December 10, 2004

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

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS RE: PRESSURE-TEMPERATURE LIMITS (TAC NOS. MB7795 AND MB7796)

Dear Mr. Crane:

The U.S. Nuclear Regulatory Commission (NRC or Commission) has issued the enclosed Amendment No. 170 to Facility Operating License No. NPF-11 and Amendment No. 156 to Facility Operating License No. NPF-18 for the LaSalle County Station (LSCS), Units 1 and 2, respectively. The amendments are in response to your application dated January 31, 2003 (RS-03-010), and supplemented by letters dated July 7 (RS-04-099) and November 15, 2004 (RS-04-169).

The LSCS, Unit 2 submittal also contained an analysis of the impact of 32 effective full-power years (EFPY) of operation on the pressure-temperature (P-T) limits. The low pressure coolant injection (LPCI) nozzles in LSCS, Unit 2 have higher adjusted reference temperature (ART) values at 32 EFPY than the reference temperature (RT_{NDT}) for the feedwater nozzles and will have less fracture resistance at 32 EFPY than the feedwater nozzles. Hence, it has not been demonstrated that the feedwater nozzles are more limiting at 32 EFPY for LaSalle Unit 2. Therefore, when LSCS, Unit 2 submits the 32 EFPY limit curves for staff review and approval, the licensee must perform a quantitative evaluation to demonstrate that the feedwater nozzles are more limiting than the LSCS, Unit 2 LPCI nozzles or provide P-T limit curves based on the ART for the LPCI nozzles. Since the ART values for the LPCI nozzle forgings at 20 EFPY are less than the RT_{NDT} values for the feedwater nozzles, the LPCI nozzles have greater fracture resistance than the feedwater nozzles and this is not a concern for the 20 EFPY P-T limits.

The amendments provide new P-T limits for the technical specifications that are valid to 20 effective full power years for each unit. The changes to the P-T curves are based, in part, on the American Society of Mechanical Engineers Code Case N-640, “Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1,” which was reviewed and approved by NRC staff for use by the LSCS in a letter dated November 8, 2000.

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/

Stephen P. Sands, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:
1. Amendment No. 170 to NPF-11
2. Amendment No. 156 to NPF-18
3. Safety Evaluation

cc w/encls: See next page

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

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EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
License No. NPF-11

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated January 31, 2003, and supplemented by letters dated July 7 and November 15, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 170, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the 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

/RA/

Gene Y. Suh, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: December 10, 2004

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 156
License No. NPF-18

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated January 31, 2003, and supplemented by letters dated July 7 and November 15, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 156, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the 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

/RA/

Gene Y. Suh, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: December 10, 2004

ATTACHMENT TO LICENSE AMENDMENT NOS. 170 AND 156

FACILITY OPERATING LICENSE NOS. NPF-11 AND NPF-18

DOCKET NOS. 50-373 AND 50-374

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove Pages	Insert Pages
3.4.11-3	3.4.11-3
3.4.11-6	3.4.11-6
3.4-11-7	3.4.11-7
3.4.11-8	3.4.11-8
3.4.11-9	3.4.11-9
3.4.11-10	3.3.11-10
3.4.11-11	3.4.11-11

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. NPF-11

AND AMENDMENT NO. 156 TO FACILITY OPERATING LICENSE NO. NPF-18

EXELON GENERATION COMPANY, LLC

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By application dated January 31, 2003 [ADAMS Accession No. ML030410027] and supplemented by letters dated July 7 [ADAMS Accession No. ML041950136] and November 15, 2004 [ADAMS Accession No. ML043210296], (Note: Public access to ADAMS has been temporarily suspended so that security reviews of publicly available documents may be performed and potentially sensitive information removed. Please check the NRC Web site for updates on the resumption of ADAMS access), Exelon Generation Company (the licensee) requested changes to the Technical Specifications (TSs) for the LaSalle County Station, Units 1 and 2 (LaSalle). The supplements dated July 7 and November 15, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 1, 2003.

The licensee revised the pressure-temperature (P-T) limits to provide new limits that are valid to 20 effective full power years (EFPY) for each unit. The proposed changes to the P-T curves are based, in part, on the use of the American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1," which was reviewed and approved by NRC staff for use by the LaSalle facility in a letter dated November 8, 2000.

The proposed changes are as follows:

- TS Surveillance Requirement (SR) 3.4.11.1 and SR 3.4.11.2 are modified to reference the revised P-T curves to be used to perform the surveillances. The revised surveillances with the modified wording in bolded text are as follows.

SR 3.4.11.1 a. Reactor coolant system (RCS) pressure and RCS temperature are within the applicable limits specified in Figures 3.4.11-1, 3.4.11-2, and 3.4.11-3 for Unit 1 **up to 20 EFPY**, and Figures 3.4.11-4, 3.4.11-5, and 3.4-11.6 for Unit 2 **up to 20 EFPY**;

b. RCS heatup and cooldown rates are # 100EF in any 1 hour period; and

c. RCS temperature change during system leakage and hydrostatic testing is #20EF in any one hour period when the RCS pressure and RCS temperature are not within the limits of Figure 3.4.11-2 for Unit 1 **up to 20 EFPY**, and Figure 3.4-11-5 for Unit 2 **up to 20 EFPY**.

SR 3.4.11.2 Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.11-3 for Unit 1 **up to 20 EFPY**, and Figure 3.4.11 6 for Unit 2 **up to 20 EFPY.**

- Replace the current TS Figures 3.4.11-1 through Figure 3.4.11-6 with revised TS Figures 3.4.11-1 through 3.4.11-6. The revised TS Figures are applicable to 20 EFPY.

2.0 REGULATORY EVALUATION

The NRC staff evaluates the acceptability of a facility's proposed P-T limits based on the following regulations and guidance:

10 CFR Part 50.60(a) states: *Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications under §50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant program pressure boundary set forth in appendices G and H to this part.*

Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Requirements," establishes requirements related to facility reactor pressure vessel (RPV) material surveillance programs. Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence. RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," contains methodologies for determining the increase in transition temperature resulting from neutron radiation.

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," requires that facility P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Code. The most recent version of Appendix G to Section XI of the ASME Code, which has been endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G, is the 1998 Edition through the 2000 Addenda of the ASME Code. This edition of the Appendix incorporates the provisions of ASME Code Cases N-588 and N-640. Additionally,

Appendix G to 10 CFR Part 50 imposes minimum reactor vessel head flange temperatures when system pressure is at or above 20% of the preservice hydrostatic test pressure.

Generic Letter (GL) 92-01, Revision 1, requested that licensees submit RPV material property data for their plants to the NRC staff for review, and GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

NUREG-0800, “Standard Review Plan,” Section 5.3.2, “Pressure Temperature Limits,” provides guidance on using these regulations and documents in the NRC staff’s review. Additionally, Section 5.3.2 provides guidance to the NRC staff in performing check calculations of the licensee’s submittal.

The methodology of Appendix G to the ASME Code postulates the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum applied stress. For materials in the beltline, upper and lower head regions of the RPV, the maximum flaw size is postulated to have a depth that is equal to one-fourth of the thickness and a length equal to 1.5 times the thickness. Thus, the critical locations in the RPV beltline and head regions are the 1/4-thickness (1/4T) and 3/4-thickness (3/4T) locations, which correspond to the points of the crack tips if the flaws are initiated and grown from the inside and outside surfaces of the vessel, respectively. For the case of evaluating RPV nozzles, the surface flaw is postulated to propagate parallel to the axis of the nozzle’s corner radius. The basic parameter in Appendix G to the ASME Code for calculating P-T limit curves is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. The 1999 Addenda to the 1998 Edition to Section XI of the ASME Code requires that licensees determine the reference stress intensity (K_{Ic}) factors. K_{Ic} is determined from Figure G-2210-1 in Appendix G to the ASME Code. The axis in Figure G-2210-1 are K_{Ic} and T-RT_{NDT}, where T is temperature and RT_{NDT} is the reference temperature of the material. For beltline materials, the RT_{NDT} value is increased due to neutron radiation embrittlement. This value is described as an adjusted reference temperature (ART) value, which is described later in this section.

The methodology of Appendix G requires that P-T curves must satisfy a safety factor of 2.0 on stress intensities arising from primary membrane and bending stresses during normal plant operations (including heatups, cooldowns, and transient operating conditions), and a safety factor of 1.5 on stress intensities arising from primary membrane and bending stresses when leak rate or hydrostatic pressure tests are performed on the RCS. Table 1 of 10 CFR Part 50, Appendix G, provides the NRC staff’s criteria for meeting the P-T limit requirements of Appendix G to the ASME Code and the minimum temperature requirements of the rule for bolting up the vessel during normal and pressure testing operations. Table 1 of 10 CFR Part 50, Appendix G also identifies P-T limits based on the RT_{NDT} value of the materials in the closure flange region that is highly stressed by the bolt preload.

The Appendix G, ASME Code methodology requires that licensees determine the ART value at the maximum postulated flaw depth for beltline materials. The ART value is defined as the sum of the initial (unirradiated) reference temperature value (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term. The ΔRT_{NDT} value is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2 or from surveillance data. The fluence factor is dependent upon

the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} value is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2 or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2 describes the methodology to be used in calculating the margin term.

3.0 TECHNICAL EVALUATION

On November 8, 2000, pursuant to 10 CFR 50.12, the NRC granted an exemption to allow the licensee to deviate from the requirements of 10 CFR Part 50, Appendix G, and to use Code Case N-640 as the part of the basis for generating the LaSalle P-T limit curves (ADAMS Accession No. ML003771016). The NRC staff's evaluation of the proposed P-T limit curves is, in part, based on this exemption. Code Case N-640 allows the use of K_{IC} of Figure A-4200-1 of Appendix A of the ASME Code in lieu of K_{Ia} to determine P-T limits.

In the licensee's letter dated July 7, 2004, the licensee provided the P-T limit curves for non-nuclear inservice leak and hydrostatic testing; for non-nuclear heatup/cooldown; and critical operation conditions effective to 20 EFPY. For non-nuclear inservice leak and hydrostatic testing and for non-nuclear heatup/cooldown conditions, the proposed limits contain curves for the bottom head and a composite curve for the upper vessel and the beltline regions. For core critical operation conditions, the proposed limit is a single curve for the beltline, upper and lower vessel. This letter contains two General Electric (GE) Nuclear Energy (NE) reports: GE-NE-0000-0003-5526-02R1, Revision 1, (Reference 1) and GE-NE-000-0003-5526-01R1, Revision 1, (Reference 2). These reports are applicable for LaSalle Units 1 and 2, respectively.

The licensee has proposed to implement the P-T limit curves based upon limiting RT_{NDT} values for the low alloy steel components in the reactor vessel. The RT_{NDT} value is defined in ASME Code Section III, Subsection NB-2300 and was initially contained in the Summer 1972 Addenda. Section III.A. of 10 CFR Part 50, Appendix G permits licensees with RPVs constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition to determine the RT_{NDT} value differently than that specified in the ASME Code provided the method is approved by the Director, Office of Nuclear Reactor Regulation (NRR). The LaSalle RPVs were procured to earlier ASME Code requirements. Therefore, the material test data is not in accordance with the Summer 1972 Addenda. In 1994, the Boiling Water Reactor (BWR) Owner's Group (Reference 3) proposed a method of estimating the initial RT_{NDT} value that was approved by the staff for generic use in Reference 4. The staff has reviewed the data provided by the licensee and concludes that the values of RT_{NDT} are consistent with the methodology contained in Reference 3. Since the methodology has been approved by NRR staff, the RT_{NDT} values meet 10 CFR Part 50, Appendix G.

Bottom Head Curves

Bottom head curves are utilized because the water in the reactor vessel lower head is separated from the water in contact with the vessel beltline and upper head regions by the reactor baffle plates. The water in the regions above the baffle plate is heated by decay heat from the reactor core, while the water in the lower head is at a lower temperature due to the injection of control rod drive water for vessel pressurization. With little or no circulation through

the recirculation pump loops, these regions are maintained at different temperatures during non-nuclear inservice leak and hydrostatic testing and non-nuclear heatup/cooldown conditions.

The applied stress intensity factors, K_I, for the bottom head curves were determined using the primary and secondary stresses from a control rod drive (CRD)/bottom head finite element analysis that was performed by a BWR reactor vessel vendor in the early 1970's and a membrane stress intensity factor, $M_{m,}$ based on paragraph G-2214.1 in Appendix G, Section XI of the ASME Code. The stress analysis used commonly accepted practices and their applications are consistent with analyses performed to demonstrate conformance with ASME Code Section III.

The pressures and temperatures for the bottom head curves were determined using: (a) the K_I described above; (b) the material fracture toughness described in Code Case N-640; (c) an RT_{NDT} value of 47 °F for the limiting low alloy steel component (bottom head plate C6003-3) in Unit 1 and an RT_{NDT} value of 44 °F for the limiting low alloy steel component (bottom head plate C9306-2) in Unit 2; (d) an adjustment in the RT_{NDT} for heatup/cooldown curves based on a revised finite element analysis that is described in Appendix H of the General Electric Nuclear Energy (GENE) reports (References 1 and 2); and (e) an adjustment in the RT_{NDT} value of the bottom head plate C9306-2 to ensure the Unit 2 P-T limits for the pressure test bounds all nozzles.

Analyses were done to determine whether the nozzle location analyzed in the earlier finite element evaluation was the limiting location. The location analyzed for Unit 1 was limiting, but the location analyzed for Unit 2 was not limiting. Therefore, based on the results of the analyses, the RT_{NDT} of the Unit 2 bottom head plate C9306-2 was increased to account for the difference in fracture properties between the limiting nozzle and the nozzle evaluated in the earlier finite element analysis.

To determine whether the analysis and assumptions used in the earlier CRD/bottom head finite element analysis were applicable, General Electric (GE), the vendor that developed the P-T limit curves, performed additional CRD/bottom head finite element analysis that is described in Appendix H of the GENE reports. This new finite element analysis determined that the earlier analysis was non-conservative for the assumptions used in the analysis. The applied stress intensity factors, K_I, for the finite element analysis described in Appendix H of the reports were determined using the pressure and thermal load cases from a bounding transient and using an new advanced finite element analysis developed by GE for the BWR bottom head geometry with a part through crack, with a depth of ¼ of the vessel wall thickness. GE developed new finite element analyses using ANSYS Version 6.1 (Reference 5). ANSYS is a computer code that is commonly used for performing finite element analyses. The peak stress intensities for the pressure and thermal load cases are used as inputs into the ASME Code Appendix G evaluation methodology to calculate a T-RT_{NDT} value. The adjustment in the RT_{NDT} value for bottom head heatup/cooldown curves is the difference between the T-RT_{NDT} values in the old and new analyses. The new finite element analysis was benchmarked using existing solutions and analyses methods and against many cases with a variety of geometries, loadings and material properties. GE concludes that the results of these bench-marking studies have demonstrated the accuracy of this method. Since the new finite element analysis has been bench-marked against existing solutions, analysis methods, and under a variety of conditions, the staff considers its use in this application acceptable.

Upper Vessel, Flange and Beltline Region Curves

The P-T limits for non-nuclear inservice leak and hydrostatic testing, and non-nuclear heatup/cooldown operations include a curve based on the material properties for the upper vessel (including feedwater nozzle), vessel flange and vessel beltline regions. The P-T limits for core critical operation conditions include a curve based on the material properties for the bottom head, upper vessel, vessel flange and vessel beltline regions. Since the bottom head curves are less limiting than the upper vessel, vessel flange, and beltline region curves, the bottom head curves are not utilized for developing the core critical operation curve. Using the highest RT_{NDT} value for the materials in the beltline, upper vessel, and closure flange regions, the licensee developed P-T limits to meet the criteria in 10 CFR Part 50, Appendix G and Appendix G of Section XI of the ASME Code.

The upper vessel region P-T limits are based on analysis of the feedwater nozzle and beltline regions. The K_I for the feedwater nozzle during pressure testing was computed using the methods from Weld Research Council (WRC) Bulletin 175 together with the geometry from a generic 251-inch BWR/6 feedwater nozzle. Since Appendix G of the ASME Code indicates that the methods from WRC 175 provide approximate methods for analyzing the inside corner of a nozzle and cylindrical shell for elastic stresses due to internal pressure stress, the method of analysis proposed by the licensee for the upper vessel and feedwater nozzle will satisfy 10 CFR Part 50, Appendix G.

The applied stress intensity factors, K_I, for the upper vessel curve during normal operation were determined using the primary and secondary stresses from a feedwater nozzle finite element analysis that was performed by a BWR reactor vessel vendor in the early 1970's and a membrane stress intensity factor, $M_{m,}$ based on the values identified in the 1995 ASME Code paragraph G-2214.1 for a postulated defect normal to the direction of maximum stress. The pressures and temperatures for the upper vessel curve were determined using: (a) the K_I described above, (b) the material fracture toughness described in Code Case N-640, (c) the methods in Appendix 5 of WRC Bulletin 175 and (d) the limiting feedwater transient for normal and upset conditions.

The beltline region P-T limits are based on the ART values for the limiting materials in the beltline of the LaSalle RPVs. The limiting material for the LaSalle Unit 1 RPV is the middle shell axial weld fabricated using weld wire heat 1P3571, which has an ART value at the ¼T location of 84 °F at 20 EFPY. The limiting material for the LaSalle Unit 2 RPV is the lower-intermediate shell plate with heat number C9404-2, which has an ART value at the ¼T location of 77 °F at 20 EFPY. The ART values for each beltline material are calculated based on the neutron flux that was determined for the currently licensed power of 3489 MW_t , using a conservative power distribution and is conservatively used from the beginning to the end of 20 EFPY. The neutron fluxes for the RPVs are calculated using a method consistent with RG 1.190. The methods used to calculate the neutron flux were in accordance with GE Licensing Topical report NEDC-32983P, which was approved by the NRC staff in Reference 6.

The P-T limits apply for both heatup/cooldown and for both ¼T and ¾T locations because the maximum tensile stress for either heatup or cooldown is applied at the ¼T location. For the beltline curves this approach has added conservatism because irradiation effects cause the

allowable K_{IC} at ¼T to be less than at the ¾T for a given temperature. As a result, the ¼T location is limiting at all temperatures. The staff's assessment also included an independent calculation of the ART value for the ¼T location of the LaSalle RPVs beltline regions based on the neutron fluence specified in the submittal for 20 EFPY. For the evaluation of the limiting beltline materials, the staff confirmed that the ART values were based on the methodology of RG 1.99, Rev. 2.

The staff has compared the chemical composition of the materials in the beltline region in Table 4-4 of Reference 1 to those reported by the licensee in their prior power uprate submittal {Table 3-1 of GE Nuclear Energy Report NEDC-32701P, Revision 2, July 1999 identifies the chemical composition (amounts of copper and nickel) for all vertical and girth welds in the beltline of LaSalle Unit 1. This report was submitted in a letter from the licensee dated July 14, 1999}. These chemical compositions are different from those reported in Table 4-4 of Reference 1. The licensee, in a response to a staff request for additional information (RAI), indicates:

> The best-estimate values for copper and nickel for vertical and girth welds in the beltline of LaSalle County Station, Units 1 and 2, were provided in a R. M. Krich to U.S. NRC letter, "Response to Request for Additional Information Regarding Reactor Pressure Vessel Integrity," dated July 30, 1998. These best-estimate values are identical to those values provided in GE Nuclear Energy Report GE-NE-0000-0003-5526-02R1, Revision 1, May 2004, which was submitted to the NRC with the proposed pressure-temperature limits amendment. The copper and nickel values provided in the GE Nuclear Energy Report NEDC-32701P, Revision 2, July 1999, are the non best-estimate chemistry values that were also listed in the July 30, 1998, Generic Letter 92-01 response.

The best-estimate values for the LaSalle Unit 1 beltline vertical and girth welds are based on data reported in a Combustion Engineering Owners Group (CEOG) report, "Best Estimate Copper and Nickel Values in CE Fabricated Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997. This report is applicable to LaSalle Unit 1 because its vessel was fabricated by CE. The staff confirmed that the best-estimate values of copper and nickel in LaSalle Unit 1 beltline vertical and girth welds are identical to those in CE NPSD-1039, Revision 2 and are identical to the values in Reference 1.

The difference between the best-estimate values for the LaSalle Unit 2 beltline vertical and girth welds reported in Reference 2 and NEDC-32701P, Revision 2, are insignificant.

The LaSalle Unit 1 beltline region contains a transition discontinuity between the lower and lower-intermediate shells. The licensee has performed an analysis that demonstrates that the increase in stresses resulting from the transition discontinuity does not result in the materials in this location being more limiting than middle shell axial weld fabricated using weld wire heat 1P3571. Since the axial weld in the middle shell is more limiting than the materials in the transition discontinuity region, the staff agrees that the proposed P-T limits do not need any adjustment due to the higher stresses in the transition discontinuity region. The LaSalle Unit 2 beltline region does not contain a transition discontinuity.

The beltline in both units also contains low-pressure coolant injection nozzles (LPCI nozzles) that the licensee indicates are not limiting. The initial report identified the impact of neutron radiation embrittlement on the LPCI nozzles; but not on the welds of the nozzles to the reactor vessels. In response to a staff RAI, the licensee identified the amount of copper and nickel,

and the unirradiated reference temperature for the welds between the nozzles and the reactor vessels. The values reported are the limiting values for all the nozzles and welds in each reactor vessel. Using these values and the neutron fluence for the LPCI nozzles and welds, the licensee determined the impact of neutron radiation embrittlement on the welds between the nozzles to the reactor vessels. For 20 EFPY and 32 EFPY neutron fluence conditions, the LPCI nozzles were more limiting than the welds of the nozzles to the reactor vessels because the ART values for the LPCI nozzles are greater than the ART values for the welds.

The licensee used the analysis for the feedwater nozzles to represent all the nozzles in the upper shell region. The feedwater nozzle was selected to represent these nozzles because the stress conditions for the feedwater nozzles are the most severe experienced in the vessel. In response to a staff request for additional information (RAI), the licensee provided an analysis to demonstrate that the feedwater nozzles were more limiting than the LPCI nozzles, which are in the upper shell region and the beltline. The impact of pressure was quantitatively evaluated and the impact of the change in temperature during the limiting cooldown transients were qualitatively evaluated. The quantitative evaluation followed the methodology described in Appendix 5 of WRC Bulletin 175. The qualitative evaluation included the impact of the limiting transients and the nozzle dimensions on the thermal stresses. These analyses indicate that the stresses resulting from pressure and thermal transient conditions are greater for the feedwater nozzle than the LPCI nozzle. Since the LPCI nozzle forgings are in the beltline, they will have their RT_{NDT} values increased based on the neutron fluence at 20 EFPY and the amounts of copper and nickel in the nozzle forgings. However, since the feedwater nozzles are not in the beltline, their RT_{NDT} values are not affected by neutron radiation. Since the ART values for the LPCI nozzle forgings at 20 EFPY are less than the RT_{NDT} values for the feedwater nozzles, the LPCI nozzles have greater fracture resistance than the feedwater nozzles. Therefore, since the LPCI nozzles have greater fracture resistance at 20 EFPY and the stresses on the LPCI nozzles are less than those in the feedwater nozzles for the limiting transients, the LPCI nozzles are less limiting than the feedwater nozzles at 20 EFPY. The LPCI nozzles in LaSalle Unit 2 have higher ART values at 32 EFPY than the RT_{NDT} value for the feedwater nozzle and will have less fracture resistance at 32 EFPY than the feedwater nozzles. Hence, it has not been demonstrated that the feedwater nozzles are more limiting at 32 EFPY. Therefore, when the licensee submits the 32 EFPY limit curves for staff review and approval, the licensee must perform a quantitative evaluation to demonstrate that the feedwater nozzles are more limiting than the LaSalle Unit 2 LPCI nozzles or provide P-T limit curves based on the ART values for the LPCI nozzles.

Table 1 in 10 CFR Part 50, Appendix G, establishes additional P-T limits for the closure flange region that is dependent upon the RT_{NDT} value for the limiting closure flange material. For LaSalle Unit 1 and Unit 2, the limiting RT_{NDT} values for the closure flange region is 12 °F and 26 °F, respectively. The staff has confirmed that the proposed P-T limits satisfy the closure flange limits of 10 CFR Part 50, Appendix G.

Based on the NRC staff's review and evaluation of LaSalle proposed P-T limit curves for Units 1 and 2, the staff has determined that the proposed P-T limit curves are consistent with the alternate assessment criteria and methods of ASME Code Case N-640, and satisfy (1) the requirements of 10 CFR 50.60(a), "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," (2) Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," and (3) Appendix G to Section XI of the ASME Code, as exempted by the methods of analyses in the Code Case. On the basis of the above regulatory

and technical evaluations of the licensee’s justifications for TS changes, the staff concludes that the licensee’s proposed TS changes are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (68 FR 15759). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: December 10, 2004

7.0 REFERENCES

1. GE Nuclear Energy, GE-NE-0000-0003-5526-02R1, Revision 1, "Pressure-Temperature Curves for Exelon LaSalle Unit 1," San Jose, California, May 2004

2. GE Nuclear Energy, GE-NE-0000-0003-5526-01R1, Revision 1, "Pressure-Temperature Curves for Exelon LaSalle Unit 2," San Jose, California, May 2004

3. GE Nuclear Energy, NEDC-32399-P, "Basis for GE RT_{NDT} Estimation Method," Report for BWR Owners Group, San Jose, California, September 1994 (GE Proprietary)

4. Letter from B. Sheron, USNRC, to R. A. Pinelli, "Safety Assessment of Report NEDC-32399-P, Basis for GE RT_{NDT} Estimation Method," USNRC, December 16, 1994

5 ANSYS User's Manual, Version 6.1

6. Letter from S. A. Richard, USNRC, to J. F. Klapproth, GE-NE, "Safety Evaluation for NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (TAC No. MA9891)," MFN 01-050, September 14, 2001