage 13 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, and whether initiated on primary or secondary side for each service induced flaw detected 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 5.5.9 c.4.i,

k. For Unit 2, following completion of an inspection performed in Refueling Outage 13 (and any inspections performed in the subsequent operating

cycle), the operational primary to secondary leakage rate observed (greater than three gallons per day) in each steam generator (if it is not practical to assign the leakage to an individual steam generator, 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

l. For Unit 2, following completion of an inspection performed in Refueling Outage 13 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the lowermost 4-inches of tubing 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 interim TS changes under Amendment No. 141 exempted the lower 4-inch portion of the tube within the 21-inch-deep tubesheet from an inspection and exempted tubes with flaw indications in this region from being removed from service (i.e., plugged). The requested amendment, 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. The amendment takes 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 requested amendment that is the subject of this SE differs fundamentally from Amendment No. 141 and is a more conservative approach. The requested 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 requested 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 5.5.9.c, “Provisions for SG tube repair criteria”

The 40 percent depth-based tube repair criterion in TS 5.5.9.c is intended to ensure, in conjunction with other elements of TS 5.5.9, that tubes accepted for continued service (i.e., not plugged) satisfy the performance criteria for structural integrity in TS 5.5.9.b.1 and the performance criteria for accident leakage integrity in TS 5.5.9.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 request are alternatives to this 40 percent depth-based criterion.

4.2.1.1 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 the 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 IARC 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 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 react 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 stated 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 Braidwood will be inspected at Refueling Outage 14, should any crack indications be found during Refueling Outage 13. In addition, the 203- and 94-degree criteria conservatively take no credit for the effects of friction between the tube and tubesheet in any portion of the tube-to-tubesheet joint, in reacting out a portion of the axial end cap load before it reaches the cracked cross-section. Thus, the NRC 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 a 0.7-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% through wall, irrespective of the actual depth of the flaw. With this assumption, the license 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 from one another 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, because 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 IARC 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 would continue to include the current interim requirement (Amendment No. 141) 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 because 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 Limiting Condition for Operation 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 leakage performance criteria in TS 5.5.9.b.2, including the leakage values assumed in the plant licensing basis accident analyses. The licensee stated that this is ensured for Braidwood by limiting primary-to-secondary leakage to 0.50 gallon per minute in the faulted SG during an MSLB accident. In this regard, the MSLB accident is the most limiting of the accidents affected by the requested amendment.

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 stated 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, it has been assumed to be constant with joint contact pressure at a value that 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 because 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.

The licensee stated in its March 27, 2008, response to an NRC staff RAI 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 stated 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 leakage and that an administrative limit will be established to not exceed the calculated value. Because 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.

In its letter dated April 9, 2008, the licensee submitted a regulatory commitment that stated the 2.5 factor would be used in the completion of its CM and OA upon implementation of the IARC in this amendment. This is an IARC because it applies only to Refueling Outage 13 and the subsequent operating cycle.

The NRC staff has previously found 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 (ADAMS Accession No. ML072000343). 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 5.5.9.d, “Provisions for SG tube inspections”

With the plant entry into Refueling Outage 13, the sentence added to TS 5.5.9.d in Amendment No. 141 is no longer applicable. This is the statement that, "For Unit 2 only, during Refueling Outage 12 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded." Therefore, in Refueling Outage 13, the inspection requirements of TS 5.5.9.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 5.5.9.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. This is to say that the licensee is proposing to inspect the entire tube in the tubesheet in the requested amendment, whereas in Amendment No. 141, the licensee had proposed and the NRC accepted that the lower 4 inches of the tube in the tubesheet did not have to be inspected.

4.2.3 Proposed Change to TS 5.6.9, “Steam Generator (SG) 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 requested 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 STS and the Braidwood TSs state specifically that the tube-to-tubesheet welds are not part of the tube. Therefore, the requirements of TS 5.5.9 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, 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 Braidwood. An industry peer review was recently conducted for the Catawba, Unit 2, 2007 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.

4.3 Summary

Based on the above evaluation, the NRC staff finds that the requested license amendment, which is applicable only to Braidwood, Unit 2 for Refueling Outage 13 and the subsequent operating cycle of Braidwood, Unit 2, 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 requested amendment meets 10 CFR 50.36 and, thus, the requested amendment is acceptable.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed no Significant Hazards Consideration Determination, and Opportunity for Hearing for this amendment was published in the *Federal Register* on March 11, 2008 (73 FR 13029). Therefore, this amendment is being issued after the 30-day public comment period has expired, but before the 60-day hearing request period has expired.

The Commission may issue the license amendment before the expiration of the 60-day hearing period provided that its final determination is that the amendment involves no significant hazards consideration. Because this amendment is being issued prior to the expiration of the 60-day period, the NRC staff has made a final finding of no significant hazards consideration, which is given below.

In its application, the licensee made a determination that the amendment request involved no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this determination 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 provided its analysis of the issue of no significant hazards consideration in Attachment I to its February 25, 2008, application, 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 the steam generator tube rupture (SGTR), 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, because 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 D5 steam generators has shown that axial loading of the tubes is negligible during an SSE.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below 17 inches from the top of the tubesheet 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 is maintained by limiting the allowable ligament size for a circumferential crack to remain in service to 214 degrees below 17 inches from the top of the tubesheet. 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 214 degrees, which takes into account eddy current uncertainty and crack growth rate. It has been shown that a circumferential crack with an azimuthal extent of 214 degrees meets the performance criteria of NEI [Nuclear Energy Institute] 97-06, Rev. 2, "Steam Generator Program Guidelines" and the Draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Likewise, a visual inspection will be conducted as necessary to confirm that a circumferential crack of greater than 294 degrees does not remain in service in the tube-to-tubesheet weld metal in any tube thereby mitigating the potential for tube pullout. 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 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. Because normal operating leakage is limited to 0.10 gallons per minute (gpm) (or 150 gallons per day (gpd)), 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 0.5 gpm. This value is within the accident analysis assumptions for the limiting design basis accident for Braidwood, Unit 2, which is the postulated SLB event.

Based on the above, the performance criteria of NEI-97-06, Rev. 2 and RG 1.121 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 kind of 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, Rev. 2 and RG 1.121 are used as the basis in the development of the interim alternate repair criteria (IARC) methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 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 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-tubeshet [tubesheet] weld, the Westinghouse analysis, provided in report "LTR-CDME-08-11 P-Attachment," defines 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 IARC will not result in unacceptable primary-to-secondary leakage during all plant conditions, including transients and postulated accident conditions.

Based on the above, it is concluded that the proposed changes do not result in any reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis of no significant hazards consideration given above. Although the licensee revised the allowable ligament sizes for a circumferential crack in its supplemental letter dated March 27, 2008, and modified the proposed new TS 5.6.9 reporting requirements in its supplemental letter dated April 9, 2008, the new sizes are shorter (i.e., more conservative) than the values proposed in the February 25, 2008, request, and the modified reporting requirements are for the purpose of clarity and consistency with other information in the request. Therefore, these changes do not affect the above no significant hazards consideration analysis. In addition, the NRC staff finds that the licensee did not need to consider whether the requested amendment would increase the probability or consequences of the locked rotor or control rod ejection accidents because the licensee determined that the change would not increase the probability or consequences of the SLB accident and that the SLB is the limiting accident affected by the requested amendment.

Based on its review of the above analysis, the licensee's letters, and the reduced proposed allowable ligament sizes, the NRC staff concludes that the three standards of 10 CFR 50.92 are satisfied by the above analysis. Therefore, the NRC staff has determined that the amendment involves no significant hazards consideration.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility's component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards finding with respect to this amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) involve a significant increase in the probability or consequences of an accident previously evaluated; or (b) create the possibility of a new or different kind of accident from any accident previously evaluated; or (c) involve a significant reduction in a margin of safety; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (3) such activities will be conducted in compliance with the Commission's regulations; and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Johnson, NRR

Date: April 18, 2008