1717 Wakonade Drive
Welch, MN 55089

June 13, 2019

L-PI-19-002
10 CFR 50.55a

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

Prairie Island Nuclear Generating Plant, Units 1 and 2
Docket Nos. 50-282 and 50-306
Renewed Facility Operating License Nos. DPR-42 and DPR-60

10 CFR 50.55a Requests Nos. 1-RR-5-10 and 2-RR-5-10, Proposed Alternative to Reactor Vessel Inservice Inspection (ISI) Intervals for Prairie Island Unit 1 and Unit 2

Pursuant to 10 CFR 50.55a(z)(1), Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), requests U.S. Nuclear Regulatory Commission (NRC) authorization of an alternative to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2007 Edition through 2008 Addenda for the Prairie Island Nuclear Generating Plant (PINGP) Unit 1 and Unit 2.

NSPM is requesting NRC authorization of the proposed alternative for the fifth and sixth ten-year intervals of the PINGP Inservice Inspection (ISI) Program. Specifically, NSPM is requesting authorization to extend the Unit 1 and Unit 2 reactor pressure vessel ISI intervals. The enclosed relief request for Unit 1 (1-RR-5-10) and Unit 2 (2RR-5-10) provides the basis and supporting information for the proposed alternative.

PINGP is currently in the fifth ten-year interval, which began on December 21, 2014, and is currently scheduled to end December 20, 2024. NSPM requests authorization of these 10 CFR 50.55a requests by July 15, 2020, to support planning milestones for the Unit 2 refueling outage (2R33) scheduled to commence fall of 2023.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

If there are any questions or if additional information is needed, please contact Mr. Jeff Kivi at 612-330-5788.

Scott Sharp
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure:

1. 10 CFR 50.55a Request Nos. 1-RR-5-10 and 2-RR-5-10, Proposed Alternative to Reactor Vessel Inservice Inspection (ISI) Intervals for Prairie Island Unit 1 and Unit 2

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC

10 CFR 50.55a Request 1-RR-5-10 (PINGP Unit 1)
10 CFR 50.55a Request 2-RR-5-10 (PINGP Unit 2)
Proposed Alternative to Reactor Vessel Inservice Inspection (ISI) Intervals

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1)

Provides an Acceptable Level of Quality and Safety

1. American Society of Mechanical Engineers (ASME) Code Component(s) Affected

The affected components are the Prairie Island Nuclear Generating Plant (PINGP) Unit 1 and Unit 2 reactor vessels (RV's), specifically, the following ASME Boiler and Pressure Vessel (BPV) Code, Section XI (Reference 1) examination categories and item numbers covering examinations of the RV's. These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME BPV 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 Number	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. Applicable Code Edition and Addenda

ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 2007 Edition though 2008 Addenda (Reference 1).

3. Applicable Code Requirement

IWB-2411, Inspection Program, requires volumetric examination of essentially 100% of reactor vessel pressure-retaining welds identified in Table IWB-2500-1 once each ten-year interval. The PINGP Unit 1 and Unit 2 fifth ten-year inservice inspection (ISI) interval is scheduled to end on December 20, 2024. The applicable Code for the sixth ten-year ISI interval will be selected in accordance with the requirements of 10 CFR 50.55a.

4. Reason for Request

An alternative is requested from the requirement of IWB-2411, Inspection Program, that volumetric examination of essentially 100% of reactor vessel pressure-retaining Examination Category B-A and B-D welds be performed once each ten-year interval. Extension of the interval between examinations of Category B-A and B-D welds from 10 years to up to 20 years will reduce man-rem exposure and examination costs.

5. Proposed Alternative and Basis for Use

Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), proposes not to perform the ASME Code required volumetric examinations of the PINGP Unit 1 and Unit 2 reactor vessels full penetration pressure-retaining Examination Category B-A and B-D welds for the fifth inservice inspection interval, currently scheduled for 2024 and 2025[1], respectively. NSPM will perform the fifth ASME Code required volumetric examination of the PINGP Unit 1 and Unit 2 reactor vessels full penetration pressure-retaining Examination Category B-A and B-D welds in the sixth inservice inspection interval in 2033 and 2034, respectively. The proposed inspection dates for PINGP Unit 1 and Unit 2 are consistent with the implementation plan presented in 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" (Reference 2).

In accordance with 10 CFR 50.55a(z)(1), an alternate inspection interval is requested on the basis that the current interval can be revised with negligible change in risk by satisfying the risk criteria specified in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis", (Reference 3). Thus, the alternative provides an acceptable level of quality and safety.

The methodology used to conduct this analysis is based on that defined in the study WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval" (Reference 4). This study focuses on risk assessments of materials within the beltline region of the RV wall. The results of the calculations for PINGP Unit 1 and Unit 2 were compared to those obtained from the Westinghouse pilot plant evaluated in WCAP-16168-NP-A, Revision 3. Appendix A of the WCAP identifies the parameters to be compared. Demonstrating that the parameters for PINGP Unit 1 and Unit 2 are bounded by the results of the Westinghouse pilot plant qualifies PINGP Unit 1 and Unit 2 for an ISI interval extension.

Table 1a below lists the critical parameters investigated in the WCAP and compares the results of the Westinghouse pilot plant to those of PINGP Unit 1. Table 1b below lists the critical parameters investigated in the WCAP and compares the results of the Westinghouse

[1] NSPM's current plan is to complete the examination either in the fall of 2023 or in the fall of 2025 by extending the fifth interval for the Unit 2 vessel ISI in accordance with IWA-2430.

pilot plant to those of PINGP Unit 2. Tables 2a, 2b, 3a and 3b provide additional information that was requested by the NRC and included in Appendix A of Reference 4.

Table 1a
Critical Parameters for the Application of Bounding Analysis for PINGP Unit 1

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 5)	PTS Generalization Study (Reference 6)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 4)	7.86E-14 Events per year (Calculated per Reference 4)	No
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

Table 1b
Critical Parameters for the Application of Bounding Analysis for PINGP Unit 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 5)	PTS Generalization Study (Reference 6)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 4)	2.82E-14 Events per year (Calculated per Reference 4)	No
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 2a and 2b below provide summaries of the latest reactor vessel inspection for PINGP Unit 1 and Unit 2, respectively, and an evaluation of the recorded indications. This information confirms that satisfactory examinations have been performed on the PINGP Unit 1 and Unit 2 reactor vessels.

Table 2a
Additional Information Pertaining to Reactor Vessel Inspection for PINGP Unit 1

<table>
<tr><td>Inspection methodology:</td><td>The latest ISI examinations of PINGP Unit 1 Category B-A and B-D welds were performed to ASME Section XI, Appendix VIII, 1998 Edition with 2000 Addenda as modified by the PDI program and Federal Register, Part II, NRC 10 CFR Part 50, Industry Codes and Standards, amended requirements. Future inservice inspections will be performed to ASME Section XI, Appendix VIII methodology.</td></tr>
<tr><td>Number of past inspections:</td><td>Four ten-year inservice inspections have been performed.</td></tr>
<tr><td>Number of indications found:</td><td>There were 54 indications identified in the beltline region of the RV during the last ISI. These subsurface indications are located in the nozzle to intermediate circumferential weld seam (Item 4 in Table 3a) and the intermediate to lower shell circumferential weld seam (Item 5 in Table 3a). These indications are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. Twenty-three of these indications are within the inner 1/10th or 1 inch of the reactor vessel thickness. These indications are acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 7). A disposition of the twenty-three flaws against the limits of the Alternate PTS Rule is shown in the tables below.

The following indications are located within the weld material of the reactor vessel beltline.
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in)</th><th rowspan="2">Scaled maximum number of weld flaws</th><th rowspan="2">Number of Flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0</td><td>0.075</td><td>No Limit</td><td>0</td></tr>
<tr><td>0.075</td><td>0.475</td><td>139</td><td>6 (0/6)</td></tr>
<tr><td>0.125</td><td>0.475</td><td>76</td><td>5 (0/5)</td></tr>
<tr><td>0.175</td><td>0.475</td><td>19</td><td>3 (0/3)</td></tr>
<tr><td>0.225</td><td>0.475</td><td>8</td><td>2 (0/2)</td></tr>
<tr><td>0.275</td><td>0.475</td><td>4</td><td>1 (0/1)</td></tr>
<tr><td>0.325</td><td>0.475</td><td>3</td><td>0</td></tr>
</table>
The following indications are located within the forging material of the reactor vessel beltline.
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in)</th><th rowspan="2">Scaled maximum number of forging flaws</th><th rowspan="2">Number of Flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0</td><td>0.075</td><td>No Limit</td><td>0</td></tr>
<tr><td>0.075</td><td>0.375</td><td>45</td><td>17 (0/17)</td></tr>
<tr><td>0.125</td><td>0.375</td><td>18</td><td>10 (0/10)</td></tr>
<tr><td>0.175</td><td>0.375</td><td>5</td><td>0</td></tr>
</table>
</td></tr>
<tr><td>Proposed inspection schedule for balance of unit life:</td><td>The fifth inservice inspection is currently scheduled for 2024. This inspection will be performed in 2033. The proposed inspection date is consistent with the latest revised implementation plan, OG-10-238 (Reference 2).</td></tr>
</table>

Table 2b
Additional Information Pertaining to Reactor Vessel Inspection for PINGP Unit 2

<table>
<tr><td>Inspection methodology:</td><td>The latest ISI examinations of PINGP Unit 2 Category B-A and B-D welds were performed to ASME Section XI, Appendix VIII, 1998 Edition with 2000 Addenda as modified by the PDI program and Federal Register, Part II, NRC 10 CFR Part 50, Industry Codes and Standards, amended requirements. Future inservice inspections will be performed to ASME Section XI, Appendix VIII methodology.</td></tr>
<tr><td>Number of past inspections:</td><td>Four ten-year inservice inspections have been performed.</td></tr>
<tr><td>Number of indications found:</td><td>There were 17 indications identified in the beltline region of the RV during the last ISI. These subsurface indications are located in the nozzle to intermediate shell circumferential weld seam (Item 4 in Table 3b), and are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. Sixteen of these indications are within the inner $1/10^{th}$ or 1 inch of the reactor vessel thickness. These indications are acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 7). A disposition of the sixteen flaws against the limits of the Alternate PTS Rule is shown in the tables below.

The following indications are located within the weld material of the reactor vessel beltline.
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in)</th><th rowspan="2">Scaled maximum number of weld flaws</th><th rowspan="2">Number of Flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0</td><td>0.075</td><td>No Limit</td><td>0</td></tr>
<tr><td>0.075</td><td>0.475</td><td>139</td><td>12 (0/12)</td></tr>
<tr><td>0.125</td><td>0.475</td><td>76</td><td>7 (0/7)</td></tr>
<tr><td>0.175</td><td>0.475</td><td>19</td><td>2 (0/2)</td></tr>
<tr><td>0.225</td><td>0.475</td><td>8</td><td>2 (0/2)</td></tr>
<tr><td>0.275</td><td>0.475</td><td>4</td><td>1 (0/1)</td></tr>
<tr><td>0.325</td><td>0.475</td><td>3</td><td>0</td></tr>
</table>

The following indications are located within the forging material of the reactor vessel beltline.
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in)</th><th rowspan="2">Scaled maximum number of forging flaws</th><th rowspan="2">Number of Flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0</td><td>0.075</td><td>No Limit</td><td>0</td></tr>
<tr><td>0.075</td><td>0.375</td><td>45</td><td>4 (0/4)</td></tr>
<tr><td>0.125</td><td>0.375</td><td>18</td><td>3 (0/3)</td></tr>
<tr><td>0.175</td><td>0.375</td><td>5</td><td>1 (0/1)</td></tr>
<tr><td>0.225</td><td>0.375</td><td>2</td><td>0</td></tr>
</table>
</td></tr>
<tr><td>Proposed inspection schedule for balance of unit life:</td><td>The fifth inservice inspection is currently scheduled for 2025. This inspection will be performed in 2034. The proposed inspection date is consistent with the latest revised implementation plan, OG-10-238 (Reference 2).</td></tr>
</table>

Tables 3a and 3b summarize the inputs and outputs for the calculation of through-wall cracking frequency (TWCF) for Unit 1 and Unit 2, respectively.

Table 3a
Details of TWCF Calculation for PINGP Unit 1 at 54 Effective Full Power Years (EFPY)

Inputs(1)									
Reactor Coolant System Temperature, T_C [°F]: N/A								**T_{wall} [inches]:**	6.889
Number	**Region and Component Description**	**Material ID**	**Material Heat Number**	**Cu [wt%]**	**Ni [wt%]**	**R.G. 1.99 Pos.**	**CF [ºF]**	**$RT_{NDT(U)}$ [ºF]**	**Fluence [n/cm^2, E > 1.0 MeV]**
1	Nozzle Shell Forging	B	-	0.08	0.68	1.1	51.0	-4°F	1.9843E+19
2	Intermediate Shell Forging	C	-	0.07	0.80	2.1	55.1	14°F	5.7484E+19
3	Lower Shell Forging	D	-	0.07	0.66	1.1	44.0	-4°F	5.6462E+19
4	Nozzle Shell to Intermediate Shell Circumferential Weld	W2	2269	0.15	0.15	1.1	79.5	0°F	1.9843E+19
5	Intermediate Shell to Lower Shell Circumferential Weld	W3	1752	0.13	0.13	2.1	81.4	-13°F	5.5890E+19

Outputs							
Methodology Used to Calculate ΔT_{30}: Regulatory Guide 1.99, Revision 2 (Reference 8)							
	Controlling Material Region Number (From Above)	**α_{XX}**	**RT_{MAX-XX} [°R]**	**Fluence [n/cm^2, E >1.0 MeV]**	**FF (Fluence Factor)**	**ΔT_{30} [ºF]**	**$TWCF_{95-XX}$**
Limiting Forging - FO	2	2.5000	552.40	5.7484E+19	1.4288	78.73	3.145E-14
Limiting Circumferential Weld - CW	5	2.5000	562.55	5.5890E+19	1.4236	102.88	0.000E+00
$TWCF_{95-TOTAL} = (\alpha_{FO}TWCF_{95-FO} + \alpha_{CW}TWCF_{95-CW})$:							**7.86E-14**

(1) Material properties and fluence inputs are based on plant-specific assessment.

Table 3b
Details of TWCF Calculation for PINGP Unit 2 at 54 Effective Full Power Years (EFPY)

Inputs[1]									
Reactor Coolant System Temperature, T_C [°F]: N/A								**T_{wall} [inches]:**	6.889
Number	**Region and Component Description**	**Material ID**	**Material Heat Number**	**Cu [wt%]**	**Ni [wt%]**	**R.G. 1.99 Pos.**	**CF [ºF]**	**$RT_{NDT(U)}$ [ºF]**	**Fluence [n/cm^2, E > 1.0 MeV]**
1	Nozzle Shell Forging	B	-	0.07	0.73	1.1	44.0	-13°F	1.9497E+19
2	Intermediate Shell Forging	C	-	0.07	0.75	1.1	44.0	14°F	5.7003E+19
3	Lower Shell Forging	D	-	0.08	0.67	2.1	60.4	-4°F	5.6303E+19
4	Nozzle Shell to Intermediate Shell Circumferential Weld	W2	1752	0.13	0.13	2.1	81.4	-13°F	1.9497E+19
5	Intermediate Shell to Lower Shell Circumferential Weld	W3	2721	0.09	0.11	2.1	81.2	-31°F	5.5656E+19

Outputs							
Methodology Used to Calculate ΔT_{30}: Regulatory Guide 1.99, Revision 2 (Reference 8)							
	Controlling Material Region Number (From Above)	**α_{XX}**	**RT_{MAX-XX} [°R]**	**Fluence [n/cm^2, E >1.0 MeV]**	**FF (Fluence Factor)**	**ΔT_{30} [ºF]**	**$TWCF_{95-XX}$**
Limiting Forging - FO	3	2.5000	541.74	5.6303E+19	1.4250	86.07	1.129E-14
Limiting Circumferential Weld - CW	5	2.5000	544.21	5.5656E+19	1.4229	115.54	0.000E+00
$TWCF_{95-TOTAL} = (\alpha_{FO}TWCF_{95-FO} + \alpha_{CW}TWCF_{95-CW})$:							**2.82E-14**

(1) Material properties and fluence inputs are based on plant-specific assessment.

6. Duration of Proposed Alternative

This request is applicable to the PINGP Unit 1 and Unit 2 inservice inspection program for the fifth and sixth ten-year inspection intervals.

7. Precedents

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

2. "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.

3. "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.

4. "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.

5. "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.

6. "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.

7. "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.

8. "Braidwood Station, Units 1 and 2 – Request for Relief from the Requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) (CAC Nos. MF8191 and MF8192)", dated March 15, 2017, ADAMS Accession Number ML17054C255.

9. "South Texas Project, Units 1 and 2 – Relief from the Requirements of the ASME Code Regarding the Third 10-Year Inservice Inspection Program Interval (EPID L-2018-LLR-0010)", dated July 24, 2018, ADAMS Accession Number ML18177A425.

8. References

1. ASME Boiler and Pressure Vessel Code, Section XI, 2007 Edition through 2008 Addenda, American Society of Mechanical Engineers, New York.
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, (ADAMS Accession Number ML003740133).
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 ML11306A084).
5. NUREG-1874, “Recommended Screening Limits for Pressurized Thermal Shock (PTS)”, U.S. Nuclear Regulatory Commission, March 2010, (ADAMS Accession Number ML15222A848).
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. Code of Federal Regulations, 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, Number 1, dated January 4, 2010 and Number 22 with corrections to part (g) dated February 3, 2010, March 8, 2010, and November 26, 2010.
8. NRC Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials”, U.S. Nuclear Regulatory Commission, May 1988, (ADAMS Accession Number ML003740284).