Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

February 15, 2018
NOC-AE-18003547
10 CFR 50.55a

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

South Texas Project Units 1 and 2
Docket No. STN 50-498 and STN 50-499
Proposed Alternative to Reactor Vessel Inservice Inspection Intervals
(Relief Request RR-ENG-3-14)

In accordance with the provisions of 10 CFR 50.55a(z)(1), STP Nuclear Operating Company (STPNOC) requests approval for South Texas Project (STP) Units 1 and 2 to extend the Reactor Vessel (RV) Third Inservice Inspection (ISI) Interval of the RV welds from 2020 (Units 1 and 2) to 2026 (Unit 1) and 2027 (Unit 2).

STPNOC proposes to implement an alternative to the requirement of ASME Section XI IWB-2412, Inspection Program B, that volumetric examination of RV Examination Categories B-A and B-D be performed once each 10-year ISI interval. The current third ISI interval ends September 24, 2020 and October 18, 2020, respectively. STPNOC proposes to perform the third ASME Section XI Category B-A and B-D examinations in the fourth ISI interval no later than 2026 and 2027, respectively.

STPNOC requests NRC review and approval of this alternative request by February 2019, to support the use of the proposed alternative.

The enclosed Relief Request RR-ENG-3-14 provides the basis and supporting information for the proposed alternative.

There are no commitments in this letter.

If there are any questions, please contact Craig Younger at 361-972-8186, or Kyle Wallis at 361-972-4687.

Michael Rage
General Manager
of Engineering

rjg

Enclosure: Proposed Alternative to ASME Section XI Reactor Vessel Inservice Inspection Intervals (Relief Request RR-ENG-3-14)

STI: 34596186

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Enclosure

Proposed Alternative to ASME Section XI Reactor Vessel Inservice Inspection Intervals (Relief Request RR-ENG-3-14)

Proposed Alternative to ASME Section XI Reactor Vessel Inservice Inspection Intervals (Relief Request RR-ENG-3-14)

A. ASME Code Component(s) Affected

The affected components are the South Texas Project (STP) Unit 1 and Unit 2 reactor vessels (RV), specifically, the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, 2004 Edition (Reference 1) examination categories and item numbers covering examinations of the RV. 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 No.	Description
B-A	B1.10	Shell Welds
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.20	Head Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional 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

(Throughout this request the above examination categories are referred to as “the subject examinations” and the ASME BPV Code, Section XI, is referred to as “the Code.”)

B. Applicable ASME Code Edition and Addenda

ASME Code Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” 2004 Edition.

Table 1 - Applicable ASME Code Edition and Addenda

Unit	Interval	Edition	Start	End
Unit 1	3	2004	September 25, 2010	September 24, 2020
Unit 2	3	2004	October 19, 2010	October 18, 2020
Unit 1	4	To Be Determined	September 25, 2020	August 20, 2027
Unit 2	4	To Be Determined	October 19, 2020	December 15, 2028

C. Applicable ASME Code Requirement

IWB-2412, Inspection Program B, requires volumetric examination of essentially 100% of reactor vessel pressure retaining welds identified in Table IWB-2500-1 once each 10-year interval. The STP Unit 1 and STP Unit 2 third 10-year inservice inspection (ISI) interval is scheduled to end on September 24, 2020 and October 18, 2020, respectively. The applicable Code for the fourth 10-year ISI interval will be selected in accordance with the requirements of 10 CFR 50.55a.

D. Reason for Relief from Code Requirements

An alternative is requested from the requirement of IWB-2412, Inspection Program B, that volumetric examination of essentially 100% of reactor vessel pressure retaining examination category B-A and B-D welds be performed once each 10-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 result in a reduction in man-rem exposure and examination costs.

E. Proposed Alternative and Basis for Use:

South Texas Project Nuclear Operating Company (STPNOC) proposes not to perform the ASME Code required volumetric examination of the STP Unit 1 and STP Unit 2 reactor vessels subject examinations for the third inservice inspection, currently scheduled for 2020 and 2019, respectively. STPNOC will perform the third ASME Code required volumetric examination of the STP Unit 1 and Unit 2 reactor vessels subject examinations in the fourth inservice inspection interval no later than 2026 and 2027, respectively. The proposed inspection dates for STP Unit 1 and Unit 2 are a deviation from implementation plan presented in OG-10-238 (Reference 2). For Unit 1, the impact to the implementation plan in OG-10-238 would increase the number of inspections in 2026 from two to three, and decrease the number of inspections in 2029 from five to four. For Unit 2, the impact to the implementation plan in OG-10-238 would increase the number of inspections in 2027 from seven to eight, and decrease the number of inspections in 2030 from five to four. Based on Figures 3 and 4 of OG-10-238, this proposed inspection schedule is considered to have a minor impact on the inspection plan and the distribution of inspections over time.

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 (Reference 3).

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 STP 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. The parameters for STP Unit 1 and Unit 2 are bounded by the results of the Westinghouse pilot plant and qualifies STP Unit 1 and Unit 2 for ISI interval extensions.

Table 1a below lists the critical parameters investigated in the WCAP and compares the results of the Westinghouse pilot plant to those of STP Unit 1. Tables 2a and 3a 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 STP 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)	8.27E-16 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 2a below provides a summary of the latest reactor vessel inspection for STP Unit 1 and an evaluation of the recorded indications. This information confirms that satisfactory examinations have been performed on the STP Unit 1 reactor vessel.

Table 2a: Additional Information Pertaining to Reactor Vessel Inspection for STP Unit 1	
Inspection methodology:	The latest ISI for STP Unit 1 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, 1995 Edition with 1996 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 methodology.
Number of past inspections:	Two 10-year inservice inspections have been performed.
Number of indications found:	One indication identified in the beltline region of the RV was recorded during the last ISI. This subsurface indication is located in the intermediate to lower shell circumferential weld seam (Item 9 in Table 3a), and is allowable per Table IWB-3510-1 of Section XI of the ASME Code. This indication is not within the inner 1/10th or 1 inch of the reactor vessel thickness. Therefore, it is inherently acceptable per the requirements of the Alternate Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61a (Reference 7).
Proposed inspection schedule for balance of plant life:	The third inservice inspection is currently scheduled for 2020. This inspection will be performed no later than 2026 refueling outage. The proposed inspection date is a deviation from the latest revised implementation plan, OG-10-238 (Reference 2). The impact to the implementation plan in OG-10-238 would increase the number of inspections in 2026 from two to three, and decrease the number of inspections in 2029 from five to four. Based on Figures 3 and 4 of OG-10-238, this proposed inspection schedule is considered to have a minor impact on the future inspection plan and the distribution of inspections over time.

Table 3a summarizes the inputs and outputs for the calculation of through-wall cracking frequency for STP Unit 1.

Table 3a: Details of TWCF Calculation for STP Unit 1 at 34 Effective Full Power Years (EFPY)

Inputs									
Reactor Coolant System Temperature. T_C [°F]: N/A								T_{wall} [inches]:	9.10 (measured)
No.	Region and Component Description	Material ID	Material Heat No.	Cu [wt%]	Ni [wt%]	R.G. 1.99 Pos.	CF [°F]	$RT_{NDT(U)}$ [°F]	Fluence [n/cm², E > 1.0 MeV][(1)]
1	Intermediate Shell Plate	R1606-1	B-8120-2	0.04	0.63	1.1	26.0	10	2.51E+19
2	Intermediate Shell Plate	R1606-2	B-8120-1	0.04	0.61	2.1	29.6	0	2.51E+19
3	Intermediate Shell Plate	R1606-3	C-4326-2	0.05	0.62	1.1	31.0	10	2.51E+19
4	Lower Shell Plate	R1622-1	B-9566-2	0.05	0.61	1.1	31.0	-30	2.51E+19
5	Lower Shell Plate	R1622-2	B-9575-2	0.07	0.64	1.1	44.0	-30	2.51E+19
6	Lower Shell Plate	R1622-3	B-9575-1	0.05	0.66	1.1	31.0	-30	2.51E+19
7	Intermediate Shell Longitudinal Weld Seams	101-124A, B, C	Heat # 89476, Linde 0091 Flux, Lot # 0145	0.022	0.071	2.1	31.2	-50	2.51E+19
8	Lower Shell Longitudinal Weld Seams	101-142A, B, C	Heat # 89476, Linde 0091 Flux, Lot # 0145	0.022	0.071	2.1	31.2	-50	2.51E+19
9	Intermediate to Lower Shell Circumferential Weld Seam	101-171	Heat # 89476, Linde 124 Flux, Lot # 1061	0.022	0.071	2.1	31.2	-70	2.51E+19

Outputs							
Methodology Used to Calculate ΔT_{30}: Regulatory Guide 1.99, Revision 2 (Reference 8)							
	Controlling Material Region No. (From Above)	α_{XX}	RT_{MAX-XX} [°R]	Fluence [n/cm², E >1.0 MeV]	FF (Fluence Factor)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Axial Weld - AW	3	2.5	508	2.51E+19	1.2472	38.66	0.000E+00
Limiting Plate - PL	3	2.5	508	2.51E+19	1.2472	38.66	3.308E-16
Limiting Circumferential Weld - CW	3	2.5	508	2.51E+19	1.2472	38.66	0.000E+00
$TWCF_{95-TOTAL} = (\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW})$:							**8.27E-16**

(1) Fluence values based on plant-specific analysis of record, WCAP-17482-NP (Reference 9)

Table 1b below lists the critical parameters investigated in the WCAP and compares the results of the Westinghouse pilot plant to those of STP Unit 2. Tables 2b and 3b provide additional information that was requested by the NRC and included in Appendix A of Reference 4.

Table 1b: Critical Parameters for the Application of Bounding Analysis for STP 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)	1.09E-16 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 2b below provides a summary of the latest reactor vessel inspection for STP Unit 2 and an evaluation of the recorded indications. This information confirms that satisfactory examinations have been performed on the STP Unit 2 reactor vessel.

Table 2b: Additional Information Pertaining to Reactor Vessel Inspection for STP Unit 2	
Inspection methodology:	The latest ISI for STP Unit 2 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, 1995 Edition with 1996 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 methodology.
Number of past inspections:	Two 10-year inservice inspections have been performed.
Number of indications found:	One indication identified in the beltline region of the RV was recorded during the last ISI. This subsurface indication is located in a lower longitudinal weld seam (Item 8 in Table 3b), and is allowable per Table IWB-3510-1 of Section XI of the ASME Code. This indication is not within the inner 1/10th or 1 inch of the reactor vessel thickness. Therefore, it is 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 is currently scheduled for 2019. This inspection will be performed no later than 2027 refueling outage. The proposed inspection date is a deviation from the latest revised implementation plan, OG-10-238 (Reference 2). The impact to the implementation plan in OG-10-238 would increase the number of inspections in 2027 from seven to eight, and decrease the number of inspections in 2030 from five to four. Based on Figures 3 and 4 of OG-10-238, this proposed inspection schedule is considered to have a minor impact on the future inspection plan and the distribution of inspections over time.

Table 3b summarizes the inputs and outputs for the calculation of through-wall cracking frequency for STP Unit 2.

Table 3b: Details of TWCF Calculation for STP Unit 2 at 34 Effective Full Power Years (EFPY)									
Inputs									
Reactor Coolant System Temperature, T_C [°F]: N/A								T_{wall} [inches]:	9.10 (measured)
No.	Region and Component Description	Material ID	Material Heat No.	Cu [wt%]	Ni [wt%]	R.G. 1.99 Pos.	CF [°F]	$RT_{NDT(U)}$ [°F]	Fluence [n/cm^2, E > 1.0 MeV][(1)]
1	Intermediate Shell Plate	R2507-1	NR 62067-1	0.04	0.65	2.1	33.4	-10	2.50E+19
2	Intermediate Shell Plate	R2507-2	NR 62230-1	0.05	0.64	1.1	31.0	-10	2.50E+19
3	Intermediate Shell Plate	R2507-3	NR 62248-1	0.05	0.61	1.1	31.0	-40	2.50E+19
4	Lower Shell Plate	R3022-1	NR 64647-1	0.03	0.63	1.1	20.0	-30	2.50E+19
5	Lower Shell Plate	R3022-2	NR 64627-1	0.04	0.61	1.1	26.0	-40	2.50E+19
6	Lower Shell Plate	R3022-3	NR 64445-1	0.04	0.60	1.1	26.0	-40	2.50E+19
7	Intermediate Shell Longitudinal Weld Seams	101-124A, B, C	Heat # 90209, Linde 0091 Flux, Lot # 1054	0.044	0.126	1.1	37.8	-70	2.50E+19
8	Lower Shell Longitudinal Weld Seams	101-142A, B, C	Heat # 90209, Linde 124 Flux, Lot # 1061	0.044	0.126	1.1	37.8	-70	2.50E+19
9	Intermediate to Lower Shell Circumferential Weld Seam	101-171	Heat # 90209, Linde 124 Flux, Lot # 1061	0.044	0.126	1.1	37.8	-70	2.50E+19

Outputs							
Methodology Used to Calculate ΔT_{30}: Regulatory Guide 1.99, Revision 2 (Reference 8)							
	Controlling Material Region No. (From Above)	α_{xx}	RT_{MAX-XX} [°R]	Fluence [n/cm^2, E >1.0 MeV]	FF (Fluence Factor)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Axial Weld - AW	1	2.5	491	2.50E+19	1.2462	41.62	0.000E+00
Limiting Plate - PL	1	2.5	491	2.50E+19	1.2462	41.62	4.365E-17
Limiting Circumferential Weld - CW	1	2.5	491	2.50E+19	1.2462	41.62	0.000E+00
$TWCF_{95-TOTAL} = (\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW})$:							1.09E-16

(1) Fluence values based on plant-specific analysis of record, WCAP-17636-NP (Reference 10)

F. Duration of Proposed Alternative

This request is applicable to the STP Unit 1 and Unit 2 inservice inspection programs for the third and fourth 10-year inspection intervals.

G. Precedents

Relief from this examination requirement to apply the proposed alternative at the South Texas Project Unit 1 and Unit 2 was previously approved by the NRC for the following (with ADAMS Accession No. references):

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.

H. References

1. ASME Boiler and Pressure Vessel Code, Section XI, 2004 Edition, Rules for Inservice Inspection of Nuclear Power Plants, 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 ML023240437).

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, (ADAMS Accession No. 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, 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. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988, (ADAMS Accession No. ML003740284).

9. Westinghouse Report, WCAP-17482-NP, Revision 0, "Analysis of Capsule W from the South Texas Project Nuclear Operating Company South Texas Unit 1 Reactor Vessel Radiation Surveillance Program," May 11, 2012.

10. Westinghouse Report, WCAP-17636-NP, Revision 0, "Analysis of Capsule W from the South Texas Project Nuclear Operating Company South Texas Unit 2 Reactor Vessel Radiation Surveillance Program," October 22, 2012.