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RS-18-030

10 CFR 50.55a(z)

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

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

Subject: Braidwood Station, Units 1 and 2, Relief Requests Associated with the Fourth Inservice Inspection Interval

In accordance with 10 CFR 50.55a, "Codes and standards," paragraphs (z)(1) and (z)(2), Exelon Generation Company, LLC (EGC) requests NRC approval of the attached relief requests associated with the fourth Inservice Inspection (ISI) interval for Braidwood Station, Units 1 and 2. The fourth interval of the Braidwood ISI Program is currently scheduled to begin on August 29, 2018, and October 17, 2018, for Unit 1 and Unit 2, respectively, subject to the allowable changes for inspection intervals in IWA-2430, "Inspection Intervals," and will comply with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2013 Edition. EGC proposes the following relief requests for the Braidwood fourth 10-year ISI interval:

- I4R-01 requests authorization of alternate risk-informed selection and examination criteria for Category B-F, B-J, C-F-1, and C-F-2 pressure retaining piping welds.
- I4R-02 requests authorization of an alternative method for pressure testing Post Accident Hydrogen Monitoring System piping, Process Sampling System piping
- I4R-03 requests authorization of alternative requirements for repair/replacement of Control Rod Drive Mechanism Canopy Seal Welds in accordance with IWA-4000.
- I4R-04 requests authorization of use of ASME Code Case N-513-4, Evaluation Criteria for Temporary Acceptance Flaws in Moderate Energy Class 2 or 3 Piping.
- I4R-05 requests authorization of an alternative for examination of ASME Section XI, Examination Category B-G-1, Item Number B6.40, Threads in Flange.
- I4R-06 requests authorization of an alternative for the use of Encoded Phased Array Ultrasonic Examination Techniques in lieu of Radiography.

The bases for these relief requests are provided in Attachments 1 through 6, respectively.

Relief requests similar or identical to all listed above have previously been approved for use at Braidwood Station. EGC requests approval of these requests by March 19, 2019, to support implementation of the Braidwood Station fourth 10-year ISI interval.

There are no regulatory commitments contained within this letter.

Should you have any questions concerning this letter, please contact Mr. Ryan M. Sprengel at (630) 657-2814.

Respectfully,

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

Attachments:

1. 10 CFR 50.55a Request Number I4R-01
2. 10 CFR 50.55a Request Number I4R-02
3. 10 CFR 50.55a Request Number I4R-03
4. 10 CFR 50.55a Request Number I4R-04
5. 10 CFR 50.55a Request Number I4R-05
6. 10 CFR 50.55a Request Number I4R-06

ATTACHMENT 1
10 CFR 50.55a Request Number I4R-01
Proposed Alternative In Accordance with 10 CFR 50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--

1. ASME Code Component(s) Affected

Code Class:	1 and 2
Reference:	Table IWB-2500-1, Table IWC-2500-1
Examination Category:	B-F, B-J, C-F-1, and C-F-2
Item Number:	B5.10, B5.40, B5.70, B9.11, B9.21, B9.22, B9.31, B9.32, B9.40, C5.11, C5.21, C5.30, C5.41, C5.51, C5.61, and C5.81
Description:	Alternate Risk-Informed Selection and Examination Criteria for Examination Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds
Component Number:	Unit 1 and Unit 2 Pressure Retaining Piping

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Braidwood Station, Units 1 and 2, Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2013 Edition.

3. Applicable Code Requirement

Table IWB-2500-1, Examination Category B-F, requires volumetric and surface examinations on all welds for Item Numbers B5.10, B5.40, and B5.70.

Table IWB-2500-1, Examination Category B-J, requires volumetric and surface examinations on a sample of welds for Item Numbers B9.11 and B9.31, volumetric examinations on a sample of welds for Item Number B9.22, and surface examinations on a sample of welds for Item Numbers B9.21, B9.32, and B9.40. The weld population selected for inspection is specified in Notes (2) and (7):

Note (2) *Examinations shall include the following:*
- *(a) All terminal ends in each pipe or branch run connected to vessels.*
- *(b) All terminal ends and joints in each pipe or branch run connected to other components where the stress levels exceed either of the following limits under loads associated with specific seismic events and operational conditions:*
  - *(1) primary plus secondary stress intensity range of 2.4 S_m for ferritic steel and austenitic steel.*
  - *(2) cumulative usage factor U of 0.4.*
- *(c) All dissimilar metal welds not covered under Examination Category B-F.*
- *(d) Additional piping welds so that the total number of circumferential butt welds (or branch connection or socket welds) selected for examination equals 25% of the circumferential butt welds (or branch connection or socket welds) in the reactor coolant piping system. This total does not include welds exempted by IWB-1220 or welds in Item Number B9.22. These additional welds may be located as follows:*
  - *(1) For PWR plants*
    - *(a) one hot-leg and one cold-leg in one reactor coolant piping loop,*

(b) one branch, representative of an essentially symmetric piping configuration among each group of branch runs that are connected to reactor coolant loops and that perform similar system functions, and

(c) each piping and branch run exclusive of the categories of loop and runs that are part of system piping of (a) and (b) above.

Note (7) A 10% sample of PWR high pressure safety injection system circumferential welds in piping ≥ NPS 1½ and < NPS 4 shall be selected for examination. This sample shall be selected from locations determined by the Owner as most likely to be subject to thermal fatigue. Thermal fatigue may be caused by conditions such as valve leakage or turbulence effects.

Table IWC-2500-1, Examination Categories C-F-1 and C-F-2 require volumetric and surface examinations on a sample of welds for Item Numbers C5.11, C5.21, C5.51, and C5.61 and surface examinations on a sample of welds for Item Numbers C5.30, C5.41, and C5.81. The weld population selected for inspection is specified in Note (2) for both Examination Categories:

Note (2) The welds selected for examination shall include 7.5%, but not less than 28 welds, of all dissimilar metal, austenitic stainless steel or high alloy welds (Examination Category C-F-1) or of all carbon and low alloy steel welds (Examination Category C-F-2) not exempted by IWC-1220. (Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Categories C-F-1 and C-F-2. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.) The examinations shall be distributed as follows:

(a) the examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt dissimilar metal, austenitic stainless steel, and high alloy welds (Examination Category C-F-1) or carbon and low alloy welds (Examination Category C-F-2) in each system;

(b) within a system, the examinations shall be distributed among terminal ends, dissimilar metal welds, and structural discontinuities prorated, to the degree practicable, on the number of nonexempt terminal ends, dissimilar metal welds, and structural discontinuities in the system; and

(c) within each system, examinations shall be distributed between line sizes prorated to the degree practicable.

4. <u>Reason for Request</u>

In accordance with 10 CFR 50.55a(z)(1), relief is requested on the basis that the proposed alternative utilizing Electric Power Research Institute (EPRI) Topical Report (TR) 112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A (Reference 1) along with two enhancements from ASME Code Case N-578-1 (N-578-1), "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B, Section XI, Division 1," (Reference 4) will provide an acceptable level of quality and safety.

As stated in "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)" (Reference 2):

"The staff concludes that the proposed RI-ISI Program as described in EPRI TR-112657, Revision B, is a sound technical approach and will provide an acceptable level of quality and safety pursuant to 10 CFR 50.55a for the proposed alternative to the piping ISI requirements with regard to the number of locations, locations of inspections, and methods of inspection."

The initial Braidwood Station Risk-Informed Inservice Inspection (RI-ISI) Program was submitted during the first period of the second ISI interval for both units. This initial RI-ISI Program was developed in accordance with EPRI TR-112657, Revision B-A, as supplemented by N-578-1. The initial program was approved for use by the Nuclear Regulatory Commission (NRC) via a safety evaluation (SE) as transmitted to Exelon Generation Company, LLC (EGC) on February 20, 2002 (Reference 5).

The Braidwood Station RI-ISI Program was resubmitted using the same approach during the third ISI interval for both Units. The program was approved for use by the NRC via SE as transmitted to EGC on November 5, 2009 (Reference 6).

The transition from the 2001 Edition through the 2003 Addenda to the 2013 Edition of ASME Section XI for Braidwood Station's fourth ISI interval does not impact the currently approved RI-ISI evaluation methods and process used in the third ISI interval, and the requirements of the new Code Edition will be implemented as detailed in the Braidwood Station ISI Program Plan. Therefore, with the exception of specific weld locations that may have changed due to maintenance or modification activities (e.g., Fukushima FLEX modification) and the addition of an Alloy 600 Augmented Examination Program, the proposed alternative RI-ISI Program for the fourth ISI interval is the same program methodology as approved in Reference 6 for the third ISI interval.

The Risk Impact Assessment completed as part of the initial baseline RI-ISI Program was an implementation/transition check on the initial impact of converting from a traditional ASME Section XI Program to the new RI-ISI methodology. For the fourth interval ISI update, there is no transition occurring between two different methodologies, but rather, the previously approved RI-ISI methodology and evaluation will be maintained for the new interval. The initial methodology of the evaluation has not changed, and the change in risk was simply re-assessed using the initial 1989 Edition with No Addenda ASME Section XI Program prior to RI-ISI and the new element selection for the fourth ISI interval RI-ISI Program. This same process has been maintained in each revision to the Braidwood Station RI-ISI assessment that has been performed to date.

Based on the fourth ISI interval update of this risk impact assessment, the change in risk from the pre-RI-ISI Section XI Program to the fourth interval RI-ISI Program was within the 1.00E-06 and 1.00E-07 acceptance criteria for delta-core damage frequency (Delta-CDF) and delta-large early release frequency (Delta-LERF) as described in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-

Informed Decisions on Plant-Specific Changes to the Licensing Basis." The Delta-CDF and Delta-LERF values for Braidwood Station, Units 1 and 2, are listed in the following table.

Change in Risk from Braidwood Station Pre-RI-ISI Section XI Program to Fourth Interval RI-ISI Program		
Unit No.	**ΔCDF**	**ΔLERF**
Unit 1	7.71E-08	1.53E-08
Unit 2	6.85E-08	1.37E-08

The following tables document the Delta-CDF and Delta-LERF for Braidwood Station Units 1 and 2 over the initial ASME Section XI Program for the fourth ISI interval. The first two tables provide results for Unit 1. The results for Unit 1 are provided in the first table by system and the second table for only the Break Exclusion Region (BER) weld population. The next two tables provide the equivalent results for Unit 2.

Braidwood Station Unit 1 Delta-CDF and Delta-LERF by System

System	ΔCDF Events/Reactor-Year		ΔLERF Events/Reactor-Year	
	RI-ISI	Acceptance Criteria	RI-ISI	Acceptance Criteria
AF	-2.22E-09	1.00E-07	-1.11E-09	1.00E-08
CS	3.52E-11	1.00E-07	5.28E-13	1.00E-08
CV	1.08E-10	1.00E-07	-1.22E-12	1.00E-08
FW	1.50E-08	1.00E-07	9.57E-09	1.00E-08
MS	5.45E-09	1.00E-07	2.72E-09	1.00E-08
RC	5.80E-08	1.00E-07	3.31E-09	1.00E-08
RH	4.38E-11	1.00E-07	1.18E-12	1.00E-08
SI	4.59E-10	1.00E-07	5.23E-10	1.00E-08
SX	2.63E-10	1.00E-07	2.56E-10	1.00E-08
VQ	0.00E+00	1.00E-07	0.00E+00	1.00E-08
Total	7.71E-08	<1.00E-06	1.53E-08	<1.00E-07

Braidwood Station Unit 1 BER Weld Delta-CDF and Delta-LERF by System

System	ΔCDF Events/Reactor-Year		ΔLERF Events/Reactor-Year	
	RI-ISI	Acceptance Criteria	RI-ISI	Acceptance Criteria
FW	2.00E-09	1.00E-07	9.99E-10	1.00E-08
MS	1.69E-09	1.00E-07	8.46E-10	1.00E-08
Total	3.69E-09	<1.00E-06	1.84E-09	<1.00E-07

Braidwood Station Unit 2 Delta-CDF and Delta-LERF by System

System	ΔCDF		ΔLERF	
	RI-ISI	Acceptance Criteria	RI-ISI	Acceptance Criteria
AF	-2.28E-09	1.00E-07	-1.14E-09	1.00E-08
CS	3.23E-11	1.00E-07	4.84E-13	1.00E-08
CV	-9.16E-11	1.00E-07	-7.35E-12	1.00E-08
FW	3.70E-09	1.00E-07	8.12E-09	1.00E-08
MS	5.46E-09	1.00E-07	2.72E-09	1.00E-08
RC	6.11E-08	1.00E-07	3.46E-09	1.00E-08
RH	4.38E-11	1.00E-07	9.31E-13	1.00E-08
SI	2.26E-10	1.00E-07	1.77E-10	1.00E-08
SX	3.30E-10	1.00E-07	3.15E-10	1.00E-08
VQ	0.00E+00	1.00E-07	0.00E+00	1.00E-08
Total	6.85E-08	<1.00E-06	1.37E-08	<1.00E-07

Braidwood Station Unit 2 BER Weld Delta-CDF and Delta-LERF by System

System	ΔCDF Events/Reactor-Year		ΔLERF Events/Reactor-Year	
	RI-ISI	Acceptance Criteria	RI-ISI	Acceptance Criteria
FW	1.97E-09	1.00E-07	9.84E-10	1.00E-08
MS	1.62E-09	1.00E-07	8.10E-10	1.00E-08
Total	3.59E-09	<1.00E-06	1.79E-09	<1.00E-07

The actual "evaluation and ranking" procedure including the Consequence Evaluation and Degradation Mechanism Assessment processes of the currently approved (Reference 6) RI-ISI Program remain unchanged and are continually applied to maintain the Risk Categorization and Element Selection methods of EPRI TR-112657, Revision B-A. These requirements of the RI-ISI Program have been and will continue to be reevaluated and revised as major revisions of the site Probabilistic Risk Assessment (PRA) occur and modifications to plant configuration are made. The Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, Element Selection, and Risk Impact Assessment steps encompass the complete *living program* process applied under the Braidwood Station RI-ISI Program. All Risk Categories and Element Selections were validated as part of the new Fourth Interval Program development, and the living program process will be maintained throughout the interval.

5. <u>Proposed Alternative and Basis for Use</u>

The proposed alternative initially implemented in the Braidwood Station, Units 1 and 2, "Risk-Informed Inservice Inspection Evaluation," (Reference 3), along with the two enhancements noted below, provide an acceptable level of quality and safety as required by 10 CFR 50.55a(z)(1). This initial program, along with these enhancements, was resubmitted and is currently approved for Braidwood Station third ISI interval as documented in Reference 6.

The fourth ISI interval RI-ISI Program will be a continuation of the current application and will continue to be a living program as described in Section 4 of this relief request. No changes to the evaluation methodology as currently implemented under EPRI TR-112657, Revision B-A, are required as part of this interval update. The following two enhancements will continue to be implemented.

a. In lieu of the evaluation and sample expansion requirements in Section 3.6.6.2, "RI-ISI Selected Examinations" of EPRI TR-112657, Braidwood Station will utilize the requirements of Paragraph -2430, "Additional Examinations" contained in N-578-1 (Reference 4). The alternative criteria for additional examinations contained in N-578-1 provides a more refined methodology for implementing necessary additional examinations. The reason for this selection is that the guidance discussed in EPRI TR-112657 includes requirements for additional examinations at a high level, based on service conditions, degradation mechanisms, and the performance of evaluations to determine the scope of additional examinations, whereas N-578-1 provides more specific and clearer guidance regarding the requirements for additional examinations that is structured similar to the guidance provided in ASME Section XI, Paragraphs IWB-2430 and IWC-2430. Additionally, similar to the current requirements of ASME Section XI, Braidwood Station intends to perform additional examinations that are required due to the identification of flaws or relevant conditions exceeding the acceptance standards, during the outage the flaws are identified.

b. To supplement the requirements listed in EPRI TR-112657, Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination Methods," Braidwood Station will utilize the provisions listed in Table 1, Examination Category R-A, "Risk-Informed Piping Examinations," contained in N-578-1 (Reference 4). To

implement Note 10 of this table, paragraphs and figures from the 2013 Edition of ASME Section XI (i.e., Braidwood Station's Code of Record for the fourth ISI interval) will be utilized which parallel those referenced in the code case. Table 1 of N-578-1 will be used as it provides a detailed breakdown for "Examination Method" and "Categorization of Parts to be Examined." Based on these methods and categorization, the examination figures specified in EPRI TR-112657, Section 4 will then be used to determine the examination volume/area based on the degradation mechanism and component configuration. For piping structural elements not subject to a degradation mechanism, N-578-1, Table 1, Note 1 will be applied using the expanded examination volume.

In addition, Alloy 600/82/182 piping structural elements potentially subject to primary water stress corrosion cracking (PWSCC) with full structural weld overlay applied will be removed from the RI-ISI Program and examined under the Braidwood Station Alloy 600 Augmented Examination Program under ASME Code Case N-770-2 (N-770-2) (Reference 9); and non-Alloy 600/82/182 piping structural elements with full-structural weld overlay applied will be removed from the RI-ISI Program and examined under ASME Section XI, Non-Mandatory Appendix Q. Alloy 600/82/182 piping structural elements potentially subject to PWSCC that have been mitigated with a mechanical stress improvement process will remain in the RI-ISI Program and will also be addressed under the Braidwood Station Alloy 600 Augmented Examination Program under N-770-2.

Piping examinations under this augmented examination program are currently performed in accordance with the criteria below. This augmented examination program is subject to change and will be maintained in accordance with the latest regulations relative to PWSCC and the Braidwood Station Alloy 600 Augmented Examination Program. For piping structural elements evaluated under RI-ISI to only be subject to the PWSCC degradation mechanism and all piping structural elements with full structural weld overlay applied, the requirements incorporated in the augmented examination program are:

1. Weld Locations with Full-Structural Overlays
   Alloy 600/82/182 locations with applied full-structural weld overlays where the degradation mechanism assessment of the overlaid weld identified PWSCC, or PWSCC and another degradation mechanism as determined by the RI-ISI Program, will be removed from the RI-ISI Program and administered solely under the Braidwood Station Alloy 600 Augmented Examination Program. These locations will receive examinations as specified under N-770-2 separate from the RI-ISI Program in order to maintain compliance with 10 CFR 50.55a(g)(6)(ii)(F).

   Non Alloy 600/82/182 locations with applied full-structural weld overlays will be removed from the RI-ISI Program and treated solely under the requirements of ASME Section XI, 2013 Edition, Non-Mandatory Appendix Q.

2. Weld Locations Mitigated with Mechanical Stress Improvement Process
   For Alloy 600/82/182 locations where the PWSCC degradation mechanism has been mitigated with a Mechanical Stress Improvement Process, the piping

structural elements will remain in the RI-ISI Program and are subject to the normal RI-ISI element selection process. These welds will also be governed by the Braidwood Station Alloy 600 Augmented Examination Program under N-770-2. The selection and examination of these welds will comply with both RI-ISI and the N-770-2 requirements. The examinations in the fourth ISI interval may be credited to both programs as applicable.

The Braidwood Station RI-ISI Program, as developed in accordance with EPRI TR-112657, Rev. B-A (Reference 1), requires that 25% of the piping structural elements that are categorized as "High" risk (i.e., Risk Category 1, 2, and 3) and 10% of the piping structural elements that are categorized as "Medium" risk (i.e., Risk Categories 4 and 5) be selected for inspection. For this application, the guidance for the examination volume for a given degradation mechanism is provided by the EPRI TR-112657 while the guidance for the examination method and categorization of parts to be examined are provided by the EPRI TR-112657 as supplemented by N-578-1.

For NRC staff consideration in the evaluation of this alternative RI-ISI Program, Enclosure BW-MISC-037, Revision 0 to this relief request contains a summary of the Regulatory Guide 1.200, Revision 2 (Reference 7) evaluation performed on Byron/Braidwood BB-PRA-014, Rev. BB016a (Reference 8), and the impact of the identified gaps on the technical adequacy of the Braidwood Station PRA Model to support this RI-ISI application (see Enclosure, Table 1).

In addition to this risk-informed evaluation, selection, and examination procedure, all ASME Section XI piping components, regardless of risk classification, will continue to receive Code-required system pressure testing as part of the current ASME Section XI Program. VT-2 visual examinations are scheduled in accordance with the Braidwood Station System Pressure Testing Program, which remains unaffected by the RI-ISI program.

6. Duration of Proposed Alternative

Relief is requested for the fourth ISI interval for Braidwood Station, Units 1 and 2.

7. Precedents

- Braidwood Station, Units 1 and 2, third ISI interval Relief Request I3R-01 was authorized by NRC SE dated November 5, 2009 (ADAMS Accession No. ML093070271). This relief request for the Braidwood Station, Units 1 and 2, fourth ISI interval, utilizes a similar RI-ISI methodology to the previously approved relief request.
- Relief Request I4R-01 was authorized for Byron Station, Units 1 and 2, by NRC SE dated December 20, 2016 (ADAMS Accession No. ML16327A396).
- Relief Request I4R-01 was authorized for Limerick Generating Station, Units 1 and 2, by NRC SE dated December 29, 2016 (ADAMS Accession No. ML16344A324).

- Relief Request RR-7 was authorized for St. Lucie Plant, Unit 2 by NRC SE dated August 10, 2015 (ADAMS Accession No. ML15196A623).

- Relief Request 4RR-01 was authorized for Susquehanna Steam Electric Station, Units 1 and 2, by NRC SE dated April 28, 2015 (ADAMS Accession No. ML15098A478).

8. References

1. Electric Power Research Institute (EPRI) Topical Report (TR) 112657 "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Rev. B-A, dated December 1999.

2. Letter from W. H. Bateman (NRC) to G. L. Vine (EPRI), "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)," dated October 28, 1999 (ADAMS Accession Nos. ML993190460 and ML993190474).

3. Initial Risk-Informed Inservice Inspection Evaluation, Revision 0 - Braidwood Station, Units 1 and 2, dated July 2000. (Letter BW000102 from Timothy Tulon (Commonwealth Edison Company) to the NRC, "Braidwood Station Interval 2 Inservice Inspection Program: Relief Request I2R-39, Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and Class 2 Piping Welds," dated October 16, 2000.)

4. ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B, Section XI, Division 1," dated March 28, 2000.

5. Letter from A. J. Mendiola (NRC) to O. D. Kingsley (EGC), "Braidwood Station, Units 1 and 2 - Interval 2 Inservice Inspection Program - Relief Request I2R-39, Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and Class 2 Piping Welds (TAC Nos. MB0506 and MB0507)," dated February 20, 2002 (ADAMS Accession No. ML020350153).

6. Letter from S. Campbell (NRC) to C. G. Pardee (EGC), "Braidwood Station, Units 1 and 2 - Risk-Informed Relief Request I3R-01 for Certain Pressure Retaining Piping Welds (TAC Nos. ME0225 and ME0226)," dated November 5, 2009 (ADAMS Accession No. ML093070271).

7. NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009 (ADAMS Accession No. ML090410014).

8. BB-PRA-014, "Byron and Braidwood Stations, Quantification Notebook," Rev. BB016a, dated March 2017.

9. ASME Code Case N-770-2, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities Section XI, Division 1." (Approval Date: June 9, 2011).

9. Enclosure

BW-MISC-037, Rev. 0, Braidwood Nuclear Generating Station Units 1 and 2, "PRA Capability Assessment for RI-ISI," Attachment 1, dated July 2017.

ENCLOSURE

BW-MISC-037, Revision 0, "PRA Capability Assessment for RI-ISI," Attachment 1, "Summary Statement of Braidwood PRA Model Capability for use in Risk-Informed Inservice Inspection Applications," dated July 2017.

Introduction

Exelon Generation Company (EGC) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating EGC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Braidwood PRA.

PRA Maintenance and Update

The EGC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the EGC Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation procedures. EGC procedure ER-AA-600-1015, "FPIE PRA Model Update" delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating EGC nuclear generation sites. The overall EGC Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on Core Damage Frequency is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities for equipment that can have a significant impact on the PRA model are updated approximately every four years.

In addition to these activities, EGC risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for EGC nuclear generation sites.

- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65(a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant.

The most recent update of the Braidwood PRA model (designated the BB016a model) [6] was completed in March 2017 as a result of a regularly scheduled update to the previous BB011b PRA model [8]. The BB016a model is the most recent evaluation of the risk profile at Braidwood for internal event challenges, including internal flooding. The Braidwood PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the Braidwood PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

PRA Peer Review

Several assessments of technical capability have been made, and continue to be planned for the Braidwood PRA model. A chronological list of the assessments performed includes the following:

- The Braidwood PRA was subjected to a Westinghouse Owners Group (WOG) Peer Review in September 1999.
- The Byron PRA was subjected to a separate Peer Review in July 2000.
- The Byron and Braidwood PRA BB011b version of the model, was Peer Reviewed in July 2013 for internal events and internal flooding [7]. That Peer Review of the Byron and Braidwood PRA was against the requirements of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard [1] and any Clarifications and Qualifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Revision 2 to Regulatory Guide (RG) 1.200 [2]. Of the 320 supporting requirements reviewed 302 were considered met at Capability Category I/II or greater. Ten were considered Capability Category I and only six Not-Met.
- The Byron and Braidwood PRA model that was completed as BB016a was subjected to a Finding-Level Fact and Observation (F&O) peer review [10] in February 2017 to perform an independent assessment of the close-out of Finding-Level F&Os of record from the 2013 peer review. As a result of the closure of F&Os during this review, all Supporting Requirements are now considered as Met at Capability Category II or higher.

In addition, plant changes made since the last PRA update have been reviewed and determined to not have a significant PRA impact. These items are documented in the Updating Requirements Evaluation (URE) database for consideration in future PRA updates, as appropriate.

Guidance from EPRI Report on PRA Technical Adequacy for RI-ISI

EPRI report TR-1021467-A [9] provides guidance on the PRA Standard Capability Category necessary to support RI-ISI. This report received a Safety Evaluation (SE) from the NRC in January 2012. Reg. Guide 1.200 considers it a good practice to have, in general, SRs meet Capability Category II for applications. However, according to the EPRI report not all SRs require Capability Category II to adequately support RI-ISI applications.

Per the F&O Closure Review [10], all Byron and Braidwood Supporting Requirements are now considered as Met at Capability Category II or higher. There is one remaining open F&O (SY-B12-01) associated with an SR that is met, however. This F&O asserts that the lack of modeling of an electrical and battery room cooling dependency is not adequately justified. A room cooling dependency becomes particularly important in RI-ISI for steam line break initiators outside containment. This F&O has been partially addressed in the current model by adding a room cooling dependency to the Miscellaneous Electrical Equipment Room (MEER). However, for the remaining electrical rooms, the resolution of the room cooling dependency in the model of record was not fully accepted by the F&O Closure Review. To address this outstanding F&O, a sensitivity model was created to explore the impact of adding additional room cooling dependency. The net effect does not change the importance of steam line breaks outside containment since the addition of the MEER room cooling by itself was sufficient to make the consequence risk high. Therefore, this open F&O has negligible impact on the RI-ISI consequence rankings. Furthermore, according to EPRI TR-1021467-A and the associated NRC SE, the Braidwood PRA model BB016a remains adequate for use in the RI-ISI application.

General Conclusion Regarding PRA Capability

The Braidwood PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk-informed inservice inspection applications.

Conclusion Regarding PRA Capability for Risk-Informed ISI

The Braidwood PRA model continues to be suitable for use in the risk-informed inservice inspection application. This conclusion is based on:

- PRA maintenance and update processes in place, and
- PRA technical capability evaluations that have been performed.

In support of the PRA analyses for the Braidwood 10-year interval evaluation using the BB016a PRA model, no remaining gaps to the PRA standard exist that would merit any additional RI-ISI-specific sensitivity studies, beyond the sensitivity discussed in Section 4.0, in the presentation of the application results.

References

1. ASME/American Nuclear Society, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-Sa-2009, March 2009.

2. An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment results for Risk-Informed Activities, Regulatory Guide 1.200, U.S. Nuclear Regulatory Commission, March 2009 Revision 2.

3. Deleted.

4. Deleted.

5. *Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-informed Inservice Inspection Programs*, EPRI TR-1021467-A, June 2012.

6. BB-PRA-014, Rev. BB016a, "Quantification Notebook", March 2017.

7. Byron/Braidwood Nuclear Plants RG 1.200 Internal Events and Flooding PRA Peer Review Report, December 2013.

8. BB-PRA-014, Rev. BB011b, "Quantification Notebook", November 2012.

9. Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-informed Inservice Inspection Programs, EPRI TR-1021467-A, June 2012.

10. Byron and Braidwood Nuclear Power Plants PRA Finding Level Fact and Observation Technical Review, Report # 032299-RPT-05, Rev. 0, June 2017.

ATTACHMENT 2
10 CFR 50.55a Request No. I4R-02
Proposed Alternative In Accordance with 10 CFR 50.55a(z)(2)
--Hardship or Unusual Difficulty Without Compensating Increase
In Level of Quality or Safety--

1. ASME Code Component(s) Affected

Code Class:	2
Reference:	IWC-5200, "System Test Requirements"
Examination Category:	C-H
Item Number:	C7.10
Description:	Alternative Method for Pressure Testing Post Accident Hydrogen Monitoring System Piping, Process Sampling (PS) System Piping
Component Numbers:	ASME Section XI Class 2 Piping Outside of Containment Between Valves 1(2)PS228A(B) and 1(2)PS230A(B). [Reference Drawings M-68 Sheet 7 (Unit 1) and M-140 Sheet 6 (Unit 2)]

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Braidwood Station, Units 1 and 2, Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2013 Edition.

3. Applicable Code Requirement

Table IWC-2500-1, Examination Category C-H, Item Number C7.10 requires that all Class 2 piping be tested using the VT-2 visual examination method at a frequency of each inspection period. The portion of the Process Sampling (PS) System containing the affected piping is not required to operate under normal plant operating conditions; therefore, as required by IWA-5210 and IWC-5221, a system leakage test is required in accordance with IWA-5211(a).

IWC-5210(b)(1) states the contained fluid in the system shall serve as the pressurizing medium and where air is used, the test procedure shall permit the detection and location of through-wall leakages in components of the system tested.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(2), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The specified piping serves as the supply flow path from the containment to the Hydrogen Monitors via 1/4" tubing connections and the return flow path from the Hydrogen Monitors (via 1/4" tubing connections) back to the containment. The system medium is air. The system is comprised of two separate trains for each unit. The subject piping is 1/2" NPS (nominal pipe size) and/or 1/4" stainless steel piping/tubing (SA 312 TP 304 pipe along with SA 213 TP 304 or 316 tubing). The system design pressure is 60 psig. The approximate length of piping/tubing per train (supply and return piping combined) is 275' for 1A, 225' for 1B, 245' for 2A, and 185' for 2B. The nominal system operating pressure ranges across the system from vacuum on the suction piping to a

maximum of 10 psig at the pump discharge, which decreases for the remainder of the piping. In the past, the piping was tested by pressurizing the volume and then performing a soap bubble or "snoop" test on all welds and piping. During the review of surveillance results in 2005, Braidwood Station determined a portion (approximately 50' of supply and return piping combined) of the piping on the 1A train is located in a pipe tunnel and is physically inaccessible for VT-2 visual examination due to the close proximity of adjacent piping and the pipe tunnel wall. Due to the interferences and congestion in the area, the examiner could not physically get close enough to the associated piping to apply the soap bubble solution that is necessary to meet the IWC-5210(b) examination requirements. The use of an ultrasonic sound gun was considered for the inaccessible piping, but the obstructions surrounding the area of interest significantly reduce the ability to detect and pinpoint a leak.

In addition to the limitations associated with the 1A train, for all trains there are significant portions of the piping outside the pipe tunnel located at upper elevations (approximately 30 feet above the floor) where the performance of the VT-2 visual examination using soap solution creates a personal safety hazard. In order to meet the Code requirements for the examination, the examiner would have to perform a hand over hand walk down while using fall protection along with a retractable lanyard to get close enough to the piping to apply the soap bubble solution and perform the VT-2 visual examination required by ASME Section XI. Due to the congestion from other piping in the area, scaffolding cannot be erected to provide access to the piping.

As stated previously, the subject piping is a maximum 1/2" NPS stainless steel pipe. The majority of the piping connections are socket welded with only the connections for the 1/4" diameter tubing having threaded connections. For piping 1" NPS and less, IWA-4540(b)(5) of the 2013 Edition of ASME Section XI excludes hydrostatic testing and system leakage testing (VT-2 visual examination) of piping and components after welded replacement; ASME Section XI would not require any pressure testing of replacement of piping and valves for this system.

5. <u>Proposed Alternative and Basis for Use</u>

As an alternate examination to the Code-required surface examination, Braidwood Station proposes to use an alternate method of testing for system piping outside of containment [piping between valves 1(2)PS228A to 1(2)PS230A and 1(2)PS228B to 1(2)PS230B] for ASME Section XI periodic pressure testing.

The safety related ASME Class 2 sections of piping and valves associated with the PS system at other containment penetrations in the system, where the balance of the system is non-safety related (i.e., Penetration P-70), are tested in accordance with the requirements of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," and are not required to be pressure tested per IWA-5110(c). The proposed alternative is to apply the Appendix J testing method (which is already required for the containment isolation valves at Penetrations P-36 and P-45) on the remaining portion of the ASME Class 2 piping outside of Penetrations P-36 and P-45.

The remaining portion of Class 2 piping outside of the primary containment examination boundary will be examined by pressurizing the remainder of the system to at least the applicable peak accident pressure, which is higher than the system nominal operating pressure, and applying the Appendix J acceptance criteria for the solenoid valves associated with Penetrations P-36 and P-45 to the remainder of the system located outside of containment. The applicable acceptance criteria used for the Appendix J test surveillances (currently ≤ 10 standard cubic feet per hour) would be applied independently to the supply and return piping for each hydrogen monitor train, and subsequent corrective actions would be applied to the remainder of the system. This proposed method of testing is consistent with the requirements of Appendix J and will provide a leak detection method equivalent to the soap bubble solution along with the VT-2 visual examination method for the subject piping.

As with the Appendix J volumes, if test results indicate leakage above the criteria used on the containment penetrations, an Issue Report will be initiated in accordance with the Exelon Generation Company, LLC (EGC) Corrective Action Program and the appropriate corrective actions would be employed to identify the source of leakage. The source of leakage for the piping outside of containment would most likely be attributed to valve packing or threaded tubing connections, since the majority of the system is socket welded stainless steel piping with no known degradation mechanism or previous history of failure.

6. Duration of Proposed Alternative

Relief is requested for the fourth ISI interval for Braidwood Station, Units 1 and 2.

7. Precedents

- Braidwood Station, Units 1 and 2, third ISI interval Relief Request I3R-05 was authorized per Nuclear Regulatory Commission (NRC) SE dated July 20, 2010 (ADAMS Accession No. ML101970288). This relief request for the Braidwood Station, Units 1 and 2, fourth ISI interval, utilizes a similar approach to the previously approved relief request.

8. References

None.

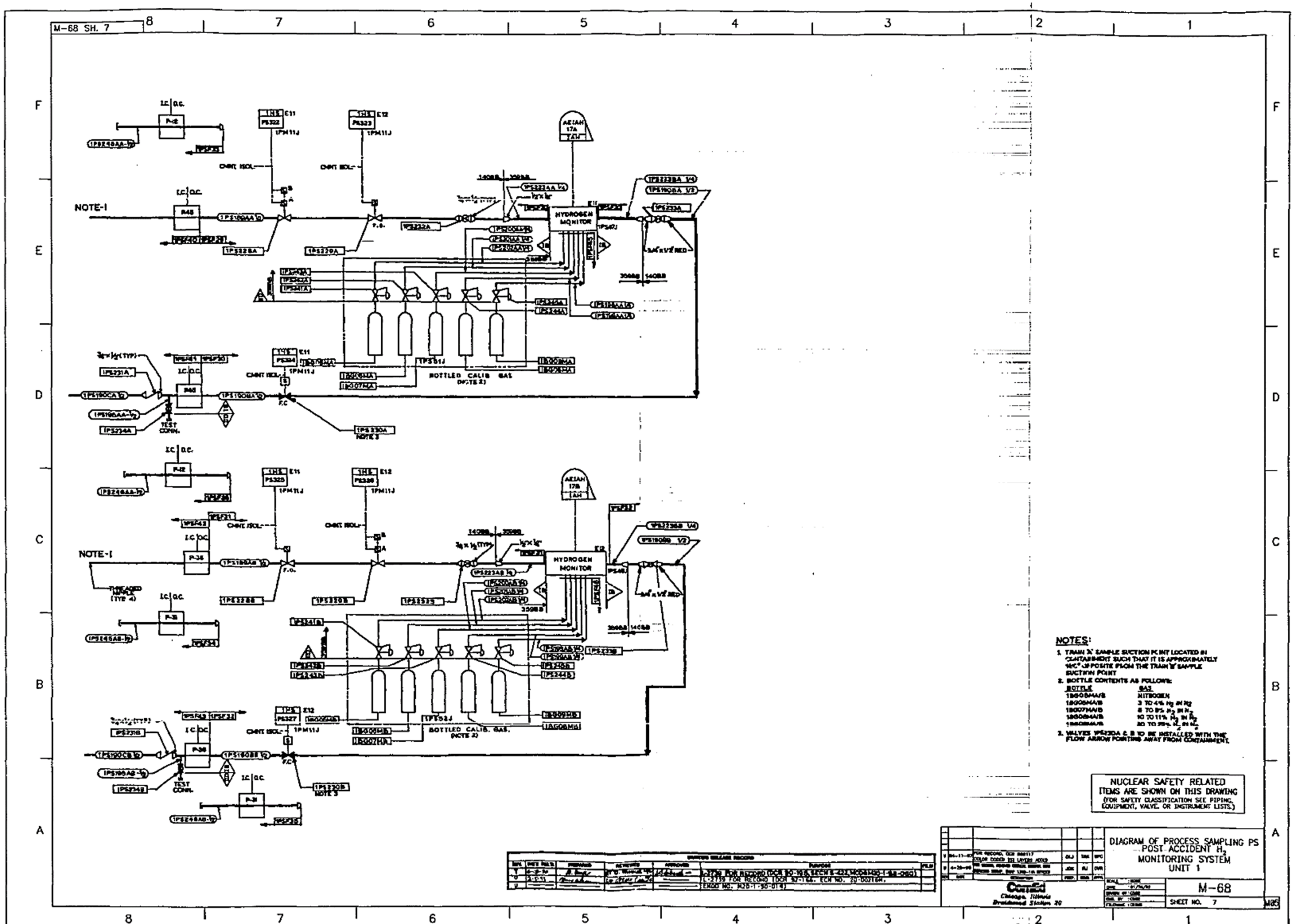
10. Enclosure (For Information Only)

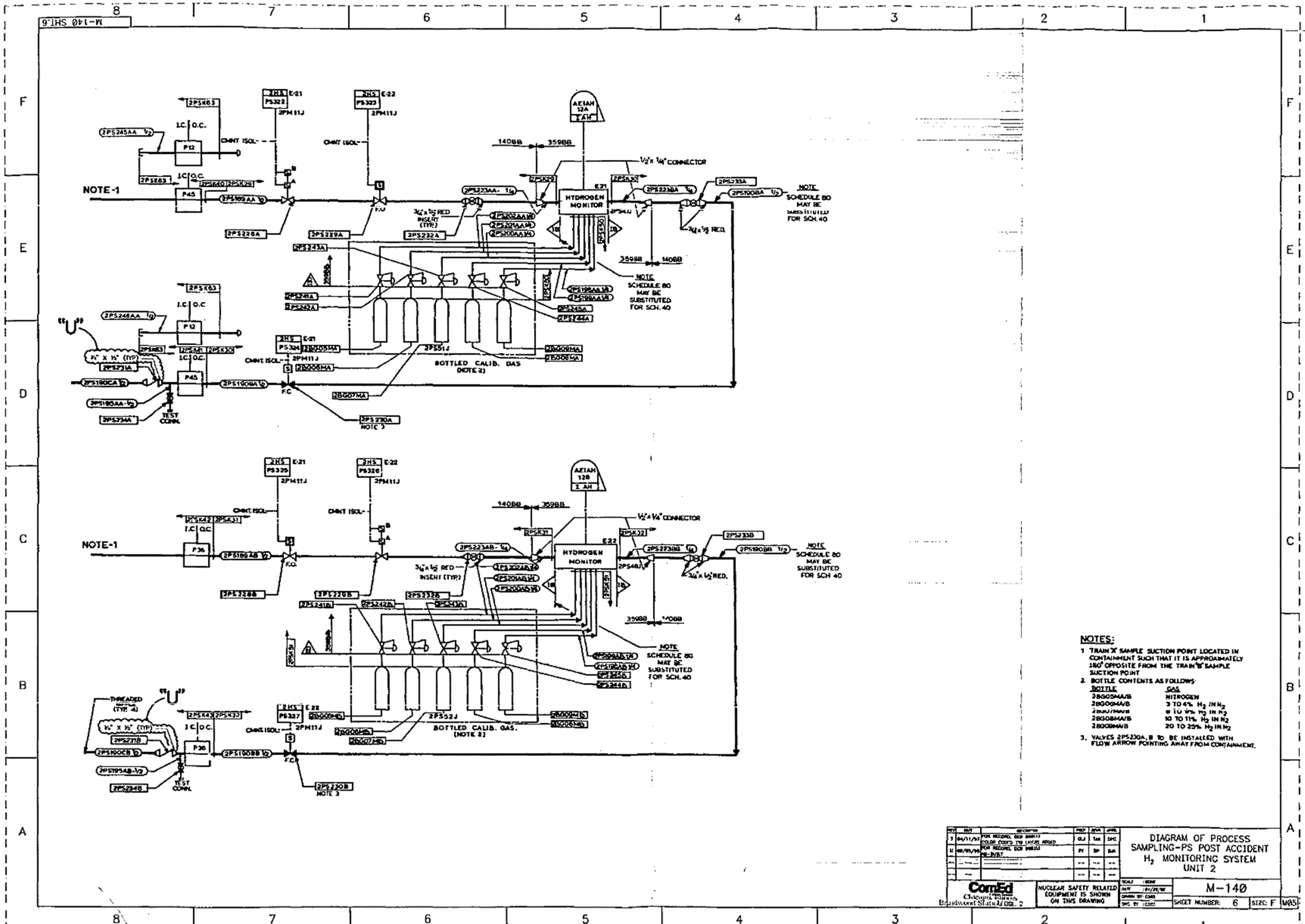
Piping & Instrument Diagrams: M-68, Sheet 7 (Unit 1) and M-140, Sheet 6 (Unit 2). For Information Only.

ATTACHMENT 2
10 CFR 50.55a Request No. I4R-02
Proposed Alternative In Accordance with 10 CFR 50.55a(z)(2)
--Hardship or Unusual Difficulty Without Compensating Increase
In Level of Quality or Safety--

Enclosure
(For Information Only)

Piping & Instrument Diagrams:
M-68, Sheet 7 (Unit 1)
and
M-140, Sheet 6 (Unit 2)


M-68 SH. 7
NOTE-1
HYDROGEN MONITOR
BOTTLED CALIB. GAS (NOTE 2)
NOTES:
NUCLEAR SAFETY RELATED ITEMS ARE SHOWN ON THIS DRAWING
DIAGRAM OF PROCESS SAMPLING PS POST ACCIDENT H2 MONITORING SYSTEM UNIT 1
M-68
SHEET NO. 7


NOTE-1
HYDROGEN MONITOR
BOTTLED CALIB. GAS (NOTE 2)
NOTES:
1 TRAIN 'A' SAMPLE SUCTION POINT LOCATED IN CONTAINMENT SUCH THAT IT IS APPROXIMATELY 180° OPPOSITE FROM THE TRAIN 'B' SAMPLE SUCTION POINT
2. BOTTLE CONTENTS AS FOLLOWS:
BOTTLE GAS
2BG05MA/B NITROGEN
2BG06MA/B 3 TO 4% H2 IN N2
2BG08MA/B 10 TO 11% H2 IN N2
2BG09MA/B 20 TO 25% H2 IN N2
3. VALVES 2PS230A, B TO BE INSTALLED WITH FLOW ARROW POINTING AWAY FROM CONTAINMENT.
DIAGRAM OF PROCESS SAMPLING-PS POST ACCIDENT H2 MONITORING SYSTEM UNIT 2
NUCLEAR SAFETY RELATED EQUIPMENT IS SHOWN ON THIS DRAWING
M-140
SHEET NUMBER: 6
SIZE: F

ATTACHMENT 3
10 CFR 50.55a Request No. I4R-03
Proposed Alternative In Accordance with 10 CFR 50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--

1. ASME Code Component(s) Affected

Code Class:	1
Reference:	IWA-4000
Examination Category:	NA
Item Number:	NA
Description:	Alternative Requirements for Repair/Replacement of Control Rod Drive Mechanism (CRDM) Canopy Seal Welds in Accordance with IWA-4000
Component Number:	Reactor CRDM Canopy Seal Welds - Class 1 Appurtenance to the Reactor Vessel

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Braidwood Station, Units 1 and 2, Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2013 Edition.

3. Applicable Code Requirement

The CRDM assemblies were designed and fabricated to the ASME Section III, 1974 Edition through Summer 1974 Addenda.

IWA-4000 of ASME Section XI requires that repairs be performed in accordance with the owner's original construction Code of the component or system, or later editions and addenda of the Code. The canopy seal weld is described in Section III, and a repair to this weld would require the following activities:

a. Excavation of the rejectable indications,
b. A surface examination of the excavated areas,
c. Re-welding and restoration to the original configuration and materials, and
d. Final surface examination.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(1), relief is requested on the basis that the proposed alternative provides an acceptable level of quality and safety.

The principal issues leading to this relief request are the excavation of indications contained within the existing weld, the accompanying dose received during the excavation and examination activities, and the weld material used for the repair or replacement.

Due to the nature of the flaw in the subject canopy seal weld, the excavation of the leaking portion of the weld would result in a cavity that extends completely through wall. A liquid penetrant (PT) examination of this cavity is required to verify the removal of the rejectable flaw or to verify that the flaw is removed or reduced to an acceptable size. This PT examination would deposit the penetrant materials onto the inner surfaces of the component. This material would not be readily removed prior to re-welding due to the

inaccessibility of the inside surface. The remaining penetrant material would introduce contaminants to the new weld metal and reduce the quality of the repair weld. The configuration of the canopy assembly would prevent the establishment and maintenance of an adequate back-purge during the welding process and would further reduce the quality of the repair weld.

The CRDM canopy seal welds are located above the Reactor Vessel Closure Head, which is highly congested and subject to high radiation levels. The high radiological dose associated with a CRDM canopy seal weld repair in strict compliance to these ASME Code requirements would be contrary to the intent of the as low as reasonably achievable (ALARA) radiological controls program. In order to reduce the exposure to personnel involved in the welding process, most of the repair activities would be performed remotely using robotic equipment to the extent practical. However, the required excavation and PT examinations would necessitate hands on access to the canopy weld. Based on expected radiation dose levels and time estimates to perform the excavation and PT examination for a single CRDM repair, the estimated total dose for these activities is estimated to be in excess of 0.600 person-Rem. This dose estimate is consistent with industry experience for similar activities.

IWA-4200 requires that the repair material conform to the original Design specification or ASME Section III. In this case, the replacement material would have the same resistance to stress corrosion cracking as the original material. Use of the original material does not guarantee that the repaired component will continue to maintain leakage integrity throughout the intended life of the item.

In lieu of performance of PT examinations of CRDM seal weld repairs or replacement, a 5X or better magnification visual examination will be performed after the welding is completed. In addition, Alloy 52/52M nickel-based weld repair material will be used rather than austenitic stainless steel.

Alloy 52/52M nickel-based weld repair material was selected for the repair because of its resistance to stress corrosion cracking. The suitability of the replacement material will be evaluated for each application and determined to be compatible with the existing component and will provide a leakage barrier for the remainder of the intended life of the CRDM.

The alternative method of repair is being requested to facilitate contingency repair efforts during future outages within the fourth ISI interval. The alternative nondestructive examination method is being requested to facilitate examination of a repair of a CRDM canopy seal weld during the fourth ISI interval.

Industry experience with failure analyses performed on leaking canopy seal welds removed from service at other plants has attributed the majority of the cases to transgranular stress corrosion cracking (SCC). The size of the opening where the leakage occurs has been extremely small, normally a few thousandths of an inch. The crack orientations vary, but often radiate outward such that a pinhole appears on the surface, as opposed to a long crack. The SCC results from exposure of a susceptible material to residual stress, which is often concentrated by weld discontinuities, and to a

corrosive environment, such as water trapped in the cavity behind the seal weld that is mixed with the air initially in the cavity, resulting in higher oxygen content than is in the bulk primary coolant.

5. Proposed Alternative and Basis for Use

The CRDM canopy seal weld flaws will not be removed, but an analysis of the repaired weldment will be performed, prior to entering Mode 4, to assure that the remaining flaw will not propagate unacceptably. The canopy seal weld is not a structural weld, nor a pressure-retaining weld, but provides a seal to prevent reactor coolant leakage if the mechanical joint leaks. The weld buildup is considered a repair in accordance with IWA-4110. Applicability of the original Code of construction or design specification is mandated because the weld is performed on an appurtenance to a pressure-retaining component. The alternative CRDM canopy seal weld repair uses a Gas Tungsten Arc Welding (GTAW) process controlled remotely. Should the need arise, a manual GTAW repair may be utilized.

A visual examination of the repaired/replaced weld will be performed using methods and personnel qualified to the standards of ASME Section XI VT-1 visual examination requirements. The VT-1 visual examination will be performed using a camera/viewing system with 5X or better magnification within several inches of the weld, qualified to ensure identification of significant flaws to assure an adequate margin of safety is maintained. The examination technique will be demonstrated to resolve a 0.001" thick wire against the surface of the weld. The repaired/replaced weld will be examined for quality of workmanship and discontinuities will be evaluated and dispositioned to ensure the adequacy of the new leakage barrier.

The automated GTAW weld repair and alternate VT-1 visual examination methods result in significantly lower radiation exposure because the equipment is remotely operated after setup. A post-maintenance VT-2 visual examination will be performed at normal operating temperature and pressure during the System Leakage Test.

Repair/Replacement activities, using the process described in this relief request, shall be documented on the required forms (i.e., NIS-2, "Form NIS-2 Owner's Report for Repair/Replacement Activity," or NIS-2A, "Form NIS-2A Repair/Replacement Certification Record.") This relief request will be identified on the NIS-2 / NIS-2A forms. The repair documents will be reviewed by the Authorized Nuclear Inspector, and maintained in accordance with the requirements for archiving permanent plant records.

6. Duration of Proposed Alternative

Relief is requested for the fourth ISI interval for Braidwood Station, Units 1 and 2.

7. Precedents

Braidwood Station, Units 1 and 2, third ISI interval Relief Request I3R-11 was authorized per Nuclear Regulatory Commission (NRC) SE dated April 28, 2014 (ADAMS Accession

No. ML14084A549). This relief request for the Braidwood Station, Units 1 and 2, fourth ISI interval, utilizes a similar approach to the previously approved relief request.

8. References

None.

ATTACHMENT 4
10 CFR 50.55a Request No. I4R-04
Proposed Alternative for Use of ASME Code Case N-513-4, Evaluation Criteria for Temporary Acceptance Flaws in Moderate Energy Class 2 or 3 Piping in Accordance with 10 CFR 50.55a(z)(2)
--Hardship or Unusual Difficulty without Compensating Increase in Level of Quality or Safety—

1. ASME Code Component(s) Affected

All American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV), Section XI, Class 2 and 3 components that meet the operational and configuration limitations of ASME Code Case N-513-4 (N-513-4), "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping, Section XI, Division 1," paragraphs 1(a), 1(b), 1(c), and 1(d).

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Braidwood Station, Units 1 and 2, Inservice Inspection (ISI) Program is based on the ASME B&PV Code, Section XI, 2013 Edition.

3. Applicable Code Requirements

IWC-3120 and IWD-3120 of ASME Section XI, require that flaws exceeding the defined acceptance criteria be corrected by repair/replacement activities or evaluated and accepted by analytical evaluation. IWC-3130 and IWD-3130 of ASME Section XI, requires that relevant conditions be subject to supplemental examination, corrective measures or repair/replacement activities, or evaluated and accepted by analytical evaluation.

4. Reason for the Request

In accordance with 10 CFR 50.55a(z)(2), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Exelon Generation Company, LLC (EGC) is requesting a proposed alternative from the requirement to perform repair/replacement activities for degraded Class 2 and 3 piping whose maximum operating temperature does not exceed 200°F and whose maximum operating pressure does not exceed 275 psig for Braidwood Station. Moderately degraded piping could require a plant shutdown within the required action statement timeframes to repair observed degradation. Plant shutdown activities result in additional dose and plant risk that would be inappropriate when a degraded condition is demonstrated to retain adequate margin to complete the component's function. The use of an acceptable alternative analysis method in lieu of immediate action for a degraded condition will allow EGC to perform additional extent of condition examinations on the affected systems while allowing time for safe and orderly long term repair actions if necessary. Actions to remove degraded piping from service could have a detrimental overall risk impact by requiring a plant shutdown, thus requiring use of a system that is in standby during normal operation. Accordingly, compliance with the current Code requirements results in a hardship without a compensating increase in the level of quality and safety.

ASME Code Case N-513-3 (N-513-3) does not allow evaluation of flaws located away from attaching circumferential piping welds that are in elbows, bent pipe, reducers, expanders, and branch tees. N-513-3 also does not allow evaluation of flaws located in heat exchanger external tubing or piping. N-513-4 provides guidance for evaluation of flaws in these locations.

5. Proposed Alternative and Basis for Use

EGC is requesting approval to apply the evaluation methods of N-513-4 to Class 2 and 3 components that meet the operational and configuration limitations of N-513-4, paragraphs 1(a), 1(b), 1(c), and 1(d) for Braidwood Station in order to avoid accruing additional personnel radiation exposure and increased plant risk associated with a plant shutdown to comply with the cited Code requirements.

The Nuclear Regulatory Commission (NRC) issued Generic Letter 90-05 (Reference 1), "Guidance for Performing Temporary Non-Code Repair of Class 1, 2, and 3 Piping (Generic Letter 90-05)," to address the acceptability of limited degradation in moderate energy piping. The generic letter defines conditions that would be acceptable to utilize temporary non-Code repairs with NRC approval. The ASME recognized that relatively small flaws could remain in service without risk to the structural integrity of a piping system and developed ASME Code Case N-513 (N-513). NRC approval of N-513 versions in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 18 (Reference 4), allows acceptance of partial through-wall or through-wall leaks for an operating cycle provided all conditions of the code case and NRC conditions are met. The code case also requires the Owner to demonstrate system operability due to leakage.

The ASME recognized that the limitations in N-513-3 were preventing needed use in piping components such as elbows, bent pipe, reducers, expanders, and branch tees and external tubing or piping attached to heat exchangers. N-513-4 was approved by the ASME to expand use on these locations and to revise several other areas of the code case. Attachment 2 of the Reference 2 letter provides a marked-up N-513-3 version of the code case to highlight the changes compared to the NRC approved N-513-3 version. Attachment 3 of the Reference 2 letter provides the ASME approved N-513-4. The following provides a high level overview of the N-513-4 changes:

1) Revised the maximum allowed time of use from no longer than 26 months to the next scheduled refueling outage.
2) Added applicability to piping elbows, bent pipe, reducers, expanders, and branch tees where the flaw is located more than $(R_o t)^{1/2}$ from the centerline of the attaching circumferential piping weld.
3) Expanded use to external tubing or piping attached to heat exchangers.
4) Revised to limit the use to liquid systems.
5) Revised to clarify treatment of Service Level load combinations.
6) Revised to address treatment of flaws in austenitic pipe flux welds.

7) Revised to require minimum wall thickness acceptance criteria to consider longitudinal stress in addition to hoop stress.
8) Other minor editorial changes to improve the clarity of the code case.

Detailed discussion of significant changes in N-513-4 when compared to NRC approved N-513-3 is provided in Attachment 4 of the Reference 2 letter.

The design basis is considered for each leak and evaluated using the EGC Operability Evaluation process. The evaluation process must consider requirements or commitments established for the system, continued degradation and potential consequences, operating experience, and engineering judgement. As required by the code case, the evaluation process considers but is not limited to system make-up capacity, containment integrity with the leak not isolated, effects on adjacent equipment, and the potential for room flooding.

Leakage rate is not typically a good indicator of overall structural stability in moderate energy systems, where the allowable through-wall flaw sizes are often on the order of inches. The periodic inspection interval defined using paragraph 2(e) of N-513-4 provides evidence that a leaking flaw continues to meet the flaw acceptance criteria and that the flaw growth rate is such that the flaw will not grow to an unacceptable size.

The effects of leakage may impact the operability determination or the plant flooding analyses specified in paragraph 1(f). For a leaking flaw, the allowable leakage rate will be determined by dividing the critical leakage rate by a safety factor of four (4). The critical leakage rate is determined as the lowest leakage rate that can be tolerated and may be based on the allowable loss of inventory or the maximum leakage than can be tolerated relative to room flooding, among others. The safety factor of four (4) on leakage is based upon ASME Code Case N-705 (N-705) (Reference 3), which is accepted without condition in Regulatory Guide 1.147, Revision 18. Paragraph 2.2(e) of N-705 requires a safety factor of two (2) on flaw size when estimating the flaw size from the leakage rate. This corresponds to a safety factor of four (4) on leakage for nonplanar flaws. Although the use of a safety factor for determination of an unknown flaw is considered conservative when the actual flaw size is known, this approach is deemed acceptable based upon the precedent of N-705. Note that the alternative herein does not propose to use any portion of N-705 and that citation of N-705 is intended only to provide technical basis for the safety factor on leakage.

During the temporary acceptance period, leaking flaws will be monitored daily as required by paragraph 2(f) of N-513-4 to confirm the analysis conditions used in the evaluation remain valid. Significant change in the leakage rate is reason to question that the analysis conditions remain valid, and would require re-inspection per paragraph 2(f) of the code case. Any re-inspection must be performed in accordance with paragraph 2(a) of the code case.

The leakage limit provides quantitative measurable limits which ensure the operability of the system and early identification of issues that could erode defense-in-depth and lead to adverse consequences.

In summary, EGC will apply N-513-4 to the evaluation of Class 2 and 3 components that are within the scope of the code case. N-513-4 utilizes technical evaluation approaches that are based on principals that are accepted in other Code documents already acceptable to the NRC. The application of this code case, in conjunction with safety factors on leakage limits, will maintain acceptable structural and leakage integrity while minimizing plant risk and personnel exposure by minimizing the number of plant transients that could be incurred if degradation is required to be repaired based on ASME Section XI acceptance criteria only.

6. Duration of Proposed Alternative

The proposed alternative is for use of N-513-4 for Class 2 and Class 3 components within the scope of the code case. An ASME Section XI compliant repair/replacement will be completed prior to exceeding the next refueling outage or allowable flaw size, whichever comes first. Relief is requested for the fourth ISI interval for Braidwood Station, Units 1 and 2, or such time as the NRC approves N-513-4, or a later revision, in Regulatory Guide 1.147 or other document. If a flaw is evaluated near the end of the interval for Braidwood Station and the next refueling outage is in the subsequent interval, the flaw may remain in service under this relief request until the next refueling outage.

7. Precedents

Braidwood Station, Units 1 and 2, third ISI interval relief request was authorized by NRC Safety Evaluation (SE) dated September 6, 2016 (Reference 5). This Braidwood Station relief request was part of an EGC fleet-wide submittal, and the use of N-513-4 was authorized for various stations. This relief request for the Braidwood Station, Units 1 and 2, fourth ISI interval, utilizes a similar approach to the previously approved relief request.

8. References

1) NRC Generic Letter 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping (Generic Letter 90-05)," dated June 15, 1990.

2) Letter from D. T. Gudger (Exelon Generation Company, LLC) to NRC, "Proposed Alternative to Utilize Code Case N-513-4, 'Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1,'" dated January 28, 2016.

3) ASME Boiler and Pressure Vessel Code, Code Case N-705, "Evaluation Criteria for Temporary Acceptance of Degradation in Moderate Energy Class 2 or 3 Vessels and Tanks Section XI, Division 1," dated October 12, 2006.

4) NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 18, dated March, 2017.

5) Letter from G. E. Miller (NRC) to B. C. Hanson (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2; Byron Station Unit Nos. 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Oyster Creek Nuclear Generating Station; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; R. E. Ginna Nuclear Power Plant; and Three Mile Island Nuclear Station, Unit 1 – Proposed Alternative to Use ASME Code Case N-513-4 (CAC Nos. MF7301-MF7322)," dated September 6, 2016 (ADAMS Accession No. ML16230A237).

ATTACHMENT 5
10 CFR 50.55a Request No. I4R-05
Proposed Alternative in Accordance With 10 CFR 50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--

1. ASME Code Component(s) Affected

Code Class:	1
Reference:	Table IWB-2500-1
Examination Category:	B-G-1
Item Number:	B6.40
Description:	Alternative for Examination of ASME Section XI, Examination Category B-G-1, Item Number B6.40, Threads in Flange
Component Number:	54 RPV threads in flange for Unit 1 (1RV-02-038-01 to 54) 53 RPV threads in flange for Unit 2 (2RV-02-38-01 to 54).

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Braidwood Station, Units 1 and 2, Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2013 Edition.

3. Applicable Code Requirements

The Reactor Pressure Vessel (RPV) threads in flange, Examination Category B-G-1, Item Number B6.40, are examined using a volumetric examination technique with 100% of the flange threaded stud holes examined every ISI interval. The examination area is the one-inch area around each RPV stud hole, as shown on Figure IWB-2500-12.

4. Reason for the Request

In accordance with 10 CFR 50.55a(z)(1), Exelon Generation Company, LLC (EGC) is requesting a proposed alternative from the requirement to perform inservice ultrasonic examinations of Examination Category B-G-1, Item Number B6.40, Threads in Flange for Braidwood Station. EGC has worked with the industry to evaluate eliminating the RPV threads in flange examination requirement. Licensees in the U.S. and internationally have worked with the Electric Power Research Institute (EPRI) to produce Technical Report No. 3002007626, "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements" (Reference 1), which provides the basis for elimination of the requirement. The report includes a survey of inspection results from over 168 units, a review of operating experience related to RPV flange/bolting, and a flaw tolerance evaluation. The conclusion from this evaluation is that the current requirements are not commensurate with the associated burden (worker exposure, personnel safety, radwaste, critical path time, and additional time at reduced water inventory) of the examination. The technical basis for this alternative is discussed in more detail below.

<u>Potential Degradation Mechanisms</u>

An evaluation of potential degradation mechanisms that could impact flange/threads reliability was performed as part of Reference 1. Potential types of degradation

evaluated included pitting, intergranular attack, corrosion fatigue, stress corrosion cracking, crevice corrosion, velocity phenomena, dealloying corrosion and general corrosion, stress relaxation, creep, mechanical wear, and mechanical/thermal fatigue. Other than the potential for mechanical/thermal fatigue, there are no active degradation mechanisms identified for the threads in flange component.

The EPRI report notes a general conclusion from Reference 2, (which includes work supported by the Nuclear Regulatory Commission (NRC)) that when a component item has no active degradation mechanism present, and a preservice inspection has confirmed that the inspection volume is in good condition (i.e., no flaws / indications), then subsequent inservice inspections do not provide additional value going forward. As discussed in the Operating Experience review summary below, the RPV flange ligaments have received the required preservice examinations and over 10,000 inservice inspections, with no relevant findings.

To address the potential for mechanical/thermal fatigue, Reference 1 documents a stress analysis and flaw tolerance evaluation of the flange thread area to assess mechanical/thermal fatigue potential. The evaluation consists of two parts. In the first part, a stress analysis is performed considering all applicable loads on the threads in flange component. In the second part, the stresses at the critical locations of the component are used in a fracture mechanics evaluation to determine the allowable flaw size for the component as well as how much time it will take for a postulated initial flaw to grow to the allowable flaw size using guidelines in ASME Section XI, IWB-3500. The Pressurized Water Reactor (PWR) design was selected because of its higher design pressure and temperature. A representative geometry for the finite element model used the largest PWR RPV diameter along with the largest bolts and the highest number of bolts. The larger and more numerous bolt configuration results in less flange material between bolt holes, whereas the larger RPV diameter results in higher pressure and thermal stresses.

<u>Stress Analysis</u>

A stress analysis was performed in Reference 1 to determine the stresses at critical regions of the thread in flange component as input to a flaw tolerance evaluation. Sixteen nuclear plant units (ten PWRs and six Boiling Water Reactors (BWRs)) were considered in the analysis. The evaluation was performed using a geometric configuration that bounds the sixteen units considered in this effort. The details of the RPV parameters for Braidwood Station as compared to the values used in the evaluation of the bounding preload stress are shown in Table 1. The preload stresses for both units are bounded by the Reference 1 report. Specifically, the Reference 1 preload stress is 42,338 psi, whereas the preload stresses are 33,323 psi at Braidwood Station, Units 1 and 2. The Braidwood Station stresses are bounded by the Reference 1 report which demonstrates that the report remains applicable to this relief request. Dimensions of the analyzed geometry are shown in Figure I4R-05-1.

Table 1: Braidwood Station Parameters Compared to Values Used in Bounding Analysis

Unit	No. of Studs Currently Installed	Minimum No. of Studs Evaluated	Stud Nominal Diameter (inches)	RPV Inside Diameter at Stud Hole (inches)	Flange Thickness at Stud Hole (inches)	Design Pressure (psig)	Preload Stress (psi)
1	54	53	6.75	171.06	16.97	2500	33,323
2	53	53	6.75	171.06	16.97	2500	33,323
Values Used in Bounding Analysis	**54**	**54**	**6.0**	**173**	**16**	**2500**	**42,338**

The analytical model is shown in Figures I4R-05-2 and I4R-05-3. The loads considered in the analysis consisted of:

- A design pressure of 2500 psia at an operating temperature of 600°F was applied to all internal surface exposed to internal pressure.
- Bolt/stud preload – The preload on the geometry is calculated as:

  $$P_{preload} = \frac{C \cdot P \cdot ID^2}{S \cdot D^2} = \frac{1.1 \cdot 2500 \cdot 173^2}{54 \cdot 6^2} = 42{,}338 \text{ psi}$$

  where:

$P_{preload}$	=	Preload pressure to be applied on modeled bolt (psi)
P	=	Internal pressure (psi)
ID	=	Largest inside diameter of RPV (in.)
C	=	Bolt-up contingencies (+10%)
S	=	Least number of studs
D	=	Smallest stud diameter (in.)

- Thermal stresses - The only significant transient affecting the bolting flange is heat-up/cooldown. This transient typically consists of a steady 100ºF/hour ramp up to the operating temperature, with a corresponding pressure ramp up to the operating pressure.

The ANSYS finite element analysis program was used to determine the stresses in the thread in flange component for the three loads described above.

<u>Flaw Tolerance Evaluation</u>

A flaw tolerance evaluation was performed using the results of the stress analysis to determine how long it would take an initial postulated flaw to reach the ASME Section XI allowable flaw size. A linear elastic fracture mechanics evaluation consistent with ASME Section XI, IWB-3600 was performed.

Stress intensity factors (K's) at four flaw depths of a 360° inside-surface-connected, partial-through-wall circumferential flaws are calculated using finite element analysis techniques with the model described above. The maximum stress intensity factor (K) values around the bolt hole circumference for each flaw depth (*a*) are extracted and used to perform the crack growth calculations. The circumferential flaw is modeled to start between the 10th and 11th flange threads from the top end of the flange because that is where the largest tensile axial stress occurs. The modeled flaw depth–to–wall thickness ratios (a/t) are 0.02, 0.29, 0.55, and 0.77, as measured in any direction from the stud hole. This creates an ellipsoidal flaw shape around the circumference of the flange, as shown in Figure I4R-05-4 for the flaw model with a/t = 0.77 a/t crack model. The crack tip mesh for the other flaw depths follows the same pattern. When preload is not being applied, the stud, stud threads, and flange threads are not modeled. The model is otherwise unchanged between load cases.

The maximum K results are summarized in Table 2 for the four crack depths. From Table 2, the maximum K occurs at operating conditions (preload + heatup + pressure). Because the crack tip varies in depth around the circumference, the maximum K from all locations at each crack size is conservatively used for the *K vs. a* profile.

Table 2: Maximum K vs. a/t

Load	K at Crack Depth (ksi√in)			
	0.02 a/t	0.29 a/t	0.55 a/t	0.77 a/t
Preload	11.2	17.4	15.5	13.9
Preload + Heatup + Pressure	13.0	19.8	16.1	16.3

The allowable stress intensity factor was determined based on the acceptance criteria in ASME Section XI, IWB-3610/Appendix A which states that:

$K_I < K_{Ic}/\sqrt{10} = 69.6$ ksi√in

Where,

K_I = Allowable stress intensity factor (ksi√in)
K_{Ic} = Lower bound fracture toughness at operating temperature (220 ksi√in)

As can be seen from Table 2, the allowable stress intensity factor is not exceeded for all crack depths up to the deepest analyzed flaw of a/t = 0.77. Hence the allowable flaw depth of the 360° circumferential flaw is at least 77% of the thickness of the flange. The allowable flaw depth is assumed to be equal to the deepest modeled crack for the purposes of this analysis.

For Braidwood Station Unit 2, the RPV closure head in service currently has one missing stud. The thread in flange configuration has much redundancy. As seen from the stress intensity factor (K) calculation documented in Table 6-1 of Reference 1 (reproduced in Table 2 above), the maximum K is 19.8 ksi√in. The allowable K calculated in Section 6.2.2 of the report is 69.6 ksi√in, significantly higher than the calculated value. Since the Unit 2 RPV flange has 53 studs and one missing stud, the increase in K is about 1.9%

resulting in a maximum K of about 20.17 ksi√in which is still significantly less than the allowable value.

For the crack growth evaluation, an initial postulated flaw size of 0.2 in. (5.08 mm) is chosen consistent with the ASME Section XI, IWB-3500 flaw acceptance standards. The deepest flaw analyzed is a/t = 0.77 because of the inherent limits of the model. Two load cases are considered for fatigue crack growth: heat-up/cooldown and bolt preload. The heat-up/cooldown load case includes the stresses due to thermal and internal pressure loads and is conservatively assumed to occur 50 times per year. The bolt preload is assumed to be present and constant during the load cycling of the heat-up/cooldown load case. The bolt preload load case is conservatively assumed to occur five times per year, and these cycles do not include thermal or internal pressure. The resulting crack growth was determined to be negligible due to the small delta K and the relatively low number of cycles associated with the transients evaluated. Because the crack growth is insignificant, the allowable flaw size will not be reached and the integrity of the component is not challenged for at least 80 years (original 40-year design life plus additional 40 years of plant life extension).

An evaluation was also performed to determine the acceptability at preload condition. Table 3 below provides the RPV flange RT_{NDT} values and the bolt-up temperatures for Braidwood Station, Units 1 and 2. As this table shows, the information was obtained from plant records as well as the NRC RVID2 database.

The values of ($T\text{-}RT_{NDT}$) for the RPV flanges for Braidwood Station, Units 1 and 2 are shown in Table 3. These were determined using the RT_{NDT} value from plant records. As can be seen from this table, the minimum ($T\text{-}RT_{NDT}$) is 70˚F and 40˚F, corresponding to Braidwood Station, Units 1 and 2, respectively. From the equations in paragraph A–4200 of ASME Section XI, Appendix A, the corresponding values of K_{Ic} are 117 and 79 ksi√in. Using a structural factor of √10, the allowable K_{Ic} value is 37.0 and 25.1 ksi√in. This value is more than the maximum stress intensity factor (K_I) for the preload condition of 17.4 ksi√in shown in Table 2, thus the report evaluation is bounding for Braidwood Station, Units 1 and 2.

Table 3: RPV Flange RT_{NDT} and Bolt-Up Temperature

Plant Name	Flange RT_{NDT} (°F)		Preload Temp (°F)	Minimum $T\text{-}RT_{NDT}$ (°F)
	From Plant Records	From NRC RVID2 Database		
Braidwood, Unit 1	-10	-10	≥60	70
Braidwood, Unit 2	20	20	≥60	40

The stress analysis / flaw tolerance evaluation presented above shows that the thread in flange component is very flaw tolerant and can operate for 80 years without violating ASME Section XI safety margins. This clearly demonstrates that the thread in flange examinations can be eliminated without affecting the safety of the RPV.

Operating Experience Review Summary

As discussed above, the results of the survey, which includes results from Braidwood Station, confirmed that the RPV threads in flange examination are adversely impacting outage activities (worker exposure, personnel safety, radwaste, critical path time, and additional time at reduced water inventory) while not identifying any service induced degradations. Specifically, for the U.S. fleet, a total of 94 units have responded to date and none of these units have identified any type of degradation. As can be seen in Table 4 below, the data is encompassing. The 94 units represent data from 33 BWRs and 61 PWRs. For the BWR units, a total 3,793 examinations were conducted and for the PWR units a total of 6,869 examinations were conducted, with no service-induced degradation identified. The response data includes information from all of the plant designs in operation in the U.S. and includes BWR-2, -3, -4, -5, and -6 designs. The PWR plants include the 2-loop, 3-loop, and 4-loop designs and each of the PWR NSSS designs (i.e., Babcock & Wilcox, Combustion Engineering, and Westinghouse).

Table 4: Summary of Survey Results – U.S. Fleet

Plant Type	Number of Units	Number of Examinations	Number of Reportable Indications
BWR	33	3,793	0
PWR	61	6,869	0
Total	94	10,662	0

<u>Related RPV Assessments</u>

In addition to the examination history and flaw tolerance discussed above, Reference 1 discusses studies conducted in response to the issuance of the Anticipated Transient Without Scram (ATWS) Rule by the NRC. This rule was issued to require design changes to reduce expected ATWS frequency and consequences. Many studies have been conducted to understand the ATWS phenomena and key contributors to successful response to an ATWS event. In particular, the reactor coolant system (RCS) and its individual components were reviewed to determine weak links. As an example, even though significant structural margin was identified in NRC SECY-83-293 for PWRs, the ASME Service Level C pressure of 3200 psig was assumed to be an unacceptable plant condition. While a higher ASME service level might be defensible for major RCS components, other portions of the RCS could deform to the point of inoperability. Additionally, there was the concern that steam generator tubes might fail before other RCS components, with a resultant bypass of containment. The key take-away for these studies is that the RPV flange ligament was not identified as a weak link and other RCS components were significantly more limiting. Thus, there is substantial structural margin associated with the RPV flange.

In summary, Reference 1 identifies that the RPV threads in flange are performing with very high reliability based on operating and examination experience. This is due to the

robust design and a relatively benign operating environment (e.g., the number and magnitude of transients is small, generally not in contact with primary water at plant operating temperatures/pressures, etc.) The robust design is manifested in that plant operation has been allowed at several plants even with a bolt/stud assumed to be out of service. As such, significant degradation of multiple bolts/threads would be needed prior to any RCS leakage.

5. Proposed Alternative and Basis for Use

In lieu of the inservice requirements for a volumetric ultrasonic examination, Braidwood Station proposes that the industry report (Reference 1) provides an acceptable technical basis for eliminating the requirement for this examination because the alternative maintains an acceptable level of quality and safety.

This report provides the basis for the elimination of the RPV threads in flange examination requirement (ASME Section XI Examination Category B-G-1, Item Number B6.40). This report was developed because evidence had suggested that there have been no occurrences of service-induced degradation and there are negative impacts on worker dose, personnel safety, radwaste, critical path time for these examinations, and additional time at reduced water inventory.

Since there is reasonable assurance that the proposed alternative is an acceptable alternate approach to the performance of the ultrasonic examinations, Braidwood Station requests authorization to use the proposed alternative pursuant to 10 CFR 50.55a(z)(1) on the basis that use of the alternative provides an acceptable level of quality and safety.

To protect against non-service related degradation, Braidwood Station uses detailed procedures for the care and visual inspection of the RPV studs and the threads in flange each time the RPV closure head is removed. Care is taken to inspect the RPV threads for damage and to protect threads from damage when the studs are removed. Prior to reinstallation, the studs and stud holes are cleaned and lubricated. The studs are then replaced and tensioned into the RPV flange. This activity is performed each time the closure head is removed, and the procedure documents each step. These controlled maintenance activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service.

The requirements in this relief request are based upon ASME Section XI Code Case N-864 (N-864) (Reference 5) and will apply to Examination Category B-G-1, Item Number B6.40, Reactor Vessel Threads in Flange. N-864 was approved by ASME Board on Nuclear Codes and Standards on July 28, 2017; however, it has not been incorporated into NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and thus, is not available for application at nuclear power plants without specific NRC approval.

6. Duration of Proposed Alternative

Relief is requested for the fourth ISI interval for Braidwood Station, Units 1 and 2, or until the NRC approves N-864, or a later revision, in Regulatory Guide 1.147 or other document during the interval.

7. Precedents

Braidwood Station, Units 1 and 2, third ISI interval relief request was authorized by NRC Safety Evaluation (SE) dated June 26, 2017 (Reference 3). This Braidwood Station relief request was part of an EGC fleet-wide submittal, and the alternative for examination of ASME Section XI, Examination Category B-G-1, Item Number B6.40, threads in flange was authorized for various stations. This relief request for the Braidwood Station, Units 1 and 2, fourth ISI interval, utilizes a similar approach to the previously approved relief request.

Relief request was authorized for Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1 by NRC SE dated January 26, 2017 (Reference 4) (ADAMS Accession No. ML17006A109).

8. References

1) Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements. EPRI, Palo Alto, CA: 2016. 3002007626 (ADAMS Accession No. ML16221A068).

2) American Society of Mechanical Engineers, Risk-Based Inspection: Development of Guidelines, Volume 2-Part 1 and Volume 2-Part 2, Light Water Reactor (LWR) Nuclear Power Plant Components. CRTD-Vols. 20-2 and 20-4, ASME Research Task Force on Risk-Based Inspection Guidelines, Washington, D.C., 1992 and 1998.

3) Letter from D. J. Wrona (NRC) to B. C. Hanson (EGC) regarding "Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; R. E. Ginna Nuclear Power Plant; and Three Mile Island Nuclear Station, Unit 1 – Proposed Alternative to Eliminate Examination of Threads in Reactor Pressure Vessel Flange (CAC Nos. MF8712-MF8729 and MF9548)," dated June 26, 2017 (ADAMS Accession No. ML17170A013).

4) Letter from M. T. Markley (NRC) to C. R. Pierce (Southern Nuclear Operating Co. Inc.) regarding "Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1 -Alternative to Inservice Inspection Regarding Reactor Pressure Vessel Threads Inflange Inspection (CAC Nos. MF8061, MF8062, MF8070)," dated January 26, 2017 (ADAMS Accession No. ML17006A109).

5) ASME Section XI Code Case N-864, "Reactor Vessel Threads in Flange Examination," Section XI, Division 1. ASME Approval Date: July 28, 2017.

Figure I4R-05-1
Modeled Dimensions

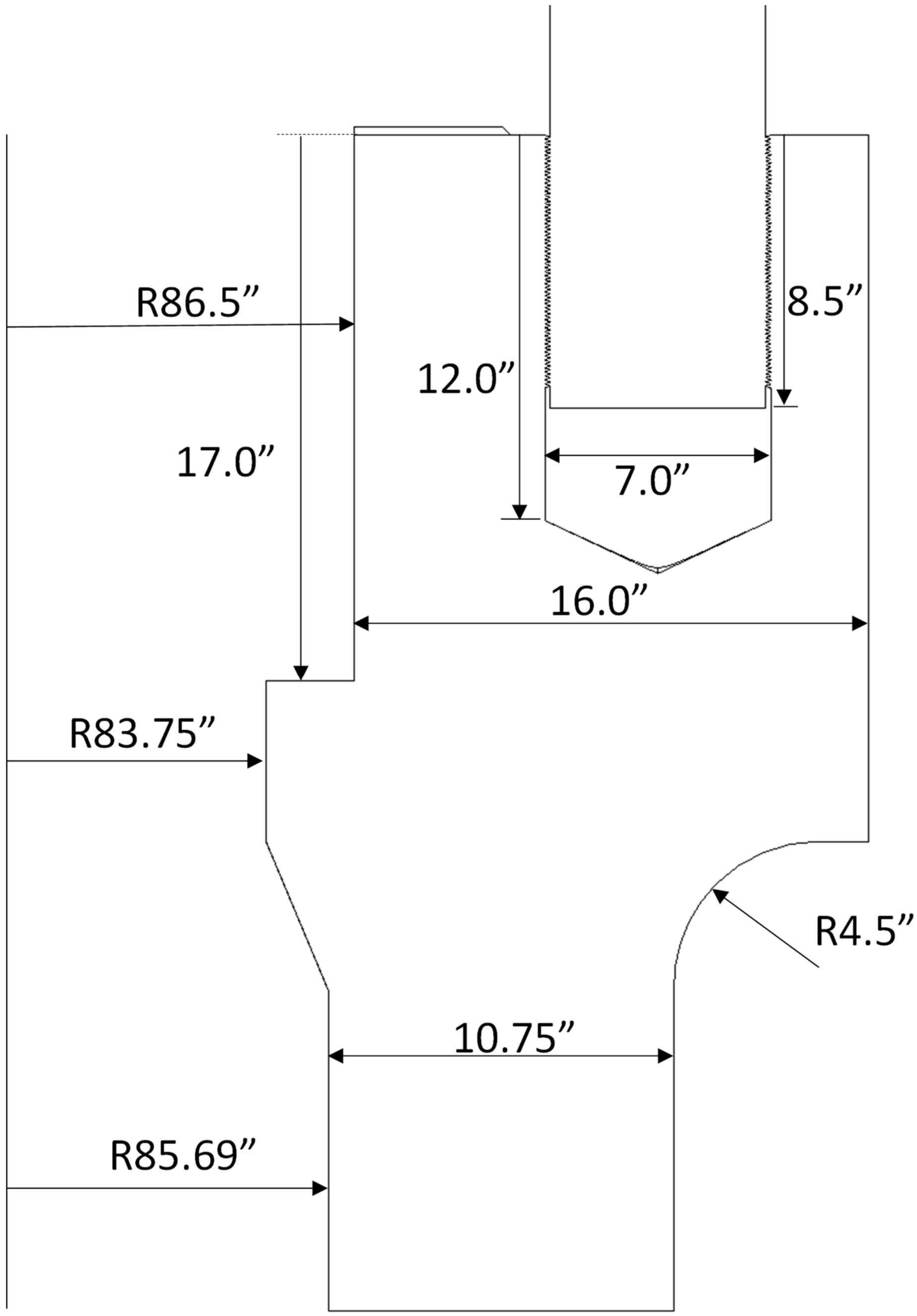


Figure I4R-05-2
Finite Element Model Showing Bolt and Flange Connection

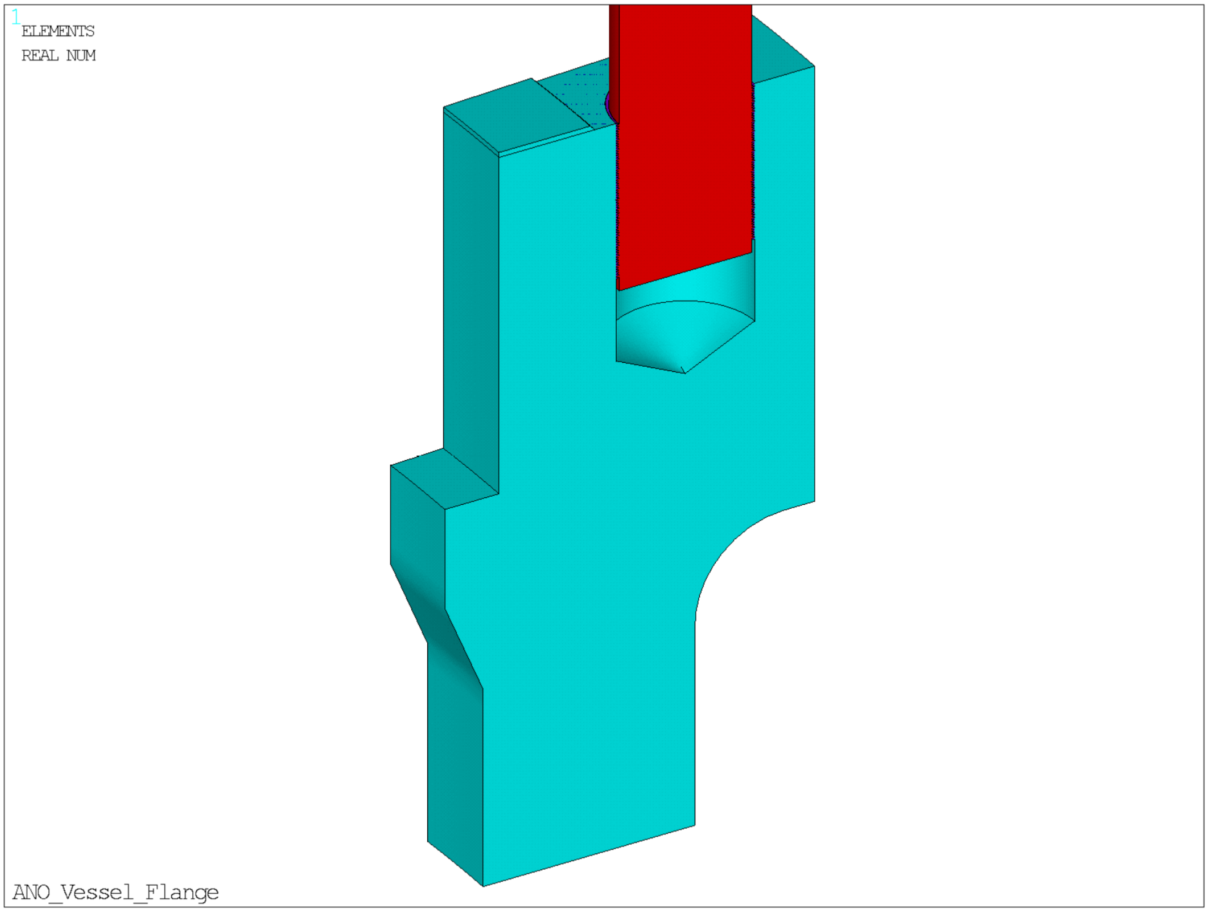


Figure I4R-05-3
Finite Element Model Mesh with Detail at Thread Location

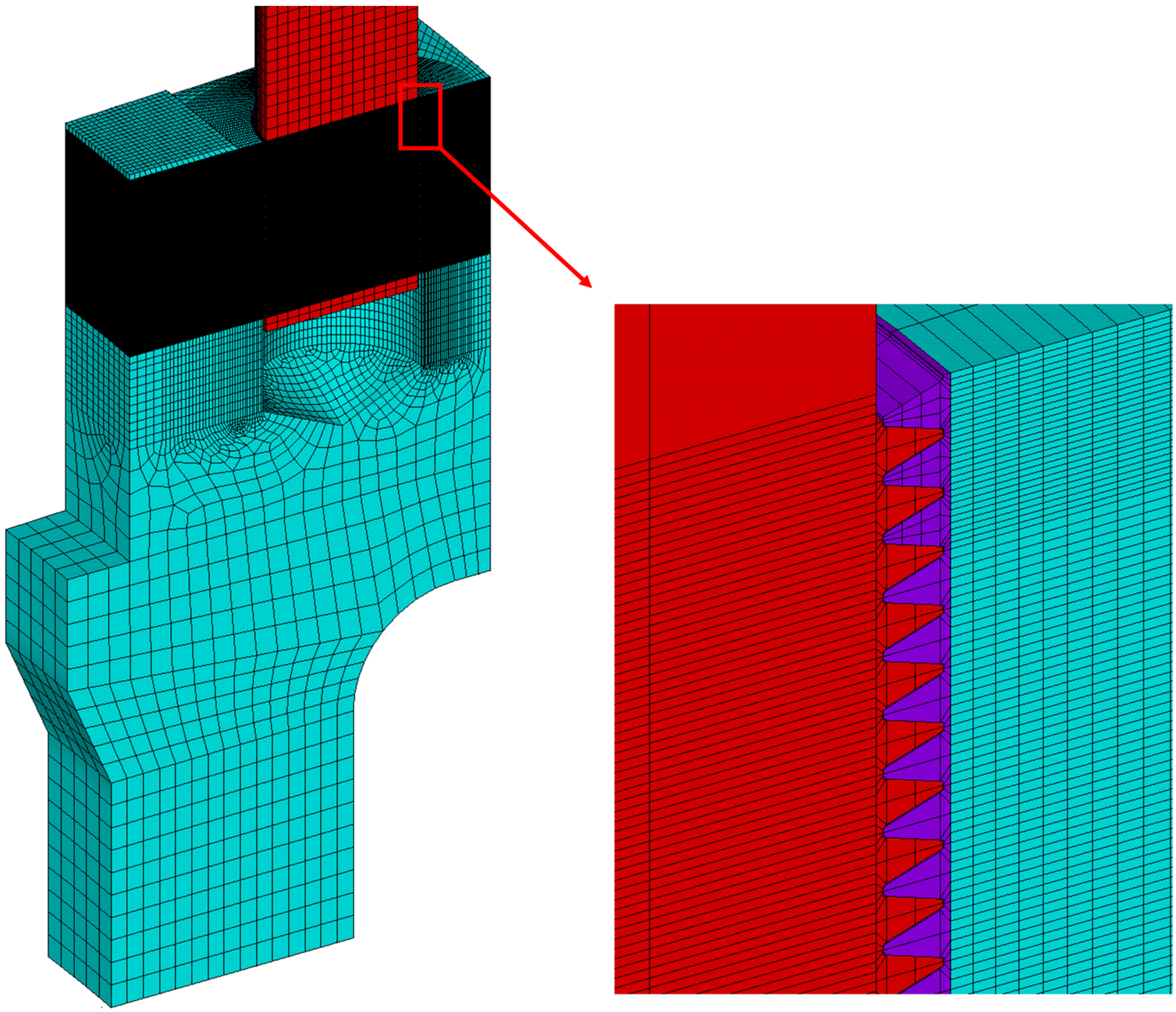
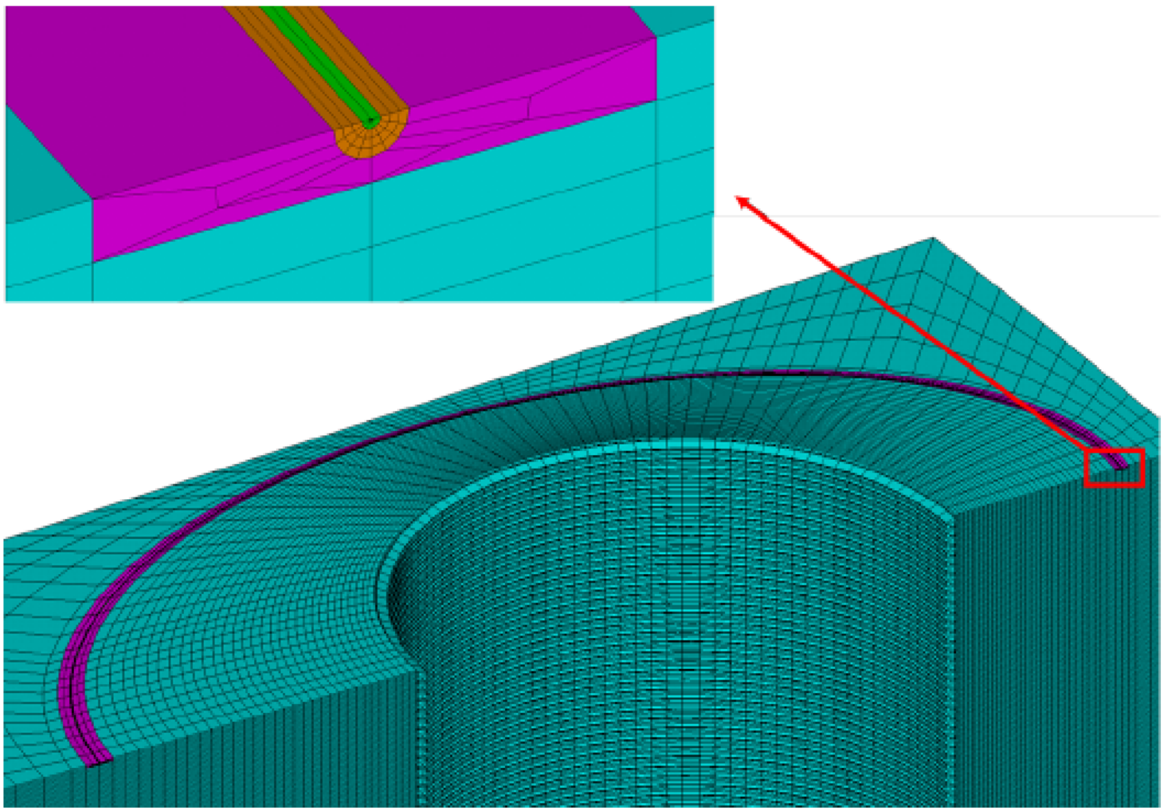


Figure I4R-05-4
Cross Section of Circumferential Flaw with Crack Tip Elements Inserted After 10th Thread from Top of Flange



1. ASME Code Component(s) Affected

All American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel (B&PV) Code, Section XI, ISI ferritic piping butt welds requiring radiography during repair/replacement activities.

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Braidwood Station, Units 1 and 2, Inservice Inspection (ISI) Program is based on the ASME B&PV Code, Section XI, 2013 Edition.

3. Applicable Code Requirements

10 CFR 50.55a(b)(2)(xx)(B) requires that "The NDE provision in IWA-4540(a)(2) of the 2002 Addenda of ASME Section XI must be applied when performing system leakage tests after repair and replacement activities performed by welding or brazing on a pressure retaining boundary using the 2003 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section." IWA-4540(a)(2) of the 2002 Addenda of ASME Section XI requires that the nondestructive examination method and acceptance criteria of the 1992 Edition or later of ASME Section III be met prior to return to service in order to perform a system leakage test in lieu of a system hydrostatic test. The examination requirements for ASME Section III, circumferential butt welds are contained in ASME Section III, Subarticles NB-5200, NC-5200, and ND-5200. The acceptance standards for radiographic examination are specified in ASME Section III, Subarticles NB-5300, NC-5300, and ND-5300.

IWA-4221 requires that items used for repair/replacement activities meet the applicable Owner's Requirements and Construction Code requirements when performing repair/replacement activities. IWA-4520 requires that welded joints made for installation of items be examined in accordance with the Construction Code identified in the Repair/Replacement Plan.

4. Reason for the Request

In accordance with 10 CFR 50.55a(z)(1), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

Replacement of piping is periodically performed in support of the Flow Accelerated Corrosion (FAC) program as well as other repair and replacement activities. The use of encoded Phased Array Ultrasonic Examination Techniques (PAUT) in lieu of radiography (RT) to perform the required examinations of the replaced welds would eliminate the safety risk associated with performing RT, which includes the planned exposure and the potential for accidental personnel exposure. PAUT minimizes the impact on other outage activities normally involved with performing RT such as limited access to work locations and the need to control system fill status because RT would require a line to remain fluid empty in order to obtain adequate examination sensitivity and resolution. In addition, encoded PAUT has been demonstrated to be adequate for detecting and sizing critical flaws.

Exelon Generation Company, LLC (EGC) requests approval of this proposed alternative to support anticipated piping repair and replacement activities for Braidwood Station during the Fourth ISI Interval.

5. <u>Proposed Alternative and Basis for Use</u>

Braidwood Station is proposing the use of encoded PAUT in lieu of the Code-required RT examinations for ASME ferritic piping repair/replacement welds. Similar techniques are being used throughout the nuclear industry for examination of dissimilar metal welds, and overlaid welds, as well as other applications including ASME B31.1 piping replacements. This proposed alternative request includes requirements that provide an acceptable level of quality and safety that satisfy the requirements of 10 CFR 50.55a(z)(1). The capability of the alternative technique is comparable to the examination methods documented in ASME Sections III, VIII, and IX, and associated code cases (References 1, 3, 5, 6, 7, 8, 9, 10, 11, and 12) related to using ultrasonic examination techniques for weld acceptance. The examinations will be performed using personnel and procedures qualified with the requirements of Section 5.1 below.

The electronic data files for the PAUT examinations will be stored as part of the archival-quality records. In addition, hard copy prints of the data will also be included as part of the PAUT examination records to allow viewing without the use of hardware or software.

5.1 Proposed Alternative

Braidwood Station is proposing to perform encoded PAUT examination techniques using demonstrated procedures, equipment, and personnel in accordance with the process documented below:

(1) The welds to be examined shall meet the surface conditioning requirements of the demonstrated ultrasonic procedure.

(2) The welds to be examined shall be conditioned such that transducers properly couple with the scanning surface with no more than a 1/32 in. (0.8 mm) gap between the search unit and the scanning surface.

(3) The ultrasonic examination shall be performed with equipment, procedures, and personnel qualified by performance demonstration.

(4) The examination volume shall include essentially 100% of the weld volume and the weld-to-base-metal interface.
   (a) Angle beam examination of the complete examination volume for fabrication flaws oriented parallel to the weld joint shall be performed.
   (b) Angle beam examination for fabrication flaws oriented transverse to the weld joint shall be performed to the extent practical. Scan restrictions that limit complete coverage shall be documented.
   (c) A supplemental straight beam examination shall be performed on the volume of base metal through which the angle beams will travel to locate any

reflectors that can limit the ability of the angle beam to examine the weld. Detected reflectors that may limit the angle beam examination shall be recorded and evaluated for impact on examination coverage. The straight beam examination procedure, or portion of the procedure, is required to be qualified in accordance with ASME Section V, Article 4 and may be performed using non-encoded techniques.

(5) All detected flaw indications from (4)(a) and (4)(b) above shall be considered planar flaws and compared to the preservice acceptance standards for volumetric examination in accordance with IWB-3000, IWC-3000, or IWD-3000. Preservice acceptance standards shall be applied. Analytical evaluation for acceptance of flaws in accordance with IWB-3600, IWC-3600 or IWD-3600 is permitted for flaws that exceed the applicable acceptance standards and are confirmed by surface or volumetric examination to be non-surface connected.

(6) Flaws exceeding the applicable acceptance standards and when analytical evaluation has not been performed for acceptance, shall be reduced to an acceptable size or removed and repaired, and the location of the repair shall be reexamined using the same ultrasonic examination procedure that detected the flaw.

(7) The ultrasonic examination shall be performed using encoded UT technology that produces an electronic record of the ultrasonic responses indexed to the probe position, permitting off-line analysis of images built from the combined data.
    (a) Where component configuration does not allow for effective examination for transverse flaws, (e.g., pipe-to-valve, tapered weld transition, weld shrinkage, etc.) the use of non-encoded UT technology may be used for transverse flaws. The basis for the non-encoded examination shall be documented.

(8) A written ultrasonic examination procedure qualified by performance demonstration shall be used. The qualification shall be applicable to the scope of the procedure, e.g., flaw detection and/or sizing (length or through-wall height), encoded or non-encoded, single and/or dual side access, etc. The procedure shall:
    (a) contain a statement of scope that specifically defines the limits of procedure applicability (e.g., minimum and maximum thickness, minimum and maximum diameter, scanning access);
    (b) specify which parameters are considered essential variables, and a single value, a range of values or criteria for selecting each of the essential variables;
    (c) list the examination equipment, including manufacturer and model or series;
    (d) define the scanning requirements; such as beam angles, scan patterns, beam direction, maximum scan speed, extent of scanning, and access;
    (e) contain a description of the calibration method (i.e., actions required to ensure that the sensitivity and accuracy of the signal amplitude and time outputs of the examination system, whether displayed, recorded, or automatically processed, are repeated from examination to examination);
    (f) describe the method and criteria for discrimination of indications (e.g., geometric indications versus indications of flaws and surface versus subsurface indications); and

(g) describe the surface preparation requirements.

(9) Performance demonstration specimens shall conform to the following requirements:
- (a) The specimens shall be fabricated from ferritic material with the same inside surface cladding process, if applicable, with the following exceptions:
  - (i) Demonstration with shielded metal arc weld (SMAW) single-wire cladding is transferable to multiple-wire or strip-clad processes;
  - (ii) Demonstration with multiple-wire or strip-clad process is considered equivalent but is not transferable to SMAW type cladding processes.
- (b) The demonstration specimens shall contain a weld representative of the joint to be ultrasonically examined, including the same welding processes.
- (c) The demonstration set shall include specimens not thicker than 0.1 in. (2.5 mm) more than the minimum thickness, nor thinner than 0.5 in. (13 mm) less than the maximum thickness for which the examination procedure is applicable. The demonstration set shall include the minimum, within ½ inch of the nominal pipe size (NPS), and maximum pipe diameters for which the examination procedure is applicable. If the procedure is applicable to outside diameter (O.D.) piping of 24 in. (600 mm) or larger, the specimen set must include at least one specimen 24 in. O.D. (600 mm) or larger but need not include the maximum diameter.
- (d) The demonstration specimen scanning and weld surfaces shall be representative of the surfaces to be examined.
- (e) The demonstration specimen set shall include geometric conditions that require discrimination from flaws (e.g., counterbore, weld root conditions, or weld crowns) and limited scanning surface conditions for single-side access, when applicable.
- (f) The demonstration specimens shall include both planar and volumetric fabrication flaws (e.g., lack of fusion, crack, incomplete penetration, slag inclusions) representative of the welding process or processes of the welds to be examined. The flaws shall be distributed throughout the examination volume.
- (g) Specimens shall be divided into flawed and unflawed grading units.
  - (i) Flawed grading units shall be the actual flaw length, plus a minimum of 0.25 in. (6 mm) on each end of the flaw. Unflawed grading units shall be at least 1 in. (25 mm).
  - (ii) The number of unflawed grading units shall be at least 1-1/2 times the number of flawed grading units.
- (h) Demonstration specimen set flaw distribution shall be as follows:
  - (i) For thickness greater than 0.50 in. (13 mm); at least 20% of the flaws shall be distributed in the outer third of the specimen wall thickness, at least 20% of the flaws shall be distributed in the middle third of the specimen wall thickness and at least 40% of the flaws shall be distributed in the inner third of the specimen wall thickness. For thickness 0.50 in. (13mm) and less, at least 20% of the flaws shall be distributed in the outer half of the specimen wall thickness and at least 40% of the flaws shall be distributed in the inner half of the specimen wall thickness.

(ii) At least 30% of the flaws shall be classified as surface planar flaws in accordance with IWA-3310. At least 40% of the flaws shall be classified as subsurface planar flaws in accordance with IWA-3320.

(iii) At least 50% of the flaws shall be planar flaws, such as lack of fusion, incomplete penetration, or cracks. At least 20% of the flaws shall be volumetric flaws, such as slag inclusions.

(iv) The flaw through-wall heights shall be based on the applicable acceptance standards for volumetric examination in accordance with IWB-3400, IWC-3400, or IWD-3400. At least 30% of the flaws shall be classified as acceptable planar flaws, with the smallest flaws being at least 50% of the maximum allowable size based on the applicable a/l aspect ratio for the flaw. Additional smaller flaws may be included in the specimens to assist in establishing a detection threshold, but shall not be counted as a missed detection if not detected. At least 30% of the flaws shall be classified as unacceptable in accordance with the applicable acceptance standards. Welding fabrication flaws are typically confined to a height of a single weld pass. Flaw through-wall height distribution shall range from approximately one to four weld pass thicknesses, based on the welding process used.

(v) If applicable, at least two flaws, but no more than 30% of the flaws, shall be oriented perpendicular to the weld fusion line and the remaining flaws shall be circumferentially oriented.

(vi) For demonstration of single-side-access capabilities, at least 30% of the flaws shall be located on the far side of the weld centerline and at least 30% of the planar flaws shall be located on the near side of the weld centerline. The remaining flaws shall be distributed on either side of the weld.

(10) Ultrasonic examination procedures shall be qualified by performance demonstration in accordance with the following requirements.

(a) The procedure shall be demonstrated using either a blind or a non-blind demonstration.

(b) The non-blind performance demonstration is used to assist in optimizing the examination procedure. When applying the non-blind performance demonstration process, personnel have access to limited knowledge of specimen flaw information during the demonstration process. The non-blind performance demonstration process consists of an initial demonstration without any flaw information, an assessment of the results and feedback on the performance provided to the qualifying candidate. After an assessment of the initial demonstration results, limited flaw information may be shared with the candidate as part of the feedback process to assist in enhancing the examination procedure to improve the procedure performance. In order to maintain the integrity of the specimens for blind personnel demonstrations, only generalities of the flaw information may be provided to the candidate. Procedure modifications or enhancements made to the procedure, based on the feedback process, shall be applied to all applicable specimens based on the scope of the changes.

(c) Objective evidence of a flaw's detection, length, and through-wall height sizing, in accordance with the procedure requirements, shall be provided to the organization administering the performance demonstration.
(d) The procedure demonstration specimen set shall be representative of the procedure scope and limitations (e.g., thickness range, diameter range, material, access, surface condition).
(e) The demonstration set shall include specimens to represent the minimum and maximum diameter and thickness covered by the procedure. If the procedure spans a range of diameters and thicknesses, additional specimens shall be included in the set to demonstrate the effectiveness of the procedure throughout the entire range.
(f) The procedure demonstration specimen set shall include at least 30 flaws and shall meet the requirements of (9) above.
(g) Procedure performance demonstration acceptance criteria
    (i) To be qualified for flaw detection, all flaws in the demonstration set that are not less than 50% of the maximum allowable size, based on the applicable *a/l* aspect ratio for the flaw, shall be detected. In addition, when performing blind procedure demonstrations, no more than 20% of the non-flawed grading units may contain a false call. Any non-flaw condition (e.g., geometry) reported as a flaw shall be considered a false call.
    (ii) To be qualified for flaw length sizing, the root mean square (RMS) error of the flaw lengths estimated by ultrasonics, as compared with the true lengths, shall not exceed 0.25 in. (6 mm) for diameters of NPS 6.0 in. (DN150) and smaller, and 0.75 in. (18 mm) for diameters greater than NPS 6.0 in. (DN150).
    (iii) To be qualified for flaw through-wall height sizing, the RMS error of the flaw through-wall heights estimated by ultrasonics, as compared with the true through-wall heights, shall not exceed 0.125 in. (3 mm).
    (iv) RMS error shall be calculated as follows:

$$RMS = \left[\frac{\sum_{n}^{i=1} \left(m_i - t_i \right)^2}{n} \right]^{1/2}$$

where:
m_i = measured flaw size
n = number of flaws measured
t_i = true flaw size
(h) Essential variables may be changed during successive personnel performance demonstrations. Each examiner need not demonstrate qualification over the entire range of every essential variable.

(11) Ultrasonic examination personnel shall be qualified in accordance with IWA-2300. In addition, examination personnel shall demonstrate their capability to detect and

size flaws by performance demonstration using the qualified procedure in accordance with the following requirements:

(a) The personnel performance demonstration shall be conducted in a blind fashion (flaw information is not provided).

(i) The demonstration specimen set shall contain at least 10 flaws and shall meet the flaw distribution requirements of (9)(h) above, with the exception of (9)(h)(v). When applicable, at least one flaw, but no more than 20% of the flaws, shall be oriented perpendicular to the weld fusion line and the remaining flaws shall be circumferentially oriented.

(b) Personnel performance demonstration acceptance criteria:

(i) To be qualified for flaw detection, personnel performance demonstration shall meet the requirements of the following table for both detection and false calls. Any non-flaw condition (e.g., geometry) reported as a flaw shall be considered a false call.

Performance Demonstration Detection Test Acceptance Criteria			
Detection Test Acceptance Criteria		**False Call Test Acceptance Criteria**	
No. of Flawed Grading Units	**Minimum Detection Criteria**	**No. of Unflawed Grading Units**	**Maximum Number of False Calls**
10	8	15	2
11	9	17	3
12	9	18	3
13	10	20	3
14	10	21	3
15	11	23	3
16	12	24	4
17	12	26	4
18	13	27	4
19	13	29	4
20	14	30	5
Note 1: Flaws $\geq$ 50% of the maximum allowable size, based on the applicable *a/ℓ* aspect ratio for the flaw.			

(ii) To be qualified for flaw length sizing, the RMS error of the flaw lengths estimated by ultrasonics, as compared with the true lengths, shall not exceed 0.25 in. (6 mm) for NPS 6.0 in. (DN150) and smaller, and 0.75 in. (18 mm) for diameters larger than NPS 6.0 in. (DN150).

(iii) To be qualified for flaw through-wall height sizing, the RMS error of the flaw through-wall heights estimated by ultrasonics, as compared with the true through-wall heights, shall not exceed 0.125 in. (3 mm).

(12) Documentation of the qualifications of procedures and personnel shall be maintained. Documentation shall include identification of personnel, NDE procedures, equipment and specimens used during qualification, and results of the performance demonstration.

(13) The preservice examinations will be performed per ASME Section XI (Reference 4).

5.2 Basis for use

The overall basis for this proposed alternative is that encoded PAUT is equivalent or superior to RT for detecting and sizing critical (planar) flaws. In this regard, the basis for the proposed alternative was developed from numerous codes, code cases, associated industry experience, articles, and the results of RT and encoded PAUT examinations. It has been shown that PAUT provides an equally effective examination for identifying the presence of fabrication flaws in carbon steel welds compared to RT (Reference 2). The examination procedure and personnel performing examinations are qualified using representative piping conditions and flaws that demonstrate the ability to detect and size flaws that are both acceptable and unacceptable to the defined acceptance standards. The demonstrated ability of the examination procedure and personnel to appropriately detect and size flaws provides an acceptable level of quality and safety alternative as allowed by 10 CFR 50.55a(z)(1).

The requirements in this relief request are based upon ASME Section XI Code Case N-831 (N-831) (Reference 14) and will apply to ISI ferritic piping butt welds requiring radiography during repair/replacement activities. N-831 was approved by ASME Board on Nuclear Codes and Standards on October 20, 2016; however, it has not been incorporated into NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and thus, is not available for application at nuclear power plants without specific NRC approval.

6. <u>Duration of Proposed Alternative</u>

Relief is requested for the fourth ISI interval for Braidwood Station, Units 1 and 2, or until the NRC approves N-831, or a later revision, in Regulatory Guide 1.147 or other document during the interval.

7. <u>Precedents</u>

Braidwood Station, Units 1 and 2, third ISI interval relief request was authorized by NRC Safety Evaluation (SE) dated June 5, 2017 (Reference 13). This Braidwood Station relief request was part of an EGC fleet-wide submittal, and the use of encoded phased array ultrasonic examination techniques in lieu of radiography was authorized for various stations. This relief request for the Braidwood Station, Units 1 and 2, fourth ISI interval, utilizes a similar approach to the previously approved relief request.

Oconee Request for Relief No. 2006-ON-01, dated February 2, 2006, requested an alternative for examination of butt welds between the Pressurizer Level and Sample Tap nozzles and their respective Safe Ends. The reason for the request was based on the

difficulty to perform the Code-required radiography. The alternative was to perform ultrasonic examination per similar requirements to ASME Code Case N-659-1. (ADAMS Accession No. ML060450464).

Wolf Creek 10CFR50.55a Request ET 06-0029, dated September 1, 2006, requested an alternative for examination of Main Steam and Feedwater piping welds being replaced due to flow assisted corrosion. The reason for the request was based on the acceptability of the proposed ultrasonic examination alternative process, radiation exposure reduction, outage costs and duration, and radiography exposure risk. (ADAMS Accession No. ML062500093).

Palo Verde Nuclear Generating Station Relief Request 48, dated August 1, 2012, ADAMS Accession No. ML12229A046. NRC approval dated April 12, 2013, ADAMS Accession No. ML13091A177.

Letter from Michael T. Markley, US NRC, to Daniel G. Stoddard, Dominion Energy, Subject: Millstone Power Station, Units 1 and 2, and Surry Power Station, Units 1 and 2; Proposed Alternative for the Use of Encoded Phased Array Ultrasonic Examination (CAC Nos. MF9923, MF9924, MF9925, MF9926, MF9927, and MF9928; EPID L-2017-LLR-0060) (Accession No. ML18019A195).

8. References

1) ASME Section III Code Case N-659-2, "Use of Ultrasonic Examination in Lieu of Radiography for Weld Examination Section III, Divisions 1 and 3," dated June 9, 2008.

2) US NRC, NUREG/CR-7204, "Applying Ultrasonic Testing in Lieu of Radiography for Volumetric Examination of Carbon Steel Piping" (ML15253A674).

3) ASME B31.1 Case 168, "Use of Ultrasonic Examination in Lieu of Radiography for B31.1 Application," dated June 1997.

4) ASME Boiler and Pressure Vessel Code, Section XI, Division 1, “Inservice Inspection of Nuclear Power Plant Components,” 2013 Edition.

5) ASME Section III Code Case N-818, "Use of Analytical Evaluation approach for Acceptance of Full Penetration Butt Welds in Lieu of Weld Repair," dated December 6, 2011.

6) ASME Code Case 2235-9, "Use of Ultrasonic Examination in Lieu of Radiography Section I, Section VIII, Divisions 1 and 2, and Section XII," dated October 11, 2005.

7) ASME Code Case 2326, "Ultrasonic Examination in Lieu of Radiographic Examination for Welder Qualification Test Coupons Section IX," dated January 20, 2000.

8) ASME Code Case 2541, "Use of Manual Phased Array Ultrasonic Examination Section V," dated January 19, 2006.

9) ASME Code Case 2558, "Use of Manual Phased Array E-Scan Ultrasonic Examination per Article 4 Section V," dated December 30, 2006.

10) ASME Code Case 2599, "Use of Linear Phased Array E-Scan Ultrasonic Examination per Article 4 Section V," dated January 29, 2008.

11) ASME Code Case 2600, "Use of Linear Phased Array S-Scan Ultrasonic Examination Per Article 4 Section V," dated January 29, 2008.

12) ASME Code Case N-713, "Ultrasonic Examination in Lieu of Radiography Section XI, Division 1," dated November 10, 2008.

13) Letter from D. J. Wrona (NRC) to B. C. Hanson (EGC) regarding "Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; R. E. Ginna Nuclear Power Plant; and Three Mile Island Nuclear Station, Unit 1 – Proposed Alternative to Use Encoded Phased Array Ultrasonic Examination Techniques (CAC Nos. MF8763-MF8782 and MF9395)," dated June 5, 2017 (ADAMS Accession No. ML17150A091).

14) ASME Section XI Code Case N-831, "Ultrasonic Examination in Lieu of Radiography for Welds in Ferritic Pipe Section XI, Division 1," ASME Approval Date: October 20, 2016.