October 24, 2006

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

SUBJECT: BRAIDWOOD STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENTS RE: STEAM GENERATOR INSPECTION CRITERIA (TAC NO. MC8969)

Dear Mr. Crane:

The Commission has issued the enclosed Amendment No. 141 to Facility Operating License No. NPF-77 for the Braidwood Station (Braidwood), Unit No. 2. The amendment is in response to your application dated November 18, 2005, as supplemented by letters dated August 18 and September 28, 2006.

The amendment revises Technical Specification 5.5.9, "Steam Generator (SG) Tube Surveillance Program," regarding the required SG inspection scope for Braidwood, Unit No. 2, during refueling outage 12 and the subsequent operating cycle. This is a partial approval of the amendment request submitted on November 18, 2005. The NRC staff is continuing its review of the original amendment request which will be addressed in future NRC correspondence.

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 No. STN 50-457

Enclosures:
1. Amendment No. 141 to NPF-77
2. Safety Evaluation

cc w/encls: See next page

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EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
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, 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. 141 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 within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Daniel S. Collins, 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: October 24, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 141

FACILITY OPERATING LICENSE NO NPF-77

DOCKET NO. STN 50-457

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 2 License Page 3	Unit 2 License Page 3
5.5-8	5.5-8
5.5-9	5.5-9
5.5-12	5.5-12
5.5-13	5.5-13
5.5-14	5.5-14

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNIT NO. 2

DOCKET NO. STN 50-457

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated November 18, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML053320303), as supplemented by letters dated August 18 (ADAMS Accession Number ML062400348) and September 28, 2006 (ADAMS Accession Number ML062710559), Exelon Generation Company, LLC (the licensee) requested changes to the technical specifications (TSs), for the Braidwood Station (Braidwood), Unit No. 2. The proposed changes would revise TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," regarding the required SG inspection scope for Braidwood, Unit 2, during refueling outage (RFO) 12 and the subsequent operating cycle.

The August 18 and September 28, 2006 supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The November 18, 2005 amendment request was modeled after the NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4. In addition, the November 18, 2005 amendment request included special SG tube inspection and plugging requirements applicable to the hot-leg tubesheet region for Braidwood, Unit No. 2, and Byron Station (Byron), Unit No. 2. These special requirements would exclude the lowermost 4 inches of the tubes within the hot-leg tubesheet from the TS tube inspection and repair requirements for these units. As acknowledged in the licensee's September 28, 2006 letter, the NRC staff concluded that further review and evaluation is needed before the requested special inspection and plugging requirements in the tubesheet region of these units can be approved on a permanent basis. For this reason, the licensee requested a partial approval of the November 18, 2005 amendment request, which would apply only to the special inspection and plugging requirements in the hot-leg tubesheet region of Braidwood, Unit No. 2, and which would only be applicable during RFO 12 and the subsequent operating cycle.

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

Under the plant TSs SG surveillance program requirements, the licensee is required to monitor the condition of the SG tubing and to plug or repair tubes as necessary. Specifically, the licensee is required to perform periodic inspections of and to repair or remove from service by plugging all tubes found to contain flaws with sizes exceeding the acceptance limit, termed "plugging limit." The tube plugging limits were developed with the intent of ensuring that degraded tubes (1) maintain factors of safety against gross rupture consistent with the plant design basis (i.e., consistent with the stress limits of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section III), and (2) maintain leakage integrity consistent with the plant licensing basis while, at the same time, allowing for potential flaw size measurement error and flaw growth between SG inspections. The required frequency and scope of tubing examinations and the tube plugging limits are specified in TS 5.5.9.

The subject partial amendment request concerns the portions of the tubing that are subject to the TS SG tube surveillance requirements, including any necessary plugging or repairs, and the inspection methods to be employed. TS 5.5.9 defines a tube inspection as an inspection of the SG tube from the point of entry (hot-leg side) completely around the U-bend to the top support of the cold leg. This includes the full length of tubing within the thickness of the tubesheet on the hot-leg side.

The proposed license amendment (applicable during RFO 12 and the subsequent operating cycle) would limit the required inspections and any resulting plugging on the hot-leg side of the 21-inch thick tubesheet region to the upper 17 inches of the tubesheet region at Braidwood, Unit No. 2, and is similar to the one-time amendment approved for RFO 11 and the subsequent operating cycle (with minor differences discussed at the end of Section 3.2 of this SE) for Braidwood, Unit No. 2, and Byron, Unit No. 2, respectively, and to amendments approved for the Wolf Creek and Vogtle Unit 2 nuclear power plants.

1.1 <u>Proposed TS Changes</u>

1.1.1 TS 5.5.9.e.6, "Plugging or Repair Limit"

Paragraphs 2 and 3 of this specification currently state:

> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, this definition does not apply to degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet. Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.
>
> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, degradation 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.

These paragraphs would be revised to indicate RFO 12 instead of RFO 11.

1.1.2 TS 5.5.9.e.8, “Tube Inspection”

Paragraph 2 of this specification currently states:

> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded.

This paragraph would be revised to indicate RFO 12 instead of RFO 11.

1.1.3 TS 5.5.9.b, “SG Tube Sample Selection and Inspection”

TS 5.5.9.b.5 currently states:

> For Unit 2 during Refueling Outage 11, a 20% minimum sample of all inservice tubes from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be inspected by rotating probe. This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet.

TS 5.5.9.b.5 would be deleted (i.e., the specification would not be revised to address RFO 12 and the subsequent operating cycle).

1.1.4 TS 5.5.9.e, “Acceptance Criteria”

TS 5.5.9.e.12 currently states:

> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle:
>
> Bulge refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin coil probe; and
>
> Overexpansion refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin coil probe.

TS 5.5.9.e.12 would be deleted (i.e., the specification would not be revised to address RFO 12 and the subsequent operating cycle).

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 state that the RCPB shall have “an extremely low probability of abnormal leakage...and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards possible" (GDC 30), and shall be designed to permit "periodic inspection and testing ... to assess ... structural and 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 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 (DBAs) such as an SG tube rupture (SGTR) and main steamline break (MSLB). These analyses consider the primary-to-secondary leakage through the tubing which may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR 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 some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits).

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee’s regulatory and technical analyses in support of its proposed license amendments which are described in Section 4.0 of the licensee’s submittal. The detailed evaluation below will support the conclusion 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Braidwood, Unit No. 2, has four model D5 SGs designed and fabricated by Westinghouse Electric Company (Westinghouse). The thermally-treated Alloy 600 SG tubes have an outside diameter of 0.75 inches and a nominal wall thickness of 0.043 inches. The tubes are hydraulically expanded for the full depth of the tubesheet at each end and are welded to the tubesheet at the bottom of each expansion. Braidwood, Unit No. 2, operates with a hot-leg temperature of 611-degrees Fahrenheit (611 °F) and had operated for approximately 14.2 effective full power years (EFPY) as of its most recent inspection in April 2005 (RFO 11).

The licensee has been using bobbin probes for inspecting the length of tubing within the tubesheet. However, the bobbin probe is not capable of reliably detecting stress corrosion cracks (SCC) in the tubesheet region should such cracks be present. For this reason, the licensee has been supplementing the bobbin probe inspections with rotating coil probes 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 zone located at the TTS. The expansion transition contains significant residual stress and was considered a likely location for SCC should it ever develop. Until the fall of 2004, there had not been any reported instances of SCC affecting the tubesheet region of thermally-treated alloy 600 tubing, either at Braidwood, Unit No. 2 or elsewhere in the U.S.

In the fall of 2004, crack-like indications were found in tubes in the tubesheet region at the Catawba Unit 2 nuclear power plant, whose SGs are of similar design to those of Braidwood, Unit No. 2, (i.e., model D5 SGs with thermally-treated alloy 600 tubing) and which has accumulated a comparable operating time (14.7 EFPY) at a comparable operating temperature. These crack-like indications were found in bulges (or over-expansions) in the tubesheet region, in the tack roll region, and in the tube-to-tubesheet weld (The tack expansion is an initial 0.7-inch long expansion at each tube end and is formed prior to the hydraulic expansion over the full tubesheet depth. Its purpose was to facilitate performing the tube-to-tubesheet weld.). Crack-like indications were found in a bulge in one tube and in the tack expansion in nine tubes. Approximately 6 of the 196 tube-to-tubesheet weld indications extended into the parent tube.

During RFO 11, as a result of the Catawba findings, the licensee expanded the scope of previous rotating coil inspections for Braidwood, Unit No. 2, to address the potential for cracks within the thickness of the tubesheet down to 17 inches below the TTS. However, the licensee believes that any flaws located at elevations more than 17 inches below the TTS (i.e., in the bottom 4 inches of the tubesheet region, including the tack expansion region and the tubing in the vicinity of the welds) have no potential to impair tube integrity and, thus, do not pose a safety concern. At the licensee's request, the NRC staff reviewed and approved License Amendment 135 for Braidwood, Unit No. 1, and License Amendment 135 for Braidwood, Unit No. 2, by letter dated April 25, 2005 (ADAMS Accession Number ML051110573), modifying the Braidwood, Unit No. 2 inspection and plugging requirements for portions of the SG tubing within the hot-leg tubesheet region to make these requirements applicable only to the portion of tubing within the upper 17 inches of the tubesheet thickness. The portion of Braidwood, Unit No. 2, tubing below 17 inches from the TTS is excluded from these inspection and plugging requirements. These license amendments applied only to RFO 11 and the subsequent operating cycle for Braidwood, Unit No. 2. The licensee is now requesting, in its September 28, 2006 letter, that the previously approved one-time amendments be extended to apply to RFO 12 and the subsequent operating cycle for Braidwood, Unit No. 2.

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 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 licensee, in effect, is proposing to exempt, during RFO 12 and the subsequent operating cycle, the lower 4 inches of the 21-inch deep tubesheet region from a tube inspection and to exempt tubes with flaw indications in the lower 4-inch zone from the need to plug or repair. These exemptions currently apply only to RFO 11 and the subsequent operating cycle (which ends at the beginning of RFO 12). The latter part of this proposal (i.e., to exempt tubes from plugging or repair) is needed as a practical matter since although rotating coil probe inspections will not be performed in this region, the bobbin probe will necessarily be recording any signals produced in this zone. This proposal, in effect, redefines the pressure boundary at the tube-to-tubesheet joint as consisting of a friction or expansion joint with the tube assumed to be hydraulically expanded against the 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 and weld are assumed not to exist.

The regulatory standard by which the NRC staff has evaluated the subject license amendment is that the amended TS 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 3.1 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. Consistent with industry guidelines in Nuclear Energy Institute (NEI) 97-06, “Steam Generator Tube Integrity” (ADAMS Accession Number ML052710007), the total amount of potential accident-induced leakage must be maintained within the licensing basis assumptions.) 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 3.1 and 3.2 of this SE, respectively.

The licensee is also proposing, on a one-time basis to plug or repair on detection, any flaw indication found in the upper 17-inch region of the tubesheet region of the tubes, irrespective of whether the flaw exceeds the TS 40 percent plugging limit. The NRC staff finds this acceptable since it is more conservative than the current TS 40-percent plugging limit and will provide added assurance that the length of tubing along the entire proposed 17-inch inspection zone, will be effective in resisting tube pullout under tube end cap pressure loads and in resisting primary- to-secondary leakage between the tube and tubesheet. Non-degraded tubing in the upper 17-inch zone is also consistent with the technical basis supporting this request.

3.1 Joint Structural Integrity

Westinghouse has conducted analyses and testings to establish the engagement (embedment) length of hydraulically expanded tubing inside the tubesheet that is necessary to resist pullout under normal operating and DBA 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 DBA 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 conservatively treated as tube slippage relative to the tubesheet of 0.25 inches. The NRC staff concurs that these are the appropriate safety factors to apply to demonstrate structural integrity. As documented in a detailed SE (ADAMS Accession Number ML042570427) accompanying the NRC staff’s approval of new performance-based SG TS for Farley Units 1 and 2, dated September 10, 2004, the NRC staff has concluded that these safety factor criteria are consistent with the design basis; namely the stress limit criteria in the ASME Code, Section III.

The resistance to pullout is the axial friction force developed between the expanded tube and the tubesheet over the engagement distance. The friction force is a function of the radial contact pressure between the expanded tube and the tubesheet. The radial contact pressure derives from several contributors including (1) the contact pressure associated directly with the hydraulic expansion process itself, (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, and (4) 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, to evaluate these contributors. Based on these analyses and tests, Westinghouse concludes that the required engagement distances to ensure the safety factor criteria against pullout are achieved 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.

The NRC staff has not reviewed the Westinghouse analyses in detail and, thus, has not reached a conclusion with respect to whether 3 to 8.6 inches of engagement (termed H* criterion by Westinghouse) 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 to ensure a minimum of 17 inches of effective engagement, well in excess of the 3 to 8.6 inches that the Westinghouse analyses indicate are needed. Based on the following considerations, the NRC staff concludes the proposed 17-inch engagement length is acceptable to ensure the structural integrity of the tubesheet joint:

- Pullout tests demonstrate that the radial contact pressure produced by the hydraulic expansion alone requires an engagement distance of 6 inches to ensure the appropriate safety margins against pullout. This estimate is a mean minus one standard deviation estimate based on nine pull tests. This estimate ignores the effect on needed engagement distance from differential thermal expansion, internal primary pressure in the tube, and tubesheet bore dilations associated with tubesheet bow.

- Radial differential thermal expansion between the tube and tubesheet under hot operating and accident conditions will act to further tighten the joint (i.e., increase radial contact pressure) and to reduce the necessary engagement distance relative to room temperature conditions. The radial differential thermal expansion arises from the facts that the Alloy 600 tubing has a slightly higher (by 6 percent) coefficient of thermal expansion than does the SA-508 Class 2a tubesheet material and that the tubes are a little hotter than the tubesheet.

- The internal primary pressure inside the tube under normal operating and accident conditions also acts to tighten the joint relative to unpressurized conditions, thus reducing the necessary engagement distance.

- 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 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 TTS), 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 the radial differential thermal expansion and primary pressure inside the tube, contact pressure over at least a 6.5-inch distance should be considerably higher than the contact pressure simulated in the above-mentioned pullout 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 pullout tests. Thus, the NRC staff concludes that the proposed 17-inch engagement distance (or inspection zone) is acceptable to ensure the structural integrity of the tubesheet joint.

3.2 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 limiting condition for operation (LCO) limits. However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during DBAs 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 gallons per minute (gpm) in the faulted SG during a MSLB.

To support its H* criterion, 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. For the proposed one-time 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 a SG leaking at the TS LCO limit (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 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 is 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 the Bernoulli equation 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. In Section 3.1 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 pressured 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 Westinghouse estimate that contact pressures, and, thus, leak flow resistance, always increase over at least an 8-inch distance appears reasonable to the NRC staff.

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.

Based on the above, the NRC staff concludes that there is reasonable assurance that the proposed one-time exclusion of the lower 4 inches of the tubes in the tubesheet region from the tube inspection and plugging and repair requirements will not impair the leakage integrity of the tube-to-tubesheet joint for RFO 12 and the subsequent operating cycle.

3.3 TS Changes Evaluation

3.3.1 TS 5.5.9.e.6, "Plugging or Repair Limit"

Paragraphs 2 and 3 of this specification currently state:

> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, this definition does not apply to degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet. Degradation found in th portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.
>
> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, degradation 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.

These paragraphs would be revised to indicate RFO 12 instead of RFO 11. Based on the above SE, the NRC staff finds this TS change acceptable.

3.3.2 TS 5.5.9.e.8, "Tube Inspection"

Paragraph 2 of this specification currently states:

> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded.

This paragraph would be revised to indicate RFO 12 instead of RFO 11. Based on the above SE, the NRC staff finds this TS change acceptable.

3.3.3 TS 5.5.9.b, "SG Tube Sample Selection and Inspection"

TS 5.5.9.b.5 currently states:

> For Unit 2 during Refueling Outage 11, a 20% minimum sample of all inservice tubes from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be inspected by rotating probe. This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet.

The licensee is proposing to delete (i.e., not to extend) the one time (i.e., RFO 11 and the subsequent operating cycle) requirement in TS 5.5.9.b.5 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 this 17-inch zone. The NRC staff notes that all other inspection sampling requirements of TS 5.5.9.b will continue to be applicable to the 17-inch inspection zone. Thus, the 17-inch inspection zone will continue to be subject to the same inspection requirements as all other portions of tubing subject to the TS inspection requirements. Through NEI 97-06 and NRC Generic Letter 04-01, "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 consistent with ensuring tube integrity consistent with the design and licensing basis. No cracking activity was detected in the 17-inch tubesheet inspection zone during the most recent inspection (i.e., RFO 11 in spring 2005). During the upcoming RFO 12 inspection, the NRC staff will monitor the scope, methods, and results of this inspection as part of existing SG oversight programs. For these reasons, the NRC staff finds the deletion of TS 5.5.9.b.5 to be acceptable.

3.3.4 TS 5.5.9.e, "Acceptance Criteria"

TS 5.5.9.e.12 currently states:

> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle:
>
> <u>Bulge</u> refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin coil probe; and
>
> <u>Overexpansion</u> refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin coil probe.

Consistent with the proposed deletion of TS 5.5.9.b.5, the licensee is proposing to delete the one-time definitions in TS 5.5.9.e.12 for the words "bulge" and "over-expansion." These definitions are no longer needed since TS 5.5.9.b.5 was the only place they were used. Therefore, the NRC staff finds this TS change acceptable.

4.0 <u>STATE CONSULTATION</u>

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 requirements with respect to installation or use of a facility's components located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. 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

Date: October 24, 2006

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. 141, 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.