October 17, 2005

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

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING PRESSURE AND TEMPERATURE LIMITS (TAC NOS. MC5160, MC5161, MC5162 AND MC5163)

Dear Mr. Crane:

The Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 217 to Facility Operating License No. DPR-19, and Amendment No. 209 to Facility Operating License No. DPR-25 for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Amendment No. 228 to Facility Operating License No. DPR-29, and Amendment No. 223 to Facility Operating License No. DPR-30 for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The amendments are in response to your application dated November 4, 2004, supplemented by your letters dated March 8, May 25 and July 8, 2005. The supplements 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 February 1, 2005 (70 FR 5244).

The amendments revise Technical Specification (TS) Section 3.4.9, "Reactor Coolant System Pressure and Temperature (P/T) Limits," by incorporating revisions to the P/T limit curves for 54 effective full power years (extending to the end of the renewed license). The amendments also resolve a non-conservative condition for TS 3.4.9, Figure 3.4.9-2, "Non-Nuclear Heatup/Cooldown Curve," for QCNPS.

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/

Maitri Banerjee, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-254 and 50-265

Enclosures:
1. Amendment No. 217 to DPR-19
2. Amendment No. 209 to DPR-25
3. Amendment No. 228 to DPR-29
4. Amendment No. 223 to DPR-30
5. Safety Evaluation

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/
Maitri Banerjee, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.217
License No. DPR-19

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated November 4, 2004, as supplemented by letters dated March 8, May 25 and July 8, 2005, 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. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 217, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by J.Hopkins for/

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: October 17, 2005

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.209
License No. DPR-25

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated November 4, 2004, as supplemented by letters dated March 8, May 25 and July 8, 2005, 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 3.B. of Facility Operating License No. DPR-25 is hereby amended to read as follows:

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 209, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by J.Hopkins for/

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: October 17, 2005

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 228
License No. DPR-29

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated November 4, 2004, as supplemented by letters dated March 8, May 25 and July 8, 2005, 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 3.B. of Facility Operating License No. DPR-29 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 228, are hereby incorporated into the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by J.Hopkins for/

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: October 17, 2005

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 223
License No. DPR-30

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated November 4, 2004, as supplemented by letters dated March 8, May 25 and July 8, 2005, 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 3.B. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 223, are hereby incorporated into the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by J.Hopkins for/

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: October 17, 2005

ATTACHMENT TO LICENSE AMENDMENT NOS. 217 AND 209

FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 50-249

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

Remove Pages	Insert Pages
3.4.9-6	3.4.9-6
3.4.9-7	3.4.9-7
3.4.9-8	3.4.9-8

ATTACHMENT TO LICENSE AMENDMENT NOS. 228 AND 223

FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

DOCKET NOS. 50-254 AND 50-265

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

Remove Pages	Insert Pages
3.4.9-6	3.4.9-6
3.4.9-7	3.4.9-7
3.4.9-8	3.4.9-8

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 217 TO FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 209 TO FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 228 TO FACILITY OPERATING LICENSE NO. DPR-29,

AND AMENDMENT NO. 223 TO FACILITY OPERATING LICENSE NO. DPR-30

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, AND

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-237, 50-249, 50-254 AND 50-265

1.0 INTRODUCTION

By application dated November 4, 2004, as supplemented by letters dated March 8, May 25 and July 8, 2005 (Reference 1, Reference 2, Reference 3, Reference 4), Exelon Generation Company, LLC (the licensee) requested changes to the Technical Specifications (TSs) for the the Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The supplements dated March 8, May 25 and July 8, 2005 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 February 1, 2005 (70 FR 5244).

The proposed changes would revise Technical Specifications (TS) Section 3.4.9, "Reactor Coolant System Pressure and Temperature (P/T) Limits." Specifically the proposed changes would revise the P/T limit curves for 54 effective full power year (EFPY) to support additional 20 years of operation under the renewed license and resolve a non-conservative condition for TS 3.4.9, Figure 3.4.9-2, "Non-Nuclear Heatup/Cooldown Curve," for QCNPS.

The proposed changes to the P-T curves are based, in part, on the use of the American Society of Mechanical Engineers (ASME) Code Cases N–640 and N–588. These code cases were approved by the staff for use by QCNPS, Units 1 and 2 and DNPS, Units 2 and 3 in letters to O. D. Kingsley dated February 4, 2000, and August 25, 2000, respectively. Subsequently, the application of the Code Cases N–640 and N–588 was generically approved in Regulatory Guide

(RG) 1.147, Revision 13, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1."

The NRC also evaluated the acceptability of the vessel fluence methodology and the fluence values for the calculation of the P-T limit curves for 54 EFPYs. The NRC staff evaluated the acceptability of the proposed P-T limits based on regulations and guidance discussed below.

2.0 REGULATORY EVALUATION

Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Reactor Vessel Neutron Fluence," March 2001, outlines the attributes of neutron transport methodologies acceptable to the staff for the calculation of pressure vessel fluence. The General Electric Company has a staff approved methodology (Reference 5) for vessel fluence calculations that address the guidance in RG 1.190.

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 required under §50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.

Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," establishes requirements related to facility reactor pressure vessel (RPV) material surveillance programs. 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 in IV.A.2.b that facility P-T limit curves to be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code. Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is above 20% of the preservice hydrostatic test pressure.

Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," requested that licensees submit their RPV 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 staff finds that the licensee in section 5 of its submittal titled "Regulatory Analysis," identified the applicable regulatory requirements.

3.0 TECHNICAL EVALUATION

The staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment which are described in Sections 4 and 5, Appendices A and B, of the licensee's submittal. The detailed evaluation below supports the conclusion that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

3.1 Neutron Fluence Calculation

Based on operational data, GE selected Dresden Unit 3 to calculate the bounding fluence values for all four units of DNPS and QCNPS. The calculations were carried out using the DORT discrete ordinates program with a P_3 Legendre scattering approximation and an appropriate value for the angular quadrature. Neutron source strength was derived from power distribution at the cycle midterm. The downcomer region was represented as a mixture of metal and water at appropriate ratios. For post power uprate operation the rated power level of 2957 MW_t was taken into account. In this manner the peak vessel inside diameter bounding fluence value for all DNPS and QCNPS units is 5.59×10^{17} n/cm^2.

The GE fluence methodology included a limitation on vessel shroud fluence. The licensee's submittal included shroud fluence calculations, however, the licensee stated (Reference 4) that these values will not be used in the context of this license amendment action. Since the licensee's submittal, the limitation for the shroud has been removed from the GE fluence methodology.

TS 3.4.9, "RCS Pressure and Temperature (P/T) Limits," Figures 3.4.9-1 to 3.4.9-3 have been substituted to reflect the revised limits and range of applicability to 54 EFPYs for all four units of DNPS and QCNPS. Regarding vessel fluence values for 54 EFPYs, the staff finds that the licensee used a staff approved methodology with approximations in the range of the recommendations of RG 1.190, and cross sections, sources and source distributions that adhere to the guidance in RG 1.190. Therefore, the proposed values are acceptable.

3.2 P/T Limits Calculations

The methodology of Appendix G to Section XI of the ASME Code, as modified by Code Case N-588, postulates the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum applied stress for plates, forgings and axial welds and parallel to the welding direction for circumferential welds. For materials in the beltline and in 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 Section XI of 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 Section XI of the ASME Code. The axes 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} is increased due to neutron irradiation embrittlement. This value is described as an adjusted reference temperature (ART), which is described later in this section.

The methodology of Appendix G to Section XI of the ASME Code 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 reactor coolant system (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 of Section XI 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} of the materials in the closure flange region that is highly stressed by the bolt preload.

The ASME Code, Section XI, Appendix G methodology requires that licensees determine the effect of radiation when establishing the relationship between K_{IC} and T-RT_{NDT}. RG 1.99, Revision 2 defines the adjusted reference temperature (ART), which replaces RT_{NDT} and accounts for the effects of radiation on the material properties of beltline materials. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ÎRT_{NDT}), and a margin (M) term. The ÎRT_{NDT} 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, Revision 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} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Revision 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, Revision 2 also describes the methodology to be used in calculating the margin term.

The licensee provided the P-T limit curves for non-nuclear inservice leak and hydrostatic testing; for non-nuclear heatup/cooldown; and for core critical operation conditions effective to 54 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 operations conditions the proposed limit is a single curve for the beltline, upper and lower vessel. The November 4, 2004, letter (Reference 1) contains four General Electric (GE) Nuclear Energy (NE) reports: GE-NE-0000-0002-9629-01R1, Revision 1, (Reference 6), GE-NE-0000-0002-9600-01R2, Revision 2, (Reference 7), GE-NE-0000-0002-9600-02R2, Revision 2, (Reference 8), and GE-NE-0000-0002-9600-03R2, Revision 2, (Reference 9). These reports are applicable for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, respectively.

The licensee has proposed to implement the P-T limit curves based upon limiting ART or RT_{NDT} for the low alloy steel components in the reactor vessel. The RT_{NDT} 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} differently than that specified in the ASME Code, provided the method is approved by the Director, Office of Nuclear Reactor Regulation (NRR). The DNPS/QCNPS RPVs were procured to earlier ASME Code requirements. Therefore, the material test data is not in accordance the Summer 1972 Addenda. In 1994, the BWR Owner's Group (Reference 10) proposed a method of estimating the initial RT_{NDT} that was approved by the staff for generic use in Reference 11. 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 10. 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 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 therefore 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 1970s and a membrane stress intensity factor, M_m, based on paragraph G-2214.1 in Appendix G to Section XI of the ASME Code, 1995 Edition with Addenda through 1996. 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 40E F for the limiting low alloy steel component (bottom head plates) in DNPS, Unit 2, an RT_{NDT} value of 44E F for the limiting low alloy steel component (bottom head plate C1173-2) in DNPS, Unit 3, an RT_{NDT} value of 60E F for the limiting low alloy steel component (bottom head plate C1478-3) in QCNPS, Unit 1 and an RT_{NDT} value of 46E F for the limiting low alloy steel component (bottom head plate A1899-1) in QCNPS, 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 F of the GENE reports; and (e) adjustments in the RT_{NDT} of the limiting bottom head plates for DNPS, Units 2 and 3, and QCNPS, Unit 2, to ensure the P-T limits for the pressure test bounds all nozzles. The adjustment for QCNPS, Unit 1 was not necessary because the limiting plate, C1478-3, had a high RT_{NDT} relative to other nozzles.

To determine whether the analysis and assumptions used in the earlier CRD/bottom head finite element analysis were applicable, General Electric, the vendor that developed the P-T limit curves, performed additional CRD/bottom head finite element analysis that is described in

Appendix F 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 F of the reports were determined using the pressure and thermal load cases from a bounding transient and using a new advanced finite element analysis developed by GE for the BWR bottom head geometry with a part-through crack, 1/4 of the vessel wall thickness deep. The new GE finite element analyses were done using ANSYS Version 6.1 (Reference 12). 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, Section XI, Appendix G evaluation methodology to calculate a T-RT_{NDT} value. The adjustment in the RT_{NDT} 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 benchmarkings studies have demonstrated the accuracy of this method. Since the new finite element analysis has been benchmarked against existing solutions and analyses methods and under a variety of conditions, the staff considers its use in this application acceptable.

The CRD/bottom head finite element analysis determined that the bottom head region of the nonnuclear heatup/cooldown curve was non-conservative for QCNPS. Specifically, the current P-T curve for the reactor pressure vessel bottom head region of Figure 3.4.9-2, "Non-nuclear Heatup/Cooldown Curve," was determined to be non-conservative by 14.6 degrees Fahrenheit (E F) for QCNPS, Unit 1 and 11.6E F for QCNPS, Unit 2 during a bounding anticipated operational occurrence for the improper start of an idle recirculation pump, which is more severe than the normal upset condition (i.e., loss of feedwater transient). The current QCNPS P-T limit curves were submitted to the NRC in a letter dated November 12, 1999, and supplemented by letter dated January 10, 2000. Since the P-T limits are established considering the safety factors of ASME Section XI, Appendix G, this non-conservativism in the region of the bottom head curve implies that a safety factor of two may not be maintained if the plant is conducting operations with the RPV bottom head pressure and temperature near the existing curve in TS Figure 3.4.9-2. This nonconservatism does not apply to DNPS, Units 2 and 3 because the P-T limit curves for these units, which were submitted for staff review in letters dated February 27, 2003, as supplemented on July 17, July 31, September 11 and November 25, 2003, were prepared using analyses equivalent to those in Appendix F. The staff review of these DNPS, Units 2 and 3 P-T limit curves is contained in a letter to J. L. Skolds dated November 26, 2003 (Reference 13).

The licensee has informed the staff in Attachment 1 to a letter dated May 25, 2005 (Reference 3), that GE performed three evaluations to determine the impact of the non-conservative condition. The three evaluations were: 1) an evaluation using Appendix E to Section XI of the ASME Code, 2) an evaluation using Appendix G to Section XI of the ASME Code with a reduced postulated flaw size, and 3) an evaluation using Appendix G to Section XI of the ASME Code with a reduced safety factor. The licensee, in Attachment 1 to the May 25, 2005, letter (Reference 3) states:

> The evaluation using Appendix E to Section XI of the ASME Boiler & Pressure Vessel Code, 1995 Edition with Addenda through 1996 (current commitment) demonstrates that the existing Technical Specification Figure 3.4.9-2 P/T limit curve has adequate structural integrity. The evaluation using Appendix G to

> Section XI of the ASME Boiler & Pressure Code, 1995 Edition with Addenda through 1996 demonstrates that the existing Technical Specification Figure 3.4.9-2 P/T limit curve has adequate structural integrity with a reduced postulated flaw size of 13% from 1/4T or with a reduced safety factor from 2.0 to 1.8 on the primary stress. Therefore, since all three evaluations demonstrate that the core-not-critical bottom head (CRD Penetration) curve in report GE Nuclear Energy GE-NE-B13-02057-00-02R1 (i.e. Technical Specification Figure 3.4.9-2) ensures adequate structural integrity of the reactor pressure vessel, the impact of a 14.6EF shift for Quad Cities, Unit 1 would not have caused permanent damage to the RPV, had the limits been violated (within the 14.6EF shift) during the bounding improper start of an idle Recirculation Pump transient. As a result, the RPV operability with respect to the P/T limit of the bottom head non-conforming condition is supported by the above evaluations.

In addition, the licensee compared the actual cooldown pressures and temperatures during the last three years to the TS limits. The licensee reported that the actual bottom head temperatures were well above the curve for the given pressures, and there does not appear to be any issue with previous history relative to the new limit. Based on this review and the results from the GE evaluation, the staff concludes that the use of non-conservative P-T limits did not impact structural integrity of the RPVs at QCNPS, Units 1 and 2.

Upper Vessel, Flange and Beltline Region Curves

The P-T limits for non-nuclear inservice leak and hydrostatic testing and for 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 operations 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 operations curve. Using the highest RT_{NDT} 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 test 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. This methodology was previously evaluated by the staff in a March 23, 2001, letter (Reference 14) to Exelon Generation Company. Since Appendix G to Section XI of the ASME Code indicates that the methods from WRC 175 provide acceptable 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 1970s and a membrane stress intensity factor, M_m, based on the values identified in paragraph G-2214.1 in Appendix G to Section XI of the 1995 Edition with Addenda through 1996 of the ASME Code 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 for the limiting materials in the beltline of the DNPS and QCNPS RPVs. The limiting material in the RPVs for DNPS, Units 2 and 3 and the QCNPS, Units 1 and 2 is the lower-intermediate axial shell electoslag welds, which have an ART at the 1/4T location of 104E F at 54 EFPY. The ART values for each beltline material are calculated based on the neutron flux that was determined for pre-EPU (Extended Power Uprate) conditions of 2527 megawatts-thermal (MW_t) and EPU conditions of 2957 MW_t. 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 5.

The P-T limits apply for both heatup/cooldown and for both 1/4T and 3/4T locations because the maximum tensile stress for either heatup or cooldown is applied at the 1/4T location. For the beltline curves this approach has added conservatism because irradiation effects causes the allowable K_{IC} at 1/4T to be less than at the 3/4T for a given temperature. As a result, the 1/4T location is limiting at all temperatures. The staff's assessment also included an independent calculation of the ART values for the 1/4T location of the DNPS and QCNPS RPVs beltline regions based on the neutron fluence specified in the submittal for 54 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, Revision 2.

The staff has compared the chemical composition for the materials in the beltline region in Table 4-4 of GE Nuclear Energy reports to the values previously reported by the licensee and documented in the Reactor Vessel Integrity Data Base (RVID). The chemical composition data in the GE Nuclear Energy reports are consistent with data documented in the RVID.

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} for the limiting closure flange material. For DNPS, Units 2 and 3 and QCNPS, Units 1 and 2, the limiting RT_{NDT} for the closure flange region is 23E F. 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 DNPS and QCNPS proposed P-T limit curves, the staff has determined that the proposed P-T limit curves are consistent with the alternate assessment criteria and methods of ASME Code Cases N–640 and N–588, 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 modified by the methods of analyses in the Code Cases. On the basis of the above technical evaluations of the licensee's justifications for TS changes, the staff concludes that the licensee's proposed TS changes are acceptable. In addition, the staff has verified that the licensee has performed sufficient review of its operating history and evaluation of structural integrity to confirm that the use of non-conservative P-T limits did not impact the structural integrity of the RPVs at QCNPS, Units 1 and 2.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to 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 (70 FR 5244). 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.

7.0 REFERENCES

1. Letter from P. R. Simpson, Exelon Generation Company to U.S. Nuclear Regulatory Commission "Request for Changes Related to Technical Specification Section 3.4.9, Reactor Coolant System Pressure and Temperature (P/T) Limits," November 4, 2004.

2. Letter from P. R. Simpson, Exelon Generation Company to U.S. Nuclear Regulatory Commission "Request for Additional Information - Pressure and Temperature Limits Curves for 54 Effective Full-Power Years," March 8, 2005.

3. Letter from P. R. Simpson, Exelon Generation Company to U.S. Nuclear Regulatory Commission "Additional Information Related to Dresden and Quad Cities Nuclear Power Stations 54 Effective Full Power Years Pressure and Temperature Limit Curves," May 25, 2005.

4. Letter from P. R. Simpson, Exelon Generation Company to U.S. Nuclear Regulatory Commission "Response to NRC Request for Additional Information - Pressure and Temperature Limit Curves for 54 Effective Full Power Years," July 8, 2005.

5. Letter from S. A. Richards, U.S. Nuclear Regulatory Commission, to J. F. Klapproth, GE Nuclear Energy, "Safety Evaluation for NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation,"" September 14, 2001.

6. GE Nuclear Energy, GE-NE-0000-0002-9629-01R1, Revision 1, "Pressure-Temperature Curves for Exelon Dresden Unit 2," San Jose, California, May 2004.

7. GE Nuclear Energy, GE-NE-0000-0002-9600-01R2, Revision 2, "Pressure-Temperature Curves for Exelon Dresden Unit 3," San Jose, California, May 2004.

8. GE Nuclear Energy, GE-NE-0000-0002-9600-02R2, Revision 2, "Pressure-Temperature Curves for Exelon Quad Cities Unit 1," San Jose, California, May 2004.

9. GE Nuclear Energy, GE-NE-0000-0002-9600-03R2, Revision 2, "Pressure-Temperature Curves for Exelon Quad Cities Unit 2," San Jose, California, May 2004.

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

11. Letter from B. Sheron, U.S. Nuclear Regulatory Commission, to R. A. Pinelli, "Safety Assessment of Report NEDC-32399-P, Basis for GE RT_{NDT} Estimation Method," December 16, 1994.

12. ANSYS User's Manual, Version 6.1

13. Letter from M. Banerjee, U.S. Nuclear Regulatory Commission, to J. L. Skolds, Exelon Generation Company, "Dresden Nuclear Power Station Units 2 and 3 - Issuance of Amendments Regarding Pressure and Temperature Limits," November 26, 2003.

14. Letter from C. Gratton, U.S. Nuclear Regulatory Commission, to J. A. Hutton, Exelon Generation Company, "Limerick Generating Station, Unit 2 - Issuance of Amendment Re: Update the Pressure-Temperature Limit Curves for Limerick Generating Station," March 23, 2001.

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