Exelon Generation®

4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

Braidwood Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-72 and NFP-77
NRC Docket Nos. STN-50-456 and STN 50-457

Subject: Response to Request for Additional Information Regarding Relief Request I3R-17 to Extend the Reactor Pressure Vessel Inservice Inspection Interval

References:
1) Letter from D. M. Gullott (Exelon Generation Company, LLC) to NRC, "Relief Request I3R-17 to Extend the Reactor Pressure Vessel Inservice Inspection Interval," dated July 21, 2016
2) Email form J. Wiebe (NRC) to J. A. Bauer (Exelon Generation Company, LLC), "Preliminary RAIs for Braidwood Station Relief Request I3R-17," dated October 6, 2016
3) WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," dated October 2011
4) Letter from R. A. Nelson (NRC) to W. A. Nowinowski (PWROG), "Revised Final Safety Evaluation by the Office of Nuclear Reactor Regulation Regarding Pressurized Water Reactor Owners Group Topical Report WCAP-16168-NP-A, Revision 2, 'Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval,'" dated July 26, 2011

In the Reference 1, Exelon Generation Company, LLC (EGC) requested approval of Relief Request I3R-17 in accordance with 10 CFR 50.55a, "Codes and standards," paragraph (z)(1). Relief Request I3R-17 is associated with the Third 10-year Inservice Inspection (ISI) Program Interval for Braidwood Station, Units 1 and 2. The third interval of the Braidwood Station ISI program complies with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition through the 2003 Addenda. The Braidwood Station Third 10-year ISI intervals are currently scheduled to end on July 28, 2018 for Unit 1 and October 16, 2018 for Unit 2.

NRC approval of Relief Request I3R-17 submitted in Reference 1 is requested based on the justification provided in topical report WCAP-16168-NP-A, Revision 3 (Reference 3). This topical report demonstrated that extending the ASME B&PV Code, Section XI ISI interval from the current 10 years to 20 years for reactor pressure vessel (RPV) pressure containing welds is an alternative inspection interval that provides an acceptable level of quality and safety. NRC approval of WCAP-16168 is documented in Reference 4.

In Reference 2, the NRC requested additional information related to Relief Request I3R-17. It was agreed that the response to the request for additional information would be submitted to the NRC on or before November 28, 2016. The subject response is presented in Attachment 1.

As noted in Reference 1, EGC requested approval of this relief request by July 21, 2017.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Joseph A. Bauer at (630) 657-2804.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachment 1: Response to Request for Additional Information Regarding Relief Request I3R-17 to Extend the Reactor Pressure Vessel Inservice Inspection Interval

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, Braidwood Station
NRR Project Manager, Braidwood Station
Illinois Emergency Management Agency – Division of Nuclear Safety

In the Reference 1, Exelon Generation Company, LLC (EGC) requested approval of Relief Request I3R-17 in accordance with 10 CFR 50.55a, "Codes and standards," paragraph (z)(1). Relief Request I3R-17 is associated with the Third 10-year Inservice Inspection (ISI) Program Interval for Braidwood Station, Units 1 and 2. The third interval of the Braidwood Station ISI program complies with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition through the 2003 Addenda. The Braidwood Station Third 10-year ISI intervals are currently scheduled to end on July 28, 2018 for Unit 1 and October 16, 2018 for Unit 2.

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In Reference 2, the NRC requested additional information related to Relief Request I3R-17. It was agreed that the response to the request for additional information would be submitted to the NRC on or before November 28, 2016. The subject response is presented below.

NRC Request for Additional Information

In reviewing the Exelon Generation Company, LLC's (Exelon's) submittal dated July 21, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16203A081), related to request for alternative I3R-17, for the Braidwood Station (Braidwood), Units 1 and 2, the NRC staff has determined that the following information is needed in order to complete its review:

RAI-1

Request for Alternative I3R-17 is based on WCAP-16168, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (ADAMS Accession No. ML113060207). The revised safety evaluation (SE) for WCAP-16168 dated July 26, 2011 (ADAMS Accession Number ML111600303), states in required plant-specific information Item 3 that, "The 20-year inspection interval is a maximum interval."

The licensee stated in its July 21, 2016, request for alternative I3R-17, "Upon approval of this relief request, these required [reactor pressure vessel (RPV) weld] examinations would be performed during the Fourth ISI Interval for each Unit in 2027 plus or minus one refueling outage." The licensee also states that, these dates are consistent with those provided in Pressurized Water Reactor Owners Group (PWROG) letter OG-06-356 (ADAMS Accession No. ML082210245) and the latest implementation plan provided in PWROG letter OG-10-238 (ADAMS Accession Number ML11153A033). If this request for an alternative is approved, the third ISI interval will be extended to July 27, 2028 for Unit 1 and October 15, 2028 for Unit 2. As such, the proposed inspection date of 2027, plus or minus one refueling outage, could be within *the implementation plan dates but potentially be beyond the requested extended intervals, which would put them beyond the 20 year maximum interval. Confirm that the required RPV inspections will be performed before the approved extension dates of July 27, 2028 for Unit 1 and October 15, 2028 for Unit 2.*

Response to RAI-1

Based on the current refueling outage dates projection, Braidwood Station, Unit 1 is scheduled to have a refueling outage in Spring 2027 (A1R26); and Unit 2 is scheduled to have a refueling outage in Fall 2027 (A2R26). As noted above, conducting the required weld examinations in 2027 plus one refueling outage (i.e., Fall 2028 (A1R27) for Unit 1; and Spring 2029 (A2R27) for Unit 2) would extend the examination date beyond the 20 year maximum interval (i.e., July 27, 2028 for Unit 1 and October 15, 2028 for Unit 2).

EGC confirms that the subject weld examinations will be performed no later than the completion of the Spring 2027 refueling outage (A1R26) for Unit 1; and Fall 2027 refueling outage (A2R26) for Unit 2.

RAI-2

The licensee in Tables 3-1 and 3-2 of the July 21, 2016, request for alternative provides information pertaining to RPV inspections for Braidwood, Units 1 and 2. It was stated in both tables that the indications (3 for Unit 1 and 2 for Unit 2) are acceptable per Table IWB-2510-1of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. Also, Table 3-2 listed flaw numbers under "Scaled maximum number of forging flaws" along with the detected number of forging flaws.

a. *Provide the dimensions of the 5 indications mentioned above to allow the staff to confirm that they are acceptable per Table IWB-2510-1.*

b. *Provide information on the methodology used to determine the scaled maximum number of forging flaws.*

Response to RAI-2a

The dimensions of the three Unit 1 indications, identified in Table 3-1; and the two Unit 2 indications, identified in Table 3-2, are presented below.

Braidwood Unit 1 (Three Indications)

Indication 1 (Weld Number 1RV-01-004), Upper Shell to Intermediate Circumferential (Circ) Weld

The indication was classified as a planar subsurface indication 0.5" long, 0.57" from the reactor pressure vessel (RPV) outside diameter (OD) surface with a through wall extent of 0.19" in an area with a design thickness of 8.63".

Indication 2 (Weld Number 1RV-01-003), Intermediate Circ Weld to Lower Circ Weld

The indication was classified as a planar subsurface indication 1.5" long, 3.89" from the RPV OD surface with a through wall extent of 0.16" in an area with a design thickness of 8.63".

Indication 3 (Weld Number 1RV-01-003), Intermediate Circ Weld to Lower Circ Weld

The indication was classified as a planar subsurface indication 1.0" long, 3.44" from the RPV inside diameter (ID) surface with a through wall extent of 0.27" in an area with a design thickness of 8.63".

Braidwood Unit 2 (Two Indications)

Indication 1 (Weld Number 2RV-01-004), Upper Shell to Intermediate Circ Weld

The indication was classified as a planar subsurface indication 4.5" long, 0.51" from the RPV OD surface with a through wall extent of 0.32" in an area with a design thickness of 8.63" (actual thickness is 9.09", but the more conservative thickness was used for the evaluation).

Indication 2 (Weld Number 2RV-01-003), Intermediate Circ Weld to Lower Circ Weld

The indication was classified as a planar subsurface indication 2.1" long, 0.1" from the RPV ID surface with an assigned through wall extent of 0.125" (based on the Ultrasonic Testing (UT) data, the indication did not have any through wall extent; when this occurs the procedure assigns the resolution capability of the transducer as the through wall extent). The thickness based on design information equals 8.63".

Response to RAI-2b

The methodology used to determine the scaled maximum number of forging flaws is discussed in a Westinghouse Electric Company, LLC (i.e., Westinghouse) technical report prepared for Braidwood Station, Units 1 and 2. This report addressed the implementation of NRC-approved technical report, WCAP-16168-NP-A, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," Revision 3, dated October 2011 (Reference 3). The methodology is described below.

Reactor vessel inservice inspection data must be reviewed to confirm that satisfactory inspections have been performed on the reactor vessel and that the flaw distributions used in the pilot plant analyses are bounding. In accordance with the analyses in NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," March 2010, which is the probabilistic fracture mechanics basis for the Alternate PTS Rule (i.e., 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events"), the only flaws of concern are those that are within the inner 3/8th of the reactor vessel thickness and in the beltline region adjacent to the reactor core.

These flaws must meet the allowable sizes in Table IWB-3510-1 of Section XI of the ASME Code. Furthermore, only those flaws, found in an ASME Section XI, Appendix VIII inspection, in the inner 1/10th thickness or 1 inch, whichever is greater, must meet the allowable flaw limits of Tables 4.5-1 and 4.5-2.

The following process is followed for each flaw (recordable indication) identified in one of the welds that is at least partially in the beltline region:

- To determine if the flaw is in the inner 3/8th thickness of the reactor vessel: The flaw "S" dimension is compared to the RV thickness (at the location of the flaw) multiplied by 0.375.

- To determine if the limits of Tables 4.5-1 and 4.5-2 apply, the flaw "S" dimension is compared to the greater of 1/10th of the reactor vessel thickness or 1 inch.

- To determine if the flaw is in the weld or the adjacent forging material (only required if the ISI report does not explicitly state the location of the flaw):

    - Axial Welds - The difference between the angular position of the flaw and the weld centerline is determined. This difference in angular position is used along with the reactor vessel inside diameter and flaw depth to determine the offset of the flaw from the weld centerline. This offset is then compared to ½ of the weld width. If the offset is greater than ½ of the weld width, the flaw is determined to be in the forging material. If the offset is less than ½ of the weld width, the flaw is determined to be in the weld material.

    - Circumferential Welds – The offset or difference between the height of the flaw and the weld centerline is determined. This offset is then compared to ½ of the weld width. If the offset is greater than ½ of the weld width, the flaw is determined to be in the forging material. If the offset is less than ½ of the weld width, the flaw is determined to be in the weld material.

- To determine the orientation of the flaw: For axial welds, ISI scan direction UP/DN yields a parallel scan and CW/CCW yields a perpendicular scan and, for circumferential welds, ISI scan direction UP/DN yields a perpendicular scan and CW/CCW yields a parallel scan. The designations are defined as follows: UP is Up, DN is Down, CW is Clockwise, and CCW is Counter-Clockwise. The direction of the ultrasonic beam/scan was then used to determine the orientation of the flaw. A scan perpendicular to an axial weld or parallel to a circumferential weld is capable of detecting flaws that are oriented axially with respect to the RV. A scan parallel to an axial weld or perpendicular to a circumferential weld is capable of detecting flaws that are oriented circumferentially with respect to the RV.

  The flaws that are determined to be in the beltline region and within the inner 1/10th thickness or 1 inch of the reactor vessel, whichever is greater, are then to be evaluated against the limits of Tables 4.5-1 and 4.5-2. The limits of these tables are expressed in terms of number of flaws per 1000 inches of length or square inches of area. Therefore, in order to make use of Table 4.5-1, the total length of welds inspected in the beltline is determined based on the plant-specific dimensions of the inspected beltline region. This weld length is used to determine plant-specific weld flaw limits. In order to make use of Table 4.5-2, the total area of forgings or plates inspected in the beltline region is also determined. This is also done using the plant-specific dimensions including the RV thickness and diameter along with the requirements for examination volume in Figures IWB-2500-1 and IWB-2500-2 of Section XI of the ASME Code. This area is used to determine plant-specific forging flaw limits.

Table 4.5-1: Allowable Number of Flaws in Welds (Reference 5)

Through-Wall Extent (TWE) of Flaw (in)		Maximum number of flaws per 1000 inches of weld length in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}
TWE_{MIN}	TWE_{MAX}	
0	0.075	No Limit
0.075	0.475	166.70
0.125	0.475	90.80
0.175	0.475	22.82
0.225	0.475	8.66
0.275	0.475	4.01
0.325	0.475	3.01
0.375	0.475	1.49
0.425	0.475	1.00
0.475	Infinite	0.00

Table 4.5-2: Allowable Number of Flaws in Plates or Forgings (Reference 5)

Through-Wall Extent (TWE) of Flaw (in)		Maximum number of flaws per 1000 square inches of inside surface area in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}
TWE_{MIN}	TWE_{MAX}	
0	0.075	No Limit
0.075	0.375	8.05
0.125	0.375	3.15
0.175	0.375	0.85
0.225	0.375	0.29
0.275	0.375	0.08
0.325	0.375	0.01
0.375	Infinite	0.00

Note that the data in Tables 4.5-1 and 4.5-2 above are taken from 10 CFR 50.61a (i.e., Reference 5).

The data in Table 5.5.2-2 below (excerpted from the Westinghouse report) was used to calculate the total forging area (in the beltline) of 9401 in^2.

Table 5.5.2-2: Inspection Area - Braidwood Unit 2	
Inside Diameter of RV (including cladding thickness)	173 in
Cladding Thickness	0.19 in[(a)]
Inside Diameter of Weld Inspection Volume	173.38 in
RV base metal thickness	8.63 in[(a)]
Number of Circumferential Welds	2
Circumferential Weld Length	544.7 in
Total Forging Area (in beltline)	**9401 in^2**

Note:
(a) Value reported in Braidwood Unit 2 ISI report (Reference 11).

The total forging area (in the beltline) is then multiplied by each of the flaw acceptance criteria values in Table 4.5-2 to determine the values shown in Table 5.5.2-3 below.

Table 5.5.2-3: Alternate PTS Rule Allowable Number of Flaws in Plates and Forgings – Scaled for Braidwood Unit 2			
Through-Wall Extent, TWE (in)		Scaled Maximum number of flaws per 9401 square-inches of inside surface area in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}. This flaw density does not include underclad cracks in forgings	Number of Flaws (Axial/Circ.)
TWE_{MIN}	TWE_{MAX}		
0	0.075	No Limit	0
0.075	0.375	76	1 (0/1)
0.125	0.375	30	1 (0/1)
0.175	0.375	8	0
0.225	0.375	3	0
0.275	0.375	1	0
0.325	0.375	1	0
0.375	Infinite	0	0

The sizes of the identified flaws were then compared to the acceptance criteria, in terms of Through-Wall Extent. No Unit 1 flaws were required to be evaluated using the 10 CFR 50.61a flaw acceptance criteria as shown in Table 5.5.1-1 below. One Unit 2 flaw was determined to be within the inner 1 inch of the Unit 2 vessel and thus required an evaluation using the 10 CFR 50.61a flaw acceptance criteria as shown in Table 5.5.2-1 below.

Table 5.5.1-1: Reactor Vessel ISI Information for Beltline Flaws – Braidwood Unit 1								
Weld ISI No.	Indication No.	TWE (in)[a]	Location (Forging/Weld)	Beltline?	Inner (3/8)t?	Inner (1/10)t or 1"?	Flaw Orientation	Flaw Limits Evaluation Required?
1RV-01-004	1	0.19	Forging	Yes	No	No	Circ.	No
1RV-01-003	1	0.16	Forging	Yes	No	No	Circ.	No
1RV-01-003	2	0.27	Weld	Yes	No	No	Circ.	No

Note:
(a) 2a dimension from Table 4.6.1-3

Table 5.5.2-1: Reactor Vessel ISI Information for Beltline Flaws – Braidwood Unit 2								
Weld ISI No.	Indication No.	TWE (in)[a]	Location (Forging/Weld)	Beltline?	Inner (3/8)t?	Inner (1/10)t or 1"?	Flaw Orientation	Flaw Limits Evaluation Required?
2-RV-01-004	1	0.32	Weld	Yes	No	No	Circ.	No
2-RV-01-003	1	0.125	Forging	Yes	Yes	Yes	Circ.	Yes

Note:
(a) 2a dimension from Table 4.6.2-3

This flaw (in weld 2-RV-01-003) was found to be acceptable in accordance with the Alternate PTS Rule flaw acceptance criteria; therefore, Braidwood Station Unit 2 demonstrated that the flaw distributions used in the pilot plant analyses were bounding.

REFERENCES

1. Letter from D. M. Gullott (Exelon Generation Company, LLC) to NRC, " Relief Request I3R-17 to Extend the Reactor Pressure Vessel Inservice Inspection Interval," dated July 21, 2016

2. Email form J. Wiebe (NRC) to J. A. Bauer (Exelon Generation Company, LLC), "Preliminary RAIs for Braidwood Station Relief Request I3R-17," dated October 6, 2016

3. WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," dated October 2011

4. Letter from R. A. Nelson (NRC) to W. A. Nowinowski (PWROG), "Revised Final Safety Evaluation by the Office of Nuclear Reactor Regulation Regarding Pressurized Water Reactor Owners Group Topical Report WCAP-16168-NP-A, Revision 2, 'Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval,'" dated July 26, 2011

5. 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events"