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Docket Nos.: 50-424
50-425

NL-12-1446

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

Vogtle Electric Generating Plant – Units 1 & 2
Proposed Inservice Inspection Alternatives VEGP-ISI-ALT-05 and
VEGP-ISI-ALT-06 for Implementation of
Extended Reactor Vessel Inservice Inspection Intervals

Ladies and Gentlemen:

In accordance with 10 CFR 50.55a(a)(3)(i), Southern Nuclear Operating Company (SNC) hereby requests Nuclear Regulatory Commission (NRC) approval of proposed inservice inspection (ISI) Alternatives VEGP-ISI-ALT-05, Version 1 and VEGP-ISI-ALT-06, Version 1. These alternatives propose one-time extensions of reactor vessel examinations from a ten-year examination period to a twenty-year examination period for their respective unit. Specifically, these alternatives request one-time extensions to selected ASME Examination Category B-A reactor vessel welds and selected ASME Examination Category B-D reactor vessel nozzle welds and reactor vessel inside radius sections.

The approval of VEGP-ISI-ALT-05 will move examinations currently scheduled for Unit 1 Reload 19 (1R19, Fall 2015) to either 1R25 (Fall 2024) or 1R26 (Spring 2026). The approval of VEGP-ISI-ALT-06 will move examinations currently scheduled for Unit 2 Reload (2R18, Spring 2016) to either 2R24 (Spring 2025) or 2R25 (Fall 2026). Approval of these alternatives is requested by September 1, 2013.

This letter contains no NRC commitments. If you have any questions, please contact me at (205) 992-7673.

Respectfully submitted,

Mark J Ajluni

M. J. Ajluni
Nuclear Licensing Director

MJA/RMJ/lac

Enclosures:
1. Proposed Alternative VEGP-ISI-ALT-05, Version 1.0, in Accordance With 10 CFR 50.55a(a)(3)(i)
2. Proposed Alternative VEGP-ISI-ALT-06, Version 1.0, in Accordance With 10 CFR 50.55a(a)(3)(i)

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. T. E. Tynan, Vice President – Vogtle
Mr. B. L. Ivey, Vice President – Regulatory Affairs
Mr. B. J. Adams, Vice President – Fleet Operations
RType: CVC7000

U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
Mr. R. E. Martin, NRR Senior Project Manager - Vogtle
Mr. L. M. Cain, Senior Resident Inspector – Vogtle

Vogtle Electric Generating Plant – Unit 1
Proposed Alternative for the Third ISI Interval

Enclosure 1

Proposed Alternative VEGP-ISI-ALT-05 Version 1.0,
in Accordance with 10 CFR 50.55a(a)(3)(i)

Enclosure 1

Proposed Alternative VEGP-ISI-ALT-05 Version 1.0,
in Accordance with 10 CFR 50.55a(a)(3)(i)

Plant Site-Unit:	Vogtle Electric Generating Plant (VEGP) - Unit 1.
Interval Dates:	3rd Inservice Inspection (ISI) Interval – May 31, 2007 through May 30, 2017.
Requested Date for Approval :	Approval is requested by September 1, 2013.
ASME Code Components Affected:	The affected components are Examination Category B-A, Items B1.11, B1.12, B1.21, B1.22, B1.30, and B1.40 reactor vessel (RV) welds, and Examination Category B-D, Items B3.90 and B3.100 RV nozzle welds and nozzle inside radius section. The specific components are provided in Table 4.
Applicable Code Edition and Addenda:	The applicable code edition and addenda (for the 3rd ISI interval) is ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition with the 2003 addenda (Reference 1).
Applicable Code Requirements:	Table IWB-2500-1 requires volumetric examination of the affected RV components once each ten-year interval.
Reason for Request:	The Westinghouse pilot plant RV analysis defined in WCAP-16168-NP-A, Revision 3 (Reference 2) utilizes probabilistic fracture mechanics and risk analysis methods to justify extending the ISI interval for reactor vessel welds (Examination Category B-A) and nozzle-to-vessel welds and nozzle inside radius section (Examination Category B-D) from 10 years to 20 years. An analysis has been performed showing that VEGP - Unit 1, which is a Westinghouse 4-Loop plant, is bounded by the pilot plant parameters defined in Reference 2. Therefore, Southern Nuclear Operating Company (SNC) is requesting approval of this alternative to allow the use of the ISI interval extension for the affected VEGP - Unit 1 components.
Proposed Alternative:	SNC is requesting a one-time extension of the ISI interval from 10 years to 20 years for VEGP - Unit 1 Examination Category B-A reactor vessel welds and Examination Category B-D nozzle-to-vessel welds and nozzle inside radius section. Specifically, this proposed alternative would permit the deferral of the ASME Code required Examination Category B-A and B-D volumetric examinations currently scheduled for the Fall of 2015 (3rd period of 3rd interval) until 2026, plus or minus one refueling cycle (3rd period of 4th interval). The required examinations would subsequently be performed using the Section XI Code in effect for the 4th interval.

Enclosure 1

**Proposed Alternative VEGP-ISI-ALT-05 Version 1.0,
in Accordance with 10 CFR 50.55a(a)(3)(i)**

Basis for Use:	The methodology used to demonstrate the acceptability of extending the inspection intervals for Examination Category B-A and B-D components is contained in WCAP-16168-NP-A, Revision 3 (Reference 2). This methodology was used to develop a pilot plant risk analysis for Westinghouse (W), Combustion Engineering (CE), and Babcock and Wilcox (B&W) RV designs and is an extension of the work that was performed as part of the NRC Pressurized Thermal Shock (PTS) Risk Re-Evaluation (Reference 3). Reference 2 used the estimated through wall cracking frequency (TWCF) as a measure of the risk of RV failure, and it was demonstrated that the inspection interval for the affected components can be extended from 10 years to 20 years while meeting the change in risk guidelines found in Regulatory Guide 1.174 (Reference 4). Reference 2 was approved by the NRC in a July 26, 2011 safety evaluation (ML111600303). Section 3.4 of the safety evaluation provides the requirements for a utility to submit an alternative in accordance with 10 CFR 50.55a(a)(3)(i) to use the Reference 2 for a plant specific evaluation. These requirements are addressed below: 1. Licensees must demonstrate that the embrittlement of their RV is within the envelope used in the supporting analyses. A plant specific analysis was performed that demonstrated that VEGP – Unit 1 RV parameters are bounded by corresponding pilot plant parameters. The critical parameters are identified in Table 1. Table 3 provides detailed information relative to the calculation of the TWCF. 2. Licensees must report whether the frequency of the limiting design basis transients during prior operation are less than the frequency identified in the PWROG (Reference 2) fatigue analysis. As shown in Table 1, the frequency of the VEGP – Unit 1 limiting design basis transients are bounded by the frequency identified in the PWROG (Reference 2) fatigue analysis. 3. Licensees must report the results of prior ISI of RV welds and the proposed schedule for the next 20 year ISI interval. The results of the previous RV inspections for VEGP – Unit 1 are provided in Table 2. This information confirms that satisfactory examinations have been performed on the VEGP – Unit 1 RV. 4. In the request for an alternative, each licensee shall identify the years in which the future inspections will be performed. The VEGP – Unit 1 RV examinations currently scheduled for 2015 will be deferred until 2026, plus or minus one refueling cycle. The dates provided must be within plus or minus one refueling cycle of the date identified in PWROG letter OG-10-238, dated July 12, 2010 (Reference 5).

Enclosure 1

**Proposed Alternative VEGP-ISI-ALT-05 Version 1.0,
in Accordance with 10 CFR 50.55a(a)(3)(i)**

	VEGP – Unit 1 is bounded by the pilot plant application because the total TWCF for VEGP – Unit 1 was calculated as 7.66E-14. This value is less than the value of 1.76E-08 which was calculated for the Westinghouse pilot plant in Reference 2; therefore, the use of this proposed alternative will provide an acceptable level of quality and safety. Therefore, it is requested that the NRC authorize this proposed alternative in accordance with 10 CFR 50.55a(a)(3)(i).
Duration of Proposed Alternative:	The 3rd and 4th Intervals.
Precedents:	1. Donald C. Cook Nuclear Plant, Unit 2 – Evaluation of Relief Request (ISIR-29) to Extend the Third 10-Year Inservice Inspection Interval for Reactor Vessel Weld Examination (TAC MD9934) Donald C. Cook Nuclear Power Plant, Unit 2 – NRC Safety Evaluation Report dated June 8, 2009 (ML091260163). 2. Safety Evaluation for Relief Requests ISI-090 & 021 Reactor Vessel Weld Examination Extension – Calvert Cliffs Nuclear Power Plant, Unit 2 (TAC Nos. MD9773 and MD9774). Calvert Cliffs Nuclear Power Plant Unit No. 2 – NRC Safety Evaluation Report dated April 8, 2009 (ML090920077). 3. Palisades Plant – Evaluation of Relief Request to Extend the Third 10-Year Inservice Inspection Interval For Reactor Weld Examinations (TAC NO. MD9265) Palisades Plant – NRC Safety Evaluation Report dated February 11, 2009 (ML 090120896). 4. R.E. Ginna Nuclear Power Plant: Safety Evaluation for Relief Request No. 18, Reactor Vessel Weld Examination Extension (TAC NO. MD9962) – R.E. Ginna Plant – NRC Safety Evaluation Report dated July 31, 2009 (ML092080229). 5. Kalyanam, N., NRC, to Vice President, Entergy Operations, "Waterford Steam Electric Station, Unit 3 – Withdrawal of an Amendment Request (TAC NO. MD9669)," dated June 12, 2009 (ML091600132 and ML091600158). 6. 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), NRC Safety Evaluation Report dated July 12, 2010 (ML101750402).
References:	1. American Society of Mechanical Engineers. ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition with the 2003 Addenda. 2. WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," October 2011. (Approved under NRC SER

	ML111600303). 3. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock," March 2010. 4. 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," November 2002. 5. 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. 6. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004. 7. Nuclear Regulatory Commission, Code of Federal Regulations, 10 CFR Part 50.61a, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," 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," May 1988. 9. WCAP-17076-NP, Revision 0, "Vogtle Unit 1, Heatup and Cooldown Limit Curves for Normal Operation," July 2009.
Status:	Awaiting NRC approval.

Enclosure 1

Proposed Alternative VEGP-ISI-ALT-05 Version 1.0, in Accordance with 10 CFR 50.55a(a)(3)(i)

Table 1
Critical Parameters for the Application of the Bounding Analysis as Applied to VEGP – Unit 1

Parameter	Pilot Plant Basis	VEGP – Unit 1 Basis	Additional Evaluation Required?
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are applicable	NRC PTS Risk Study (Reference 3)	PTS Generalization Study (Reference 6)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 2)	7.66E-14 Events per year (Calculated using Reference 2)	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldown cycles per year (Reference 2)	Bounded by 7 heatup/cooldown cycles per year (1)	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 2)	Single Layer	No

Note:

(1) Per the VEGP Application for License Renewal, after 60 years of operation, the projected number of design basis transients is below the number specified in the 40-year design bases. As a result, VEGP – Unit 1 is conservatively bounded by 7 heatup/cooldown cycles per year.

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Proposed Alternative VEGP-ISI-ALT-05 Version 1.0, in Accordance with 10 CFR 50.55a(a)(3)(i)

Table 2
Additional Information Pertaining to Reactor Vessel Inspections for VEGP – Unit 1

<table>
<tr><td>Inspection methodology:</td><td>The latest ISI was conducted in accordance with the ASME Code, Section XI 1989 Edition. Examinations of Category B-A and B-D welds were performed to ASME Section XI Appendix VIII, 1995 Edition with the 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 requirements.</td></tr>
<tr><td>Number of past inspections:</td><td>Two 10-Year inservice inspections have been performed.</td></tr>
<tr><td>Number of indications found:</td><td>There were two indications identified in the beltline region during the most recent inservice inspection. These indications are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. Only one indication is within the inner 1/10th or 1 inch of the reactor vessel thickness. This indication is located along the axial weld fusion line but is conservatively considered to be in intermediate shell plate B8805-3. This indication is acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 7), since the flaw 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.
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in)</th><th rowspan="2">Scaled maximum number of plate flaws</th><th rowspan="2">Number of plate flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0.075</td><td>0.375</td><td>73</td><td>1 (1/0)</td></tr>
<tr><td>0.125</td><td>0.375</td><td>29</td><td>1 (1/0)</td></tr>
<tr><td>0.175</td><td>0.375</td><td>8</td><td>1 (1/0)</td></tr>
<tr><td>0.225</td><td>0.375</td><td>3</td><td>1 (1/0)</td></tr>
<tr><td>0.275</td><td>0.375</td><td>1</td><td>1 (1/0)</td></tr>
</table>
This indication is 1.6 inches in length, 0.32 inches in through-wall extent (2a dimension), and is embedded with an 'S' dimension of 0.55 inches as measured from the cladding-to-base-metal interface.

This indication was observed in the second ISI interval inspection only; it was not recorded during the first ISI interval examinations. The second ISI interval examination was an Appendix VIII (PDI) examination with a much higher sensitivity than the first ISI interval Section XI/Regulatory Guide 1.150 examination.</td></tr>
<tr><td>Proposed inspection schedule for balance of plant life:</td><td>The third inservice inspection is scheduled for 2015. This inspection will be performed in 2026, plus or minus one refueling cycle. The proposed inspection date is consistent with the latest revised implementation plan, OG-10-238 (Reference 5), plus or minus one refueling outage.</td></tr>
</table>

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Proposed Alternative VEGP-ISI-ALT-05 Version 1.0, in Accordance with 10 CFR 50.55a(a)(3)(i)

Table 3
Details of the Through-Wall Cracking Frequency Calculation for VEGP - Unit 1 at 56.3 EFPY

Inputs

Reactor Coolant System Temperature, T_C[°F]:	N/A	T_{wall} [inches]:	8.781

No.	Region and Component Description	Material Heat No.	Cu[1] [wt%]	Ni[1] [wt%]	R.G. 1.99 Pos.[1]	CF[1] [°F]	$RT_{NDT(u)}$[1] [°F]	Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]
1	Intermediate Shell Plate B8805-1	C-0613-1	0.083	0.597	1.1	53.1	0	3.53
2	Intermediate Shell Plate B8805-2	C-0613-2	0.083	0.610	1.1	53.1	20	3.53
3	Intermediate Shell Plate B8805-3	C-0623-1	0.062	0.598	2.1	47.0	30	3.53
4	Lower Shell Plate B8606-1	C-2146-1	0.053	0.593	1.1	32.8	20	3.53
5	Lower Shell Plate B8606-2	C-2146-2	0.057	0.60	1.1	35.2	20	3.53
6	Lower Shell Plate B8606-3	C-2085-2	0.067	0.623	1.1	41.9	10	3.53
7	Inter. Shell Long. Weld 101-124A	83653	0.042	0.102	2.1	23.3	-80	1.67
8	Inter. Shell Long. Weld 101-124B	83653	0.042	0.102	2.1	23.3	-80	3.17
9	Inter. Shell Long. Weld 101-124C	83653	0.042	0.102	2.1	23.3	-80	3.17
10	Lower Shell Long. Weld 101-142A	83653	0.042	0.102	2.1	23.3	-80	3.17
11	Lower Shell Long. Weld 101-142B	83653	0.042	0.102	2.1	23.3	-80	1.67
12	Lower Shell Long. Weld 101-142C	83653	0.042	0.102	2.1	23.3	-80	3.17
13	Inter. To Lower Shell Circ. Weld 101-171	83653	0.042	0.102	2.1	23.3	-80	3.53

Outputs

Methodology Used to Calculate ΔT_{30}:	Regulatory Guide 1.99, Revision 2 (Reference 8)

	Controlling Material Region No. (From Above)	RT_{MAX-XX} [°R]	Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]	FF (Fluence Factor)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Axial Weld - AW	3	550.96	3.17	1.304	61.29	0.00E+00
Limiting Plate - PL	3	552.12	3.53	1.329	62.45	3.06E-14
Circumferential Weld - CW	3	552.12	3.53	1.329	62.45	0.00E+00
$TWCF_{95-TOTAL}(\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW})$:						7.66E-14

Note:

(1) Reference 9

Enclosure 1

Proposed Alternative VEGP-ISI-ALT-05 Version 1.0, in Accordance with 10 CFR 50.55a(a)(3)(i)

Table 4
List of Affected Components for VEGP – Unit 1

ASME CATEGORY	ASME ITEM NUMBER	COMPONENT ID	DESCRIPTION
B-A	B1.11	11201-V6-001-W04	UPPER SHELL TO INTERMEDIATE SHELL WELD
B-A	B1.11	11201-V6-001-W05	INTERMEDIATE SHELL TO LOWER SHELL WELD
B-A	B1.11	11201-V6-001-W06	LOWER SHELL TO BOTTOM HEAD TORUS WELD
B-A	B1.12	11201-V6-001-W12	UPPER SHELL LONGITUDINAL WELD AT 42 DEGREES
B-A	B1.12	11201-V6-001-W13	UPPER SHELL LONGITUDINAL WELD AT 162 DEGREES
B-A	B1.12	11201-V6-001-W14	UPPER SHELL LONGITUDINAL WELD AT 282 DEGREES
B-A	B1.12	11201-V6-001-W15	INTERMEDIATE SHELL LONGITUDINAL WELD AT 0 DEGREES
B-A	B1.12	11201-V6-001-W16	INTERMEDIATE SHELL LONGITUDINAL WELD AT 120 DEGREE
B-A	B1.12	11201-V6-001-W17	INTERMEDIATE SHELL LONGITUDINAL WELD AT 240 DEGREE
B-A	B1.12	11201-V6-001-W18	LOWER SHELL LONGITUDINAL WELD AT 60 DEGREES
B-A	B1.12	11201-V6-001-W19	LOWER SHELL LONGITUDINAL WELD AT 180 DEGREES
B-A	B1.12	11201-V6-001-W20	LOWER SHELL LONGITUDINAL WELD AT 300 DEGREES
B-A	B1.21	11201-V6-001-W01	CLOSURE HEAD DOME TO TORUS WELD
B-A	B1.21	11201-V6-001-W07	BOTTOM HEAD TORUS TO BOTTOM HEAD DOME WELD
B-A	B1.22	11201-V6-001-W08	CLOSURE HEAD TORUS MERIDIONAL WELD AT 45 DEGREES
B-A	B1.22	11201-V6-001-W09	CLOSURE HEAD TORUS MERIDIONAL WELD AT 135 DEGREES
B-A	B1.22	11201-V6-001-W10	CLOSURE HEAD TORUS MERIDIONAL WELD AT 225 DEGREES
B-A	B1.22	11201-V6-001-W11	CLOSURE HEAD TORUS MERIDIONAL WELD AT 315 DEGREES
B-A	B1.22	11201-V6-001-W21	BOTTOM HEAD TORUS MERIDIONAL WELD AT 0 DEGREES
B-A	B1.22	11201-V6-001-W22	BOTTOM HEAD TORUS MERIDIONAL WELD AT 90 DEGREES
B-A	B1.22	11201-V6-001-W23	BOTTOM HEAD TORUS MERIDIONAL WELD AT 180 DEGREES
B-A	B1.22	11201-V6-001-W24	BOTTOM HEAD TORUS MERIDIONAL WELD AT 270 DEGREES
B-A	B1.30	11201-V6-001-W03	VESSEL SHELL TO FLANGE WELD
B-A	B1.40	11201-V6-001-W02	CLOSURE HEAD TORUS TO FLANGE WELD
B-D	B3.90	11201-V6-001-W25	VESSEL TO OUTLET NOZZLE N1 WELD AT 22 DEGREES
B-D	B3.90	11201-V6-001-W26	VESSEL TO INLET NOZZLE N2 WELD AT 67 DEGREES
B-D	B3.90	11201-V6-001-W27	VESSEL TO INLET NOZZLE N3 WELD AT 113 DEGREES
B-D	B3.90	11201-V6-001-W28	VESSEL TO OUTLET NOZZLE N4 WELD AT 158 DEGREES
B-D	B3.90	11201-V6-001-W29	VESSEL TO OUTLET NOZZLE N5 WELD AT 202 DEGREES
B-D	B3.90	11201-V6-001-W30	VESSEL TO INLET NOZZLE N6 WELD AT 247 DEGREES
B-D	B3.90	11201-V6-001-W31	VESSEL TO INLET NOZZLE N7 WELD AT 293 DEGREES
B-D	B3.90	11201-V6-001-W32	VESSEL TO OUTLET NOZZLE N8 WELD AT 338 DEGREES
B-D	B3.100	11201-V6-001-IR01	OUTLET NOZZLE N1 INNER RADIUS AT 22 DEGREES
B-D	B3.100	11201-V6-001-IR02	INLET NOZZLE N2 INNER RADIUS AT 67 DEGREES
B-D	B3.100	11201-V6-001-IR03	INLET NOZZLE N3 INNER RADIUS AT 113 DEGREES
B-D	B3.100	11201-V6-001-IR04	OUTLET NOZZLE N4 INNER RADIUS AT 158 DEGREES
B-D	B3.100	11201-V6-001-IR05	OUTLET NOZZLE N5 INNER RADIUS AT 202 DEGREES
B-D	B3.100	11201-V6-001-IR06	INLET NOZZLE N6 INNER RADIUS AT 247 DEGREES
B-D	B3.100	11201-V6-001-IR07	INLET NOZZLE N7 INNER RADIUS AT 293 DEGREES
B-D	B3.100	11201-V6-001-IR08	OUTLET NOZZLE N8 INNER RADIUS AT 338 DEGREES

Vogtle Electric Generating Plant – Unit 2
Proposed Alternative for the Third ISI Interval

Enclosure 2

Proposed Alternative VEGP-ISI-ALT-06 Version 1.0,
in Accordance with 10 CFR 50.55a(a)(3)(i)

Enclosure 2

Proposed Alternative VEGP-ISI-ALT-06 Version 1.0,
in Accordance with 10 CFR 50.55a(a)(3)(i)

Plant Site-Unit:	Vogtle Electric Generating Plant (VEGP) - Unit 2.
Interval Dates:	3rd Inservice Inspection (ISI) Interval – May 31, 2007 through May 30, 2017.
Requested Date for Approval :	Approval is requested by September 1, 2013.
ASME Code Components Affected:	The affected components are Examination Category B-A, Items B1.11, B1.12, B1.21, B1.22, B1.30, and B1.40 reactor vessel (RV) welds, and Examination Category B-D, Items B3.90 and B3.100 RV nozzle welds and nozzle inside radius section. The specific components are provided in Table 4.
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Reason for Request:	The Westinghouse pilot plant RV analysis defined in WCAP-16168-NP-A, Revision 3 (Reference 2) utilizes probabilistic fracture mechanics and risk analysis methods to justify extending the ISI interval for reactor vessel welds (Examination Category B-A) and nozzle-to-vessel welds and nozzle inside radius section (Examination Category B-D) from 10 years to 20 years. An analysis has been performed showing that VEGP - Unit 2, which is a Westinghouse 4-Loop plant, is bounded by the pilot plant parameters defined in Reference 2. Therefore, Southern Nuclear Operating Company (SNC) is requesting approval of this alternative to allow the use of the ISI interval extension for the affected VEGP - Unit 2 components.
Proposed Alternative:	SNC is requesting a one-time extension of the ISI interval from 10 years to 20 years for VEGP – Unit 2 Examination Category B-A reactor vessel welds and Examination Category B-D nozzle-to-vessel welds and nozzle inside radius section. Specifically, this proposed alternative would permit the deferral of the ASME Code required Examination Category B-A and B-D volumetric examinations currently scheduled for the Spring of 2016 (3rd period of 3rd interval) until 2026, plus or minus one refueling cycle (3rd period of 4th interval). The required examinations would subsequently be performed using the Section XI Code in effect for the 4th interval.

Enclosure 2

**Proposed Alternative VEGP-ISI-ALT-06 Version 1.0,
in Accordance with 10 CFR 50.55a(a)(3)(i)**

Basis for Use:	The methodology used to demonstrate the acceptability of extending the inspection intervals for Examination Category B-A and B-D components is contained in WCAP-16168-NP-A, Revision 3 (Reference 2). This methodology was used to develop a pilot plant risk analysis for Westinghouse (W), Combustion Engineering (CE), and Babcock and Wilcox (B&W) RV designs and is an extension of the work that was performed as part of the NRC Pressurized Thermal Shock (PTS) Risk Re-Evaluation (Reference 3). Reference 2 used the estimated through wall cracking frequency (TWCF) as a measure of the risk of RV failure, and it was demonstrated that the inspection interval for the affected components can be extended from 10 years to 20 years while meeting the change in risk guidelines found in Regulatory Guide 1.174 (Reference 4). Reference 2 was approved by the NRC in a July 26, 2011 safety evaluation (ML111600303). Section 3.4 of the safety evaluation provides the requirements for a utility to submit an alternative in accordance with 10 CFR 50.55a(a)(3)(i) to use Reference 2 for a plant specific evaluation. These requirements are addressed below: 1. Licensees must demonstrate that the embrittlement of their RV is within the envelope used in the supporting analyses. A plant specific analysis was performed that demonstrated that VEGP - Unit 2 RV parameters are bounded by corresponding pilot plant parameters. The critical parameters are identified in Table 1. Table 3 provides detailed information relative to the calculation of the TWCF. 2. Licensees must report whether the frequency of the limiting design basis transients during prior operation are less than the frequency identified in the PWROG (Reference 2) fatigue analysis. As shown in Table 1, the frequency of the VEGP – Unit 2 limiting design basis transients are bounded by the frequency identified in the PWROG (Reference 2) fatigue analysis. 3. Licensees must report the results of prior ISI of RV welds and the proposed schedule for the next 20 year ISI interval. The results of the previous RV inspections for VEGP – Unit 2 are provided in Table 2. This information confirms that satisfactory examinations have been performed on the VEGP - Unit 2 RV. 4. In the request for an alternative, each licensee shall identify the years in which the future inspections will be performed. The VEGP – Unit 2 RV examinations currently scheduled for 2016 will be deferred until 2026, plus or minus one refueling cycle. The dates provided must be within plus or minus one refueling cycle of the date identified in PWROG letter OG-10-238, dated July 12, 2010 (Reference 5).

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	VEGP – Unit 2 is bounded by the pilot plant application because the total TWCF for VEGP – Unit 2 was calculated as 1.22E-13. This value is less than the value of 1.76E-08 which was calculated for the Westinghouse pilot plant in Reference 2; therefore, the use of this proposed alternative will provide an acceptable level of quality and safety. Therefore, it is requested that the NRC authorize this proposed alternative in accordance with 10 CFR 50.55a(a)(3)(i).
Duration of Proposed Alternative:	The 3rd and 4th Intervals.
Precedents:	1. Donald C. Cook Nuclear Plant, Unit 2 – Evaluation of Relief Request (ISIR-29) to Extend the Third 10-Year Inservice Inspection Interval for Reactor Vessel Weld Examination (TAC MD9934) Donald C. Cook Nuclear Power Plant, Unit 2 – NRC Safety Evaluation Report dated June 8, 2009 (ML091260163). 2. Safety Evaluation for Relief Requests ISI-090 & 021 Reactor Vessel Weld Examination Extension – Calvert Cliffs Nuclear Power Plant, Unit 2 (TAC Nos. MD9773 and MD9774). Calvert Cliffs Nuclear Power Plant Unit No. 2 – NRC Safety Evaluation Report dated April 8, 2009 (ML090920077). 3. Palisades Plant – Evaluation of Relief Request to Extend the Third 10-Year Inservice Inspection Interval For Reactor Weld Examinations (TAC NO. MD9265) Palisades Plant – NRC Safety Evaluation Report dated February 11, 2009 (ML 090120896). 4. R.E. Ginna Nuclear Power Plant: Safety Evaluation for Relief Request No. 18, Reactor Vessel Weld Examination Extension (TAC NO. MD9962) – R.E. Ginna Plant – NRC Safety Evaluation Report dated July 31, 2009 (ML092080229). 5. Kalyanam, N., NRC, to Vice President, Entergy Operations, "Waterford Steam Electric Station, Unit 3 – Withdrawal of an Amendment Request (TAC NO. MD9669)," dated June 12, 2009 (ML091600132 and ML091600158). 6. 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), NRC Safety Evaluation Report dated July 12, 2010 (ML101750402).
References:	1. American Society of Mechanical Engineers. ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition with the 2003 Addenda. 2. WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," October 2011. (Approved under NRC SER

	ML111600303) 3. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock," March, 2007. 4. 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," November 2002. 5. 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. 6. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004. 7. Nuclear Regulatory Commission, 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," May 1988. 9. WCAP-17353-NP, Revision 0, "Vogtle Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," August 2011.
Status:	Awaiting NRC approval.

Enclosure 2

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Table 1
Critical Parameters for the Application of the Bounding Analysis as Applied to VEGP - Unit 2

Parameter	Pilot Plant Basis	VEGP – Unit 2 Basis	Additional Evaluation Required?
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are applicable	NRC PTS Risk Study (Reference 3)	PTS Generalization Study (Reference 6)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 2)	1.22E-13 Events per year (Calculated using Reference 2)	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldown cycles per year (Reference 2)	Bounded by 7 heatup/cooldown cycles per year (1)	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 2)	Single Layer	No

Note:

(1) Per the VEGP Application for License Renewal, after 60 years of operation, the projected number of design basis transients is below the number specified in the 40-year design bases. As a result, VEGP – Unit 2 is conservatively bounded by 7 heatup/cooldown cycles per year.

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Table 2 Additional Information Pertaining to Reactor Vessel Inspections for VEGP - Unit 2	
Inspection methodology:	The latest ISI was conducted in accordance with the ASME Code, Section XI, 1989 Edition. Examinations of Category B-A and B-D welds were performed to ASME Section XI Appendix VIII, 1995 Edition up to and including the 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 requirements.
Number of past inspections:	Two 10-Year inservice inspections have been performed.
Number of indications found:	There were six indications identified in the beltline region during the most recent inservice inspection. These 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 and all 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 is scheduled for 2016. This inspection will be performed in 2026, plus or minus one refueling cycle. The proposed inspection date is consistent with the latest revised implementation plan OG-10-238 (Reference 5), plus or minus one refueling outage.

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Table 3
Details of the Through Wall Cracking Frequency Calculation for VEGP - Unit 2 at 57 EFPY

Inputs

Reactor Coolant System Temperature, T_C[°F]:	N/A	T_{wall} [inches]:	8.781

No.	Region and Component Description	Material Heat No.	Cu[1] [wt%]	Ni[1] [wt%]	R.G. 1.99 Pos.[1]	CF[1] [°F]	$RT_{NDT(u)}$[1] [°F]	Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]
1	Intermediate Shell Plate R4-1	C-3527-1	0.07	0.63	1.1	44.0	10	3.19
2	Intermediate Shell Plate R4-2	C-3527-2	0.06	0.61	1.1	37.0	10	3.19
3	Intermediate Shell Plate R4-3	C-3552-1	0.05	0.60	1.1	31.0	30	3.19
4	Lower Shell Plate B8825-1	C-3500-1	0.06	0.62	1.1	37.0	40	3.19
5	Lower Shell Plate R8-1	C-4304-1	0.07	0.63	1.1	44.0	40	3.19
6	Lower Shell Plate B8628-1	C-3500-2	0.05	0.59	1.1	31.0	50	3.19
7	Inter. Shell Long. Weld 101-124A	87005	0.05	0.15	2.1	20.7	-10	1.68
8	Inter. Shell Long. Weld 101-124B	87005	0.05	0.15	2.1	20.7	-10	3.07
9	Inter. Shell Long. Weld 101-124C	87005	0.05	0.15	2.1	20.7	-10	3.07
10	Lower Shell Long. Weld 101-142A	87005	0.05	0.15	2.1	20.7	-10	1.68
11	Lower Shell Long. Weld 101-142B	87005	0.05	0.15	2.1	20.7	-10	3.07
12	Lower Shell Long. Weld 101-142C	87005	0.05	0.15	2.1	20.7	-10	3.07
13	Inter. To Lower Shell Circ. Weld 101-171	87005	0.05	0.15	2.1	20.7	-30	3.19

Outputs

Methodology Used to Calculate ΔT_{30}:	Regulatory Guide 1.99, Revision 2 (Reference 8)

	Controlling Material Region No. (From Above)	RT_{MAX-XX} [°R]	Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]	FF (Fluence Factor)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Axial Weld - AW	5	556.69	3.07	1.296	57.02	0.00E+00
Limiting Plate - PL	5	557.10	3.19	1.305	57.43	4.87E-14
Circumferential Weld - CW	5	557.10	3.19	1.305	57.43	0.00E+00
$TWCF_{95-TOTAL}(\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW})$:						1.22E-13

Note:

(1) Reference 9

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Table 4
List of Affected Components for VEGP – Unit 2

ASME CATEGORY	ASME ITEM NUMBER	COMPONENT ID	DESCRIPTION
B-A	B1.11	21201-V6-001-W04	UPPER SHELL TO INTERMEDIATE SHELL WELD
B-A	B1.11	21201-V6-001-W05	INTERMEDIATE SHELL TO LOWER SHELL WELD
B-A	B1.11	21201-V6-001-W06	LOWER SHELL TO BOTTOM HEAD TORUS WELD
B-A	B1.11	21201-V6-001-W207	NOZZLE DROP-OUT TO SHELL WELD
B-A	B1.11	21201-V6-001-W208	NOZZLE DROP OUT TO SHELL WELD
B-A	B1.12	21201-V6-001-W12	UPPER SHELL LONGITUDINAL WELD AT 42 DEGREES
B-A	B1.12	21201-V6-001-W13	UPPER SHELL LONGITUDINAL WELD AT 162 DEGREES
B-A	B1.12	21201-V6-001-W14	UPPER SHELL LONGITUDINAL WELD AT 282 DEGREES
B-A	B1.12	21201-V6-001-W15	INTERMEDIATE SHELL LONGITUDINAL WELD AT 0 DEGREES
B-A	B1.12	21201-V6-001-W16	INTERMEDIATE SHELL LONGITUDINAL WELD AT 120 DEGREE
B-A	B1.12	21201-V6-001-W17	INTERMEDIATE SHELL LONGITUDINAL WELD AT 240 DEGREE
B-A	B1.12	21201-V6-001-W18	LOWER SHELL LONGITUDINAL WELD AT 90 DEGREES
B-A	B1.12	21201-V6-001-W19	LOWER SHELL LONGITUDINAL WELD AT 210 DEGREES
B-A	B1.12	21201-V6-001-W20	LOWER SHELL LONGITUDINAL WELD AT 330 DEGREES
B-A	B1.21	21201-V6-001-W01	CLOSURE HEAD DOME TO TORUS WELD
B-A	B1.21	21201-V6-001-W07	BOTTOM HEAD TORUS TO BOTTOM HEAD DOME WELD
B-A	B1.22	21201-V6-001-W08	CLOSURE HEAD TORUS MERIDIONAL WELD AT 45 DEGREES
B-A	B1.22	21201-V6-001-W09	CLOSURE HEAD TORUS MERIDIONAL WELD AT 135 DEGREES
B-A	B1.22	21201-V6-001-W10	CLOSURE HEAD TORUS MERIDIONAL WELD AT 225 DEGREES
B-A	B1.22	21201-V6-001-W11	CLOSURE HEAD TORUS MERIDIONAL WELD AT 315 DEGREES
B-A	B1.22	21201-V6-001-W21	BOTTOM HEAD TORUS MERIDIONAL WELD AT 0 DEGREES
B-A	B1.22	21201-V6-001-W22	BOTTOM HEAD TORUS MERIDIONAL WELD AT 90 DEGREES
B-A	B1.22	21201-V6-001-W23	BOTTOM HEAD TORUS MERIDIONAL WELD AT 180 DEGREES
B-A	B1.22	21201-V6-001-W24	BOTTOM HEAD TORUS MERIDIONAL WELD AT 270 DEGREES
B-A	B1.30	21201-V6-001-W03	VESSEL FLANGE TO SHELL WELD
B-A	B1.40	21201-V6-001-W02	CLOSURE HEAD TORUS TO FLANGE WELD
B-D	B3.90	21201-V6-001-W25	VESSEL TO OUTLET NOZZLE N1 WELD
B-D	B3.90	21201-V6-001-W26	VESSEL TO INLET NOZZLE N2 WELD
B-D	B3.90	21201-V6-001-W27	VESSEL TO INLET NOZZLE N3 WELD
B-D	B3.90	21201-V6-001-W28	VESSEL TO OUTLET NOZZLE N4 WELD
B-D	B3.90	21201-V6-001-W29	VESSEL TO OUTLET NOZZLE N5 WELD
B-D	B3.90	21201-V6-001-W30	VESSEL TO INLET NOZZLE N6 WELD
B-D	B3.90	21201-V6-001-W31	VESSEL TO INLET NOZZLE N7 WELD
B-D	B3.90	21201-V6-001-W32	VESSEL TO OUTLET NOZZLE N8 WELD
B-D	B3.100	21201-V6-001-IR01	OUTLET NOZZLE N1 INNER RADIUS
B-D	B3.100	21201-V6-001-IR02	INLET NOZZLE N2 INNER RADIUS
B-D	B3.100	21201-V6-001-IR03	INLET NOZZLE N3 INNER RADIUS
B-D	B3.100	21201-V6-001-IR04	OUTLET NOZZLE N4 INNER RADIUS
B-D	B3.100	21201-V6-001-IR05	OUTLET NOZZLE N5 INNER RADIUS
B-D	B3.100	21201-V6-001-IR06	INLET NOZZLE N6 INNER RADIUS
B-D	B3.100	21201-V6-001-IR07	INLET NOZZLE N7 INNER RADIUS
B-D	B3.100	21201-V6-001-IR08	OUTLET NOZZLE N8 INNER RADIUS