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

April 13, 2011

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 AND BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CHANGES TO TECHNICAL SPECIFICATION SECTIONS 5.5.9, "STEAM GENERATOR (SG) PROGRAM" AND 5.6.9 "STEAM GENERATOR (SG) TUBE INSPECTION REPORT." (TAC NOS. ME5198, ME5199, ME5200, AND ME5201)

Dear Mr. Pacilio:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 166 to Facility Operating License No. NPF-72 and Amendment No. 166 to Facility Operating License No. NPF-77 for the, Braidwood Station, Units 1 and 2, respectively, and Amendment No. 172 to Facility Operating License No. NPF-37 and Amendment No.172 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated December 14, 2010.

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,

Nicholas J. DiFrancesco, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 166 to NPF-72
2. Amendment No. 166 to NPF-77
3. Amendment No. 172 to NPF-37
4. Amendment No. 172 to NPF-66
5. Safety Evaluation

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
License No. NPF-72

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated December 14, 2010, 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. 166 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 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael Mahoney FOR

Robert D. Carlson, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: April 13, 2011.

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
License No. NPF-77

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated December 14, 2010, 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. 166 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 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael Mahoney FOR

Robert D. Carlson, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: April 13, 2011

ATTACHMENT TO LICENSE AMENDMENT NOS. 166 AND 166

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

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

Replace the following pages of the Facility Operating Licenses and 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
License NPF-72	License NPF-72
Page 3	Page 3
License NPF-77	License NPF-77
Page 3	Page 3
TSs	TSs
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.6-7	5.6-7

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

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

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

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No.166, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Emergency Planning

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

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

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

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

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) 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. 166, 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.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed a total of 1 gpm for all SGs.

3. The operational LEAKAGE performance criteria is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired. The following alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:

For Unit 2 during Refueling Outage 15 and the subsequent operating cycle, tubes with service-induced flaws located greater than 16.95 inches below the top of the tubesheet do not require plugging or repair. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 16.95 inches below the top of the tubesheet shall be plugged or repaired upon detection.

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

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

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

5.5.9 <u>Steam Generator (SG) Program</u> (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Unit 2 during Refueling Outage 15 and the subsequent operating cycle, portions of the tube below 16.95 inches from the top of the tubesheet are excluded from this requirement.

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the Unit 1 tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

   Inspect 100% of the Unit 2 tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

5.5.9 Steam Generator (SG) Program (continued)

3. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). For Unit 2 during Refueling Outage 15 and the subsequent operating cycle, if crack indications are found in any SG tube from 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less).

   If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair.

   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.

      i. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

5.6 Reporting Requirements

5.6.9 Steam Generator (SG) Tube Inspection Report (continued)

j. For Unit 2 following completion of an inspection performed in Refueling Outage 15 (and any inspections performed in the subsequent operating cycle), the operational primary to secondary leakage rate observed (greater than three gallons per day) in each steam generator (if it is not practical to assign the leakage to an individual steam generator, the entire primary to secondary leakage should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report, and

k. For Unit 2 following completion of an inspection performed in Refueling Outage 15 (and any inspections performed in the subsequent operating cycle), the calculated accident induced leakage rate from the portion of the tubes below 16.95 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 3.11 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

l. For Unit 2 following completion of an inspection performed in Refueling Outage 15 (and any inspections performed in the subsequent operating cycle), the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
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 December 14, 2010, 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. 172 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to conducting the steam generator inspections required by Technical Specifications 5.5.9 for Byron Station, Unit No. 2, fall 2011 refueling outage (B2R16).

FOR THE NUCLEAR REGULATORY COMMISSION

Michael Mahoney FOR

Robert D. Carlson, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: April 13, 2011

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
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 December 14, 2010, 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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 172 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 prior to conducting the steam generator inspections required by Technical Specifications 5.5.9 for Byron Station, Unit No. 2, fall 2011 refueling outage (B2R16).

FOR THE NUCLEAR REGULATORY COMMISSION

Michael Mahoney FOR

Robert D. Carlson, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: April 13, 2011

ATTACHMENT TO LICENSE AMENDMENT NOS. 172 AND 172

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
License NPF-37	License NPF-37
Page 3	Page 3
License NPF-66	License NPF-66
Page 3	Page 3
TSs	TSs
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.6-7	5.6-7

(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 No172 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. 172, 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.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed a total of 1 gpm for all SGs.

3. The operational LEAKAGE performance criteria is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired. The following alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:

For Unit 2 during Refueling Outage 16 and the subsequent operating cycle, tubes with service-induced flaws located greater than 16.95 inches below the top of the tubesheet do not require plugging or repair. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 16.95 inches below the top of the tubesheet shall be plugged or repaired upon detection.

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

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

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

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Unit 2 during Refueling Outage 16 and the subsequent operating cycle, portions of the tube below 16.95 inches from the top of the tubesheet are excluded from this requirement.

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the Unit 1 tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

Inspect 100% of the Unit 2 tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

3. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). For Unit 2 during Refueling Outage 16 and the subsequent operating cycle, if crack indications are found in any SG tube from 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less).

   If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

5.6 Reporting Requirements

5.6.9 Steam Generator (SG) Tube Inspection Report (continued)

j. For Unit 2 following completion of an inspection performed in Refueling Outage 16 (and any inspections performed in the subsequent operating cycle), the operational primary to secondary leakage rate observed (greater than three gallons per day) in each steam generator (if it is not practical to assign the leakage to an individual steam generator, the entire primary to secondary leakage should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report, and

k. For Unit 2 following completion of an inspection performed in Refueling Outage 16 (and any inspections performed in the subsequent operating cycle), the calculated accident induced leakage rate from the portion of the tubes below 16.95 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 3.11 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

l. For Unit 2 following completion of an inspection performed in Refueling Outage 16 (and any inspections performed in the subsequent operating cycle), the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. NPF-72,

AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. NPF-77,

AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. NPF-37,

AND AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. NPF-66

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

BYRON STATION, UNIT NOS. 1 AND 2

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

STN 50-454, AND STN 50-455,

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission), dated December 14, 2010, Exelon Generation Company, LLC (the licensee), submitted a license amendment request to revise the technical specifications (TS) of Braidwood, Units 1 and 2 and Byron Stations, Unit Nos. 1 and 2. The request proposed changes to the inspection scope and repair requirements of TS section 5.5.9, "Steam Generator (SG) Program" and to the reporting requirements of TS section 5.6.9, "Steam Generator (SG) Tube Inspection Report." The proposed changes would establish alternate repair criteria for portions of the SG tubes within the tubesheet of the Braidwood Unit 2 and Byron Unit 2, Model D5 SGs. The proposed changes would be applicable to Braidwood Unit 2 during Refueling Outage 15 (Spring 2011) and to Byron Unit 2 during Refueling Outage 16 (Fall 2011), and their respective subsequent operating cycles. The proposed changes would establish temporary alternate repair criteria for portions of the Braidwood Unit 2 and Byron Unit 2 SG tubes within the tubesheet, and would replace similar, existing criteria that were used during the previous refueling outages (Fall 2009 - Braidwood Unit 2, Spring 2010 - Byron Unit 2). Because Braidwood and Byron Stations, Units 1 and 2, have common TSs, the licensee docketed the amendment request for all four units.

2.0 BACKGROUND

Braidwood Unit 2 and Byron Unit 2 have four Model D5 SGs each, which were designed and fabricated by Westinghouse. There are 4,570 thermally treated Alloy 600 (Alloy 600TT) tubes in each SG, each with an outside diameter of 0.750 inches and a nominal wall thickness of 0.043 inches. The tubes are hydraulically expanded for the full depth of the 21-inch thick tubesheet and are welded to the tubesheet at each tube end. Until the fall of 2004, no instances of stress corrosion cracking (SCC) affecting the tubesheet region of Alloy 600TT tubing had been reported at any nuclear power plants in the United States (U.S.).

In the fall of 2004, crack-like indications were found in tubes in the tubesheet region of Catawba Nuclear Station Unit 2 (Catawba), which has Westinghouse Model D5 SGs. Like Braidwood Unit 2 and Byron Unit 2, the Catawba SGs use Alloy 600TT tubing that is hydraulically expanded against the tubesheet. The crack-like indications at Catawba were found in a tube overexpansion (OXP), in the tack expansion region, and near the tube-to-tubesheet (T/TS) weld. An OXP is created when the tube is expanded into a tubesheet bore hole that is not perfectly round. These out-of-round conditions were created during tubesheet drilling, by conditions such as drill bit wandering or chip gouging. The approximately 1-inch long tack expansion is made at each tube end and facilitates performing the T/TS weld, which is made prior to the hydraulic expansion of the tube over the full tubesheet depth. Since the initial findings at Catawba in the fall of 2004, other nuclear plants have found crack-like indications in tubes within the tubesheet as well. These plants include Braidwood Unit 2, Byron Unit 2, Comanche Peak Unit 2, Surry Unit 2, Vogtle Unit 1, and Wolf Creek Unit 1. Most of the indications were found in the tack expansion region near the tube-end welds and were a mixture of axial and circumferential primary water stress corrosion cracking.

On February 21, 2006, Wolf Creek Nuclear Operating Corporation (WCNOC), the licensee for Wolf Creek Generating Station, submitted a limiting amendment request (LAR) that would permanently limit the scope of inspections required for tubes within the tubesheet (Reference 2). The LAR was based on an analysis performed by Westinghouse that provided a technical basis for permanently limiting the scope of inspections required for tubes within the tubesheet. After three requests for additional information (RAIs) and several meetings with WCNOC, the NRC staff informed WCNOC during a phone call on January 3, 2008, that it had not provided sufficient information to allow the NRC staff to review and approve the permanent LAR. WCNOC withdrew the LAR by letter dated February 14, 2008 (Reference 3). Other plants had submitted permanent LARs similar to that for Wolf Creek prior to 2008, which also were subsequently withdrawn. In a letter dated February 28, 2008 (Reference 4), the NRC staff identified the specific issues that needed to be addressed to support any future request for a permanent amendment, which included, but were not limited to, thermal expansion coefficients, crevice pressure assumptions, uncertainty models, acceptance standards for probabilistic assessment, and leakage resistance.

After withdrawal of the initial round of permanent LARs submitted prior to 2008, the licensees and their contractor, Westinghouse, worked with the NRC staff to address the issues posed in Reference 4. The NRC and industry held public meetings (References 5, 6, and 7) and phone calls to discuss resolution of these issues. The permanent LAR received from Braidwood Unit 2 and Byron Unit 2 on June 24, 2009 (Reference 8), resolved the issues identified by the NRC staff in Reference 4 but raised an additional technical issue that prevented approval of the permanent LAR. Responses to additional NRC staff RAIs were supplied in References 9, 10, and 11, and the licensee modified its June 24, 2009 LAR (via Reference 12*) to apply during Braidwood Unit 2 during Refueling Outage 14 (Fall 2009) and the subsequent* operating cycle, and to Byron Unit 2 during Refueling Outage 15 (Spring 2010) and the subsequent operating cycle, instead of the permanent change originally requested.

The NRC staff approved the revised amendment in Reference 12. The accompanying safety evaluation (SE) concluded that the NRC staff did not have sufficient information to determine whether the tubesheet bore displacement eccentricity had been addressed in a conservative fashion and, thus, the NRC staff did not have an adequate basis to approve a permanent H* amendment at that time. The NRC staff further concluded that despite any potential non-conservatism in the calculated H* distance that may have been associated with the eccentricity issue, there was sufficient conservatism embodied in the proposed H* distance to ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity would be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

Subsequent analyses by industry to address the NRC staff concerns revealed that tubesheet bore eccentricity did not have a significant bearing on the outcome of the H* analyses. However, these analyses also revealed a significant shortcoming in how displacements from the 3-D finite element model of the lower SG assembly were being applied to the T/TS interaction model, which was based on thick shell equations. The industry developed a new T/TS interaction model to address this shortcoming and the H* analyses were updated accordingly. This more recent background is discussed in more detail as part of the NRC staff technical evaluation in Section 4.0 of this SE. Details of these more recent analyses became available for NRC staff review too late to support applications for a permanent H* amendment in the spring or fall of 2011. For this reason, the subject amendment request by the licensee is for an interim H* amendment, applicable to Braidwood Unit 2 during refueling outage (RFO) 15 (Spring 2011) and to Byron Unit 2 during RFO 16 (Fall 2011), and their respective subsequent operating cycles.

3.0 REGULATORY EVALUATION

In Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36 "Technical specifications", the requirements related to the content of the TSs are established. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TS. In 10 CFR 50.36(c)(5), administrative controls are, "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." Programs established by the licensee, including the SG program, are listed in the administrative controls section of the TS to operate the facility in a safe manner. For Braidwood Unit 2 and Byron Unit 2, the requirements for performing SG tube inspections and repair are in TS 5.5.9, while the requirements for reporting the SG tube inspections and repair are in TS 5.6.9.

The TSs for all pressurized water reactor (PWR) plants require that an SG program be established and implemented to ensure that SG tube integrity is maintained. For Braidwood Unit 2 and Byron Unit 2, SG tube integrity is maintained by meeting the performance criteria specified TS 5.5.9.b for structural and leakage integrity, consistent with the plant design and licensing basis. Technical specification 5.5.9.a, requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. Technical specification 5.5.9.d, includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube, from the T/TS weld at the tube inlet to the T/TS weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The applicable tube repair criteria, specified in TS 5.5.9.c., are that tubes found during inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, unless the tubes are permitted to remain in service through application of the alternate repair criteria provided in TS 5.5.9.c.1.

The SG tubes are part of the reactor coolant pressure boundary (RCPB) and isolate fission products in the primary coolant from the secondary coolant. For the purposes of this safety evaluation, SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis. The General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 provide regulatory requirements, which are applicable to Braidwood Unit 2 and Byron Unit 2, and state that the RCPB shall have "an extremely low probability of abnormal leakage . . . and of gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing . . . to assess . . . structural and leaktight integrity" (GDC 32). The licensee discusses compliance with each of these GDC for the Braidwood Unit 2 and Byron Unit 2 in Section 3.1 of the Updated Final Safety Analysis Report (UFSAR). To this end, 10 CFR 50.55a specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), except as provided in 10 CFR 50.55a(c)(2), (3), and (4). Section 50.55a further requires that throughout the service life of PWR facilities (Braidwood Unit 2 and Byron Unit 2), ASME Code Class 1 components meet the Section XI requirements of the ASME Code to the extent practical, except for design and access provisions, and pre-service examination requirements. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to in-service inspection of SG tubing are augmented by additional requirements in the TS.

As part of the plant's licensing bases, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBA), such as a SG tube rupture and a main steamline break (MSLB). These analyses consider primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 50.67 accident source term, GDC 19 for control room operator doses (or some fraction thereof as appropriate to the accident), or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analyses for Braidwood Unit 2 and Byron Unit 2 are being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The use of the proposed alternate repair criteria does not impact the integrity of the SG tubes, and the SG tubes, therefore, still meet the requirements of the GDC in Appendix A to 10 CFR Part 50, and the requirements for Class 1 components in Section III of the ASME Code. The proposed changes maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated DBAs for SG tubes

License amendment Nos. 161 and 166 are currently approved at Braidwood Unit 2 and Byron Unit 2, respectively, and the amendments modified TS section 5.5.9, "Steam Generator (SG) Program" and TS section 5.6.9, "Steam Generator (SG) Tube Inspection Report," by incorporating interim alternate repair criteria and associated tube inspection and reporting requirements that are applicable to Braidwood Unit 2 during RFO 14 and to Byron Unit 2 during RFO 15, and their respective, subsequent operating cycles. The proposed subject amendment maintains the same alternate repair criteria (i.e., 16.95 inches below the top of the tubesheet (TTS)), but would be applicable to Braidwood Unit 2 during RFO 15 (Spring 2011) and to Byron Unit 2 during RFO 16 (Fall 2011), and their respective subsequent operating cycles.

4.0 TECHNICAL EVALUATION

Proposed Changes to the TS

The TSs identified in this section are based on the Braidwood Station TS. Any differences between the Braidwood and Byron TSs are identified with braces, with specific Byron TS wording identified in brackets (e.g., " . . .during {Refueling Outage **15** [Byron: Refueling Outage **16**]}").

TS 5.5.9 is being revised as follows (new text in underline and **bold**, old text ~~strikethrough~~):

5.5.9 Steam Generator (SG) Program

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired. The following alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:

   For Unit 2 during {Refueling Outage **15** [Byron: Refueling Outage **16**]} and the subsequent operating cycle, tubes with service-induced flaws located greater than 16.95 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 16.95 inches below the top of the tubesheet shall be plugged or repaired upon detection.

2. [No change/Not shown]

3. [No change/Not shown]

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Unit 2 during {Refueling Outage **15** [Byron: Refueling Outage **16**]} and the subsequent operating cycle, portions of the tube below 16.95 inches **from** ~~below~~ the top of the tubesheet are excluded from this requirement.

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. [No change/Not shown]

2. [No change/Not shown]

3. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). For Unit 2 during {Refueling Outage **15** [Byron: Refueling Outage **16**]} and the subsequent operating cycle, if crack indications are found in any SG tube from 16.95 inches below the top of the tubesheet on the hot-leg side to 16.95 inches below the T/TSet on the cold-leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less).

   If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

TS 5.6.9 is being revised as follows (new text in **bold**):

5.6.9 <u>Steam Generator (SG) Tube Inspection Report</u>

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with TS 5.5.9, SG program. The report shall include:

a. – i. [No change/Not shown]

j. For Unit 2 following completion of an inspection performed in {Refueling Outage **15** [Byron: Refueling Outage **16**]} (and any inspections performed in the subsequent operating cycle), the operational primary to secondary leakage rate observed (greater than three gallons per day) in each steam generator (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report,

k. For Unit 2 following completion of an inspection performed in {Refueling Outage **15** [Byron: Refueling Outage **16**]} (and any inspections performed in the subsequent operating cycle), the calculated accident induced leakage rate from the portion of the tubes below 16.95 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 3.11 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

l. For Unit 2 following completion of an inspection performed in {Refueling Outage **15** [Byron: Refueling Outage **16**]} (and any inspections performed in the subsequent operating cycle), the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

4.1 Technical Evaluation

The T/TS joints are part of the pressure boundary between the primary and secondary systems. Each T/TS joint consists of the tube, which is hydraulically expanded against the bore of the tubesheet, the T/TS weld located at the tube end, and the tubesheet. The joints were designed in accordance with Section III of the ASME Code as welded joints, not as friction joints. The T/TS welds were designed to transmit the tube end cap pressure loads, during normal operating and DBA conditions, from the tubes to the tubesheet with no credit taken for the friction developed between the hydraulically-expanded tube and the tubesheet. The axial force which could produce pullout comes from the primary-to-secondary pressure differentials associated with normal operating and DBA conditions and is called the end cap load. In addition, the welds serve to make the joints leak tight.

This design basis is a conservative representation of how the T/TS joints actually work, since it conservatively ignores the role of friction between the tube and tubesheet in reducing the tube end cap loads. The initial hydraulic expansion of the tubes against the tubesheet produces an "interference fit" between the tubes and the tubesheet; thus, producing a residual contact pressure (RCP) between the tubes and tubesheet, which acts normally to the outer surface of the tubes and the inner surface of the tubesheet bore holes. Additional contact pressure between the tubes and tubesheet is induced by operational conditions, as will be discussed in detail below. The amount of friction force that can be developed between the outer tube surface and the inner surface of the tubesheet bore is a direct function of the contact pressure between the tube and tubesheet times the applicable coefficient of friction.

To support the proposed TS changes, the licensee's contractor, Westinghouse, has defined a parameter called H* to be that distance below the top of the tubesheet over which sufficient frictional force, with acceptable safety margins, can be developed between each tube and the tubesheet, under tube end cap pressure loads associated with normal operating and design basis accident conditions, to prevent significant slippage or pullout of the tube from the tubesheet, assuming the tube is fully severed at the H* distance below the top of the tubesheet. For Braidwood Unit 2 and Byron Unit 2, the proposed H* distance is 16.95 inches. Given that the frictional force developed in the T/TS joint over the H* distance is sufficient to resist the tube end cap pressure loads, it is the licensee's and Westinghouse's position that the length of tubing between the H* distance and the T/TS weld is not needed to resist any portion of the tube end cap pressure loads. Thus, the licensee is proposing to change the TS to not require inspection of the tubes below the H* distance and to exclude tube flaws located below the H* distance (including flaws in the T/TS weld) from the application of the TS tube repair criteria. Under these changes, the T/TS joint would now be treated as a friction joint extending from the top of the tubesheet to a distance below the top of the tubesheet equal to H* for purposes of evaluating the structural and leakage integrity of the joint.

The regulatory standard by which the staff has evaluated the subject license amendment is that the amended TSs should continue to ensure that tube integrity will be maintained consistent with the current design basis, as defined in the UFSAR. This includes maintaining structural safety margins consistent with the structural integrity performance criteria in TS 5.5.9.b.1, as discussed in Section 4.2.1 below. In addition, this includes limiting the potential for accident-induced primary-to-secondary leakage to values that do not exceed the accident-induced leakage performance criteria in TS 5.5.9.b.2, which are consistent with values assumed in the UFSAR accident analyses. Maintaining tube integrity in this manner ensures that the amended TS are in compliance with all applicable regulations. The NRC staff evaluation of joint structural integrity and accident-induced leakage integrity is discussed in Sections 4.2 and 4.3 of this SE, respectively.

4.2 Joint Structural Integrity

4.2.1 Acceptance Criteria

Westinghouse has conducted extensive analyses to establish the necessary H* distance to resist pullout under normal operating and DBA conditions. The NRC staff finds that 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 comes from the primary-to-secondary pressure differentials associated with normal operating and DBA conditions, and is called the end cap load. Westinghouse determined the needed H* distance on the basis of maintaining a factor of 3 against pullout under normal operating conditions and a factor of 1.4 against pullout under DBA conditions. The staff finds that these are the appropriate safety factors to apply to demonstrate structural integrity. These safety factors are consistent with the safety factors embodied in the structural integrity performance criteria in TS 5.5.9.b.1 and with the design basis, including the stress limit criteria in Section III of the ASME Code.

4.2.2 3-D Finite Element Analysis

A detailed 3-D finite element analysis (FEA) of the lower SG assembly (consisting of the lower portion of the SG shell, the tubesheet, the channel head, and the divider plate separating the hot- and cold-leg inlet plenums inside the channel head) was performed to calculate tubesheet displacements due to primary pressure acting on the primary face of the tubesheet and SG channel head, secondary pressure acting on the secondary face of the tubesheet and SG shell, and the temperature distribution throughout the entire lower SG assembly. The calculated tubesheet displacements were used as input to the T/TS interaction analysis evaluated in Section 4.2.3 below.

The tubesheet bore holes were not explicitly modeled. Instead, the tubesheet was modeled as a solid structure with equivalent material property values selected such that the solid model exhibited the same stiffness properties as the actual perforated tubesheet.

A number of FEA mesh enhancements in the tubesheet region have been made since the reference analysis (Reference 14) was performed. The mesh near the plane of symmetry (perpendicular to the divider plate) was revised to permit obtaining displacements parallel to the direction of the divider plate directly from the 3-D finite element model, for application (as displacement boundary conditions) to the edges of the square cell model discussed in Section 4.2.3.2. The mesh near the top of the tubesheet was enhanced to accommodate high temperature gradients in this area during normal operating conditions. This 3-D FEA replaces the 2-D axisymmetric FEA used to support H* amendment requests submitted prior to 2008. The staff finds that the 3-D analysis adequately addresses a concern cited by the staff in Reference 4 concerning the validity of the axisymmetric model to conservatively bound significant non-axisymmetric features of the actual tubesheets. These non-axisymmetric features include the solid (non-bored) portion of the tubesheet between the hot and cold leg sides, and the divider plate which acts to connect the solid part of the tubesheet to the channel head.

Some non-U.S. units have experienced cracks in the weld between the divider plate and the stub runner attachment on the bottom of the tubesheet. Should such cracks ultimately cause the divider plate to become disconnected from the tubesheet, tubesheet vertical and radial displacements under operational conditions could be significantly increased relative to those for an intact divider plate weld. Although the industry believes that there is little likelihood that cracks such as those seen abroad could cause a failure of the divider plate weld, the 3-D FEA conservatively considered both the case of an intact divider plate weld and a detached divider plate weld to ensure a conservative analysis. The case of a detached divider plate weld was found to produce the most limiting H* values. In the reference analyses (Reference 14), a factor was applied to the 3-D FEA results to account for a non-functional divider plate, based on earlier sensitivity studies performed with the 2-D axisymmetric FEA model of the lower SG assembly. The 3-D FEA model now assumes the upper 5 inches of the divider plate to be non-existent. The staff finds this further improves the accuracy of the 3-D FEA for the assumed condition of a non-functional divider plate.

Separate 3-D FEA analyses were conducted for each loading condition considered (i.e., normal operating conditions, MSLB, feedwater line break (FLB)), rather than scaling unit load analyses to prototypic conditions as was done in analyses prior to 2008. The staff finds that this addresses (corrects) a significant source of error in analyses used by applicants to support permanent H* amendment requests submitted prior to 2008 and which were subsequently withdrawn (Reference 4). In addition, the temperature distributions throughout the lower SG assembly, including the tubesheet region, were calculated directly in the 3-D FEA from the assumed plant temperature conditions (e.g., from the assumed primary and secondary water temperatures) for each operating condition. The NRC staff finds this a more realistic approach than the reference analysis (Reference 14), where a linear distribution of temperature was assumed to exist through the thickness of the tubesheet, and an adjustment factor (based on sensitivity analyses) was applied to the H* calculations for normal operating conditions to account for the actual temperature distribution in the tubesheet.

4.2.3 T/TS Interaction Model

4.2.3.1 Thick Shell Model

The resistance to pullout is the axial friction force developed between the expanded tube and the tubesheet over the H* distance. The friction force is a function of the radial contact pressure between the expanded tube and the tubesheet. In the analyses (Reference 14) for the interim H* amendment issued on October 16, 2009, for Braidwood Unit 2 and Byron Unit 2 (Reference

12), Westinghouse used classical thick shell equations to model the interaction effects between the tubes and tubesheet under various pressure and temperature conditions for purposes of calculating contact pressure (T/TS interaction model). Calculated displacements from the 3-D FEA of the lower tubesheet assembly (see section 4.2.2 above) were applied to the thick shell model as input to account for the increment of tubesheet bore diameter change caused by the primary pressure acting on the primary face of the tubesheet and SG channel head, secondary pressure acting on the secondary face of the tubesheet and SG shell, and the temperature distribution throughout the entire lower SG assembly. However, the tubesheet bore diameter change from the 3-D FEA tended to be non-uniform (eccentric) around the bore circumference. The thick shell equations used in the T/TS interaction model are axisymmetric. Thus, the non-uniform diameter change from the 3-D FEA had to be adjusted to an equivalent uniform value before it could be used as input to the T/TS interaction analysis. A 2-D, plane stress, finite element model was used to define a relationship for determining a uniform diameter change that would produce the same change to average T/TS contact pressure as would the actual non-uniform diameter changes from the 3-D finite element analyses.

In Reference 14, Westinghouse identified a difficultly in applying this relationship to Model D5 SGs under MSLB conditions. In reviewing the reasons for this difficulty, the staff developed questions relating to the conservatism of the relationship and whether the tubesheet bore displacement eccentricities are sufficiently limited such as to ensure that T/TS contact is maintained around the entire tube circumference. This concern was applicable to all SG models with Alloy 600TT tubing. However, responses to NRC staff questions provided in References 10 and 11 did not contain sufficient information to allow the NRC staff to reach a conclusion on these matters and on the acceptability of a permanent H* amendment. However, for reasons discussed in the staff SE in Reference 12, the NRC staff concluded that there was an adequate technical basis to support issuance of an interim H* amendment.

In Reference 15, the NRC staff documented a list of questions that would need to be addressed satisfactorily before the staff would be able to approve a permanent H* amendment. These questions related to the technical justification for the eccentricity adjustment, the distribution of contact pressure around the tube circumference, and a new model under development by Westinghouse to address the aforementioned issue encountered with the Model D5 SGs.

On June 14 and 15, 2010, the NRC staff conducted an audit at the Westinghouse Waltz Mill Site (Reference 16). The purpose of the audit was to gain a better understanding of the H* analysis pertaining to eccentricity, to review draft responses to the NRC staff questions in Reference 15, and to determine which documents would need to be provided on the docket to support any future requests for a permanent H* amendment. Based on the audit, including review of pertinent draft responses to Reference 15, the NRC staff concluded that eccentricity does not appear to be a significant variable affecting either average T/TS contact pressure at a given elevation or calculated values of H*. The NRC staff found that average contact pressure at a given elevation is primarily a function of average bore diameter change at that elevation associated with the pressure and temperature loading of the tubesheet. Accordingly, the NRC staff concluded that no adjustment of computed average bore diameter change considered in the thick shell model is needed to account for eccentricities computed by the 3-D FEA. The material reviewed during the audit revealed that computed H* values from the reference analyses continued to be conservative when the eccentricity adjustment factor is not applied.

During the audit, Westinghouse presented preliminary details of a new T/TS interaction model developed as an alternative to the thick shell interaction model. This model is termed the square cell model and was developed in response to the difficulty encountered when applying the eccentricity adjustment to the Model D5 SG T/TS interaction analysis under MSLB conditions using the thick shell model. Early results with this model indicated significant differences compared to the thick shell model, irrespective of whether the eccentricity adjustment was applied to the thick shell model. The square cell model revealed a fundamental problem with how the results of the 3-D FEA model of the lower SG assembly were being applied to the tubesheet bore surfaces in the thick shell model. As discussed in Section 4.2.2 above, the perforated tubesheet is modeled in the 3-D FEA model as a solid plate whose material properties were selected such that the gross stiffness of the solid plate is equivalent to that of a perforated plate under the primary-to-secondary pressure acting across the thickness of the plate. This approach tends to smooth out the distribution of tubesheet displacements as a function of radial and circumferential location in the tubesheet, and ignores local variations of the displacements at the actual bore locations. These smoothed out displacements from the 3-D FEA results were the displacements applied to the bore surface locations in the thick shell model. The square cell model provides a means for post-processing the 3-D FEA results such as to account for localized variations of tubesheet displacement at the bore locations as part of T/TS interaction analysis. The square cell model was still under development at the time of the audit and no draft documentation of the model was available for staff review. Although the NRC staff found that objectives of the new model approach appeared reasonable, the staff was unable to provide feedback on the details of the approach at that time. The staff also observed (Reference 16) that the square cell model approach might need to be applied to the Model F, 44F, and 51F SGs to confirm that the analyses for these plants were conservative.

4.2.3.2 Square Cell Model

Documentation for the square cell model is included with the subject amendment request for an interim H* at Braidwood Unit 2 and Byron Unit 2. The square cell model is a 2-D, plane stress, finite element model of a single square cell of the tubesheet with a bore hole in the middle and each of the four sides of the cell measuring one tube pitch in length. Displacement boundary conditions are applied at the edges of the cell, based on the displacement data from the 3-D FEA model. The model also includes the tube cross-section inside the bore. Displacement compatibility between the tube outer surface and bore inner surface is enforced except at locations where a gap between the tube and bore tries to occur.

The square cell model is applied to nine different elevations, from the top to the bottom of the tubesheet, for each tube and loading case analyzed. The square cell slices at each elevation are modeled to act independently of one another. The T/TS contact pressure results from each of the nine slices are used to define the contact pressure distribution from the top to the bottom of the tubesheet.

The resisting force to the applied end cap load, which is developed over each incremental axial distance from the top of the tubesheet, is the average contact pressure over that incremental distance times the tubesheet bore surface area (equal to the tube outer diameter surface area) over the incremental axial distance times the coefficient of friction. The NRC staff reviewed the coefficient of friction used in the analysis and judges it to be a reasonable lower bound (conservative) estimate. The H* distance for each tube was determined by integrating the incremental friction forces from the top of the tubesheet to the distance below the top of the tubesheet where the friction force integral equaled the applied end cap load, times the appropriate safety factor as discussed in Section 4.2.1.

The square cell model assumes as an initial condition that each tube outer surface is in contact with the inner surface of the tubesheet bore, at room temperature and atmospheric pressure, with zero RCP associated with the hydraulic expansion process.

The NRC staff finds the assumption of zero RCP in all tubes to be a conservative assumption. The limiting tube locations in terms of H* were determined during the reference analysis to lie along the plane of symmetry perpendicular to the divider plate. The outer edges of the square cell model conform to the revised mesh pattern along this plane of symmetry in the 3-D FEA model of the lower SG assembly, as discussed in Section 4.2.2. Because the tubesheet bore holes were not explicitly modeled in the 3-D FEA, only the average displacements along each side of the square cell are known from the 3-D FEA. Three different assumptions for applying displacement boundary conditions to the edges of the square cell model were considered, to allow for a range of possibilities about how local displacements might vary along the length of each side. The most conservative assumption, in terms of maximizing the calculated H* distance, was to apply the average transverse displacement uniformly over the length of each edge of the square cell.

Primary pressure acting on the inside tube surface, and crevice pressure[1] acting on both the tube outside surface and tubesheet bore surface, are not modeled directly as in the case of the thick shell model. Instead, the primary side (inside) of the tube is assumed to have a pressure equal to the primary pressure minus the crevice pressure. Note the crevice pressure varies as a function of the elevation being analyzed, as discussed in Section 4.2.4.

The NRC staff has not completed its review of the square cell model. This review will need to be completed before the staff can approve a future request for a permanent H* amendment. However, for reasons discussed in Section 4.5, the NRC staff concludes the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

4.2.4 Crevice Pressure Evaluation

As discussed in an earlier footnote, the H* analyses postulate that interstitial spaces exist between the hydraulically expanded tubes and tubesheet bore surfaces. These interstitial spaces are assumed to act as crevices between the tubes and the tubesheet bore surfaces. The staff finds that the assumption of crevices is conservative since the pressure inside the crevices acts to push against both the tube and the tubesheet bore surfaces, thus reducing contact pressure between the tubes and tubesheet.

For tubes which do not contain through-wall flaws within the thickness of the tubesheet, the pressure inside the crevice is assumed to be equal to the secondary system pressure. For tubes that contain through-wall flaws within the thickness of the tubesheet, a leak path is

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[1]Although the tubes are in tight contact with the tubesheet bore surfaces, surface roughness effects are conservatively assumed to create interstitial spaces, which are effectively crevices, between these surfaces. See Section 4.2.4 for more information.

assumed to exist, from the primary coolant inside the tube, through the flaw, and up the crevice to the secondary system. Hydraulic tests were performed on several tube specimens that were hydraulically expanded against tubesheet collar specimens to evaluate the distribution of the crevice pressure from a location where through-wall holes had been drilled into the tubes to the top of the crevice location. The T/TS collar specimens were instrumented at several axial locations to permit direct measurement of the crevice pressures. Tests were run for both normal operating and MSLB pressure and temperature conditions.

The NRC staff finds that the use of the drilled holes, rather than through-wall cracks, is conservative since it eliminates any pressure drop between the inside of the tube and the crevice at the hole location. This maximizes the pressure in the crevice at all elevations, thus reducing contact pressure between the tubes and tubesheet. The crevice pressure data from these tests were used to develop a crevice pressure distribution as a function of normalized distance between the top of the tubesheet and the H* distance below the top of the tubesheet where the tube is assumed to be severed. These distributions were used to determine the appropriate crevice pressure for each axial slice of the T/TS interaction model. The NRC staff finds that this approach acceptably addresses the staff's concerns cited in Reference 4 concerning the use of the limiting median crevice pressure value of the normal operating and MSLB data, respectively, for each axial slice, in previous H* analyses in support of amendment applications submitted prior to 2008. The NRC staff finds the crevice pressure distributions used to support the current amendment request to be more realistic and more conservative than those used previously.

Because the crevice pressure distribution is assumed to extend from the H* location, where crevice pressure is assumed to equal primary pressure, to the top of the tubesheet, where crevice pressure equals secondary pressure, an initial guess as to the H* location must be made before solving for H* using the T/TS interaction model and 3-D finite element model. The resulting new H* estimate becomes the initial estimate for the next H* iteration.

4.2.5 <u>H* Calculation Process</u>

The calculation of H* consists of the following steps for each loading case considered:

1. Perform initial H* estimate (mean H* estimate) using the T/TS interaction and 3-D finite element models, assuming nominal geometric and material properties, and assuming that the tube is severed at the bottom of the tubesheet for purposes of defining the contact pressure distribution over the length of the T/TS crevice. This initial estimate did not consider the effect of the Poisson's contraction of the tube radius associated with application of the axial end cap load (see step 6 below).

2. In the reference analysis (Reference 14), a 0.3-inch adjustment was added to the initial H* estimate to account for uncertainty in the bottom of the tube expansion transition location relative to the T/TS, based on an uncertainty analysis on the bottom expansion transition (BET) for Model F SGs, conducted by Westinghouse. This adjustment is not included in the revised H* analysis of the subject amendment request, as discussed and evaluated in Section 4.2.5.1 if this SE.

3. In the reference analysis (Reference 14), for normal operating conditions only, an additional adjustment was added to the initial H* estimate to correct for the actual

temperature distribution in the tubesheet compared to the linear distribution assumed in the FEA. This adjustment is no longer necessary, as discussed in Section 4.2.2, since the tubesheet temperature distributions were calculated directly in the 3-D FEA, supporting the current request for an interim H* amendment.

4. Steps 1 through 3 yield a so-called "mean" estimate of H*, which is deterministically based. Step 4 involves a probabilistic analysis of the potential variability of H*, relative to the mean estimate, associated with the potential variability of key input parameters for the H* analyses. This leads to a "probabilistic" estimate of H*, which includes the mean estimate. The NRC staff evaluation of the probabilistic analysis is provided in Sections 4.2.6 and 4.2.7 of this SE.

5. Add a crevice pressure adjustment to the probabilistic estimate of H* to account for the crevice pressure distribution that results from the tube being severed at the final H* value, rather than at the bottom of the tubesheet. This step is discussed and evaluated in Section 4.2.5.2 of this SE.

6. This step has been added to the H* calculation process since the reference analysis, to support the subject interim amendment request. This step involves adding an additional adjustment to the probabilistic estimate of H* to account for the Poisson contraction of the tube radius due to the axial end cap load acting on each tube. This step is discussed and evaluated in Section 4.2.5.3 of this SE.

4.2.5.1 Bottom of Expansion Transition (BET) Considerations

In the reference H* analysis (Reference 14), a 0.3-inch adjustment was added to the initial H* estimate to account for uncertainty in the BET location, relative to the top of the tubesheet, based on a BET uncertainty analysis for Model F SGs conducted by Westinghouse. As discussed previously in Section 4.2.3, the reference analysis was based on the thick shell model and the results of that analysis did not indicate a loss of contact pressure at the T/TS during normal operating procedures or steamlinebreak conditions; therefore, this adjustment for the BET location was necessary. In response to NRC staff questions regarding the BET uncertainty analysis, Westinghouse performed an analysis (Reference 17) that showed BET locations as great as one inch below the T/TS could be tolerated at any tube location. Because the limiting calculated H* value is in the most limiting tubesheet sector, that H* value provides greater than one inch of margin for most other tubesheet sectors. For those few sectors in the tubesheet where the local H* distance was within one inch of the maximum H* distance, Westinghouse showed that the contact pressure gradient was positive with increasing depth into the tubesheet, and therefore, an H* length reduced by one inch still met the pull out resistance requirements, including appropriate safety factors.

The new analysis performed in Reference 18 has made the need for this adjustment moot, as the square cell model shows a loss of contact pressure at the T/TS that is greater than the possible variation in the BET location. The loss of contact pressure at the T/TS shown in the square cell model (which is unrelated to BET location) is compensated for by a steeper contact pressure gradient than was shown previously in the thick shell model H* analysis

4.2.5.2 Crevice Pressure Adjustment

As discussed in Section 4.2.5, steps 1 through 4 of the H* calculation process leading to a probabilistic H* estimate are performed with the assumption that the tube is severed at the bottom of the tubesheet for purposes of calculating the distribution of crevice pressure as a function of elevation. If the tube is assumed to be severed at the initially computed H* distance and steps 1 through 4 repeated, a new H* may be calculated, which will be incrementally larger than the first estimate. This process may be repeated until the change in H* becomes small (convergence). Sensitivity analyses conducted during the reference analysis with the thick shell model showed that the delta between the initial H* estimate and final (converged) estimate is a function of the initial estimate for the tube in question. This delta (i.e., the crevice pressure adjustment referred to in step 5 of Section 4.2.5) was plotted as a function of the initial H* estimate for the limiting loading case and tube radial location. The NRC staff concludes this to be an acceptable approach where the H* estimates are based on the thick shell model; however, the staff has not yet reached a conclusion regarding the applicability of this adjustment to H* estimates that are based on the square cell model. The staff will need to reach a conclusion on this point before the NRC staff can approve any request for a permanent H* amendment. However, for reasons discussed in Section 4.5, the NRC staff concludes the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

4.2.5.3 Poisson Contraction Effect

The axial end cap load acting on each tube is equal to the primary-to-secondary pressure difference times the tube cross-sectional area. For purposes of resisting tube pullout under normal and accident conditions, the end cap loads used in the H* analyses are based on the tubesheet bore diameter, which the NRC staff finds to be a conservative assumption. The axial end cap load tends to stretch the tube in the axial direction, but causes a slight contraction in the tube radius due to the Poisson's Ratio effect. This effect, by itself, tends to reduce the T/TS contact pressure and, thus, to increase the H* distance. The axial end cap force is resisted by the axial friction force developed at the T/TS joint. Thus, the axial end cap force begins to decrease with increasing distance into the tubesheet, reaching zero at a location before the H* distance is reached. This is because the H* distances are intended to resist pullout under the end cap loads with the appropriate factors of safety applied as discussed in Section 4.2.1.

This Poisson radial contraction effect was neglected in the Reference 4 analyses, but is accounted for in the analyses supporting the subject amendment request. A simplified approach was followed in the current license amendment request. First, thick shell equations were used to estimate the reduction in contact pressure associated with application of the full end cap load, assuming none of this end cap load has been reacted by the tubesheet. The T/TS contact pressure distributions determined in Step 4 of the H* calculation process in Section 4.2.5 were reduced by this amount. Second, the friction force associated with these reduced T/TS contact pressures were integrated with distance into the tubesheet, and the length of engagement necessary to react one times the end cap loading (i.e., no safety factor applied) was determined. At this distance (termed attenuation distance by Westinghouse), the entire end cap loading was assumed to have been reacted by the tubesheet, and the axial load in the tube below the attenuation distance was assumed to be zero. Thus, the T/TS contact pressures below the attenuation distance were assumed to be unaffected by the Poisson radial contraction effect. Finally, a revised H* distance was calculated, where the T/TS contact pressures from Step 4 of Section 4.2.5 were reduced only over the attenuation distance. The NRC staff has not completed its review of the applied adjustment to account for the Poisson radial contraction effect. However, for reasons discussed in Section 4.5, the staff concludes the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

4.2.6 Acceptance Standard - Probabilistic Analysis

The purpose of the probabilistic analysis is to develop a safe H* distance that ensures with a probability of 0.95 that the population of tubes will retain margins against pullout consistent with criteria evaluated in Section 4.2.1 of this safety evaluation, assuming all tubes to be completely severed at their H* distance. The NRC staff finds this probabilistic acceptance standard is consistent with what the staff has approved previously and is acceptable. For example, the upper voltage limit for the voltage based tube repair criteria in NRC Generic Letter 95-05 (Reference 19) employs a consistent criterion. The NRC staff also notes that use of the 0.95 probability criterion ensures that the probability of pullout of one or more tubes under normal operating conditions and conditional probability of pullout under accident conditions is well within tube rupture probabilities previously considered in probabilistic risk assessments (References 20 and 21).

In terms of the confidence level that should be attached to the 0.95 probability acceptance standard, it is industry practice for SG tube integrity evaluations, as embodied in industry guidelines, to calculate such probabilities at a 50 percent confidence level. The NRC staff has been encouraging the industry to revise its guidelines to call for calculating such probabilities at a 95 percent confidence level when performing operational assessments and a 50 percent confidence level when performing condition monitoring (Reference 22). In the meantime, the calculated H* distances supporting the interim amendment currently being requested have been evaluated at the 95 percent confidence level, as recommended by the NRC staff.

Another issue relating to the acceptance standard for the probabilistic analysis is determining what population of tubes needs to be analyzed. For accidents such as MSLB or FLB, the NRC staff and licensee both find that the tube population in the faulted SG is of interest, since it is the only SG that experiences a large increase in the primary-to-secondary pressure differential. For the Model D5 SGs in the subject amendment request, MSLB is the most limiting condition and the H* distances referenced herein are based on 0.95 probability/95 percent confidence estimates for the population of tubes in any one SG in the plant.

Based on the above, the NRC staff concludes that the proposed H* distance in the subject license amendment request is based on acceptable probabilistic acceptance standards evaluated at acceptable confidence levels.

4.2.7 Probabilistic Analyses

Sensitivity studies were conducted during the reference analyses (Reference 14) and demonstrated that H* was highly sensitive to the potential variability of the coefficients of thermal expansion (CTE) for the Alloy 600 tubing material and the SA-508 Class 2a tubesheet material. Given that no credit was taken in the reference H* analyses (Reference 14) for RCP associated with the tube hydraulic expansion process[2] the sensitivity of H* to other geometry and material input parameters was judged by Westinghouse to be inconsequential and were ignored, with the exception of Young's modulus of elasticity for the tube and tubesheet materials. Although the Young's modulus parameters were included in the reference H* analyses sensitivity studies, these parameters were found to have a weak effect on the computed H*. Based on its review of the analysis models and its engineering judgment, the staff concurs that the sensitivity studies adequately capture the input parameters which may significantly affect the value of H*. This conclusion is based, in part, on no credit being taken for RCP during the reference H* analyses.

These sensitivity studies were used to develop influence curves describing the change in H*, relative to the mean H* value estimate (see Section 4.2.5), as a function of the variability of each CTE parameter and Young's modulus parameter, relative to the mean values of CTE and Young's Modulus. Separate influence curves were developed for each of the four input parameters. The sensitivity studies showed that of the four input parameters, only the CTE parameters for the tube and tubesheet material had any interaction with one another. A combined set of influence curves containing this interaction effect were also created.

Two types of probabilistic analyses were performed independently in the reference analyses (Reference 14). One was a simplified statistical approach utilizing a "square root of the sum of the squares" method and the other was a detailed Monte Carlo sampling approach. The staff's review of the reference analyses relied primarily on the Monte Carlo analysis, which provides the more realistic treatment of uncertainties.

The NRC staff reviewed the implementation of probabilistic analyses in the reference analyses (Reference 14) and questioned whether the H* influence curves had been conservatively treated. To address this concern, the licensee submitted new H* analyses as documented in References 9 and 10. These analyses made direct use of the H* influence curves in a manner the NRC staff finds to be acceptable.

The revised reference analyses in References 9 and 10 divided the tubes by sector location within the tube bundle and all tubes were assumed to be at the location in their respective sectors where the initial value of H* (based on nominal values of material and geometric input parameters) was at its maximum value for that sector. The H* influence curves discussed above, developed for the most limiting tube location in the tube bundle, were conservatively used for all sectors. The revised reference analyses also addressed a question posed by the staff in Reference 4 concerning the appropriate way to sample material properties for the tubesheet, whose properties are unknown but do not vary significantly for a given SG, in contrast to the tubes whose properties tend to vary much more randomly from tube to tube in a given SG.

This issue was addressed by a staged sampling process where the tubesheet properties were sampled once and then held fixed, while the tube properties were sampled a number of times

[2]Residual contact pressures are sensitive to variability of other input parameters.

equal to the SG tube population. This process was repeated 10,000 times, and the maximum H* value from each repetition was rank ordered. The final H* value was selected from the rankordering to reflect a 0.95 probability value at the desired level of confidence for a single SG tube population or all SG population, as appropriate. The NRC concludes that this approach addresses the NRC staff question in a realistic fashion and is acceptable.

New Monte Carlo analyses using the square cell model to evaluate the statistical variability of H* due to the CTE variability for the tube and tubesheet materials were not performed in support of the subject interim amendments. Instead, the probabilistic analysis utilized the results of the Monte Carlo from the reference analyses (References 14 and 9)[3], which are based on the thick shell T/TS interaction model, to identify CTE values for the tube and tubesheet associated with the probabilistic H* values near the desired rank ordering. Tube CTE values associated with the high ranking order estimates are generally negative variations from the mean value whereas tubesheet CTE values associated with the higher ranking order estimates are generally positive variations from the mean value. For the upper 10 percent of the Monte Carlo results ranking order, a combined uncertainty parameter, "alpha," was defined as the square root of the sum of the squares of the associated tube and tubesheet CTE values for each Monte Carlo sample. Alpha was plotted as a function of the corresponding H* estimate and separately as a function of rank order. Each of these plots exhibited well defined "break lines," representing the locus of maximum H* estimates and maximum rank orders associated with a given values of alpha. From these plots, paired sets of tube and tubesheet CTE values were selected such as to maximize the H* estimate and to upper and lower bound the rank orders corresponding to the appropriate probabilistic acceptance criteria described and evaluated in Section 4.2.6. These CTE values were then input to the lower SG assembly 3-D finite element model and the square cell model to yield probabilistic H* estimates. These H* estimates were then plotted as a function of rank ordering, allowing the interpolation of H* values at the desired rank orders.

The limiting probabilistic H* value, evaluated at the appropriate acceptance standard as discussed in Section 4.2.6 and with the adjustments for crevice pressure and Poisson radial contraction effect discussed in Section 4.2.5, is bounded by the proposed H* value of 16.95 inches in the subject request for an interim amendment.

The NRC staff has not completed its evaluation of the above probabilistic analysis, which must be done before the staff can approve any request for a permanent H* amendment. However, for reasons discussed in Section 4.5, the staff concludes the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

[3] The NRC staff notes that because the reference Monte Carlo simulation for the Model D5 SGs was based on NOP conditions, Westinghouse performed an additional reference Monte Carlo simulation on the Model D5 SGs using SLB conditions, prior to performing the rank ordering of CTE values associated with the probabilistic H* values.

4.2.8 Coefficient of Thermal Expansion

During operation, a large part of contact pressure in a SG T/TS joint is derived from the difference in the CTE between the tube and tubesheet. As discussed in Section 4.2.7, the calculated value of H* is highly sensitive to the assumed values of these CTE parameters. However, CTE test data acquired by an NRC contractor, Argonne National Laboratory , suggested that CTE values may vary substantially from values listed in the ASME Code for design purposes. In Reference 4, the NRC staff highlighted the need to develop a rigorous technical basis for the CTE values, and their potential variability, to be employed in future H* analyses.In response, Westinghouse had a subcontractor review the CTE data in question, determine the cause of the variance from the ASME Code CTE values, and provide a summary report (Appendix A to Reference 14). Analysis of the CTE data in question revealed that the CTE variation with temperature had been developed using a polynomial fit to the raw data, over the full temperature range from 75 °F to 1300 °F. The polynomial fit chosen resulted in mean CTE values that were significantly different from the ASME Code values from 75 °F to about 300 °F. When the raw data was reanalyzed using the locally weighted least squares regression method, the mean CTE values determined were in good agreement with the established ASME Code values.

Westinghouse also formed a panel of licensee experts to review the available CTE data in open literature, review the ANL provided CTE data, and perform an extensive CTE testing program on Alloy 600 and SA-508 steel material to supplement the existing data base. Two additional sets of CTE test data (different from those addressed in the previous paragraph) had CTE offsets at low temperature that were not expected. Review of the test data showed that the first test, conducted in a vacuum, had proceeded to a maximum temperature of 700 °C, which changed the microstructure and the CTE of the steel during decreasing temperature conditions. As a result of the altered microstructure, the CTE test data generated in the second test, conducted in air, was also invalidated. As a result of the large "dead band" region and the altered microstructure, both data sets were excluded from the final CTE values obtained from the CTE testing program.

The test program included multiple material heats to analyze chemistry influence on CTE values and repeat tests on the same samples were performed to analyze for test apparatus influence. Because the tubes are strain hardened when they are expanded into the tubesheet, strain hardened samples were also measured to check for strain hardening influence on CTE values.

The data from the test program were combined with the ANL data that were found by the licensee to be acceptable, and with the data obtained from the open literature search. A statistical analysis of the data uncertainties was performed by comparing deviations to the mean values obtained at the applicable temperatures. The correlation coefficients obtained indicated a good fit to a normal distribution, as expected. Finally, an evaluation of within-heat variability was performed due to increased data scatter at low temperatures. The within-heat variability assessment determined that the increase in data scatter was a testing accuracy limitation that was only present at low temperature..

The testing showed that the nominal ASME Code values for Alloy 600 and SA-508 steel were both conservative relative to the mean values from all the available data. Specifically, the CTE mean value for Alloy 600 was greater than the ASME Code value and the CTE mean value for SA-508 steel was smaller than the ASME Code value. Thus, the H* analyses utilized the ASME

Code values as mean values in the H* analyses. The staff finds this to be conservative because it tends to lead to an over-prediction of the expansion of the tubesheet bore and an under-prediction of the expansion of the tube, thereby resulting in an increase in the calculated H* distance. The statistical variances of the CTE parameters from the combined data base were utilized in the H* probabilistic analysis.

Based on its review of Westinghouse CTE program, the NRC staff concludes that the CTE values used in the H* analyses are fully responsive to the concerns stated in Reference 4 and are acceptable

4.3 Accident-induced Leakage Considerations

Operational leakage integrity is assured by monitoring primary-to-secondary leakage relative to the applicable TS limiting condition of operation limits in TS 3.4.13, "RCS Operational LEAKAGE." However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during DBA to exceed the accident leakage performance criteria in TS 5.5.9.2.b.2, including the leakage values assumed in the plant licensing basis accident analyses.

If a tube is assumed to contain a 100 percent through-wall flaw some distance into the tubesheet, a potential leak path between the primary and secondary systems is introduced between the hydraulically expanded tubing and the tubesheet. The leakage path between the tube and tubesheet has been modeled by the licensee's contractor, Westinghouse, as a crevice consisting of a porous media. Using Darcy's model for flow through a porous media, leak rate is proportional to differential pressure and inversely proportional to flow resistance. Flow resistance is a direct function of viscosity, loss coefficient, and crevice length.

Westinghouse performed leak tests of T/TS joint mockups to establish loss coefficient as a function of contact pressure. A large amount of data scatter, however, precluded quantification of such a correlation. In the absence of such a correlation, Westinghouse has developed a leakage factor relationship between accident induced leak rate and operational leakage rate, where the source of leakage is from flaws located at or below the H* distance.

Using the Darcy model, the leakage factor for a given type accident is the product of four quantities. The first quantity is the ratio of the maximum primary-to-secondary pressure difference during the accident divided by that for normal operating conditions. The second quantity is the ratio of viscosity under normal operating primary water temperature divided by viscosity under the accident condition primary water temperature. The third quantity is the ratio of crevice length under normal operating conditions to crevice length under accident conditions. This ratio equals 1, provided it can be shown that positive contact pressure is maintained along the entire H* distance for both conditions. The fourth quantity is the ratio of loss coefficient under normal operating conditions to loss coefficient under the accident condition. Although the absolute value of these loss coefficients isn't known, Westinghouse has assumed that the loss coefficient is constant with contact pressure such that the ratio is equal to 1. The NRC staff agrees that this is a conservative assumption, provided there is a positive contact pressure for both conditions along the entire H* distance and provided that contact pressure increases at each axial location along the H* distance when going from normal operating to accident conditions. Both assumptions were confirmed to be valid in the original H* analyses submitted with Reference 14.

Leakage factors were calculated for design basis accidents exhibiting a significant increase in primary-to-secondary pressure differential, including MSLB, FLB, locked rotor, and control rod ejection. The design basis FLB heat-up transient was found to exhibit the highest leakage factor, 3.11, meaning that it is the transient expected to result in the largest increase in leakage relative to normal operating conditions.

The latest H* analyses by Westinghouse (Reference 18) did not show an increasing T/TS contact pressure when going from normal operating to MSLB conditions. The new analyses used the revised 3-D finite element model of the lower SG assembly and the new square cell model, discussed in Section 4.2.3.2 of this safety evaluation. Although T/TS contact pressureincreased over some sections of the tubing under SLB conditions, it decreased over other sections within the H* distance. This violated the assumed precondition for assuming that the ratio of loss coefficient under MSLB and normal operating conditions was at least equal to 1. As discussed above, the large scatter of the loss coefficient versus contact pressure data prevented direct use of this data in applying Darcy's leakage model. Instead, Westinghouse considered a number of mathematical functions that represented the potential functional relationship between loss coefficient and contact pressure. For each potential functional relationship, Westinghouse evaluated the ratio of loss coefficient under MSLB and normal operating conditions, at each elevation and radial location within the tubesheet. For each tube, this ratio was integrated over the length of the H* distance yielding a ratio of flow resistances for MSLB and normal operating conditions. This ratio, in conjunction with the differential pressure and viscosity ratios, was then used to compute the ratio of leakage under MSLB and normal operating conditions, at each radial location within the tubesheet.

None of the potential functional relationships between loss coefficient and contact pressure considered by Westinghouse resulted in a leakage ratio value exceeding the value of 3.11 calculated for FLB. The description of the revised 3-D FEA of the lower SG assembly and the square cell model was submitted to the NRC staff for evaluation, but there was insufficient time to review the new information in support of permanent H* amendments for the spring or fall of 2011. However, the staff finds that leakage is not a concern for the proposed period of the interim amendment for reasons discussed in Section 4.5 below.

In connection with Amendment Nos. 161 and 166 (i.e., the currently approved alternate repair criteria (Reference 12)) for Braidwood Unit 2 and Byron Unit 2, the licensee provided a commitment that described how the leakage factor would be used to satisfy TS 5.5.9.a for condition monitoring and TS 5.5.9.b.2 regarding performance criteria for accident induced leakage:

> *For the condition monitoring (CM) assessment, the component of operational leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.11 and added to the total accident leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference between the allowable accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.11 and compared to the observed operational leakage. An administrative operational leakage limit will be established to not exceed the calculated value.*

In the subject amendment request (Reference 1), the licensee stated the program/procedure changes needed to meet these commitments were completed in accordance with Amendment Nos. 161 and 166 and that these changes remain in place and will also apply to the subject license amendment. The staff finds these previously implemented program/procedural changes acceptable, since they provide further assurance, in addition to the licensee's operational leakage monitoring processes, that accident induced SG tube leakage will not exceed values assumed in the licensing bases accident analyses.

4.4 Proposed Change to TS 5.6.9, "Steam Generator Tube Inspection Report"

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

4.5 Technical Bases for Interim H* Amendment

The proposed H* value is based on the conservative assumption that all tubes in all steam generators are severed at the H* location. This is a bounding, but necessary assumption for purposes of supporting a permanent H* amendment because the tubes will not be inspected below the H* distance for the remaining life of the SGs, which may range up to 30 years from now depending on the plant, and because the tubes are susceptible to stress corrosion cracking below the H* distance. In addition, the proposed H* distance conservatively takes no credit for RCP associated with the tube hydraulic expansion process.

As discussed in Sections 4.2.3.2, 4.2.5.2, 4.2.5.3, 4.2.7, and 4.3, the staff has not completed its review of certain elements of the technical basis for the proposed H* distance. Thus, in spite of the significant conservatisms embodied in the proposed H* distance, the staff is unable to conclude at this time that the proposed H* distance is, on net, conservative from the standpoint of ensuring that all tubes will retain acceptable margins against pullout (i.e., structural integrity) and acceptable accident leakage integrity for the remaining lifetime of the steam generators, assuming all tubes to be severed at the H* location. The NRC staff will need to complete its review of these certain elements before it can approve any request for a permanent H* amendment. However, for the reasons below, the staff concludes the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

From a fleet-wide perspective (for all Westinghouse plants with Alloy 600TT tubes), the NRC staff has observed from operating experience that the extent of cracking is at an early stage in terms of the number of tubes affected by cracking below the H* distance and the severity of cracks, compared to the idealized assumption that all tubes are severed at the H* distance. Most of these cracks occur in the lower-most one inch of tubing, which is a region of relatively high residual stress associated with the 1-inch tack roll expansion in that region. Although the extent of cracking can be expected to increase with time, it is the NRC staff judgment based on experience that it will continue to be limited to a small percentage of tubes, mostly near the tube ends, over the next operating cycle (approximately 18 months for Braidwood Unit 2 and Byron Unit 2). The NRC staff observations are based on the review of SG tube inspection reports from throughout the PWR fleet. These reports are reviewed and the NRC staff's conclusions are typically documented within a year of each SG tube inspection. Reference 23 provides a recent example of such a review for Braidwood Unit 2 and Byron Unit 2 by the NRC staff.

The NRC staff reviewed the most recent tubing inspection reports from Braidwood Unit 2 and Byron Unit 2, and provides a summary of relevant findings below.

For Braidwood Unit 2, the licensee reported in Reference 23 that in RFO 13 (Spring 2008), 331 flaw indications were found in the lower-most 1 inch of tubing, out of 18,080 hot-leg tube ends inspected. These indications resulted in 16 tubes being plugged. In Reference 24, the licensee reported that in RFO 14 (Fall 2009), examinations of the tubes within the tubesheet (30 percent of hot-legs in all SGs from T/TS to 18 inches below the T/TS) showed no flaw indications. The only active degradation mechanisms found in the SGs were anti-vibration bar, pre-heater support plate, and foreign object wear, which resulted in the plugging of 6 tubes.

For Byron Unit 2, the licensee reported in Reference 23 that in RFO 14 (Fall 2008), 65 flaw indications were found in the lower-most 0.5 inches of hot-leg tubing, out of 21,574 hot-leg and cold-leg tube ends inspected. These indications resulted in 0 tubes being plugged. In Reference 25, the licensee reported that RFO 15 (Spring 2010), examinations of the tubes within the tubesheet (25 percent of hot-legs in all SGs from TTS to 18 inches below the TTS) showed no flaw indications. The only active degradation mechanisms found in the SGs were anti-vibration bar, pre-heater support plate, and foreign object wear, which resulted in the plugging of 1 tube.

The NRC staff finds the extent and severity of cracking at Braidwood Unit 2 and Byron Unit 2 to be limited and within the envelope of industry experience with similar units.

The NRC staff concludes that there is sufficient conservatism embodied in the proposed H* distances to ensure acceptable margins against tube pullout for at least one operating cycle for the reasons discussed above. The NRC staff also concludes there is reasonable assurance during the next operating cycle that any potential accident induced leakage will not exceed the technical specification performance criteria for accident induced leakage. This reflects current operating experience trends that cracking below the H* distance is occurring predominantly in the tack roll region near the bottom of the tube. At this location, it is the staff's judgment that the total resistance to primary-to-secondary leakage will be dominated by the resistance of any "crevice" in the roll expansion region (due to very high T/TS contact pressures in this region), such that the leakage factors discussed in Section 4.3 will remain conservative even should there be a loss of T/TS contact near the top of the tubesheet due to tubesheet bore eccentricity effects.

4.6 Technical Conclusion

The NRC staff has not completed its review of certain elements of the technical basis for the proposed H* distance and, thus, the staff does not have an adequate basis to approve a permanent H* amendment.

The proposed license amendment applies to Braidwood Unit 2 during RFO 15 and to Byron Unit 2 during RFO 16, and their respective, subsequent operating cycles. The staff concludes that there is sufficient conservatism embodied in the proposed H* distances to ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety. Based on this finding, the staff further concludes that the proposed amendment meets 10 CFR 50.36 and, thus, the proposed amendment is acceptable.

5.0 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.

6.0 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 (76 FR 5617, dated February 1, 2011). 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.

7.0 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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the one-cycle amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Exelon Generation Company LLC, Letter RS-10-197, "License Amendment Request to Revise Technical Specifications (TS) Sections 5.5.9, 'Steam Generator (SG) Program," and TS Sections 5.6.9, 'Steam Generator (SG) Tube Inspection Report,' for Temporary Alternate Repair Criteria," December 14, 2010, NRC ADAMS Accession No. ML103510345. This letter also transmitted Reference 18.

2. Wolf Creek Nuclear Operating Corporation, letter ET-06-004, "Revision to Technical Specification 5.5.9, "Steam Generator Tube Surveillance Program,"" February 21, 2006, NRC ADAMS Accession No. ML060600456.

3. Wolf Creek Nuclear Operating Corporation, letter ET-08-0010, "Withdrawal of License Amendment Request for a Permanent Alternate Repair Criteria in Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program" February 14, 2008, NRC ADAMS Accession No. ML080580201.

4. NRC letter to Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station – Withdrawal of License Amendment Request on Steam Generator tube Inspections," February 28, 2008, NRC ADAMS Accession No. ML080450185.

5. NRC Meeting minutes, "Summary of the October 29 and 30, 2008, Category 2 Public Meeting with the Nuclear Energy Institute (NEI) and Industry to Discuss Modeling Issues Pertaining to the Steam Generator Tube-to-tubesheet Joints," NRC ADAMS Accession No. ML083300422.

6. NRC Meeting minutes, "Summary of the January 9, 2009, Category 2 Public Meeting with the U.S. Nuclear Industry Representatives to Discuss Steam Generator H*/B* Issues," NRC ADAMS Accession No. ML090370945

7. NRC Meeting minutes, "Summary of the April 3, 2009, Category 2 Public Meeting with the U.S. Nuclear Industry Representatives to Discuss Steam Generator H* Issues," April 30, 2009, NRC ADAMS Accession No. ML091210437.

8. Exelon Generation Company LLC, Letter RS-09-071, "Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, – License Amendment Request to Revise Technical Specifications (TS) for Permanent Alternate Repair Criteria," June 24, 2009, NRC ADAMS Accession No. ML091770545.

9. Exelon Generation Company LLC, RS-09-108, " Additional Information Supporting License Amendment Request to Revise Technical Specifications (TS) for Steam Generator Permanent Alternate Repair Criteria," August 14, 2009. This letter also transmitted WEC letter, LTR-SGMP-09-100 P-Attachment (Proprietary) and LTR-SGMP-09-100 NP-Attachment (Non-Proprietary), Revision 0, "Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators," August 12, 2009, NRC ADAMS Accession No. ML092320377 (Non-Proprietary).

10. Exelon Generation Company LLC, Letter RS-09-117, August 31, 2009, responding to Braidwood Station Unit 2 and Byron Station Unit 2 RAI No. 4. This letter also transmitted WEC letter, LTR-SGMP-09-109-P (Proprietary) and LTR-SGMP-09-109-NP (Non-Proprietary) "Response to NRC Request for Additional Information on H*; RAI #4; Model F and D5 Steam Generators," August 25, 2009, NRC ADAMS Accession No. ML092460590 (Non-Proprietary).

11. Southern Nuclear Operating Company, Letter NL-09-1317, August 28, 2009. This letter transmitted WEC letter LTR-SGMP-09-104 P-Attachment "White Paper on Probabilistic Assessment of H*," dated August 13, 2009, NRC ADAMS Accession No. ML092450029 (Non-Proprietary).

12. Exelon Generation Company LLC, Letter RS-09-127, September 15, 2009, amending its H* application for Braidwood Station Unit 2 and Byron Station Unit 2 to apply only to the next operating cycle, NRC ADAMS Accession No. ML092600170

13. NRC letter to Exelon Generating Company LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2 - Issuance of Amendments Re: Revision to Technical Specifications for the Steam Generator Program (TAC NOS. ME1613, ME1614, ME1615, and ME1616)," October 16, 2009, NRC ADAMS Accession No. ML092520512.

14. Westinghouse Electric Company report, WCAP-17072-P (Proprietary) and WCAP-17072-NP (Non-Proprietary), Rev. 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," May 2009, NRC ADAMS Accession No. ML091670172 (Non- Proprietary).

15. NRC letter to Exelon Generation Company LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2 - Transmittal of Unresolved Issues Related to Steam Generator Permanent Alternate Repair Criteria," December 17, 2009, NRC ADAMS Accession No. ML093380167

16. NRC memorandum, R. Taylor to G. Kulesa, "Vogtle Electric Generating Plant – Audit of Steam Generator H* Amendment Reference Documents," July 9, 2010, NRC ADAMS Accession No. ML093030490.

17. Westinghouse Electric Company letter, LTR-SGMP-09-111 P-Attachment, Rev. 1 (Proprietary) and LTR-SGMP-09-111 NP-Attachment, Rev. 1 (Non-Proprietary), "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," (Proprietary/Non-Proprietary) for Review and Approval, NRC ADAMS Accession Nos. ML103400083 (Proprietary) and ML103400084 (Non-Proprietary).

18. Westinghouse Electric Company report, WCAP-17330-P (Proprietary) and WCAP-17330-NP (Non-Proprietary), Rev. 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity) Model F/Model D5)," November 2010; NRC ADAMS Accession No. ML103510338 (Non-Proprietary).

19. NRC Generic Letter 95-05,"Voltage Based Alternate Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995, NRC ADAMS Accession No. ML031070113.

20. NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," September 1988.

21. NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," March 1998.

22. NRC Meeting minutes, "Summary of the January 8, 2009, Category 2 Public Meeting with the Nuclear Energy Institute (NEI) and Industry to Discuss Steam Generator Issues," February 6, 2009, NRC ADAMS Accession No. ML090370782.

23. NRC letter to Exelon Generation Company LLC, "Braidwood Station, Unit 2 -Review of Spring 2008 Steam Generator Tube Inservice Inspection Report (TAC No. ME1029)," August 19, 2009, NRC ADAMS Accession No. ML092240032..
Exelon Generation Company LLC, letter BW080110, November 11, 2008, "Braidwood Station, Unit 2 Thirteenth Refueling Outage Steam Generator Tube Inspection Report," NRC ADAMS Accession No. ML083220444.

24. Exelon Generation Company LLC, letter BW100011, January 27, 2010, "Braidwood, Unit 2 Steam Generator Tube Inspection for Refueling Outage 14," NRC ADAMS Accession No. ML100350212.

25. Exelon Generation Company LLC, letter BYRON 2009-0003, January 20, 2009, "Byron Station Unit 2 Steam Generator Inservice Inspection Summary Report for Refueling Outage 14," NRC ADAMS Accession No. ML090330274

26. Exelon Generation Company LLC, letter BYRON 2010-0041, July 23, 2010, "Byron Station, Unit 2 Steam Generator - Inservice Inspection Summary Report for Refueling Outage 15," NRC ADAMS Accession No. ML102100454.

Principal Contributor: Andrew B. Johnson

Date of issuance: April 13, 2011

April 13, 2011

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 AND BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CHANGES TO TECHNICAL SPECIFICATION SECTIONS 5.5.9, "STEAM GENERATOR (SG) PROGRAM" AND 5.6.9 "STEAM GENERATOR (SG) TUBE INSPECTION REPORT." (TAC NOS. ME5198, ME5199, ME5200, AND ME5201)

Dear Mr. Pacilio:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No.166 to Facility Operating License No. NPF-72 and Amendment No.166 to Facility Operating License No. NPF-77 for the, Braidwood Station, Units 1 and 2, respectively, and Amendment No.172 to Facility Operating License No. NPF-37 and Amendment No.172 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated December 14, 2010.

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/
Nicholas J. DiFrancesco, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No.166 to NPF-72
2. Amendment No.166 to NPF-77
3. Amendment No.172 to NPF-37
4. Amendment No.172 to NPF-66
5. Safety Evaluation

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