April 24, 2003

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD NUCLEAR POWER PLANT UNITS 1 AND 2 - STAFF REVIEW OF 50.59 EVALUATION OF PRESSURE-TEMPERATURE LIMITS REPORTS (TAC NOS.: MB2303 AND MB2304)

Dear Mr. Skolds:

By letter dated June 6, 2001, Exelon Generation Company (EGC), the licensee, submitted revisions related to the pressure-temperature (P-T) limits in the Braidwood Nuclear Power Plant Units 1 and 2 Pressure-Temperature Limits Reports (PTLRs). The Braidwood PTLRs were revised to accommodate the change in neutron fluence expected with implementation of power uprate. Additionally, Braidwood Unit 2 PTLR incorporates revisions as a result of using updated reactor vessel material properties based on the most recent surveillance capsule results.

The methods used to revise Braidwood PTLRs have been previously reviewed and approved by the NRC in a January 21, 1998, letter from the NRC to Mr. Oliver D. Kingsley, President, Nuclear Generation Group for Commonwealth Edison Company, the previous license holder for Braidwood Station Units 1 and 2.

A teleconference was held on January 23, 2002, between the NRC staff and Exelon representatives to discuss questions raised by the staff during initial review. Questions during the staff's review led to a submittal dated August 30, 2002, which provided additional information and submittal dated October 14, 2002, which provided Revision 2 to the facilities PTLRs. Through the information provided by the licensee, the staff was able to confirm that the approved methodology was used to update Braidwood Station Units 1 and 2 PTLRs to the submitted versions.

This completes our review of the PTLRs for Braidwood Station Units 1 and 2. Attached is the staff review of the 50.59 evaluation.

Sincerely,

/RA/

Mahesh Chawla, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456 and STN 50-457

Enclosure: Staff Review of 50.59 Evaluation of PTLRs

cc w/encl: See next page

This completes our review of the PTLRs for Braidwood Station Units 1 and 2. Attached is the staff review of the 50.59 evaluation.

Sincerely,

/RA/

Mahesh Chawla, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456 and STN 50-457

Enclosure: Staff Review of 50.59 Evaluation of PTLRs

cc w/encl: See next page

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Braidwood Station Units 1 and 2

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Illinois Department of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, IL 62704

Document Control Desk-Licensing
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Ms. C. Sue Hauser, Project Manager
Westinghouse Electric Corporation
Energy Systems Business Unit
Post Office Box 355
Pittsburgh, PA 15230

Joseph Gallo
Gallo & Ross
1025 Connecticut Ave., NW, Suite 1014
Washington, DC 20036

Ms. Bridget Little Rorem
Appleseed Coordinator
117 N. Linden Street
Essex, IL 60935

Howard A. Learner
Environmental Law and Policy
Center of the Midwest
35 East Wacker Dr., Suite 1300
Chicago, IL 60601-2110

U.S. Nuclear Regulatory Commission
Braidwood Resident Inspectors Office
35100 S. Rt. 53, Suite 79
Braceville, IL 60407

Ms. Lorraine Creek
RR 1, Box 182
Manteno, IL 60950

Illinois Emergency Management
Agency
Division of Disaster Assistance &
Preparedness
110 East Adams Street
Springfield, IL 62701-1109

Chairman
Will County Board of Supervisors
Will County Board Courthouse
Joliet, IL 60434

Attorney General
500 S. Second Street
Springfield, IL 62701

George L. Edgar
Morgan, Lewis and Bockius
1800 M Street, NW
Washington, DC 20036-5869

Braidwood Station Manager
Exelon Generation Company, LLC
35100 S. Rt. 53, Suite 84
Braceville, IL 60407-9619

Site Vice President - Braidwood
Exelon Generation Company, LLC
35100 S. Rt. 53, Suite 84
Braceville, IL 60407-9619

Senior Vice President, Nuclear Services
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Vice President
Mid-West Operations Support
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Chairman, Ogle County Board
Post Office Box 357
Oregon, IL 61061

Exelon Generation Company, LLC
Regulatory Assurance Manager - Braidwood
35100 S. Rt. 53, Suite 84
Braceville, IL 60407-9619

Senior Vice President
Mid-West Regional Operating Group
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Director Licensing
Mid-West Regional Operating Group
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Senior Counsel, Nuclear
Mid-West Regional Operating Group
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Vice President - Licensing and Regulatory Affairs
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Manager Licensing
Braidwood and Byron
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE REVIEW OF THE PRESSURE-TEMPERATURE LIMIT CALCULATIONS

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

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

1.0 INTRODUCTION

By letter dated June 6, 2001, Exelon Generation Company, LLC (EGC), the licensee, submitted revisions related to the pressure-temperature (P-T) limits in the Braidwood Station Units 1 and 2, Pressure-Temperature Limits Reports (PTLRs) (Ref. 7). The licensee revised the heatup and cooldown rates for Units 1 and 2 heatup and cooldown curves. The changes were made to accommodate the change in neutron fluences expected with implementation of power uprate. Additionally, Braidwood Unit 2 PTLR incorporates revisions as a result of using updated reactor vessel material properties based on the most recent surveillance capsule results. The results were submitted through 10 CFR 50.59, "Changes, Tests, and Experiments," process and in accordance with Technical Specification 5.6.6, "Reactor Coolant System (RCS) Pressure-Temperature Limits Report (PTLR)."

By letter dated October 14, 2002, EGC submitted revision 2 related to the pressure-temperature (P-T) limits in the Braidwood Station Units 1 and 2, Pressure-Temperature Limits Reports (PTLRs) (Ref. 5). This revision included information presented in a letter dated August 30, 2002, from J. D. von Suskil to the NRC, "Braidwood Station Response to U.S. NRC Request for Additional Information Regarding the Braidwood Station Pressure-Temperature Limits Report" (Ref. 4).

2.0 REGULATION EVALUATION

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Rev. 1; GL 92-01, Rev. 1, Supplement 1; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. This data is used by the staff as the basis for the staff's review of P-T limit curves and as the basis for the staff's review of

pressurized thermal shock (PTS) assessments per 10 CFR 50.61. Appendix G to 10 CFR Part 50 requires that P-T limit curves 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) Boiler and Pressure Vessel Code. Appendix G to 10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves.

The methodology used to revise Braidwood PTLRs have been previously reviewed and approved by NRC in a January 21, 1998, letter (Ref. 1) from Robert A. Capra, NRC, to Oliver D. Kingsley, Nuclear Generation Group for Commonwealth Edison Company, the previous license holder for Braidwood Station Units 1 and 2. The PTLRs were submitted in accordance with Technical Specification 5.6.6, “Reactor Coolant System (RCS) Pressure-Temperature Limits Report (PTLR).” Revisions to the PTLRs were implemented through 10 CFR 50.59, “Changes, Tests, and Experiments.”

The approved methodology requires application and adherence to the 1989 edition of Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” to Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” of the ASME Boiler and Pressure Vessel Code and Westinghouse Topical Report WCAP-14040-NP-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” (Ref. 2) as modified by the methods of analysis in ASME Code Case N-514, “Low Temperature Overpressure, Section XI, Division 1.”

In a letter from G. F. Dick, Jr. (NRC) to O. D. Kingsley (Commonwealth Edison Company), "Exemption from Requirements of 10 CFR 50.60 - Byron, Units 1 and 2, and Braidwood Station Units 1 and 2 (TAC NOS. M98344, M98345, M98346, M98347)," dated January 16, 1998 (Ref. 6), the use of the 1995 ASME Code Edition with the 1996 Addenda to Appendix G methodology and the lower stress intensity factor correlation was approved for Braidwood Station by a 10 CFR Part 50.60 exemption.

SRP Section 5.3.2 provides an acceptable method of determining the PTLRs for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown PTLRs are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The ASME Code Section XI, Appendix G, methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). 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 (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{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, 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} is a plant-specific or a generic value and whether the chemistry factor (CF) 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 the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculated procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

3.0 TECHNICAL EVALUATION

3.1 LICENSEE'S DETERMINATION

Pursuant to 10 CFR Part 50.59, EGC has provided a change in the PTLRs for Braidwood Station Units 1 and 2. Braidwood PTLRs were revised to accommodate the change in neutron fluence expected with implementation of power uprate. Additionally, Braidwood Unit 2 PTLR incorporates revisions as a result of using updated reactor vessel material properties based on the most recent surveillance capsule results.

The Braidwood Unit 1 PTLR presented was identical to the 1995 Braidwood Unit 1 PTLR with the applicability date being decreased from 16 Effective Full Power Years (EFPY) to 14 EFPY to reflect updated chemistry and uprated fluence values. Table 4.1, Braidwood Unit 1 Calculation of Chemistry Factors Using Surveillance Capsule Data included data from Braidwood Station Units 1 and 2 Capsule W analyses. The limiting chemistry factor in Table 4.1 was calculated to be 25.7°F. In Table 4.4, Braidwood Unit 1 Calculation of Adjusted Reference Temperatures (ARTs) at 14 EFPY(b) at the Limiting Reactor Vessel Material Weld Metal (Based on Surveillance Capsule Data), used a limiting chemistry factor (CF) of 20.6°F which was used in the 1995 PTLR submittal. The (b) footnote indicated the period of applicability had been decreased from 16 EFPY to 14 EFPY to reflect the updated chemistry and uprated fluence values. The NRC staff requested clarification on the supporting methodology to ensure this reduction in EFPY applicability would adequately provide coverage for the updated CF, which was not used in the ARTs' calculation.

In response to NRC staff questions on the original submittal, EGC, submitted Rev. 2 related to the pressure-temperature (P-T) limits in the Braidwood Station Units 1 and 2, PTLRs. This revision included information presented in a letter dated August 30, 2002, from J. D. von Suskil to the NRC, "Braidwood Station Response to U.S. NRC Request for Additional Information Regarding the Braidwood Station Pressure-Temperature Limits Report" (Ref. 4). Attachment 3 of this letter, Braidwood Station Unit 1 PTLR Data Point Comparisons, provides the data correlation between the current submitted Braidwood Unit 1 PTLR, calculated using the 1989 ASME Code, Section XI, Appendix G stress intensity factors and the reevaluated 14 EFPY curves using the 1996 ASME Code, Section XI, Appendix G stress intensity factors and the 5% power uprate fluence. The use of the 1995 ASME Code with the 1996 Addenda to Appendix G methodology and the lower stress intensity factor correlation was approved for Braidwood Station by Ref. 6.

3.2 STAFF EVALUATION

In accordance with the NRC approved methodology and PTLR requirements presented in Table 1 of a January 21, 1998, letter (Ref. 3) from C. I. Grimes, NRC, to R. A. Newton, Westinghouse Owners Group, all minimum requirements to be included in the PTLR were provided.

Chemistry factors for the limiting material of Braidwood Unit 2 were updated within the proper context per Regulatory Guide 1.99, Rev. 2, Position 2.1. The limiting material's chemistry factor was adjusted from 21.1 °F to 25.7 °F due to updated surveillance information which will be included in the Reactor Vessel Integrated Database (RVID).

The staff's calculated ART values for the limiting material agreed with the licensee's calculated ART values for Braidwood Station Units 1 and 2 using the information provided by the licensee in their submittal and following the approved methodology of the Braidwood PTLR requirements.

In a letter dated August 30, 2002, (Ref. 4), Attachment 3, EGC demonstrated that the current submitted Braidwood Unit 1 PTLR, calculated using the 1989 ASME Code, Section XI, Appendix G stress intensity factors was at least as or more conservative than the reevaluated 14 EFPY curves using the 1996 ASME Code, Section XI, Appendix G stress intensity factors and the 5% power uprate fluence.

4.0 CONCLUSIONS

As the pressure-temperature limits have been removed from the technical specifications and EGC has not requested a formal review of the current PTLRs, the staff's purpose of the review is to ensure that the approved methodology was used to modify the PTLR. Through the information provided by the licensee, the staff was able to confirm that the approved methodology was used to update Braidwood Station Units 1 and 2 PTLRs to the submitted versions.

5.0 REFERENCES

1. Letter from R. A. Capra, NRC, to O. D. Kingsley, Nuclear Generation Group, Commonwealth Edison Group, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," January 21, 1998.

2. Andrachek, J. D., DiTommaso, S.M., Malone, M.J., and Rood, M.C., "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040 report, January 1996.

3. Letter from C. I. Grimes, NRC, to R. A. Newton, Westinghouse Owners Group, "Acceptance For Referencing of Topical Report WCAP-14040, Revision 1, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," October 16, 1995.

4. Letter from J. D. von Suskil, Exelon Generation Company, LLC, to the NRC, “Braidwood Station Response to U. S. NRC Request for Additional Information Regarding the Braidwood Station Pressure-Temperature Limits Report,” August 30, 2002.

5. Letter from J. D. von Suskil, Exelon Generation Company, LLC, to the NRC, “Pressure-Temperature Limits Reports (PTLRs), Revision 2, Braidwood Station, Units 1 and 2,” October 14, 2002.

6. Letter from G. F. Dick, Jr. (U.S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Exemption from Requirements of 10 CFR 50.60 - Byron, Units 1 and 2, and Braidwood, Units 1 and 2 (TAC NOS. M98344, M98345, M98346, M98347)," January 16, 1998.