Exelon Generation.

4300 Winfield Road
Warrenville, IL 60555

630 657 2000 Office

RS-17-031 10 CFR 50.55a

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

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

Subject: Correction to Relief Request I3R-17 to Extend the Reactor Pressure Vessel Inservice Inspection Interval

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

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

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

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

As noted in Reference 1, NRC approval of Relief Request I3R-17 is requested based on the justification provided in topical report WCAP-16168-NP-A, Revision 3 (Reference 2). This topical report demonstrated that extending the ASME B&PV Code, Section XI ISI interval from the current 10 years to 20 years for reactor pressure vessel (RPV) pressure containing welds is an alternative inspection interval that provides an acceptable level of quality and safety. The

NRC approved WCAP-16168 in Reference 3. Relief Request I3R-17 specifically addresses Examination Categories B-A and B-D applicable to the RPV for Braidwood Station, Units 1 and 2.

During review of the planned Unit 1 Spring 2018 (A1R20) outage scope, an error was identified in Table 1-1 (i.e., list of Unit 1 subject welds) and Table 1-2 (i.e., list of Unit 2 subject welds) of the proposed relief request. Both the upper and the lower RPV heads have a Category B-A, Item B1.21 weld which falls under the scope of WCAP-16168-NP-A, Revision 3. Both of these upper welds (i.e., 1(2)RV-03-002) and lower welds (i.e., 1(2)RV-02-001) were listed on the noted tables in an early draft revision of the relief request. During review of the draft relief request, prior to submittal, it was concluded that the upper welds should be deleted from the tables since these welds were last inspected early in the second ISI interval. Extending the next inspection of the upper welds to near the end of the fourth ISI interval, in accordance with the subject relief request, would result in an inspection interval greater than the 20-year maximum interval approved in Reference 3. When deleting the subject welds from the tables, the lower welds vice upper welds were deleted in error.

This error has been corrected in Relief Request I3R-17, Revision 1 (attached). Specifically, the upper head Category B-A, Item B1.21 welds on Table 1-1 and Table 1-2 have been deleted; and the lower head Category B-A, Item B1.21 welds (i.e., 1(2)RV-02-001, "Dutchman-to-Lower Center Disc") have been added to Table 1-1 and Table 1-2, respectively. This change, denoted by a revision bar on Table 1-1 and Table 1-2, is the only change in Revision 1 of the subject relief request and has no impact on the technical justification for this relief request since, as noted above, these lower RPV head welds are within the scope of WCAP-16168-NP-A, Revision 3.

As noted in Reference 1, EGC requested approval of this relief request by July 21, 2017 to support the Braidwood Station, Unit 1 refueling outage (A1R20) scheduled to commence on April 9, 2018.

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

Respectfully,

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

Attachment: 10 CFR 50.55a Relief Request I3R-17, Revision 1

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

1.0 ASME Code Components Affected

The affected components are the Braidwood Station, Unit 1 and Unit 2 Reactor Pressure Vessel (RPV); specifically, the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI (Reference 1) examination categories and item numbers covering examinations of the RPV. These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME B&PV Code, Section XI.

Category B-A welds are defined as "Pressure Retaining Welds in Reactor Vessel." Category B-D welds are defined as "Full Penetration Welded Nozzles in Vessels"

Examination Category	Item No.	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inside Radius Section

2.0 Applicable Code Edition and Addenda

The Third Interval Inservice Inspection (ISI) program is based on the ASME B&PV Code, Section XI, 2001 Edition through 2003 Addenda (Reference 1). Throughout this request the above examination categories are referred to as "the subject examinations" and the ASME B&PV Code, Section XI, is referred to as "the Code."

The applicable Code for the subsequent (i.e., Fourth) ISI interval will be implemented in accordance with the requirements of 10 CFR 50.55a.

3.0 Applicable Code Requirement

ASME Section XI Paragraph IWB-2412, "Inspection Program B," requires volumetric and surface examination of essentially 100% of RPV pressure retaining welds identified in Table IWB-2500-1 once each 10-year interval. The Braidwood Station Third 10-year ISI intervals are currently scheduled to end on July 28, 2018 for Unit 1 and October 16, 2018 for Unit 2.

4.0 Reason for Request

Pursuant to 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. An alternative is requested from the requirement of IWB-2412, "Inspection Program B," that volumetric examinations of essentially 100% of RPV pressure retaining, Examination Category B-A and B-D items, be performed once each 10-year interval.

Extension of the interval between examinations of Category B-A and B-D items from 10 years to a maximum of 20 years will result in a reduction in person-rem exposure and examination costs.

5.0 Proposed Alternative and Basis for Use

Exelon Generation Company, LLC (EGC) proposes to defer the ASME B&PV Code required volumetric examinations of specific Braidwood Station Unit 1 and Unit 2 RPV pressure retaining Examination Category B-A and B-D items. These examinations are currently scheduled to be performed during the Third ISI Interval in 2018. Upon approval of this relief request, these required examinations would be performed during the Fourth ISI Interval for each unit in 2027 plus or minus one refueling outage. These dates are consistent with those provided in Pressurized Water Reactor Owners Group (PWROG) letter OG-06-356 (Reference 10) and the latest implementation plan provided in PWROG letter OG-10-238 (Reference 2). Tables 1-1 and 1-2 list the applicable examination areas addressed under this relief request for Braidwood Station, Unit 1 and Unit 2 respectively.

In accordance with 10 CFR 50.55a(z)(1), an alternate inspection interval is requested on the basis that the current interval can be extended based on a negligible change in risk when compared to the risk criteria specified in Regulatory Guide 1.174 (Reference 3).

The methodology used to demonstrate the acceptability of extending the inspection intervals for Category B-A and B-D items, based on a negligible change in risk, is contained in WCAP-16168-NP-A, Revision 3, (Reference 4). This methodology was used to develop a pilot plant analysis for RPVs in Westinghouse, Combustion Engineering, and Babcock and Wilcox plant designs, and is an extension of the work that was performed as part of the NRC Pressurized Thermal Shock (PTS) Risk Study (Reference 5). The critical parameters for demonstrating that this pilot plant analysis is applicable on a plant specific basis, as identified in WCAP-16168-NP-A, Revision 3, are shown in Table 2. Table 2 lists the critical parameters investigated in WCAP-16168-NP-A, Revision 3 and compares the results of the Westinghouse pilot plant to those of Braidwood Units 1 and 2 for an ISI Interval extension.

Tables 3-1, 3-2, 4-1, and 4-2 provide additional information required by the NRC as noted in Appendix A of WCAP-16168-NP-A, Revision 3.

By demonstrating that each plant specific parameter is bounded by the corresponding pilot plant parameter, the application of the methodology for Braidwood Station, Units 1 and 2 is determined to be acceptable.

Table 1-1
Examination Category, Item Number, Component Description and Component Identification of Applicable Examinations for Braidwood Unit 1

Exam Category	Exam Item Number	Component Description	ISI Weld Number
B-A	B1.11	Reactor Vessel Shell-to-Reactor Vessel Shell	1RV-01-003
		Reactor Vessel Shell-to-Reactor Vessel Shell	1RV-01-004
		Reactor Vessel Shell-to-Dutchman	1RV-02-002
	B1.21	Dutchman-to-Lower Center Disc	1RV-02-001
	B1.30	Reactor Vessel-to-Flange	1RV-01-005
B-D	B3.90	Outlet Nozzle @ 22 Degrees	1RV-01-006
		Inlet Nozzle @ 67 Degrees	1RV-01-007
		Inlet Nozzle @ 113 Degrees	1RV-01-008
		Outlet Nozzle @ 158 Degrees	1RV-01-009
		Outlet Nozzle @ 202 Degrees	1RV-01-010
		Inlet Nozzle @ 247 Degrees	1RV-01-011
		Inlet Nozzle @ 293 Degrees	1RV-01-012
		Outlet Nozzle @ 338 Degrees	1RV-01-013
	B3.100	Outlet Nozzle Inner Radius @ 22 Degrees	1RV-01-014
		Inlet Nozzle Inner Radius @ 67 Degrees	1RV-01-015
		Inlet Nozzle Inner Radius @ 113 Degrees	1RV-01-016
		Outlet Nozzle Inner Radius @ 158 Degrees	1RV-01-017
		Outlet Nozzle Inner Radius @ 202 Degrees	1RV-01-018
		Inlet Nozzle Inner Radius @ 247 Degrees	1RV-01-019
		Inlet Nozzle Inner Radius @ 293 Degrees	1RV-01-020
		Outlet Nozzle Inner Radius @ 338 Degrees	1RV-01-021

Table 1-2
Examination Category, Item Number, Component Description and Component Identification of Applicable Examinations for Braidwood Unit 2

Exam Category	Exam Item Number	Component Description	ISI Weld Number
B-A	B1.11	Reactor Vessel Shell-to-Reactor Vessel Shell	2RV-01-003
		Reactor Vessel Shell-to-Reactor Vessel Shell	2RV-01-004
		Reactor Vessel Shell-to-Dutchman	2RV-02-002
	B1.21	Dutchman-to-Lower Center Disc	2RV-02-001
	B1.30	Reactor Vessel-to-Flange	2RV-01-005
B-D	B3.90	Outlet Nozzle @ 22 Degrees	2RV-01-006
		Inlet Nozzle @ 67 Degrees	2RV-01-007
		Inlet Nozzle @ 113 Degrees	2RV-01-008
		Outlet Nozzle @ 158 Degrees	2RV-01-009
		Outlet Nozzle @ 202 Degrees	2RV-01-010
		Inlet Nozzle @ 247 Degrees	2RV-01-011
		Inlet Nozzle @ 293 Degrees	2RV-01-012
		Outlet Nozzle @ 338 Degrees	2RV-01-013
	B3.100	Outlet Nozzle Inner Radius @ 22 Degrees	2RV-01-014
		Inlet Nozzle Inner Radius @ 67 Degrees	2RV-01-015
		Inlet Nozzle Inner Radius @ 113 Degrees	2RV-01-016
		Outlet Nozzle Inner Radius @ 158 Degrees	2RV-01-017
		Outlet Nozzle Inner Radius @ 202 Degrees	2RV-01-018
		Inlet Nozzle Inner Radius @ 247 Degrees	2RV-01-019
		Inlet Nozzle Inner Radius @ 293 Degrees	2RV-01-020
		Outlet Nozzle Inner Radius @ 338 Degrees	2RV-01-021

Table 2

Critical Parameters for the Application of Bounding Analysis for Braidwood Units 1 and 2

Parameter	Pilot Plant Basis	Plant-Specific Basis	Additional Evaluation Required
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are Applicable	NRC PTS Risk Study (Reference 7)	PTS Generalization Study (Reference 6)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 4)	Unit 1: 2.09E-16 Events per year (calculated per Reference 4)	No
		Unit 2: 2.04E-15 Events per year (calculated per Reference 4)	
Frequency and Severity of Design Basis Transients	7 heatup / cooldown cycles per year (Reference 4)	Bounded by 7 heatup / cooldown cycles per year	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 4)	Single Layer	No

Tables 3-1 and 3-2 provide a summary of the latest reactor vessel inspections for Braidwood Station, Units 1 and 2 respectively, and an evaluation of the recorded indications. This information confirms that satisfactory examinations have been performed for both Braidwood reactor vessels.

Table 3-1

Additional Information Pertaining to Reactor Vessel Inspection for Braidwood Unit 1

Inspection methodology:	The latest ISI was conducted in accordance with the ASME Code, Section XI and Section V, 1989 Edition, with no Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Examinations of Category B-A and B-D welds were performed to ASME Section XI, Appendix VIII, 1989 Edition with no Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Future inservice inspections will be performed to ASME Section XI, Appendix VIII requirements.
Number of past inspections:	Two 10-year inservice inspections have been performed.
Number of indications found:	There were three indications identified in the beltline region during the most recently completed inservice inspection. One subsurface indication is located in the nozzle shell to intermediate shell circumferential weld seam (Item 4 in Table 3-1) and two subsurface indications are located in the intermediate shell to lower shell circumferential weld seam (Item 5 in Table 3-1). All three indications are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. None of these indications are within the inner 1/10th or 1 inch of the reactor vessel thickness; therefore, they are inherently acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 7).
Proposed inspection schedule for balance of plant life:	The third inservice inspection currently is scheduled for 2018. This inspection will be performed in 2027 plus or minus one refueling outage. The proposed inspection date for Braidwood Unit 1 is consistent with the latest implementation plan presented in OG-10-238 (Reference 2).

Table 3-2

Additional Information Pertaining to Reactor Vessel Inspection for Braidwood Unit 2

Inspection methodology:	The latest ISI was conducted in accordance with the ASME Code, Section XI and Section V, 1989 Edition, with no Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Examinations of Category B-A and B-D welds were performed to ASME Section XI, Appendix VIII, 1989 Edition with no Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Future inservice inspections will be performed to ASME Section XI, Appendix VIII requirements.
Number of past inspections:	Two 10-year inservice inspections have been performed.
Number of indications found:	There were two indications identified in the beltline region during the most recently completed inservice inspection. One subsurface indication is located in the nozzle shell to intermediate shell circumferential weld seam (Item 4 in Table 3-2) and another subsurface indication is located in the intermediate shell to lower shell circumferential weld seam (Item 5 in Table 3-2). Both indications are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. One indication is within the inner 1/10th or 1" of the reactor vessel thickness. This indication is acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 7), since the number of flaws is less than the allowable number of flaws for each flaw size increment. A disposition of this flaw against the limits of the Alternate PTS Rule is shown in the table below. The following indication is located within the forging material of the reactor vessel beltline.
Proposed inspection schedule for balance of plant life:	The third inservice inspection currently is scheduled for 2018. This inspection will be performed in 2027 plus or minus one refueling outage. The proposed inspection date for Braidwood Station, Unit 2 is consistent with the latest implementation plan presented in OG-10-238 (Reference 2).

Through-Wall Extent, TWE		Scaled maximum number of forging flaws	Number of forging flaws (Axial/Circ.)
TWE_{MIN}	TWE_{MAX}		
0.075	0.375	76	1 (0/1)
0.125	0.375	30	1 (0/1)
0.175	0.375	8	0 (0/0)

Tables 4-1 and 4-2 summarize the inputs and outputs for the calculation of through-wall cracking frequency (TWCF) for Braidwood Station, Units 1 and 2 respectively.

Table 4-1:
Details of TWCF Calculation for Braidwood Unit 1 at 57 Effective Full-Power Years (EFPY)

Inputs								
Reactor Coolant System Temperature, T_{RCS} [°F]:			N/A	Twall [inches]:				8.625
No.	Region and Component Description	Material Heat No.	Cu(1) [wt%]	Ni(1) [wt%]	R.G.(1) 1.99 Pos.	CF(1) [°F]	$RT_{NDT(u)}$(1) [°F]	Fluence [10^{19} n/cm^2, E>1.0 MeV](1)
1	Nozzle Shell (NS) Forging	5P-7016	0.04	0.73	1.1	26	10	1.14
2	Intermediate Shell (IS) Forging	[49D383/49C344]-1-1	0.05	0.73	1.1	31	-30	3.19
3	Lower Shell (LS) Forging	[49D867/49C813]-1-1	0.05	0.74	1.1	31	-20	3.19
4	NS to IS Circ. Weld Seam	H4498	0.04	0.46	1.1	54	-25	1.14
5	IS to LS Circ. Weld Seam	442011	0.03	0.67	1.1	41	40	3.06

Outputs						
Methodology Used to Calculate ΔT_{30}:			Regulatory Guide 1.99, Revision 2(2)			
	Controlling Material Region No. (From Above)	RT_{MAX-XX} [°R]	Fluence [10^{19} n/cm^2, E > 1.0 MeV]	FF (Fluence Factor)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Forging – FO	1	496.62	1.14	1.037	26.95	8.38E-17
Limiting Circumferential Weld - CW	5	552.78	3.06	1.295	53.11	0.00E+00
$TWCF_{95-TOTAL}(\alpha_{FO}TWCF_{95-FO}+\alpha_{CW}TWCF_{95-CW})$:						2.09E-16

(1) Reference 8
(2) Reference 9

Table 4-2:
Details of TWCF Calculation for Braidwood Unit 2 at 57 Effective Full-Power Years (EFPY)

Inputs								
Reactor Coolant System Temperature, T_{RCS} [°F]:			N/A		T_{wall} [inches]:			8.625
No.	Region and Component Description	Material Heat No.	Cu[1] [wt%]	Ni[1] [wt%]	R.G.[1] 1.99 Pos.	CF[1] [°F]	$RT_{NDT(u)}$[1] [°F]	Fluence [10^{19} n/cm^2, E>1.0 MeV][1]
1	Nozzle Shell (NS) Forging	5P-7056	0.04	0.90	1.1	26	30	1.11
2	Intermediate Shell (IS) Forging	[49D963/49C904]-1-1	0.03	0.71	1.1	20	-30	3.16
3	Lower Shell (LS) Forging	[50D102/50C97]-1-1	0.06	0.76	1.1	37	-30	3.16
4	NS to IS Circ. Weld Seam	H4498	0.04	0.46	1.1	54	-25	1.11
5	IS to LS Circ. Weld Seam	442011	0.03	0.67	1.1	41	40	3.03

Outputs						
Methodology Used to Calculate ΔT_{30}:			Regulatory Guide 1.99, Revision 2[2]			
	Controlling Material Region No. (From Above)	RT_{MAX-XX} [°R]	Fluence [10^{19} n/cm^2, E > 1.0 MeV]	FF (Fluence Factor)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Forging – FO	1	516.43	1.11	1.029	26.76	8.18E-16
Limiting Circumferential Weld - CW	5	552.69	3.03	1.293	53.02	0.00E+00
$TWCF_{95-TOTAL}(\alpha_{FO}TWCF_{95-FO} + \alpha_{CW}TWCF_{95-CW})$:						2.04E-15

(1) Reference 8
(2) Reference 9

6.0 Duration of Proposed Alternative

This request is applicable to the Braidwood Station, Units 1 and 2 Inservice Inspection program for the third and fourth 10-year inspection intervals.

7.0 Precedents

1. "Millstone Power Station, Unit No. 3 – Alternative Request IR-3-27 for Implementation of Extended Reactor Vessel Inservice Inspection Interval (CAC NO. MF5868)," dated February 16, 2016 (ADAMS Accession Number ML16038A001)

2. "Joseph M. Farley, Unit 1, Alternative to Inservice Inspection (CAC No. MF6475)," dated February 1, 2016 (ADAMS Accession Number ML16013A348)

3. "Diablo Canyon Power Plant, Unit No. 1 - Request for Alternative RPV-U1-Extension to Allow Use of Alternate Reactor Inspection Interval Requirements (TAC No. MF4678)," dated June 19, 2015 (ADAMS Accession Number ML15168A024)

4. "Callaway Plant, Unit 1 – Request for Relief I3R-17, Alternative to ASME Code Requirements Which Extends the Reactor Vessel Inspection Interval from 10 to 20 Years (TAC No. MF3876)," dated February 10, 2015 (ADAMS Accession Number ML15035A148)

5. "Shearon Harris Nuclear Power Plant, Unit 1 – Relief from the Requirements of the ASME Code, Section XI (TAC NO. MF4113)," dated January 5, 2015 (ADAMS Accession Number ML14353A324)

6. "Byron Station, Unit No. 1 – Relief from Requirements of the ASME Code to Extend the Reactor Vessel Inservice Inspection Interval (TAC No. MF3596)," dated December 10, 2014 (ADAMS Accession Number ML14303A506, correction letter ML13113A016)

7. "Wolf Creek Generating Station – Request for Relief Nos. I3R-08 and I3R-09 for the Third 10-Year Inservice Inspection Program Interval (TAC Nos. MF3321 and MF3322)," dated December 10, 2014 (ADAMS Accession Number ML14321A864)

8. "Watts Bar Nuclear Plant, Unit 1- Request for Alternative 13-ISI-01 to Extend the Second Reactor Vessel Weld Inservice Inspection Interval (TAC No. MF2956)," dated November 28, 2014 (ADAMS Accession Number ML14314A987)

9. "Sequoyah Nuclear Plant, Units 1 and 2 – Requests for Alternatives 13-ISI-1 and 13-ISI-2 to Extend the Reactor Vessel Weld Inservice Inspection Interval (TAC Nos. MF2900 and MF2901)," dated August 1, 2014 (ADAMS Accession Number ML14188B920)

10. "Catawba Nuclear Station Units 1 and 2: Proposed Relief Request 13-CN-003, Request for Alternative to the Requirement of IWB-2500, Table IWB-2500-1, Category B-A and Category B-D for Reactor Pressure Vessel Welds (TAC Nos. MF1922 and MF1923)," dated March 26, 2014 (ADAMS Accession Number ML14079A546)

11. "Vogtle Electric Generating Plant, Units 1 and 2 – Request for Alternatives VEGP-ISI-ALT-05 and VEGP-ISI-ALT-06 (TAC Nos. MF2596 and MF2597)," dated March 20, 2014 (ADAMS Accession Number ML14030A570)

12. "Indian Point Nuclear Generating Unit No. 3 – Safety Evaluation for Relief Request IP3-ISI-RR-06 for Reactor Vesel [sic] Weld Examinations (TAC No. MF3345)," dated July 23, 2014 (ADAMS Accession Number ML14198A331)

13. "Surry Power Station Units 1 and 2 – Relief Implementing Extended Reactor Vessel Inspection Interval (TAC Nos. ME8573 and ME8574)," dated April 30, 2013 (ADAMS Accession Number ML13106A140)

14. "McGuire Nuclear Station, Unit 2, Relief 10-MN-002 to Extend the Inservice Inspection Interval for Reactor Vessel Category B-A and B-D Welds (TAC Nos. ME7329 and ME 7330)," dated September 6, 2012 (ADAMS Accession Number ML12249A175)

15. "Arkansas Nuclear One, Unit 2 – Request for Alternative ANO2-ISI-004, to Extend the Third 10-Year Inservice Inspection Interval for Reactor Vessel Weld Examinations (TAC No. ME2508)," dated September 21, 2010 (ADAMS Accession Number ML102450654)

16. "Three Mile Island Nuclear Station, Unit 1 (TMI-1) – Request to Extend the Inservice Inspection Interval for Reactor Vessel Weld and Internal Examinations, Proposed Alternative Request Nos. RR-09-01 and RR-09-02 (TAC Nos. ME2483 and ME2484)," dated September 21, 2010 (ADAMS Accession Number ML102390018)

17. "Joseph M. Farley Nuclear Plant, Unit 2 (Farley Unit 2) – Relief Request for Extension of the Reactor Vessel Inservice Inspection Date to the Year 2020 (Plus or Minus One Outage) (TAC No. ME3010)," dated July 12, 2010 (ADAMS Accession Number ML101750402)

18. "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Relief Request No. RR-40, Reactor Vessel Weld Examination Interval Extension (TAC Nos. ME1634, ME1635, and ME1636)," dated February 22, 2010 (ADAMS Accession Number ML100290415)

19. "Safety Evaluation of Relief Requests to Extend the Inservice Inspection Interval for Reactor Vessel Examinations for Salem Nuclear Generating Station, Unit Nos. 1 and 2 (TAC Nos. ME1478, ME1479, ME1480 and ME1481)," dated February 22, 2010 (ADAMS Accession Number ML100491550)

8.0 References

1. ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition through 2003 Addenda, ASME International

2. PWROG Letter OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." PA-MSC-0120," July 12, 2010 (ADAMS Accession Number ML11153A033)

3. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk **Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing** Basis," U.S. Nuclear Regulatory Commission, November 2002

4. Westinghouse Report WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval," October 2011 (ADAMS Accession Number ML113060207)

5. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," U.S. Nuclear Regulatory Commission, March 2010

6. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," U.S. Nuclear Regulatory Commission, December 14, 2004 (ADAMS Accession Number ML042880482)

7. 10 CFR Part 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Washington D. C., Federal Register, Volume 75, No. 1, dated January 4, 2010 and No. 22 with corrections to part (g) dated February 3, 2010, March 8, 2010, and November 26, 2010

8. Westinghouse Report WCAP-17607-NP, Revision 0, "Braidwood Station Units 1 and 2 Reactor Vessel Integrity Evaluation to Support License Renewal Time-Limited Aging Analysis," December 2012

9. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988

10. OG-06-356, "Plan for Plant-Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk Informed Extension of the Reactor Vessel In-Service Inspection Interval,' MUHP 5097-99, Task 2059," October 31, 2006