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

September 14, 2009

Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO RELIEF REQUEST I2R-50 (TAC NOS. ME1038 AND ME1039)

Dear Mr. Pardee:

By letter to the Nuclear Regulatory Commission (NRC) dated March 31, 2009, (Agencywide Documents Access and Management System Accession No. ML090960468), Exelon Generation Company, LLC (the licensee) submitted Relief Request (RR) I2R-50 for the second 10-year inservice inspection interval at Braidwood Station, Units 1 and 2. This RR was submitted due to the impracticality of satisfying the relevant requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI for the specified ASME Code Class 1 and 2 components.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete its review. The specific information requested is addressed in the enclosed Request for Additional Information (RAI). Your staff has agreed to provide a response to this RAI within 45 days after the date of this letter.

The NRC staff considers that timely responses to RAIs help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-1547.

Sincerely,

Christopher Gratton for

Marshall J. David, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosure:
Request for Additional Information

cc w/encl: Distribution via ListServ

REQUEST FOR ADDITIONAL INFORMATION

BRAIDWOOD STATION, UNITS 1 AND 2

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

The Nuclear Regulatory Commission (NRC) staff has reviewed the information provided by Exelon Generation Company, LLC (the licensee) in its letter dated March 31, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090960468), pertaining to Relief Request (RR) I2R-50 for the second 10-year inservice inspection (ISI) interval at Braidwood Station, Units 1 and 2. This RR was submitted due to the impracticality of satisfying the relevant requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI for the specified ASME Code Class 1 and 2 components.

The ASME Code of record for the Braidwood Station, Units 1 and 2, second 10-year interval ISI program is the 1989 Edition of the ASME Code, Section XI with no Addenda. The NRC staff requires further information to complete its assessment of RR I2R-50.

1. Reactor Vessel Head-to-Flange Welds 1RV-03-001 and 2RV-03-001

   a) The ASME Code, Section XI, requires essentially 100 percent volumetric coverage of the examination volume specified in Figure IWB-2500-5 for the reactor pressure vessel (RPV) head-to-flange welds. This examination volume includes the actual weld, as well as the adjacent base metal on either side of the weld extending to a distance of one-half the thickness of the vessel wall from the extremities of the weld crown. To obtain the total examination coverage, the licensee presents the calculated circumference in RR Attachment 1-2, "Weld 1RV-03-001," sheet 11 of 19, and RR Attachment 2-2, "Weld 2RV-03-001," sheet 12 of 13, for the 1RV-03-001 and 2RV-03-001 RPV head-to-flange circumference welds, respectively. This circumference value is used to determine the total weld metal volume, and subsequently the total inspection volume coverage. It is not clear to the NRC staff how this circumference value was calculated. Clarify the circumference calculations presented in the stated attachments.

   b) The NRC staff requests that you discuss the extent to which these RPV head-to-flange welds were examined during the first ISI interval, including the percentage of credible volumetric examination coverage that was achieved during these previous examinations. Discuss any relevant conditions or indications that were found during these previous examinations.

2. Pressurizer Spray and Relief Nozzle-to-Vessel Welds 1PZR-01-N2, 1PZR-01-N3, 2PZR-01-N2 and 2PZR-01-N3

   a) For the pressurizer spray and relief nozzle-to-vessel welds 1PZR-01-N2, 1PZR-01-N3, 2PZR-01-N2 and 2PZR-01-N3, discussed in RR Attachments 1-3, "Weld 1PZR-01-N2," 1-4, "Weld 1PZR-01-N3," 2-3, "Weld 2PZR-01-N2," and 2-4, "Weld 2PZR-01 -N3," were there any indications of the presence of unacceptable flaws or conditions found during the limited scope volumetric examination?

b) Discuss the extent to which these pressurizer spray and relief nozzle-to-vessel welds were examined during the first ISI interval, including the percentage of credible volumetric examination coverage that was achieved during these previous examinations. Discuss any relevant conditions that were found during these previous examinations.

3. Pipe-to-Valve Circumferential Weld 1RC-17-13

a) It appears to the NRC staff that there is a typographical error in RR Attachment 1-5, "Weld 1RC-17-13." Section 3 references Figure 4-3, "Examination Volume for Thermal Fatigue Cracking of Sweepolets," from EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure." Clarify the RR to reference the correct figure, or confirm that the reference to Figure 4-3 is appropriate.

b) For the pipe-to-valve circumferential weld 1RC-17-13 discussed in RR Attachments 1-5, indicate whether the limited scope volumetric examination found any presence of indications during the examinations of the subject pipe-to valve circumferential welds.

c) The NRC staff requests that you discuss the extent to which the pipe-to-valve circumferential weld 1RC-17-13 was examined during the first ISI interval, including the percentage of credible volumetric examination coverage that was achieved during these previous examinations. Discuss any relevant conditions that were found during these previous examinations. Include clarification on the consideration of this weld as a "structural discontinuity" during the first ISI inspection interval.

d) RR Attachment 1-5, Section 4 states that the subject weld had been examined during the first ISI inspection interval, and was reselected again in the second interval under the risk-informed ISI program. Why was this weld selected for the risk-informed program? Were there any other welds that could have been selected for which greater coverage could have been obtained? If so, why wasn't one of these welds selected?

4. Steam Generator Auxiliary Feedwater Safe End-to-Nozzle Weld 1SG-05-SGSE-02

a) It appears to the NRC staff that there is a typographical error in RR Attachment 1-6, "Weld 1SG-05-SGSE-02." Section 3 references Figure 4-3 from EPRI TR-112657. Clarify the RR to reference the correct figure, or confirm that the reference to Figure 4-3 is appropriate.

b) It appears to the NRC staff that there is another typographical error in RR Attachment 1-6. Section 3 references ASME Code, Section XI, Table IWB-2500-1, in lieu of Table IWC-2500-1, for categorizing this weld. Clarify the RR to reference the correct table.

c) Discuss the extent to which this steam generator auxiliary feedwater safe end-to-nozzle weld was examined during the first ISI interval, including the percentage of credible volumetric examination coverage that was achieved during these previous examinations. In addition, discuss any relevant conditions that were found during these previous examinations.

d) Indicate if any other adjacent or similar welds were examined for this system. If any, indicate the percentage of credible surface examination coverage that was achieved. If less than essentially 100 percent coverage was achieved for any of these welds, provide supplemental information justifying why compliance with the ASME Code, Section XI requirements for essentially 100 percent volumetric examination coverage of these welds was impractical.

e) For the steam generator auxiliary feedwater safe end-to-nozzle weld 1SG-05-SGSE-02, discussed in RR Attachment 1-6, indicate if any presence of indications were found in the subject auxiliary feedwater safe end-to-nozzle welds during the limited scope volumetric examinations.

5. <u>Steam Generator Tube Sheet-to-Stub Barrel Weld 2SG-01-SGC-02</u>

   a) For the steam generator tube sheet-to-stub barrel weld 2SG-01-SGC-02, discussed in RR Attachment 2-5, "Weld 2SG-01-SGC-02," indicate if any presence of indications were found in the subject steam generator tube sheet-to-stub barrel welds during the limited scope volumetric examinations.

   b) Discuss the extent to which this steam generator tube sheet-to-stub barrel weld was examined during the first ISI interval, including the percentage of credible volumetric examination coverage that was achieved during these previous examinations. Discuss any relevant conditions that were found during these previous examinations.

September 14, 2009

Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO RELIEF REQUEST I2R-50 (TAC NOS. ME1038 AND ME1039)

Dear Mr. Pardee:

By letter to the Nuclear Regulatory Commission (NRC) dated March 31, 2009, (Agencywide Documents Access and Management System Accession No. ML090960468), Exelon Generation Company, LLC (the licensee) submitted Relief Request (RR) I2R-50 for the second 10-year inservice inspection interval at Braidwood Station, Units 1 and 2. This RR was submitted due to the impracticality of satisfying the relevant requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI for the specified ASME Code Class 1 and 2 components.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete its review. The specific information requested is addressed in the enclosed Request for Additional Information (RAI). Your staff has agreed to provide a response to this RAI within 45 days after the date of this letter.

The NRC staff considers that timely responses to RAIs help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-1547.

Sincerely,
/RA by C. Gratton for M. David/
Marshall J. David, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosure:
Request for Additional Information

cc w/encl: Distribution via ListServ

DISTRIBUTION:

PUBLIC	LPL3-2 R/F	RidsNrrDorlLpl3-2 Resource
RidsNrrPMBraidwood Resource	RidsNrrLATHarris Resource	RidsNrrDciCvib Resource
RidsAcrsAcnw_MailCTR Resource	RidsRgn3MailCenter Resource	RidsOgcRp Resource
NRRRidsNrrDorlDpr Resource	TMcLellan, NRR	HGonzalez, NRR

ADAMS Accession Number: ML092510225 *Date of RAI Memo NRR-088

OFFICE	LPL3-2/PM	LPL3-2/LA	DCI/CVIB/BC	LPL3-2/BC
NAME	CGratton for MDavid	THarris	MMitchell*	SCampbell
DATE	9/14/09	9/11/09	8/31/09	9/14/09

OFFICIAL RECORD COPY