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SOUTHERN COMPANY

JUL 17 2015

Docket Nos.: 50-348

NL-15-0942

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

Joseph M. Farley Nuclear Plant - Unit 1
Proposed Inservice Inspection Alternative FNP-ISI-ALT-18, Version 1.0

Ladies and Gentlemen:

In accordance with 10CFR50.55a(z)(1), Southern Nuclear Operating Company (SNC) hereby requests Nuclear Regulatory Commission (NRC) approval of proposed inservice inspection (ISI) alternative FNP-ISI-ALT-18, Version 1.0. This alternative requests a one-time extension of the time between examinations required by IWB-2412, Inspection Program B, of Category B-A and B-D welds from 10 years to 20 years.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Sincerely,

C.R. Pierce

C.R. Pierce
Regulatory Affairs Director

CRP/JMC/lac

Enclosure: Proposed Alternative FNP-ISI-ALT-18, Version 1.0,
in Accordance with 10 CFR 50.55a(z)(1)

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Ms. C. A. Gayheart, Vice President – Farley
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U. S. Nuclear Regulatory Commission
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Joseph M. Farley Nuclear Plant – Unit 1
Proposed Inservice Inspection Alternative FNP-ISI-ALT-18, Version 1.0

Enclosure

Proposed Alternative FNP-ISI-ALT-18, Version 1.0,
in Accordance with 10 CFR 50.55a(z)(1)

Proposed Alternative FNP-ISI-ALT-18 Version 1.0,
in Accordance with 10 CFR 50.55a(z)(1)

Plant Site-Unit:	Joseph M. Farley Nuclear Plant (FNP) - Unit 1.
Interval Dates:	4th Inservice Inspection (ISI) Interval – December 1, 2007 through November 30, 2017.
Requested Date for Approval:	Approval is requested by July 31, 2016.
ASME Code Components Affected:	The affected components are Examination Category B-A, Items B1.11, B1.12, B1.21, B1.22 and B1.30 reactor vessel (RV) shell 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 4th ISI interval) is ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition through the 2003 Addenda (Reference 1).
Applicable Code Requirements:	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 FNP - Unit 1 4th 10-year inservice inspection (ISI) interval is scheduled to end on November 30, 2017.
Reason for Request:	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. The Westinghouse pilot plant RV analysis defined in WCAP-16168-NP-A, Revision 3, utilizes probabilistic fracture mechanics and risk analysis methods to justify extending the ISI interval for reactor vessel welds (Examination Category B-A), 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 FNP - Unit 1, which is a Westinghouse 3-Loop plant, is bounded by the pilot plant parameters defined in WCAP-16168-NP-A, Revision 3. Therefore, Southern Nuclear Operating Company (SNC) is requesting approval of this alternative to allow the use of the ISI interval extension for the affected FNP - Unit 1 components.

Proposed Alternative FNP-ISI-ALT-18 Version 1.0,
in Accordance with 10 CFR 50.55a(z)(1)

Proposed Alternative:	SNC is requesting a one-time extension of the ISI interval from 10 years to 20 years for FNP - Unit 1 Examination Category B-A 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 2016 (3rd period of 4th interval) until no later than the end of November 2027 (3rd period of 5th interval). The proposed inspection date for FNP - Unit 1 is within one outage of the schedule presented in the latest implementation plan, OG-10-238 (Reference 2).
Basis for Use:	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 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 4). 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 Nuclear Regulatory Commission (NRC) Pressurized Thermal Shock (PTS) Risk Re-Evaluation (Reference 5). The WCAP 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 3). Reference 4 was subsequently approved by the NRC in a July 26, 2011 revised safety evaluation. Section 3.4 of the safety evaluation provides the requirements for a utility to submit an alternative in accordance with 10 CFR 50.55a(z)(1) to use the WCAP 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 FNP - 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 4) fatigue analysis. As shown in Table 1, the frequency of the FNP - Unit 1 limiting design basis transients are bounded by the frequency identified in the PWROG (Reference 4) 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

Basis for Use (Cont.):	inspections for FNP - Unit 1 are provided in Table 2. This information confirms that satisfactory examinations have been performed on the FNP - 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 FNP - Unit 1 RV examinations currently scheduled for 2016 will be deferred until no later than the end of 2027. 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 2). The intent of the schedule identified in PWROG letter OG-10-238 is to provide for a sampling of vessel weld inspections in the PWR fleet over the 20 year interval such that any emerging degradation mechanisms are detected in a timely manner. The dates that are proposed for FNP - Unit 1 in this request for the alternative are consistent with the dates identified in PWROG letter OG-10-238. These dates will result in one fewer examination being performed in 2016. This change in dates will still provide for at least one inspection each year and will have a negligible impact on the ability of the schedule to provide for early detection of emerging degradation mechanisms. FNP - Unit 1 is bounded by the pilot plant application because the total TWCF for FNP - Unit 1 was calculated as 4.81E-11; 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(z)(1). The reactor vessel neutron fluence values used for FNP - Unit 1 are based on the pressurized thermal shock evaluation in WCAP-17506-NP (Reference 9). The fluence values in Reference 9 are based on more recent fluence evaluations than the values reported in the Farley License Renewal Application (Reference 10). The updated fluence values are also summarized in the Farley request to revise Technical Specifications associated with the Low Temperature Overpressure Protection System and the Pressure and Temperature Limits Report (Reference 11), which was accepted by the NRC in Reference 12.
Duration of Proposed Alternative:	The fourth 10 Year ISI Interval is scheduled to end in November of 2017. Granting approval of this proposed alternative will allow an extension of the fourth interval for these exams to be performed no later than the end of November 2027.
Precedents:	• "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). • "Surry Power Station Units 1 and 2 – Relief Implementing Extended Reactor Vessel Inspection Interval (TAC Nos. ME8573 and ME8574)," dated April 30,

Precedents (Cont):	2013 (ADAMS Accession Number ML13106A140). • "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). • "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). • "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). • "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). • "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). • "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).
References:	1. ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition with the 1989 Addenda up to and including the 2001 Edition with the 2003 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," November 2002. 4. Westinghouse Report WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of 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.

References (Cont):	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. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U. S. Nuclear Regulatory Commission, May 1988. 8. CE Report MISC-PENG-ER-011, Revision 0, "The Reactor Vessel Group Record Evaluation Program Phase II Final Report for the Farley 1 Reactor Pressure Vessel Plates, Forgings, Welds and Cladding," October 1995. 9. Westinghouse Report WCAP-17506-NP, Revision 0, "Farley Units 1 and 2 Pressurized Thermal Shock Evaluations," December 2011. 10. "Joseph M. Farley Nuclear Plant Application for License Renewal," Facility Operating License No. NPF-2. 11. Southern Company Letter NL-12-0868, "Joseph M. Farley Nuclear Plant – Units 1 & 2 Request to Revise Technical Specifications Associated with the Low Temperature Overpressure Protection System and the Pressure and Temperature Limits Report," Docket Nos. 50-348 and 50-364, dated August 15, 2012 (ADAMS Accession Number ML12229A521). 12. NRC Safety Evaluation Report, "Joseph M. Farley Nuclear Plant, Units 1 and 2, Issuance of Amendments Regarding Technical Specifications Revisions Associated with the Low Temperature Overpressure Protection System and the Pressure and Temperature Limits Report (TAC Nos. ME9256 and ME9257) (NL-12-0868)," dated October 2, 2013 (ADAMS Accession Number ML13249A386). 13. 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. 14. Constellation Energy Letter, Attachment 1, "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket No. 50-318 Revised Request to Extend the Inservice Inspection Interval for Reactor Vessel Weld Examinations – Relief Requests (ISI-020 and ISI-021)," dated October 1, 2008. (ADAMS Accession Numbers ML082760282 and ML082760283) 15. NRC Letter, "Safety Evaluation for Relief Requests ISI-020 and 021 Reactor Vessel Weld Examination Extension – Calvert Cliffs Nuclear Power Plant, Unit No. 2 (TAC Nos. MD9773 and MD9774)," dated April 8, 2009. (ADAMS Accession Number ML090920077)
Status:	Awaiting NRC approval.

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

Parameter	Pilot Plant Basis	FNP - 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 5)	PTS Generalization Study (Reference 6)	No
Through Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 4)	4.81E-11 Events per year (Calculated using Reference 4)	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldowns per year (Reference 4)	Bounded by 7 heatup/cooldowns per year (1)	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 4)	Single Layer	No

(1) Per the J. M. Farley Application for License Renewal (Reference 10), 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, FNP - Unit 1 is conservatively bounded by 7 heatup/cooldown events per year.

Table 2 **Additional Information Pertaining to Reactor Vessel Inspections for FNP - Unit 1**	
Inspection methodology:	The latest ISI was conducted in accordance with the ASME Code, Section XI 1989 Edition, with no Addenda. Examinations of Category B-A and B-D welds were performed to ASME Section XI Appendix VIII, 2001 Edition with the 2003 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:	Three 10-Year inservice inspections have been performed.
Number of indications found:	There were three indications identified in the beltline region during the most recent inservice inspection. These subsurface indications are located in the lower shell plate (Item 5 in Table 3). All indications are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. All three indications are within the inner 1/10th or 1" of the reactor vessel thickness. Two of these indications are acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 13), since the number of flaws is less than the allowable number of flaws for each flaw size increment. One indication, in plate material with a through-wall extent of 0.38", does not meet the requirements in the Alternate PTS Rule, 10 CFR 50.61a (Reference 13). While one flaw is outside the limits in 10 CFR 50.61a, it is not expected that this flaw would increase the FNP - Unit 1 TWCF value above that of the pilot plant for the following reasons: • The plate which the flaw is located has a maximum $RT_{NDT} + \Delta T_{30}$ of 145.50°F and is not the limiting material in the beltline region. Furthermore, the peak fluence was used to calculate the ΔT_{30} shift for this plate; however, the fluence at the position of this flaw is approximately 30% of the peak fluence. This was determined using the fluence values reported in Table 5.1-1 of WCAP-17506-NP (Reference 9) since the location of the flaw is just slightly offset from the location of the lower shell longitudinal welds. Therefore, the actual $RT_{NDT} + \Delta T_{30}$ at the specific location of this flaw is less than 145.50°F. • The total number of flaws detected in the FNP - Unit 1 beltline is far less than those allowed in the Alternate PTS Rule, 10 CFR 50.61a. • The TWCF for the FNP - Unit 1 reactor vessel is more than 2 orders of magnitude below that for the bounding pilot plant reactor vessel in WCAP-16168-NP-A, Revision 3 (Reference 4). Furthermore, the TWCF for FNP - Unit 1 was conservatively determined for 54 EFPY, corresponding to the end of license. This is conservative because FNP - Unit 1 is conservatively projected to have operated to approximately 42.5 EFPY in 2027, which is the proposed date for the next inspection. No indications are located within the weld or forging material of the reactor vessel beltline. The following indications are located within the plate material of the reactor vessel beltline.

Table 2
Additional Information Pertaining to Reactor Vessel Inspections for FNP - Unit 1

<table>
<tr><td></td><td>
<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>74</td><td>2 (2/0)</td></tr>
<tr><td>0.125</td><td>0.375</td><td>29</td><td>2 (2/0)</td></tr>
<tr><td>0.175</td><td>0.375</td><td>8</td><td>0</td></tr>
<tr><td>0.225</td><td>0.375</td><td>3</td><td>0</td></tr>
<tr><td>0.275</td><td>0.375</td><td>1</td><td>0</td></tr>
<tr><td>0.325</td><td>0.375</td><td>1</td><td>0</td></tr>
<tr><td>0.375</td><td>Infinite</td><td>0</td><td>1 (1/0)</td></tr>
</table>

Note that a reactor vessel beltline flaw with a through-wall extent of 0.60 inch was identified at Calvert Cliffs Unit 2 during the last 10-year inservice inspection. This flaw also exceeded the acceptance criteria, and was included in the Calvert Cliffs Unit 2 relief request (Reference 14) to extend the reactor vessel inservice inspection interval from 10 to 20 years for the B-A and B-D welds. The flaw in the Calvert Cliffs Unit 2 reactor vessel, which was also larger than the flaw found in the Farley Unit 1 reactor vessel, was determined to be acceptable by the NRC in Reference 15.</td></tr>
<tr><td>Proposed inspection schedule for balance of plant life:</td><td>The fourth inservice inspection originally scheduled for 2016 will be performed no later than the end of November 2027. These RPV examinations will be performed to the ASME Code in effect for the ten-year ISI interval they are performed in, which if this alternative is approved would be the fourth inservice inspection interval. The proposed inspection date is consistent with the latest revised implementation plan, OG-10-238 (Reference 2).</td></tr>
</table>

Table 3
Details of the Through Wall Cracking Frequency Calculation for FNP - Unit 1 @ 54 EFPY

Inputs

Reactor Coolant System Temperature, T_C[°F]:				N/A	Inter. & Lower Shell T_{wall} [inches]:				8.031
					Nozzle Shell T_{wall} [inches]:				9.156
No.	Region/Component Description (Ref. 9)	Material /Flux Type (Ref. 8)	Material Heat No. (Ref. 8)	Cu [wt%] (Ref. 9)	Ni [wt%] (Ref. 9)	R.G. 1.99 Pos.	CF [°F] (Ref. 9)	Un-Irradiated $RT_{NDT(u)}$ [°F] (Ref. 9)	Fluence [10^{19} Neutron/cm^2 E>1 MeV] (Ref. 9)
1	Upper Shell Forging B6914	A508 Cl. 2	123W209VA1	0.16	0.684	1.1	120.1	30	1.02
2	Intermediate Shell Plate B6903-2	A533 Gr. B Cl. 1	C6294-1	0.13	0.60	1.1	91.0	0	5.93
3	Intermediate Shell Plate B6903-3	A533 Gr. B Cl. 1	C6308-2	0.12	0.56	1.1	82.2	10	5.93
4	Lower Shell Plate B6919-1	A533 Gr. B Cl. 1	C6940-1	0.14	0.55	2.1	106.7	15	5.81
5	Lower Shell Plate B6919-2	A533 Gr. B Cl. 1	C6897-2	0.14	0.56	1.1	98.2	5	5.81
6	Inter. Shell Long. Weld 19-894 A	Linde 1092	33A277	0.258	0.165	1.1	126.3	-56	1.83
7	Inter. Shell Long. Weld 19-894 B	Linde 1092	33A277	0.258	0.165	1.1	126.3	-56	1.83
8	Lower Shell Long. Weld 20-894 A	Linde 0091	90099	0.197	0.06	1.1	91.4	-56	1.79
9	Lower Shell Long. Weld 20-894 B	Linde 0091	90099	0.197	0.06	1.1	91.4	-56	1.79
10	Upper to Inter. Shell Circ. Weld 10-894	Linde 0091	90099	0.197	0.06	1.1	91.4	-56	1.02
11	Inter. to Lower Shell Circ. Weld 11-894	Linde 0091	6329637	0.205	0.105	1.1	98.4	-56	5.81

Outputs

Methodology Used to Calculate ΔT_{30}:			Regulatory Guide 1.99, Revision 2 (Reference 7)			
	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]	TWCF95-XX
Limiting Axial Weld - AW	4	598.43	1.79	1.160	138.76	0.00E+00
Limiting Plate - PL	4	627.33	5.81	1.431	167.66	1.51E-11
Forging -FO	1	610.44	1.02	1.006	150.77	4.28E-12
Circumferential Weld - CW	4	627.33	5.81	1.431	167.66	2.37E-19
$TWCF_{95\text{-}TOTAL}(\alpha_{AW}TWCF_{95\text{-}AW} + \alpha_{PL}TWCF_{95\text{-}PL} + \alpha_{PL}TWCF_{95\text{-}FO} + \alpha_{CW}TWCF_{95\text{-}CW})$:						4.81E-11

Table 4
List of Affected Components for FNP - Unit 1

ASME Category	ASME Item Number	Component ID	Description
B-A	B1.11	ALA1-1100-2	UPPER TO MIDDLE SHELL
B-A	B1.11	ALA1-1100-5	MIDDLE TO LOWER SHELL
B-A	B1.11	ALA1-1100-8	LOWER SHELL/BOTTOM HEAD
B-A	B1.12	ALA1-1100-3	MIDDLE SHELL LONG. SEAM
B-A	B1.12	ALA1-1100-4	MIDDLE SHELL LONG. SEAM
B-A	B1.12	ALA1-1100-6	LOWER SHELL LONG. SEAM
B-A	B1.12	ALA1-1100-7	LOWER SHELL LONG. SEAM
B-A	B1.21	ALA1-1100-15	LOWER HEAD/MERIDIONAL CIR.
B-A	B1.21	ALA1-1100-16	BOTTOM HD.RING/LOWER HD. SHELL
B-A	B1.22	ALA1-1100-10	LOWER HEAD MERIDIONAL SEAM
B-A	B1.22	ALA1-1100-11	LOWER HEAD MERIDIONAL SEAM
B-A	B1.22	ALA1-1100-12	LOWER HEAD MERIDIONAL SEAM
B-A	B1.22	ALA1-1100-13	LOWER HEAD MERIDIONAL SEAM
B-A	B1.22	ALA1-1100-14	LOWER HEAD MERIDIONAL SEAM
B-A	B1.22	ALA1-1100-9	LOWER HEAD MERIDIONAL SEAM
B-A	B1.30	ALA1-1100-1	FLANGE TO UPPER SHELL
B-D	B3.100	ALA1-1100-17IR	NOZZLE-INSIDE RADIUS (OUTLET)
B-D	B3.100	ALA1-1100-18IR	NOZZLE-INSIDE RADIUS (INLET)
B-D	B3.100	ALA1-1100-19IR	NOZZLE-INSIDE RADIUS (OUTLET)
B-D	B3.100	ALA1-1100-20IR	NOZZLE-INSIDE RADIUS (INLET)
B-D	B3.100	ALA1-1100-21IR	NOZZLE-INSIDE RADIUS (OUTLET)
B-D	B3.100	ALA1-1100-22IR	NOZZLE-INSIDE RADIUS (INLET)
B-D	B3.90	ALA1-1100-17	NOZZLE/VESSEL WELD (OUTLET)
B-D	B3.90	ALA1-1100-18	NOZZLE/VESSEL WELD (INLET)
B-D	B3.90	ALA1-1100-19	NOZZLE/VESSEL WELD (OUTLET)
B-D	B3.90	ALA1-1100-20	NOZZLE/VESSEL WELD (INLET)
B-D	B3.90	ALA1-1100-21	NOZZLE/VESSEL WELD (OUTLET)
B-D	B3.90	ALA1-1100-22	NOZZLE/VESSEL WELD (INLET)