March 30, 2007

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BYRON STATION, UNIT NOS. 1 AND 2, AND BRAIDWOOD STATION, UNITS NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: STEAM GENERATOR TUBE SURVEILLANCE PROGRAM (TAC NOS. MC8966, MC8967, MC8968, AND MC8969)

Dear Mr. Crane:

The Nuclear Regulatory Commission (NRC, Commission) has issued the enclosed Amendment No. 150 to Facility Operating License No. NPF-37 and Amendment No. 150 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 144 to Facility Operating License No. NPF-72 and Amendment No. 144 to Facility Operating License No. NPF-77 for the Braidwood Station, Units 1 and 2, respectively. The amendments are in response to your application dated November 18, 2005, as supplemented by letters dated August 18 and September 28, 2006, and February 15, February 23, and March 7, 2007.

The amendments revise the existing steam generator (SG) tube surveillance program using Technical Specification Task Force Traveler No. 449 (TSTF-449), Revision 4, “Steam Generator Tube Integrity” as a basis. The amendments also revise TS 5.5.9, “Steam Generator Tube Surveillance Program,” regarding the required SG inspection scope for Byron Station, Unit No. 2, during outage number 13 and subsequent operating cycle. A similar approval was granted for Braidwood Station, Unit 2 by letter from the NRC dated October 24, 2006.

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/

Robert F. Kuntz, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 150 to NPF-37
2. Amendment No. 150 to NPF-66
3. Amendment No. 144 to NPF-72
4. Amendment No. 144 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

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/

Robert F. Kuntz, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 150 to NPF-37
2. Amendment No. 150 to NPF-66
3. Amendment No. 144 to NPF-72
4. Amendment No. 144 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

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EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
License No. NPF-37

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated November 18, 2005, as supplemented by letters dated August 18 and September 28, 2006, and February 15, February 23, and March 7, 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 2.C.(2) of Facility Operating License No. NPF-37 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 within 60 days of the date of issuance.

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: March 30, 2007

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
License No. NPF-66

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated November 18, 2005, as supplemented by letters dated August 18 and September 28, 2006, and February 15, February 23, and March 7, 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 2.C.(2) of Facility Operating License No. NPF-66 is hereby amended to read as follows:

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

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 150 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, 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 within 60 days of the date of issuance.

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: March 30, 2007

ATTACHMENT TO LICENSE AMENDMENT NOS. 150 AND 150

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

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

Remove	Insert
Unit No. 1 License Page 3	Unit No. 1 License Page 3
Unit No. 2 License Page 3	Unit No. 2 License Page 3
ii	ii
1.1-4	1.1-4
3.4.13-1	3.4.13-1
3.4.13-2	3.4.13-2
----------	3.4.19-1
----------	3.4.19-2
5.5-7	5.5-7
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	5.5-17
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	--------
5.5-23	--------
5.5-24	--------
5.5-25	--------
5.5-26	--------
5.5-27	--------
5.6-6	5.6-6
------	5.6-7

<u>EXELON GENERATION COMPANY, LLC</u>

<u>DOCKET NO. STN 50-456</u>

<u>BRAIDWOOD STATION, UNIT 1</u>

<u>AMENDMENT TO FACILITY OPERATING LICENSE</u>

Amendment No. 144
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 November 18, 2005, as supplemented by letters dated August 18 and September 28, 2006, and February 15, February 23, and March 7, 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 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. 144 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 within 60 days of the date of issuance.

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: March 30, 2007

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 144
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 November 18, 2005, as supplemented by letters dated August 18 and September 28, 2006, and February 15, February 23, and March 7, 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 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

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

The Technical Specifications contained in Appendix A as revised through Amendment No. 144 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 if its issuance and shall be implemented within 60 days of the date of issuance.

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: March 30, 2007

<u>ATTACHMENT TO LICENSE AMENDMENT NOS. 144 AND 144</u>

<u>FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77</u>

<u>DOCKET NOS. STN 50-456 AND STN 50-457</u>

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

<u>Remove</u>	<u>Insert</u>
Unit No. 1 License Page 3	Unit No. 1 License Page 3
Unit No. 2 License Page 3	Unit No. 2 License Page 3
ii	ii
1.1-4	1.1-4
3.4.13-1	3.4.13-1
3.4.19-2	3.4.13-2
----------	3.4.19-1
----------	3.4.19-2
5.5-7	5.5-7
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	5.5-17
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	--------
5.5-23	--------
5.5-24	--------
5.5-25	--------
5.5-26	--------
5.5-27	--------
5.6-6	5.6-6

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC) dated November 18, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML053320304), as supplemented by letters dated August 18 (ADAMS Accession No. ML062400348) and September 28, 2006 (ADAMS Accession No. ML062710559), February 15 (ADAMS Accession No. ML070470133), February 23 (ADAMS Accession No. ML070540421), and March 7, 2007 (ADAMS Accession No. ML070670126), Exelon Generation Company, LLC (the licensee) requested changes to the Technical Specifications (TSs) for Byron Station (Byron), Unit Nos. 1 and 2, and Braidwood Station (Braidwood), Units 1 and 2. The supplements to the November 18, 2005, application provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 23, 2006 (71 FR 29676).

The proposed changes would revise the existing steam generator (SG) tube surveillance program. The changes are modeled after TS Task Force Traveler No. 449 (TSTF-449), Revision 4, "Steam Generator Tube Integrity," and the model safety evaluation prepared by the NRC and published in the *Federal Register* on March 2, 2005 (70 FR 10298). In this regard, the scope of the application includes changes to the definition of leakage, changes to the primary-to-secondary leakage requirements, changes to the SG tube surveillance program (SG tube integrity), changes to the SG reporting requirements, and associated changes to the TS Bases.

The proposed changes would also make permanent a 17-inch inspection zone at the top of the hot leg tubesheet and associated tube repair criteria which are currently authorized in the TS for the most recent refueling outage and subsequent operating cycle for Byron, Unit No. 2 and

Braidwood, Unit 2. As documented in the following safety evaluation (SE), the NRC is only approving this change for Refueling Outage 13 and subsequent operating cycle for Byron, Unit 2 (Refueling Outage 12 and subsequent operating cycle for Braidwood Unit No. 2 was previously approved by letter from the NRC dated October 24, 2006 (ADAMS Accession No. ML062780503)).

2.0 REGULATORY EVALUATION

The NRC staff finds that the licensee, in Section 5.3 of its November 18, 2005, submittal, identified the applicable regulatory requirements. The regulatory requirements for which the NRC staff based its acceptance include the following:

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 states that the reactor coolant pressure boundary (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 leaktight integrity” (GDC 32). To this end, 10 CFR 50.55a, “Codes and standards,” specifies that components, which are part of the RCPB, must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Section 50.55a further requires, in part, that throughout the service life of a pressurized-water reactor (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 TSs.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents 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 100, “Reactor Site Criteria,” guidelines for offsite doses (or 10 CFR 50.67, “Accident source term,” as appropriate), GDC-19 criteria for control room operator doses, or the NRC-approved licensing basis (e.g., a small fraction of these limits).

2.1 Proposed Changes Modeled on TSTF-449, Revision 4

The background, description, and applicability of the proposed changes associated with the SG tube integrity issue (modeled on TSTF-449) and the applicable regulatory requirements were included in the NRC staff’s model SE published in the *Federal Register* on March 2, 2005. The “Notice of Availability of Model Application Concerning Technical Specification; Improvement To Modify Requirements Regarding Steam Generator Tube Integrity; Using the Consolidated Line Item Improvement Process,” was published in the *Federal Register* on May 6, 2005 (70 FR 24126), making the model SE available for licensees to reference.

2.2 Proposed 17-Inch Tubesheet Inspection Zone and Associated Tube Repair Criteria

The regulatory standard applicable to the NRC staff's evaluation of the licensee's proposal for the proposed 17-inch tubesheet inspection zone and associated tube repair requirement is the same as that cited in the *Federal Register* reference in Section 2.1. The NRC staff has reviewed the licensee's proposal to ensure that it meets the applicable regulations by ensuring that SG tube integrity will continue to be maintained. This means, in part, that margins against structural failure of the tube-to-tubesheet joints will be maintained consistent with the plant design and licensing basis. The design basis refers to the stress limit criteria for normal operating and faulted conditions in the Code, Section III. The licensing basis refers to the structural margins implicit in tube repair criteria in the current TSs. Maintaining tube integrity also means that primary-to-secondary leakage integrity will be maintained consistent with the current licensing basis. This means that leakage during normal operation will be limited to values less than applicable limiting condition for operation (LCO) limits in the TSs. In addition, there must be reasonable assurance that leakage during design basis accidents will not exceed the values in the licensing bases accident analyses.

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes Modeled on TSTF-449, Revision 4

In its November 18, 2006, application and August 18, 2006, and February 15, 2007, supplements, the licensee proposed changes to the TSs that are modeled after TSTF-449. The NRC staff has reviewed these proposed changes and finds they are consistent with TSTF-449 with some differences related mainly to the proposed 17-inch inspection zone and associated tube repair criteria as part of the amendment request and the inclusion of a provision for sleeve repair which is permitted under the current technical specifications. All differences between the proposed changes and TSTF-449 are evaluated below:

It is important to note that the TSs for Byron and Braidwood are common to both units (i.e., both units share the same technical specifications).

3.1.1 Proposed TS 3.4.19, "Steam Generator (SG) Tube Integrity"

This is a new specification which is numbered differently in TSTF-449 as TS 3.4.20. This is an administrative difference that is acceptable to the NRC staff.

Apart from the numbering of the TS, the proposed TS for Byron and Braidwood differs from TSTF-449, only with the use of the words "plugged or repaired" (in three different places) and "plug or repair" (in one place) where TSTF-449 uses the words "plugged [or repaired] and "plug [or repair"], respectively. The intent of TSTF-449 is that the words "or repaired" should be used in the TS only when TS 5.5.9.f provides for the use of one or more specific repair methods. In the case of Byron, TS 5.5.9.f identifies an approved repair method for Unit 2, but states there is no approved repair method for Unit 1. Similarly, for Braidwood, TS 5.5.9.f identifies an approved repair method for Unit 2, but specifically states there is no approved repair method for Unit 1. Thus, the words "or repaired" in proposed TS 3.4.19 is an option that only exists for Byron, Unit No. 2 and Braidwood, Unit 2 even though this is not explicitly stated in proposed TS 3.4.19. This is acceptable to the NRC staff since proposed TS 3.4.19 makes it clear that repair (sleeving) is only an option for Byron, Unit 2 and Braidwood, Unit 2 and is not an option for Byron, Unit No. 1 and Braidwood, Unit 1.

3.1.2 Proposed TS 5.5.9.a, “Provisions for condition monitoring assessments”

This proposed specification also differs from the TSTF with use of the words “plugging or repair” (one place) and “plugged, or repaired” (one place) where the TSTF uses the words “plugging [or repair]” and “plugged, [or repaired],” respectively. This is the same issue as discussed above for proposed specification TS 3.4.19 and is acceptable to the NRC staff for the same reasons.

3.1.3 Proposed TS 5.5.9.b.2, “Accident induced leakage performance criterion”

TSTF-449 states, in part, “Leakage is not to exceed [1 gallons per minute (gpm)] per SG, [except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program].” The exception noted in brackets does not apply to Byron and Braidwood since there are no special accident leakage criteria in proposed TS 5.5.9.c. The proposed TS 5.5.9.b.2 states “Leakage is not to exceed 1 gpm for all SGs.” The proposed “1 gpm for all SGs” is more restrictive than the “1 gpm per SG” in TSTF-449 and is, therefore, acceptable to the NRC staff.

3.1.4 Proposed TS 5.5.9.b.3 regarding operational LEAKAGE performance criteria

Proposed TS 5.5.9.b.3 states, “The operational LEAKAGE performance criteria is specified in LCO 3.4.13, “RCS [reactor coolant system] Operational LEAKAGE."“ The TSTF uses the word “criterion” rather than “criteria” since it is referring to a singular item. However, TS 5.5.9.b.3 serves as a cross reference only and is acceptable to the NRC staff as is. It does not, in any way, modify LCO 3.4.13 which is consistent with the TSTF.

3.1.5 Proposed TS 5.5.9.c, “Provisions for SG tube repair criteria”

The TSTF-449 version of TS 5.5.9.c states:

> 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 [or repaired].
>
> [The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria:
>
> 1.]

Proposed TS 5.5.9.c states (Refueling Outage X refers to Refueling Outage 13 for Byron Station, Unit No. 2 and Refueling Outage 12 for Braidwood Station, Unit 2):

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 tube 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 [X] 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 to or exceeding the following percentages of the nominal sleeve wall thickness shall be plugged:

   i. TIG [tungsten inert gas] welded sleeves (per 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 [X] 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 40 percent repair limit value for the non-sleeved region and the 32 percent value for TIG welded sleeves are consistent with the values in the current TSs and are, therefore, acceptable.

Proposed TS 5.5.9.c.3 is not in the current TSs. This proposed new requirement is more restrictive (i.e., more conservative) than the 40 percent and 32 percent repair limit values for the non-sleeved regions and sleeves, respectively, which would otherwise apply and is, therefore acceptable.

The use of the words “plugged or repaired” in TS 5.5.9.c.1 and “plugging or repair” in TS 5.5.9.c.4 is the same issue as discussed above for proposed specification TS 3.4.19 and is acceptable to the NRC staff for the same reasons.

The alternate repair criteria for the tube (ARC) referenced in TS 5.5.9.c.1 and discussed in TS 5.5.9.c.4 is consistent with the current TS with the exception that for Byron, Unit No. 2 the period of applicability is Refueling Outage 13 and the subsequent operating cycle rather than Refueling Outage 12 and the subsequent operating cycle. As evaluated in detail in Section 3.2 of this safety evaluation, the NRC staff finds the proposed ARC to be acceptable.

The requirement in TS 5.5.9.c.1 applicable to Unit No. 2 (Byron, Unit No. 2 and Braidwood, Unit 2) requiring that tubes be plugged on detection of flaws in the upper 17 inches of the tubesheet is consistent with current TS with the exception noted in the previous paragraph. This requirement is more restrictive than the 40 percent depth-based limit which would otherwise apply. This is because the technical basis for the ARC and the associated 17-inch tubesheet inspection zone (evaluated in Section 3.2 below) did not consider the presence of flaws in the 17-inch inspection zone. Thus, the NRC staff finds this restriction to be acceptable.

In summary, the NRC staff finds that all differences in proposed TS 5.5.9.c relative to TSTF-449 are acceptable.

3.1.6 Proposed TS 5.5.9.d, "Provisions for SG tube inspections"

Consistent with TSTF-449, TS 5.5.9.d states in part, "The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the tube repair criteria." However, proposed TS 5.5.9.d adds the sentence (Refueling Outage X refers to Refueling Outage 12 for Braidwood Station, Unit 2 and Refueling Outage 13 for Byron Station, Unit No. 2), "For Unit 2 only, during Refueling Outage [X] and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the tubesheet is excluded." This exclusion is not in the TSTF, but is associated with the proposed ARC in TS 5.5.9.c.4. As evaluated in Section 3.2 of this SE, the NRC staff finds the exclusion of the portion of tube below 17 inches from the top of the tubesheet to be acceptable.

Proposed TS 5.5.9.d.2 defines specific inspection frequency requirements. The TSTF-449 version is shown in brackets with three different versions applicable to plants with Alloy 600 mill annealed tubing, Alloy 600 thermally treated (TT) tubing, and Alloy 690 TT tubing, respectively. Thus, proposed TS 5.5.9.d.2 lists separate requirements for Unit No. 1 (at Byron and Braidwood), which has Alloy 690 TT tubing, versus those for Unit No. 2, which has Alloy 600 TT tubing. These proposed requirements are consistent with TSTF-449 for the Alloy 690 TT and Alloy 600 TT, respectively, and are, therefore, acceptable

3.1.7 Proposed TS 5.5.9.f, "Provisions for SG tube repair methods"

TSTF-449 places TS 5.5.9.f in brackets and states in a Reviewer's Note that this TS should not be used if there are no approved repair methods. The TSTF-449 version of TS 5.5.9.f states, in part, "All acceptable tube repair methods are listed below." For Byron and Braidwood, this sentence is replaced by the following:

1. There are no approved tube repair methods for the Unit 1 SGs.

2. All acceptable repair methods for the Unit 2 SGs are listed below.

Item 1 above is needed to clarify the words "plugged or repaired" appearing elsewhere in the proposed TS; namely repair is not an option for Unit No. 1. Therefore, the differences in proposed TS 5.5.9.f relative to TSTF-449 are acceptable.

The current TS for Byron and Braidwood define two acceptable repair methods; laser welded sleeving and TIG welded sleeving. Proposed TS 5.5.9.f.2 lists just one acceptable repair method (for Unit No. 2), which is TIG welded sleeving. The description of the TIG welded sleeving methodology is identical to that in the current TSs and, therefore, is acceptable to the NRC staff. The licensee elected not to include the laser welded sleeving methodology in proposed TS 5.5.9.f.2, since the methodology is no longer supported by the vendor (Westinghouse).

3.1.8 Renumbering of TS Section 5.5 “Programs and Manuals”

The removal of requirements from TS Section 5.5 as a part of the proposed amendment results in existing TS requirements being moved to previous pages. Therefore, these pages are reissued as part of the amendment request. This is an administrative change to the TSs which the NRC staff finds acceptable.

3.1.9 Proposed TS 5.6.9, “Steam Generator Tube Inspection Report”

This proposed TS differs from TSTF-449 with use of the words “plugged or repaired” (two places) and “plugging and tube repairs” (one place) where TSTF-449 uses the words “plugged [or repaired]” and “plugging [and tube repairs],” respectively. In addition, proposed TS 5.6.9 includes the provision for reporting “The repair method utilized and the number of tubes repaired by each repair method” which appears in brackets in TSTF-449. This is a related issue to that discussed above for proposed specification TS 3.4.19 and is acceptable to the NRC staff for the same reasons.

Proposed TS 5.6.9 adds three additional reporting requirements (relative to TSTF-449) relating to the implementation of the proposed tubesheet ARC in TS 5.9.9.c and the proposed 17-inch tubesheet inspection zone in TS 5.5.9.d as follows for Byron, Unit No. 2:

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 the primary or secondary side for each indication detected in the upper 17 inches of the tubesheet thickness.

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 SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report, 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 SG. In addition, if the calculated accident leakage rate is less than 2 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.

Proposed requirement ‘j’ will allow the NRC staff to monitor the level of degradation activity in the tubesheet. Proposed requirements ‘k’ and ‘l’ will allow the NRC staff to monitor potential accident leakage from the lowermost 4 inches of the tubesheet where the proposed ARC in TS 5.5.9.c applies and which is proposed to be excluded from the inspection requirements in TS 5.5.9.d. As discussed in Section 3.2 of this SE, the NRC staff has concluded that accident leakage in the lowermost 4 inches of the tubesheet will not increase by more than a factor of two over the normal operating value in this region immediately prior to the accident. Proposed requirement 'l' will allow the NRC staff to monitor how and on what basis the licensee is apportioning observed normal operating leakage between the lowermost 4 inches of the tubesheet and the rest of the SG. The purpose of the observation is to determine the potential accident leak rate contribution from the lowermost 4 inches of the tubesheet. For these reasons, the NRC staff concludes reporting requirements "j" through "l" in proposed TS 5.6.9 are acceptable.

3.1.10 Planned Revisions to the Technical Specification Bases

The NRC staff also reviewed for information the planned changes to the TS Bases associated with the requested TS changes. (The TS Bases are a licensee controlled document not subject to NRC approval.) The NRC staff finds that the planned changes to the TS Bases to be consistent with the TS Bases in TSTF-449 with some minor differences. These differences primarily reflect different assumptions used in the Byron and Braidwood licensing basis accident analyses versus those described in the generic TSTF-449 template. For example, the generic TSTF-449 Bases state, "The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, RCS Operational LEAKAGE, plus the leakage rate associated with a double-ended rupture of a single tube." The planned Bases for Byron and Braidwood state (in TS B 3.4.19), "The analysis of a SGTR event assumes total initial primary to secondary LEAKAGE of 1 gpm for the intact SGs plus the leakage rate associated with a double-ended rupture of a single tube." This is but one example of several statements which needed to be modified from the generic TSTF template to reflect the actual licensing basis for Byron and Braidwood.

The planned changes to the TS Bases also include some discussion of the proposed exclusion (in TS 5.5.9.d) of the zone below 17 inches from the top of the hot leg tubesheet from the inspection requirements from the TSTF template for TS 5.5.9.d.

Finally, the planned changes to the TS Bases includes the words "plugged or repaired" (or different forms of these words) consistent with the requested TS changes.

In summary, the NRC staff finds the planned changes to the TS Bases to be consistent with the TS Bases in TSTF-449, with some minor differences related to the need to be consistent with the actual plant licencing bases and with the proposed TS amendment.

3.2 <u>Proposed 17 Inch Tubesheet Inspection Zone and Associated Tube Repair Criteria</u>

The Byron, Unit No. 1 and Braidwood, Unit 1 SGs are Babcock and Wilcox Canada, replacement Model F SGs with Alloy 690 TT tubes. The Byron Station, Unit No. 2 and Braidwood Station, Unit 2 SGs are original Westinghouse equipment Model D5 SGs with Alloy 600 TT tubing. Based on recent operating experience at plants with Alloy 600 TT tubing, the Alloy 600 TT tubing at the Byron Station, Unit No. 2 and Braidwood Station, Unit 2 are potentially susceptible to stress corrosion cracking (SCC) in the tubesheet region in the near term.

The TSs for Byron, Unit No. 2 and Braidwood, Unit 2 currently specify that the portion of tubing located below 17 inches from the top of the hot leg tubesheet is excluded from inspection and tube repair limit requirements that would otherwise apply. (Thus, the lowermost 4 inches of the 21-inch thick tubesheet are excluded on the hot leg side.) For Braidwood, Unit 2, this exclusion is applicable only to the most recent refueling outage inspection in fall 2006 (i.e., Refueling Outage 12) and the subsequent (current) operating cycle. This exclusion replaced a similar exclusion in the Braidwood, Unit 2 TSs which was applicable to Refueling Outage 11 and the subsequent operating cycle. For Byron, Unit No. 2, this exclusion was applicable only to the refueling outage inspection in fall 2005 (i.e., Refueling Outage 12) and the subsequent (current) operating cycle. Similar amendments that were applicable for one or two cycles have been approved for several other plants with Westinghouse Model D5 and Model F SGs over the past two years.

At the time the amendments, with limited applicability as described above, were approved, the NRC staff SEs stated that some of the licensee's supporting analyses had not been reviewed in detail. (The NRC staff's SE accompanying issuance of License Amendment 141, by letter dated October 24, 2006, for Braidwood Unit 2, ADAMS Accesssion No. ML062780503, serves as a recent example.) Without crediting those portions of the supporting technical analyses which were not reviewed in detail, the NRC staff's SEs concluded that there was adequate technical basis to approve the exclusion on a limited basis, ranging from one to two fuel cycles.

The licensee's original amendment request on November 18, 2005, included a proposal to make the aforementioned exclusions permanent and to add an accompanying reporting requirement. These exclusions would be in the context of the proposed TSs modeled on TSTF-449, Revision 4 and would apply only to Byron, Unit No. 2 and Braidwood, Unit No. 2. The proposed exclusions were consistent with those now in effect for the most recent refueling outage, and subsequent operating cycle. Additional information, pertaining to this request, was provided by letters dated August 16, 2006, February 15, February 23, and March 7, 2007. The February 15 and February 23, 2007, letters provided revised analyses to address recent test results indicating that a fundamental assumption in the original analyses (i.e., the analyses provided with the November 18, 2005, letter) was not justified. Although the revised analyses continue to support the conservatism of the requested exclusions, the NRC staff could not complete its review of these revised analyses in time to support issuance of an amendment, without a limited applicability of the exclusion, prior to the spring 2007 outage at Byron, Unit No. 2. Accordingly, the NRC staff is partially approving the proposed amendment request. Specifically, the subject exclusions will continue to be applicable only during Refueling Outage 13 (in spring 2007) and the subsequent operating cycle for Byron, Unit No. 2 and during Refueling Outage 12 and the subsequent (current) operating cycle for Braidwood, Unit 2. This represents no change from the status quo for Braidwood, Unit 2 other than to restate the 4-inch exclusion requirements in the context of the TS changes in accordance with TSTF-449 and to add a reporting requirement. For Byron, Unit No. 2, the revised amendment request would, in effect, extend the applicability of the 17-inch tubesheet inspection zone and associated tube repair criteria for one additional refueling outage and subsequent operating cycle and add a reporting requirement.

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 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 hydraulically expanded tube and the tubesheet. In addition, the weld serves to make the joint leak tight.

For Byron, Unit No. 2 and Braidwood, Unit 2, the current TSs exempt the lower 4 inches of the 21-inch deep tubesheet region from a tube inspection and, in addition, exclude tubes with flaw indications in the lower 4-inch zone from the need to plug or repair. As discussed in Section 2.2 above, these exclusions only apply during the most recent refueling outage inspection and the current operating cycle at these units. The current operating cycle is scheduled to end in spring 2007 for Byron, Unit No. 2 and in spring 2008 for Braidwood, Unit 2. These exclusions effectively redefine the pressure boundary at the tube-to-tubesheet joint as consisting of a friction or expansion joint with the tube assumed to be hydraulically expanded against tubesheet over the top 17 inches of the tubesheet region. Under this proposal, no credit is taken for the lower 4 inches of the tube or the tube-to-tubesheet weld in contributing to the structural or leakage integrity of the joint. The lower 4 inches of the tube, tubesheet, and weld are assumed not to exist.

The licensee's proposal includes the following additional changes relative to current TS requirements concerning these exclusions:

1. The requested changes include new reporting requirements relating to the implementation of the 17-inch tubesheet inspection zone and associated tube repair criteria as evaluated in Section 3.1 of this SE.

2. The requested changes delete the current TS 5.5.9.b.5, which is applicable only to Byron, Unit No. 2 during refueling outage 12, to perform a rotating coil examination of a 20 percent sample of tubes in the upper 17-inch span of the tubesheet region, including a 20 percent sample of the bulges and over-expansions within the 17-inch tubesheet inspection zone.

3. The requested changes delete the current TS 5.5.9.e.12 that defines the words "bulges" and "over-expansions" and which is applicable only to Byron, Unit No. 2 during refueling outage 12 and the subsequent operating cycle.

The NRC staff finds item 2 to be acceptable because the 17-inch tubeheet inspection zone will be subject to the same inspection requirements as are all other portions of tubing consistent with proposed TSs in accordance with TSTF-449 (except for the lowermost 4 inches of tubing within the tubesheet which would be excluded from the inspection requirements). Through the industry's NEI 97-06, "Steam Generator Tube Integrity," initiative (ADAMS Accession No. ML052710007) and NRC Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," ample guidance is available concerning appropriate inspection techniques (including eddy current probe types) and sampling plans for ensuring effective inspections 17-inch tubesheet inspection zone in proposed TS 5.6.9 will enhance the NRC staff's ability to monitor the level of degradation activity in the tubesheet.

The NRC staff finds item 3 to be acceptable since the words “bulges” and “over-expansions” are no longer used in the TSs and, thus, no definitions of these words are needed.

The 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 plant design basis as embodied in the stress limit criteria of the ASME Code, Section III, as 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 those assumed in the licensing basis accident analyses. (This evaluation deals only with the component of potential accident-induced leakage associated with the proposed 4-inch exclusion zone, not with accident-induced leakage from tube locations not affected by this amendment). Maintaining tube integrity in this manner ensures that the amended TS continue to be consistent 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 of this SE, respectively.

3.2.1 Joint Structural Integrity

Westinghouse has conducted analysis and testing to establish 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 since 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. The licensee’s contractor, Westinghouse, determined the required engagement distance on the basis of maintaining a factor of three against pullout under normal operating conditions and a factor of 1.4 against pullout under accident conditions. Pullout was treated as tube slippage relative to the tubesheet of 0.25 inches. The NRC staff finds this to be acceptable because these safety factors are consistent with those in the structural integrity performance criteria in proposed TS 5.5.9.b.1, TSTF-449, and the design basis (i.e., the stress limit criteria in the ASME Code, Section III). The NRC staff also finds the 0.25-inch slip criterion to be acceptable, because there is still pullout resistance beyond this amount of slip.

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 contributors 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. Westinghouse employed a combination of pullout tests and analyses, including finite element analyses of the tubesheet, to evaluate these contributors. Based on these analyses and tests, Westinghouse initially concluded (licensee’s letter dated November 18, 2005, Attachment 7) that the required engagement distances to ensure the safety factor criteria against pullout vary from 3 to 8.6 inches depending on the radial location of the tube within the tube bundle, with the largest engagement distances needed toward the center of the bundle. Westinghouse refers to the required engagement distance as the H* criterion.

The February 15 and February 23, 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 November 18, 2005, letter) was not justified. Specifically, the original analysis assumed that for a through-wall 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 through wall round holes intended to represent 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 SG 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 NRC staff's request for additional information (NRC letter dated December 13, 2006 (ADAMS Accession No. ML063400204), concerning the implications of cracks being found by inspection of the welds connecting the tubesheet to divider plate at certain foreign reactors.

The revised analyses, including the assumption of no divider plate restraint against tubesheet deflection, shows that the tube to tubesheet engagement distance that is needed to provide the required margins against pullout is 6.2 to 11.5 inches, compared to 3 to 8.6 inches indicated by the original analysis. The revised engagement distances are well within the proposed 17-inch inspection zone.

The technical basis for the proposed 17-inch tubesheet inspection zone and associated tube repair criteria is based in part on pullout tests conducted on nine tube to tubesheet joint specimens. These specimens utilized cylindrical collars to simulate the actual tubesheet. These test collars were fabricated from 1018 steel rather than A508 steel from which the tubesheet is actually fabricated. When analyzing the results of the pullout tests, Westinghouse assumed that the thermal expansion coefficient (TEC) for 1018 steel was identical to that for A508 steel, consistent with the applicable nominal thermal expansion coefficients in Section II, Part D of the ASME Code. However, at the NRC staff's request, the licensee also analyzed the pullout test results using lower values of TEC published in the literature. This change affects the apportionment of the measured pullout loads to that provided by the tube hydraulic expansion process versus that provided by differential thermal expansion between the tube and tubesheet. Based on the reapportioned pullout test data, the licensee reanalyzed the required tubesheet engagement distance using the revised model described above and taking no credit for the divider plate restraint on the tubesheet. By letter dated March 7, 2007, the licensee reported that the net effect was to increase the required engagement distance at the limiting tube radial location from 11.5 inches to 12.6 inches, still well within the proposed 17-inch inspection zone.

The NRC staff has not reviewed the Westinghouse analyses in detail and, thus, has not reached a conclusion with respect to whether 12.6 inches of engagement is adequate to ensure that the necessary safety margins against pullout are maintained. The licensee, therefore, is proposing to inspect the tubes in the tubesheet region such as to ensure a minimum of 17 inches of effective engagement, well in excess of the 12.6 inches that the Westinghouse analyses indicate are needed. Pending a more detailed review of the licensee's analyses, the NRC staff concludes that it cannot approve the licensee's proposal on a permanent basis at this time. However, the NRC staff also concludes there is an adequate technical basis (discussed below) to approve the proposed 17-inch inspection zone and accompanying repair criteria for a limited time period. Specifically, the NRC staff concludes that the applicability of the 17-inch inspection zone and associated repair criteria should be extended only to Refueling Outage 13 and the subsequent operating cycle at Byron, Unit No. 2 and continue to apply only to Refueling Outage 12 and the subsequent (current) operating cycle at Braidwood, Unit 2, because a more detailed review of the revised analyses submitted recently by the licensee is required. The technical basis (from a structural integrity standpoint) supporting the adequacy of the proposed 17-inch inspection zone and associated repair criteria for this limited time period is as follows:

- Pullout tests of 9 samples 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 2 to 6.6 inches to ensure the appropriate safety margins against pullout. This 2 to 6.6 inch spread reflects very considerable scatter in the pullout data, but is well within the proposed 17-inch inspection zone. (This argument is not impacted by the thermal expansion coefficient issue discussed above since it relies on the actual pullout force data rather than inferences from that data as to the relative roles of the tube hydraulic expansion and differential thermal expansion between the tube and tubesheet in resisting pullout.)

- The primary 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. (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.)

- Tubesheet bore dilations caused by tubesheet bow under primary to secondary pressure can increase or decrease contact pressure depending on the tube location within the bundle and on location along the length of the tube in the tubesheet region. Basically, the tubesheet acts 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. The SG divider plate provides a spring support against upward displacement along a diametral mid-line. 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. The amount of 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., 10.5 inches below the top of the tubesheet), tubesheet bore dilation effects would be expected to further tighten the joint from 10 inches below the TTS, to 17 inches below the TTS, 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 6.5-inch distance will be significantly higher than the contact pressure simulated in the above mentioned pull out tests. A similar logic applied to the periphery of the tubesheet leads the NRC staff to conclude that at the top 10.5 inches of the tubesheet region, contact pressure should be considerably higher than the contact pressure simulated in the above mentioned pull out 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 to secondary is introduced between the hydraulically-expanded tubing and the tubesheet. In addition, not inspecting the tubing in the lower 4 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 which may exceed values assumed in the licensing basis accident analyses. The licensee states that this is ensured by limiting primary to secondary leakage to 0.5 gpm in the faulted SG during a 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 not reviewed or accepted this model. To support an amendment request for a one or two cycle 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)) under normal operating conditions, Westinghouse estimates that leakage would not be expected to exceed 0.21 gpm (300 gpd), significantly less than the 0.5 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 pounds per square inch (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 concurs that the factor of 2 upper bound to be reasonable, given the stated premise. The NRC staff notes that the assumed linear relationship between leak rate and differential pressure is conservative relative to alternative models such as 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 the aforementioned premise; namely 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 NRC staff in detail. Among the Westinghouse observations is that for all tubes, there is at least an 8-inch zone in the upper 17-inches of the tubesheet where there is an increase in joint contact pressure 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 letters dated February 15 and February 23, 2007, (and discussed in Section 3.2.1 of this SE) do not affect this observation. In Section 3.2.1 above, the NRC staff observed that there is at least a 6.5-inch zone over which changes in tubesheet bore dilations when going from unpressurized to pressurized conditions should result in an increase in joint contact pressure. The contact pressure due to changes in tubesheet bore dilation should increase further over this 6.5-inch zone under the increased pressure loading on the tubesheet during accident conditions. Considering the higher pressure loading in the tube when going from normal operating to accident conditions, the NRC staff finds that the Westinghouse estimate that contact pressures, and, thus, leak flow resistance, always increase over at least an 8-inch distance is reasonable.

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 4 inches are unlikely to produce leakage rates during normal operation that would approach the TS LCO operational leakage limits, 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, limiting the opening of the crack faces. In addition, little of the end cap pressure load should remain in the tube below 17 inches and, thus, 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.

In summary, the NRC staff concludes that any primary to secondary leakage existing under normal full power operating conditions in the lower 4-inch exclusion zone would not increase by more than a factor of 2 for design basis accidents such as a main steam line break. Since operating leakage is limited by the LCO limit in TS 3.4.13 to 150 gpd, the maximum possible leakage from the lowermost 4 inches inside the tubesheet will not exceed the applicable acceptance criterion of 0.5 gallons per day per SG consistent with proposed TS 5.5.9.b.2 and with the current licensing basis.

3.3 Technical Evaluation Conclusion

The NRC staff finds the proposed TS amendments acceptable except as follows. Pending a more detailed review of the licensee's structural and leakage integrity analyses of the tube to tubesheet joint, the NRC staff concludes that it cannot approve the licensee's proposed 17-inch inspection zone and accompanying plugging criteria on a permanent basis at this time. However, based on the above evaluation, the NRC staff concludes there is an adequate technical basis to approve the proposed 17-inch inspection zone and accompanying plugging criteria for a limited time period. Specifically, the NRC staff concludes that the applicability of the 17-inch inspection zone and associated plugging criteria should be extended only to Refueling Outage 13 and the subsequent operating cycle at Byron, Unit No. 2 and continue to apply only to Refueling Outage 12 and the subsequent current operating cycle at Braidwood, Unit 2, pending a more detailed review of the revised analyses submitted recently by the licensee.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to installation or use of a facility's components 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 previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (71 FR 29676; May 23, 2006). Accordingly, the 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 the 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.

Principal Contributor: E. Murphy, NRR

Date: March 30, 2007

(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) <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 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) <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.

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) 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. 144, 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) 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.

(4) 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) 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. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulation 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 not in excess of 3586.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 144 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) Deleted.

(4) Deleted.

(5) Deleted.

(6) The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the licensee's Fire Protection Report, and as approved in the SER dated February 1987 through Supplement No. 8, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(3) 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) 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) 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. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulation 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 not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 144, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

(5) Deleted.