ATTACHMENT

CALCULATION OSC-9863

Oconee Units 1, 2 and 3
License Exemption Using BAW-2308

Form 101.1 (R06-09)

CERTIFICATION OF ENGINEERING CALCULATION

Station and Unit Number: Oconee Nuclear Station Units 1, 2 and 3

Title of Calculation: Oconee Units 1, 2 and 3 License Exemption Using BAW-2308 (AREVA Document No. 86 – 9109752 – 000)

Calculation Number: OSC – 9863

Total Original Pages: A, and 1 Through 20

Total Supporting Documentation Attachments: 1 (38 pages) Total Microfiche Attachments: 0

Total Volumes: 1 Active Calculation/Analysis: Yes ☐ No ☒

Microfiche Attachment List: Yes ☐ No ☒ (See Form 101.4) If Active, is this a Type I Calculation/Analysis: Yes ☐ No ☐

These engineering Calculations cover QA Condition 1 Items. In accordance with established procedures, the quality has been assured and I certify that the above Calculation has been Originated, Checked, Inspected or Approved as noted below:

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Approved By: David E. Whitaker *David E. Whitaker* Date: 9/21/2009

Issued To DCRM: *David E. Whitaker* Date: 9/22/2009

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FIGURE - 1 CERTIFICATION OF ENGINEERING CALCULATION

REVISION DOCUMENTATION SHEET

Revision Number	Revision Description
0	Original with pages "A" (EM 4.9 Calculation Impact Assessment) added to AREVA Document No. 86 – 9109752 – 000. Calculation OSC – 9863 Revision 0 is for Limited Use and shall be used only for requesting a License Exemption using BAW-2308 from the NRC, and as inputs by AREVA in calculating new (54 EFPY) Reactor Vessel (RV) Pressure-Temperature (P-T) limits for Oconee Nuclear Station.
1	Revision 1 is a complete replacement of AREVA Document No. 86 – 9109752 – 000 with AREVA Document No. 86 – 9109752 – 002. Calculation OSC – 9863 Revision 1 is still for Limited Use and shall be used only for requesting a License Exemption using BAW-2308 from the NRC, and as inputs by AREVA in calculating new (54 EFPY) Reactor Vessel (RV) Pressure-Temperature (P-T) limits for Oconee Nuclear Station.

FIGURE ERROR! NO TEXT OF SPECIFIED STYLE IN DOCUMENT.-2 REVISION DOCUMENTATION SHEET

CALCULATION IMPACT ASSESSMENT (CIA)

Station / Unit Oconee/1, 2 and 3 | Calculation No. OSC-9863 | Rev. 1 | Page iA
PIP No. (if applicable) ____ | By David E. Whitaker | Date 6/23/2010
Prob. No. (stress &s/r use only) ____ | Checked By N/A | Date ____

NEDL reviewed to identify calculations? ☐ Yes ☒ No

Note: a NEDL search is NOT required for calculation originations (i.e., Rev.0's)

Identify in the blocks below, the groups consulted for an Impact Assessment of this calculation origination/revision.

Group	Indv. Contacted/Date	Group	Indv. Contacted/Date
☐ RES [Power, I & C, ERRT, Reactor]		☐ NGO [QA Tech. Services (ISI); Severe Accident Analysis; Elect. Sys. & Equip.; Design & Reactor Sup.; Civil-Structural; Core Mech. & T/H Analysis; Mech. Sys. & Equip.; Nuclear Design and Safety Analysis; Materials/Metallurgy & Piping]	
☐ MCE [Primary Systems, Balance of Plant, Rotating Equipment, Valves & Heat Exchangers, Civil]			
☐ MOD [Mechanical Engr., Electrical Engr., Civil Engr.]		☐ Training	
		☐ Local IT	
☐ Operations – OPS Support		☐ Regulatory Compliance	
☐ Maintenance – Tech. Support		☐ Chemistry	
☐ Work Control – Program Sup.		☐ Radiation Protection	
☐ Other Group ____		☒ **No Group required to be consulted**	

Listed below are the identified documents (ex: TECHNICAL SPECIFICATION SECTIONS, UFSAR SECTIONS, DESIGN BASIS DOCUMENTS, STATIONS PROCEDURES*, DRAWINGS, OTHER CALCULATIONS, ETC.) that may require revision as a result of the calculation origination or revision, the document owner/group and the change required (including any necessary PIP Corrective Actions).

**** Note: Any design changes, which require changes to Station Procedures, must be transmitted as Design Deliverable Documents.***

DOCUMENT	GROUP	CHANGE REQUIRED
Original		

(Attach Additional Sheets As Required)

CALCULATION IMPACT ASSESSMENT (CIA)

Station / Unit Oconee/1, 2 and 3 Calculation No. OSC-9863 Rev. 0 Page A
PIP No. (if applicable) _____ By David E. Whitaker Date 9/21/2009
Prob. No. (stress &s/r use only) _____ Checked By N/A Date _____

NEDL reviewed to identify calculations? ☐ Yes ☒ No

Note: a NEDL search is **NOT** required for calculation originations (i.e., Rev.0's)

Identify in the blocks below, the groups consulted for an Impact Assessment of this calculation origination/revision.

Group	Indv. Contacted/Date	Group	Indv. Contacted/Date
☐ RES [Power, I & C, ERRT, Reactor]		☐ NGO [QA Tech. Services (ISI); Severe Accident Analysis; Elect. Sys. & Equip.; Design & Reactor Sup.; Civil-Structural; Core Mech. & T/H Analysis; Mech. Sys. & Equip.; Nuclear Design and Safety Analysis; Materials/Metallurgy & Piping]	
☐ MCE [Primary Systems, Balance of Plant, Rotating Equipment, Valves & Heat Exchangers, Civil]			
☐ MOD [Mechanical Engr., Electrical Engr., Civil Engr.]		☐ Training	
		☐ Local IT	
☐ Operations – OPS Support		☐ Regulatory Compliance	
☐ Maintenance – Tech. Support		☐ Chemistry	
☐ Work Control – Program Sup.		☐ Radiation Protection	
☐ Other Group _____		☒ **No Group required to be consulted**	

Listed below are the identified documents (ex: TECHNICAL SPECIFICATION SECTIONS, UFSAR SECTIONS, DESIGN BASIS DOCUMENTS, STATIONS PROCEDURES*, DRAWINGS, OTHER CALCULATIONS, ETC.) that may require revision as a result of the calculation origination or revision, the document owner/group and the change required (including any necessary PIP Corrective Actions).

**** Note: Any design changes, which require changes to Station Procedures, must be transmitted as Design Deliverable Documents.***

DOCUMENT	GROUP	CHANGE REQUIRED
Original		

(Attach Additional Sheets As Required)

AREVA

CALCULATION SUMMARY SHEET (CSS)

Document No. 86 - 9109752 - 002 Safety Related: ☒ Yes ☐ No

Title Oconee Units 1, 2 and 3 License Exemption Using BAW-2308

PURPOSE AND SUMMARY OF RESULTS:

Purpose:

The purpose of this document is to summarize the pressurized thermal shock reference temperature (RT_{PTS}) [1] and adjusted reference temperature (ART) [2] calculations at 54 EFPY for Oconee Units 1, 2 and 3 when using alternate initial RT_{NDT} (nil ductility reference temperature) and σ_I (standard deviation of the initial RT_{NDT}) values from BAW-2308, Revision 2-A. This document also provides the text that Duke Energy will need in their request for exemption to 10CFR50.61 and 10CFR50 Appendix G.

The purpose of Revision 2 is to report ART values calculated at additional locations to aid the pressure-temperature limits analysis, which is performed separately.

Results:

When using the initial nil ductility reference temperature and σ_I values from BAW-2308, Revision 2-A, the RT_{PTS} values were below the screening criteria for all Oconee beltline materials. The controlling beltline material for the Oconee Unit 1 reactor vessel is the Upper Shell Longitudinal Weld, SA-1493, with a predicted RT_{PTS} value of 196.0°F. Screening criterion from this material is 270°F. The controlling beltline material for the Oconee Unit 2 reactor vessel is the Upper Shell to Lower Shell Circumferential Weld, WF-25, with a predicted RT_{PTS} value of 226.1°F. Screening criterion from this material is 300°F. The controlling beltline material for the Oconee Unit 3 reactor vessel is the Upper Shell to Lower Shell Circumferential Weld ID (75%), WF-67, with a predicted RT_{PTS} value of 222.5°F. Screening criterion from this material is 300°F.

ART values were calculated using the initial RT_{NDT} and σ_I values from BAW-2308, Revision 2-A and Regulatory Guide 1.99 Revision 2, for all the Oconee beltline materials. The results are presented in section 4.0.

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:

CODE/VERSION/REV	CODE/VERSION/REV

THE DOCUMENT CONTAINS ASSUMPTIONS THAT SHALL BE VERIFIED PRIOR TO USE

☐ YES
☒ NO

Review Method: ☒ Design Review (Detailed Check)
☐ Alternate Calculation

Signature Block

Name and Title (printed or typed)	Signature	P/R/A and LP/LR	Date	Pages/Sections Prepared/Reviewed/Approved
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Note: P/R/A designates Preparer (P), Reviewer (R), Approver (A);
LP/LR designates Lead Preparer (LP), Lead Reviewer (LR)

Project Manager Approval of Customer References (N/A if not applicable)

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Mentoring Information (not required per 0402-01)

Name (printed or typed)	Title (printed or typed)	Mentor to: (P/R)	Signature	Date
N/A	N/A	N/A	N/A	N/A

Record of Revision

Revision No.	Date	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	June 2009	All	Original Submittal
001	April 2010	Section 1.0	Removed statement about when the Oconee fluence calculations were last updated
		Section 2.0	Removed statement about the 32 EFPY fluence values being compliant with RG 1.190
		Tables 3-1, 3-2 and 3-3	Plate and forging type corrected; footnotes added
		Section 7.0	References 1, 2 and 8 updated
002	June 2010	Tables 4-2 and 4-3	Changed 55 EFPY to 54 EFPY
		Section 7.0	Reference 2 updated, added references 11-16 and renumbered all references
		Sections 2.0 and 4.0	Perform an ART calculation at an additional location on the LNB forging (Location 2) at Oconee Unit 1. Added Figures 2-1 and 2-2.
		Sections 2.0 and 4.0	Performed an ART calculation at an additional location on the LNB forging (Location 4) at Oconee Units 2 and 3.
		Tables 3-1, 3-2, and 3-3	Added Location designation to the lower nozzle belt forging

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1.0 INTRODUCTION

Lower initial RT_{NDT} values for the high copper Linde 80 welds B&W used in fabricating reactor vessels have been justified in topical report BAW-2308, Rev. 2. Technical approval by the NRC has been completed and is contained in BAW-2308 Rev. 2-A [3]. As stated in the NRC safety evaluation, each utility must submit a request for exemption per 10CFR50.12 [4] from the requirements of 10CFR50.61 [5] and/or 10CFR50 Appendix G [6] to use the lowered initial RT_{NDT} values. The submittal must include pressurized thermal shock (PTS) and/or adjusted reference temperature (ART) calculations. A significant input to these calculations is the projected fluence at the reactor vessel. Fluence projections that were submitted as part of the Oconee license renewal application for 48 EFPY are extrapolated to 54 EFPY. The use of 54 EFPY for the 60-year life was a conservative estimate and may be updated based on specific plant data at a later time. The PTS and ART values are calculated in support of the exemption request. Once the full pedigree fluence projection is completed, which takes into account the change to 2 year cycles and other fuel changes, PTS and ART values can then be recalculated for input to the pressure-temperature curve effort.

2.0 PROJECTED REACTOR VESSEL FLUENCE

The inner surface neutron fluence is the calculated value defined at the inside wetted surface of the Oconee Units 1, 2 and 3 reactor vessels. The projected 54 EFPY inner surface fluences for the Oconee Units 1, 2 and 3 reactor vessel beltline materials are listed in Table 2-1, Table 2-2 and Table 2-3. These 54 EFPY fluences were calculated by extrapolation using the previously reported projected 32 EFPY [7] and 48 EFPY [8] inner surface fluence values. The 48 EFPY fluence values were calculated in accordance with the requirements of U. S. NRC Regulatory Guide 1.190 [9], using a methodology described in detail in AREVA fluence topical report BAW-2241P-A [10]. For Oconee Unit 1, the projected fluence values for the lower nozzle belt (LNB) forging and the LNB to intermediate shell (IS) circumferential weld decreased from 32 EFPY to 48 EFPY. This was due to the change in the fluence calculation methodology for components above the core from an overly conservative method to a more accurate method. The 54 EFPY fluences for the Oconee Unit 1 LNB and the LNB to IS circumferential weld were calculated using the average percent increase in fluence from 48 EFPY to 54 EFPY of the remaining Oconee Unit 1 beltline materials.

The thickness of the lower nozzle belt (LNB) forging at Oconee Unit 1 varies from 8.44 [11] to 12.0 inches [12], as shown in Figure 2-1 [13]. The peak RPV fluence at Location 1 was calculated using the extrapolation method discussed above. The fluence at Location 2 was estimated to be equal to the fluence at Location 1, as shown in Table 2-1. This estimation is conservative because Location 1 is closer to the core. ART values were calculated for both Locations 1 and 2, as shown in Section 4.0.

The inside wetted fluence of the LNB forging at Oconee Units 2 and 3 were estimated at two locations because of the thickness change of the forging. These two locations are shown in Figure 2-2 [13]. Location 3 represents the location of the peak inside fluence of the 8.44 inches [11] thick portion of the LNB forging, which was calculated using the extrapolation method described above. The fluence at Location A is used as a conservative estimate of the fluence at Location 4; this estimate is conservative because Location 4 is farther from the core than Location A. Location A at Oconee Units 2 and 3 is at the same location as the LNB to IS circumferential weld (SA-1135) at Oconee 1, which is 11.0 feet from the reactor vessel flange mating surface. This distance of 11.0 feet is obtained from Reference 13 for Oconee Unit 1, References 14 and 15 for Oconee Unit 2 and References 15 and 16 for Oconee Unit 3. The projected 54 EFPY fluence at Oconee Unit 1 IS to US circumferential weld (SA-1229) is greater than the similarly located Oconee Unit 2 LNB to US circumferential weld (WF-154) and Oconee Unit 3 LNB to US circumferential weld (WF-200), as shown in Table 2-1, Table 2-2 and Table 2-3. These three welds are all 13.5 feet from the reactor vessel flange mating surface [13]. Based on this information, the flux at the

Oconee Unit 1 LNB to IS circumferential weld (SA-1135) will likely be equal to or greater than the flux at Location A at Oconee Unit 2 and 3. Therefore the 54 EFPY fluence at Location A for Oconee Unit 2 and 3 is projected to be 1.25×10^{18} n/cm^2, which is equivalent to the similar location at Oconee Unit 1. ART values were calculated for both Locations 3 and 4, as shown in Section 4.0.

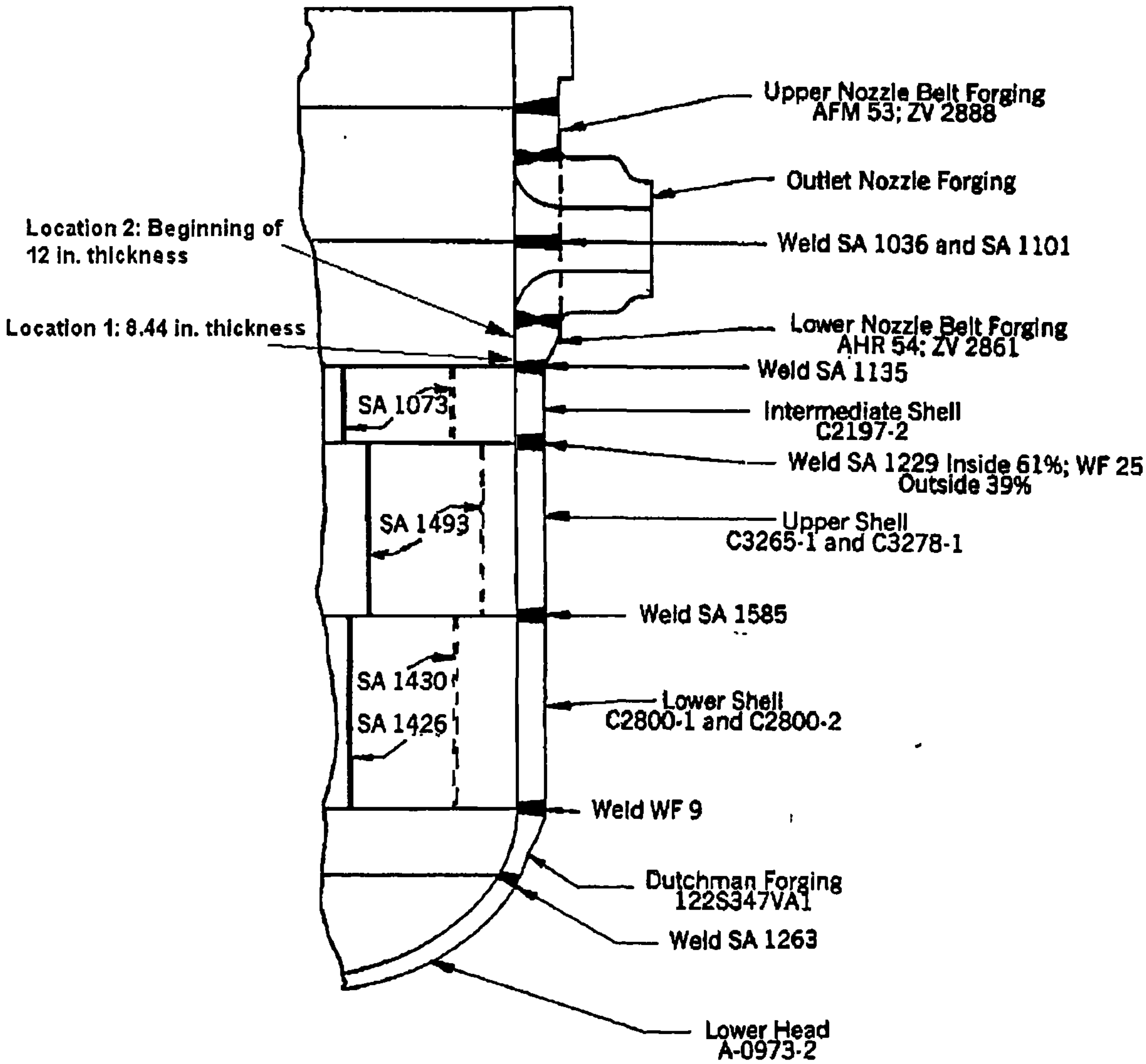

Figure 2-1: RPV Configuration for Oconee Unit 1 [13]

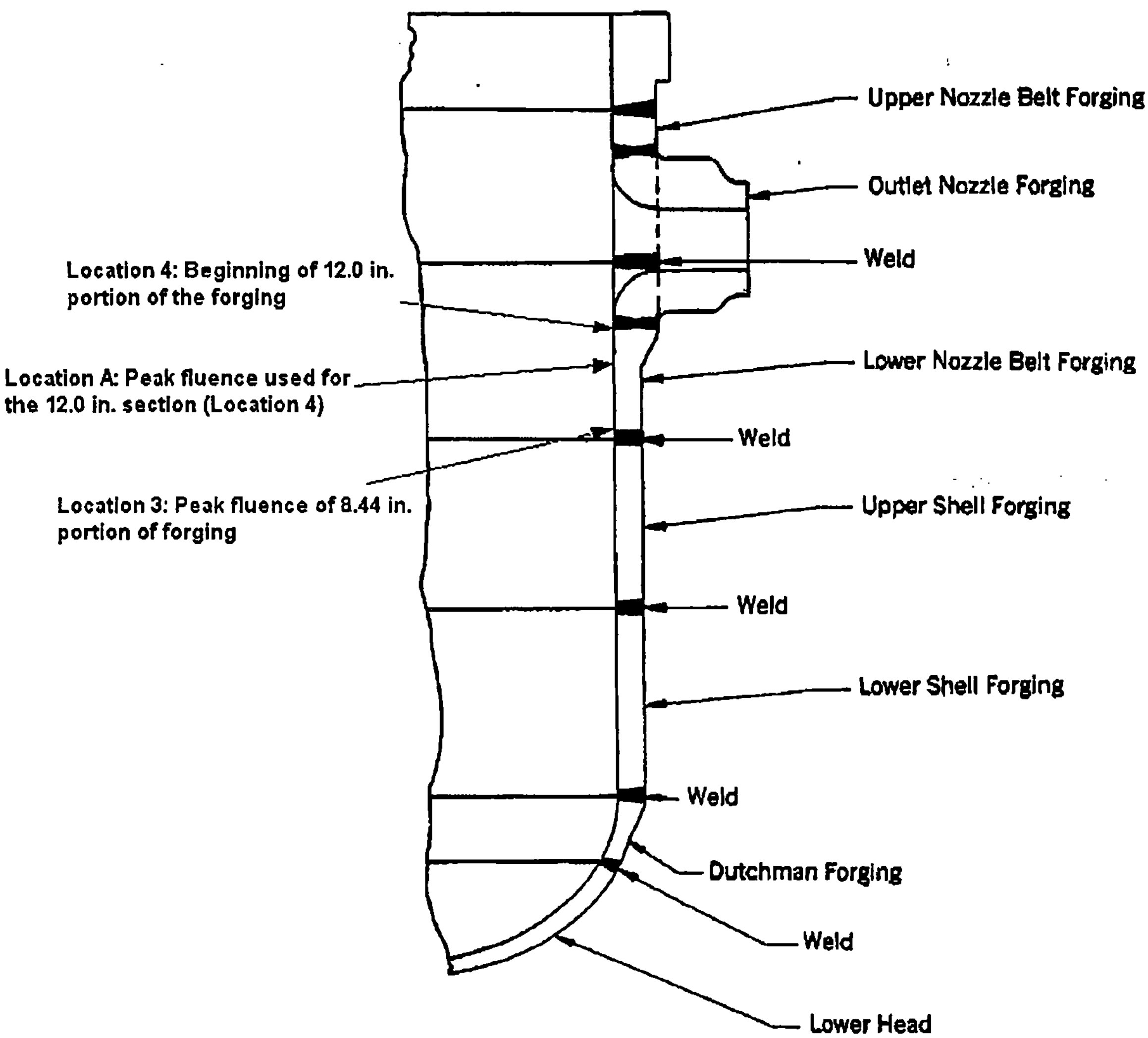

Figure 2-2: RPV Configuration for Oconee Units 2 & 3 [13]

Table 2-1: Inner Wetted Surface Fluence (E>1.0 MeV) Values for the Oconee Unit 1 Vessel Beltline Materials

Beltline Materials	Material Ident.	Inner Wetted Fluence (n/cm²)		
		32 EFPY	48 EFPY	54 EFPY
LNB Forging (Location 1)	AHR-54	1.18E+18	1.11E+18	1.25E+18*
LNB Forging (Location 2)	AHR-54	1.18E+18	1.11E+18	1.25E+18*
Intermediate Shell Plate (IS)	C2197-2	7.96E+18	1.18E+19	1.32E+19
Upper Shell Plate (US)	C3265-1	9.04E+18	1.31E+19	1.46E+19
Upper Shell Plate	C3278-1	9.04E+18	1.31E+19	1.46E+19
Lower Shell Plate (LS)	C2800-1	8.68E+18	1.31E+19	1.48E+19
Lower Shell Plate	C2800-2	8.68E+18	1.31E+19	1.48E+19
LNB to IS Circ Weld 100%	SA-1135	1.18E+18	1.11E+18	1.25E+18*
IS Long Weld 100%	SA-1073	6.28E+18	9.24E+18	1.04E+19
IS to US Circ Weld ID 61%	SA-1229	7.96E+18	1.19E+19	1.34E+19
IS to US Circ Weld OD 39%	WF-25	N/A	N/A	N/A
US Long Weld 100%	SA-1493	7.23E+18	1.12E+19	1.27E+19
US to LS Circ Weld 100%	SA-1585	8.68E+18	1.27E+19	1.42E+19
LS Long Weld 100%	SA-1426	7.29E+18	1.08E+19	1.21E+19
LS Long Weld 100%	SA-1430	7.29E+18	1.08E+19	1.21E+19

* The projected fluence values for these materials decreased from 32 EFPY to 48 EFPY. This was due to the change in the fluence calculation methodology for components above the core. The 54 EFPY fluences for these components were calculated applying the average percent increase in fluence from 48 EFPY to 54 EFPY of the remaining Oconee Unit 1 beltline materials.

Table 2-2: Inner Wetted Surface Fluence (E>1.0 MeV) Values for the Oconee Unit 2 Vessel Beltline Materials

Beltline Materials	Material Ident.	Inner Wetted Fluence (n/cm²)		
		32 EFPY	48 EFPY	54 EFPY
LNB Forging (Location 3)	AMX-77	8.42E+18	1.19E+19	1.32E+19
LNB Forging (Location 4)	AMX-77	N/A	N/A	1.25E+18
US Forging	AAW-163	9.57E+18	1.28E+19	1.40E+19
LS Forging	AWG-164	9.19E+18	1.27E+19	1.40E+19
LNB to US Circ Weld 100%	WF-154	8.42E+18	1.19E+19	1.32E+19
US to LS Circ Weld 100%	WF-25	9.19E+18	1.23E+19	1.35E+19

Table 2-3: Inner Wetted Surface Fluence (E>1.0 MeV) Values for the Oconee Unit 3 Vessel Beltline Materials

Beltline Materials	Material Ident.	Inner Wetted Fluence (n/cm²)		
		32 EFPY	48 EFPY	54 EFPY
LNB Forging (Location 3)	4680	8.26E+18	1.14E+19	1.26E+19
LNB Forging (Location 4)	4680	N/A	N/A	1.25E+18
US Forging	AWS-192	9.39E+18	1.26E+19	1.38E+19
LS Forging	ANK-191	9.01E+18	1.26E+19	1.39E+19
LNB to US Circ Weld 100%	WF-200	8.26E+18	1.14E+19	1.26E+19
US to LS Circ Weld ID 75%	WF-67	9.01E+18	1.22E+19	1.34E+19
US to LS Circ. Weld (OD 25%)	WF-70	N/A	N/A	N/A

3.0 PRESSURIZED THERMAL SHOCK

The RT_{PTS} values applicable at 60 calendar years (54 EFPY) for the Oconee Units 1, 2 and 3 reactor vessel beltline materials are listed in Table 3-1, Table 3-2 and Table 3-3. These values were calculated in accordance with the requirement specified in the Code of Federal Regulations, Title 10, Part 50.61 (10 CFR 50.61) [5] and using the initial RT_{NDT} values and the corresponding σ_I values from BAW-2308, Revision 2-A [3].

The controlling beltline material for the Oconee Unit 1 reactor vessel is the Upper Shell Longitudinal Weld, SA-1493, with a predicted RT_{PTS} value of 196.0°F. The screening criterion from this material is 270°F.

The controlling beltline material for the Oconee Unit 2 reactor vessel is the Upper Shell to Lower Shell Circumferential Weld, WF-25, with a predicted RT_{PTS} value of 226.1°F. The screening criterion from this material is 300°F.

The controlling beltline material for the Oconee Unit 3 reactor vessel is the Upper Shell to Lower Shell Circumferential Weld ID (75%), WF-67, with a predicted RT_{PTS} value of 222.5°F. The screening criterion from this material is 300°F.

Table 3-1: Oconee Unit 1 Pressurized Thermal Shock Reference Temperature at 54 EFPY

Material Description				Chemical Composition											
Reactor Vessel Beltline Region Matl.	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Chem. Factor	Initial RT_{NDT} (°F)	54 EFPY Fluence at Inside Wetted Surface (n/cm^2)	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	RT_{PTS} (°F)	Screening Criteria (°F)
LNB Forging (Location 1)[d]	ZV-2861	AHR-54	A-508, Cl. 2[a]	0.16	0.65	119.3	+3	1.25E+18	55.2	31.0	17.0	70.7	128.9	270
Intermediate Shell (IS) Plate	C2197-2	C2197-2	SA-302, Gr. B, Mod.[b]	0.15	0.50	104.5	+1	1.32E+19	112.7	26.9	17.0	63.6	177.3	270
Upper Shell (US) Plate	C3265-1	C3265-1	SA-302, Gr. B, Mod.[b]	0.10	0.50	65.0	+1	1.46E+19	71.8	26.9	17.0	63.6	136.5	270
Upper Shell Plate	C3278-1	C3278-1	SA-302, Gr. B, Mod.[b]	0.12	0.60	83.0	+1	1.46E+19	91.7	26.9	17.0	63.6	156.4	270
Lower Shell (LS) Plate	C2800-1	C2800-1	SA-302, Gr. B, Mod.[b]	0.11	0.63	74.5	+1	1.48E+19	82.5	26.9	17.0	63.6	147.2	270
Lower Shell Plate	C2800-2	C2800-2	SA-302, Gr. B, Mod.[b]	0.11	0.63	74.5	+1	1.48E+19	82.5	26.9	17.0	63.6	147.2	270
LNB to IS Circ. Weld (100%)	SA-1135	61782	Linde 80	0.23	0.52	167.0[c]	-58.5	1.25E+18	77.2	15.4	28.0	63.9	82.6	300
IS Long. Weld (100%)	SA-1073	1P0962	Linde 80	0.21	0.64	170.6	-48.6	1.04E+19	172.2	18.0	28.0	66.6	190.2	270
IS to US Circ. Weld (ID 61%)	SA-1229	71249	Linde 80	0.23	0.59	167.6	-53.5	1.34E+19	181.2	12.8	28.0	61.6	189.2	300
IS to US Circ. Weld (OD 39%)	WF-25	299L44	Linde 80	0.34	0.68	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	300
US Long. Weld (100%)	SA-1493	8T1762	Linde 80	0.19	0.57	167.0[c]	-48.6	1.27E+19	178.1	18.0	28.0	66.6	[196.0]	270
US to LS Circ. Weld (100%)	SA-1585	72445	Linde 80	0.22	0.54	167.0[c]	-72.5	1.42E+19	183.3	12.0	28.0	60.9	171.7	300
LS Long. Weld (100%)	SA-1426	8T1762	Linde 80	0.19	0.57	167.0[c]	-48.6	1.21E+19	175.9	18.0	28.0	66.6	193.9	270
LS Long. Weld (100%)	SA-1430	8T1762	Linde 80	0.19	0.57	167.0[c]	-48.6	1.21E+19	175.9	18.0	28.0	66.6	193.9	270

[ ] – Limiting reactor vessel beltline region material in accordance with 10 CFR 50.61

[a] ASTM A-508-64 Cl. 2 Mod. By ASME Code Case 1332-2

[b] ASME SA-302 Gr. B Mod. To ASME Code Case 1339

[c] Per BAW-2308 Rev. 1-A [17]

[d] See Figure 2-1

Table 3-2: Oconee Unit 2 Pressurized Thermal Shock Reference Temperature at 54 EFPY

Material Description				Chemical Composition										
Reactor Vessel Beltline Region Matl.	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Chem. Factor	Initial RT_{NDT} (°F)	54 EFPY Fluence at Inside Wetted Surface (n/cm^2)	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	RT_{PTS} (°F)	Screening Criteria (°F)
LNB Forging (Location 3)[c]	AMX 77	123T382	A-508, Cl. 2[a]	0.13	0.76	95.0	+3	1.32E+19	102.3	31.0	17.0	70.7	176.1	270
US Forging	AAW 163	3P2359	A-508, Cl. 2[b]	0.04	0.75	26.0	+20	1.40E+19	28.4	0.0	17.0	28.4	76.9	270
LS Forging	AWG 164	4P1885	A-508, Cl. 2[b]	0.02	0.80	20.0	+20	1.40E+19	21.9	0.0	17.0	21.9	63.7	270
LNB to US Circ. Weld (100%)	WF-154	406L44	Linde 80	0.27	0.59	182.6	-98.0	1.32E+19	196.7	11.6	28.0	60.6	159.3	300
US to LS Circ. Weld (100%)	WF-25	299L44	Linde 80	0.34	0.68	220.6	-74.3	1.35E+19	238.8	12.8	28.0	61.6	[226.1]	300

[ ] – Limiting reactor vessel beltline region material in accordance with 10 CFR 50.61

[a] ASTM A-508-64 Cl. 2 Mod. By ASME Code Case 1332-2

[b] ASTM A-508-64 Cl. 2 Mod. By ASME Code Case 1332-4

[c] See Figure 2-2

Table 3-3: Oconee Unit 3 Pressurized Thermal Shock Reference Temperature at 54 EFPY

Material Description				Chemical Composition										
Reactor Vessel Beltline Region Matl.	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Chem. Factor	Initial RT_{NDT} (°F)	54 EFPY Fluence at Inside Wetted Surface (n/cm²)	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	RT_{PTS} (°F)	Screening Criteria (°F)
LNB Forging (Location 3)[d]	4680	4680	A-508, Cl. 2[a]	0.13	0.91	96.0	+3	1.26E+19	102.1	31.0	17.0	70.7	175.8	270
US Forging	AWS 192	522314	A-508, Cl. 2[b]	0.01	0.73	36.0[c]	+40	1.38E+19	39.2	0.0	17.0	34.0	113.2	270
LS Forging	ANK 191	522194	A-508, Cl. 2[b]	0.02	0.76	17.4[c]	+40	1.39E+19	19.0	0.0	14.0	19.0	78.0	270
LNB to US Circ. Weld (100%)	WF-200	821T44	Linde 80	0.24	0.63	178.0	-84.2	1.26E+19	189.4	9.6	28.0	59.2	164.4	300
US to LS Circ. Weld (ID 75%)	WF-67	72442	Linde 80	0.26	0.60	180.0	-33.2	1.34E+19	194.6	12.2	28.0	61.1	[222.5]	300
US to LS Circ. Weld (OD 25%)	WF-70	72105	Linde 80	0.32	0.58	199.3	-31.1	N/A	N/A	13.7	28.0	N/A	N/A	300

[ ] – Limiting reactor vessel beltline region material in accordance with 10 CFR 50.61

[a] ASTM A-508-64 Cl. 2 Mod. By ASME Code Case 1332-3

[b] ASTM A-508-64 Cl. 2 Mod. By ASME Code Case 1332-4

[c] This Chemistry Factor was determined from surveillance data.

[d] See Figure 2-2

4.0 ADJUSTED REFERENCE TEMPERATURE

The ¼T and ¾T ART values for the Oconee Units 1, 2 and 3 reactor vessel beltline materials applicable at 60 calendar years (54 EFPY) are calculated using the alternate initial RT_{NDT} values and the corresponding σ_I values from BAW-2308, Revision 2-A [3] (for weld metals) and are listed in Table 4-1, Table 4-2 and Table 4-3. The ART values are calculated in accordance with Regulatory Guide 1.99, Revision 2 [18]. The material type, chemical compositions, chemistry factors, and initial RT_{NDT} values used in the ART calculations shown in Table 4-1, Table 4-2 and Table 4-3 are the same as shown in Table 3-1, Table 3-2 and Table 3-3 respectively and are thus not repeated.

The circumferential welds with the highest ART values for the Oconee Unit 1 reactor vessel are the Intermediate Shell to Upper Shell Circumferential Weld ID (61%), SA-1229, with an ART value at 54 EFPY of 164.2°F at the ¼T wall location and Intermediate Shell to Upper Shell Circumferential Weld OD (39%), WF-25, with an ART value at the 54 EFPY of 132.1°F at the ¾T wall location. Considering the base metal and the longitudinal welds, the materials with the highest ART values are the Upper Shell Longitudinal Weld, SA-1493, with an ART value at 54 EFPY of 171.0°F at the ¼T wall location and the Intermediate Shell Plate, C-2197-2, with an ART value at the 54 EFPY of 132.9°F at the ¾T wall location.

The circumferential weld with the highest ART values for the Oconee Unit 2 reactor vessel is the Upper Shell to Lower Shell Circumferential Weld, WF-25, with an ART value at 54 EFPY of 193.1°F at the ¼T wall location and 132.5°F at the ¾T wall location. Considering the base metal (there are no longitudinal welds in Oconee Unit 2), the material with the highest ART values is the Lower Nozzle Belt Forging, AMX-77, with an ART value at 54 EFPY of 161.8°F at the ¼T wall location and 135.7°F at the ¾T wall location. The ART values for the Lower Nozzle Belt Forging correspond to Location 3, which is the location of peak fluence for the 8.44 inches thick portion of the forging.

The circumferential welds with the highest ART values for the Oconee Unit 3 reactor vessel are the Upper Shell to Lower Shell Circumferential Weld ID (75%), WF-67, with an ART value at 54 EFPY of 195.6°F at the ¼T wall location and the Upper Shell to Lower Shell Circumferential Weld OD (25%), WF-70, with an ART value at the 54 EFPY of 162.1°F at the ¾T wall location. Considering the base metal (there are no longitudinal welds in Oconee Unit 2), the material with the highest ART values is the Lower Nozzle Belt Forging, 4680, with an ART value at 54 EFPY of 161.4°F at the ¼T wall location and 135.2°F at the ¾T wall location. The ART values for the Lower Nozzle Belt Forging correspond to Location 3, which is the location of peak fluence for the 8.44 inches thick portion of the forging.

Table 4-1: Adjusted Reference Temperature Evaluation for the Oconee Unit 1 Reactor Vessel Beltline Materials at 54 EFPY

Material Description		54 EFPY Fluence (n/cm^2)			ΔRT_{NDT} (°F) at 54 EFPY		Std. Deviation		Margin (°F)		ART (°F) at 54 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Inner Wetted Surface	¼T Location	¾T Location	¼T Location	¾T Location	σ_I	σ_Δ	¼T Location	¾T Location	¼T Location	¾T Location
LNB Forging (Location 1)[a]	AHR-54	1.25E+18	7.29E+17	2.65E+17	42.5	24.3	31.0	17.0	70.7	66.6	116.2	93.9
LNB Forging (Location 2)[a]	AHR-54	1.25E+18	5.90E+17	1.40E+17	38.2	16.4	31.0	17.0	70.7	64.1	111.9	83.5
Intermediate Shell (IS) Plate	C2197-2	1.32E+19	7.74E+18	2.81E+18	97.0	68.3	26.9	17.0	63.6	63.6	161.6	{132.9}
Upper Shell (US) Plate	C3265-1	1.46E+19	8.55E+18	3.11E+18	62.1	44.2	26.9	17.0	63.6	63.6	126.7	108.8
Upper Shell Plate	C3278-1	1.46E+19	8.55E+18	3.11E+18	79.4	56.4	26.9	17.0	63.6	63.6	144.0	121.0
Lower Shell (LS) Plate	C2800-1	1.48E+19	8.63E+18	3.13E+18	71.4	50.8	26.9	17.0	63.6	63.6	136.0	115.4
Lower Shell Plate	C2800-2	1.48E+19	8.63E+18	3.13E+18	71.4	50.8	26.9	17.0	63.6	63.6	136.0	115.4
LNB to IS Circ. Weld (100%)	SA-1135	1.25E+18	7.29E+17	2.65E+17	59.5	34.1	15.4	28.0	63.9	63.9	64.9	39.5
IS Long. Weld (100%)	SA-1073	1.04E+19	6.05E+18	2.20E+18	146.6	101.0	18.0	28.0	66.6	66.6	164.6	119.0
IS to US Circ. Weld (ID 61%)	SA-1229	1.34E+19	7.82E+18	N/A	156.1	N/A	12.8	28.0	61.6	N/A	[164.2]	N/A
IS to US Circ. Weld (OD 39%)	WF-25	1.34E+19	N/A	2.84E+18	N/A	144.8	12.8	28.0	N/A	61.6	N/A	[132.1]
US Long. Weld (100%)	SA-1493	1.27E+19	7.42E+18	2.70E+18	153.0	107.4	18.0	28.0	66.6	66.6	{171.0}	125.4
US to LS Circ. Weld (100%)	SA-1585	1.42E+19	8.31E+18	3.02E+18	158.3	112.2	12.0	28.0	60.9	60.9	146.7	100.6
LS Long. Weld (100%)	SA-1426	1.21E+19	7.09E+18	2.57E+18	150.9	105.4	18.0	28.0	66.6	66.6	168.9	123.4
LS Long. Weld (100%)	SA-1430	1.21E+19	7.09E+18	2.57E+18	150.9	105.4	18.0	28.0	66.6	66.6	168.9	123.4

[a] See Figure 2-1

[ ] –Highest values of the adjusted reference temperatures for circumferential welds.

{ } – Highest values of the adjusted reference temperatures for base metal or longitudinal welds.

Table 4-2: Adjusted Reference Temperature Evaluation for the Oconee Unit 2 Reactor Vessel Beltline Materials at 54 EFPY

Material Description		54 EFPY Fluence (n/cm²)			ΔRT_{NDT} (°F) at 54 EFPY		Std. Deviation		Margin (°F)		ART (°F) at 54 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Inner Wetted Surface	¼T Location	¾T Location	¼T Location	¾T Location	σ_I	σ_Δ	¼T Location	¾T Location	¼T Location	¾T Location
LNB Forging (Location 3)[a]	AMX 77	1.32E+19	7.72E+18	2.80E+18	88.1	62.0	31.0	17.0	70.7	70.7	{161.8}	{135.7}
LNB Forging (Location 4)[a]	AMX 77	1.25E+18	5.90E+17	1.40E+17	30.4	13.0	31.0	17.0	69.0	63.4	102.4	79.4
US Forging	AAW 163	1.40E+19	8.19E+18	2.98E+18	24.5	17.4	0.0	17.0	24.5	17.4	69.0	54.8
LS Forging	AWG 164	1.40E+19	8.20E+18	2.98E+18	18.9	13.4	0.0	17.0	18.9	13.4	57.8	46.8
LNB to US Circ. Weld (100%)	WF-154	1.32E+19	7.72E+18	2.80E+18	169.4	119.2	11.6	28.0	60.6	60.6	132.0	81.8
US to LS Circ. Weld (100%)	WF-25	1.35E+19	7.88E+18	2.86E+18	205.8	145.2	12.8	28.0	61.6	61.6	[193.1]	[132.5]

[a] See Figure 2-2

[ ] –Highest values of the adjusted reference temperatures for circumferential welds.

{ } – Highest values of the adjusted reference temperatures for base metal (no longitudinal welds in Oconee Unit 2).

Table 4-3: Adjusted Reference Temperature Evaluation for the Oconee Unit 3 Reactor Vessel Beltline Materials at 54 EFPY

Material Description		54 EFPY Fluence (n/cm^2)			ΔRT_{NDT} (°F) at 54 EFPY		Std. Deviation		Margin (°F)		ART (°F) at 54 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Inner Wetted Surface	¼T Location	¾T Location	¼T Location	¾T Location	σ_I	σ_Δ	¼T Location	¾T Location	¼T Location	¾T Location
LNB Forging (Location 3)[a]	4680	1.26E+19	7.36E+18	2.67E+18	87.7	61.5	31.0	17.0	70.7	70.7	{161.4}	{135.2}
LNB Forging (Location 4)[a]	4680	1.25E+18	5.90E+17	1.40E+17	30.7	13.2	31.0	17.0	69.2	63.4	102.9	79.5
US Forging	AWS 192	1.38E+19	8.07E+18	2.93E+18	33.8	23.9	0.0	17.0	34.0	34.0	107.8	97.9
LS Forging	ANK 191	1.39E+19	8.16E+18	2.96E+18	16.4	11.6	0.0	8.5	16.4	11.6	72.8	63.2
LNB to US Circ. Weld (100%)	WF-200	1.26E+19	7.36E+18	2.67E+18	162.7	114.0	9.6	28.0	59.2	59.2	137.7	89.0
US to LS Circ. Weld (ID 75%)	WF-67	1.34E+19	7.83E+18	N/A	167.7	N/A	12.2	28.0	61.1	N/A	[195.6]	N/A
US to LS Circ. Weld (OD 25%)	WF-70	1.34E+19	N/A	2.85E+18	N/A	130.9	13.7	28.0	N/A	62.3	N/A	[162.1]

[a] See Figure 2-2

[ ] –Controlling values of the adjusted reference temperatures for circumferential welds.

{ } – Highest values of the adjusted reference temperatures for base metal (no longitudinal welds in Oconee Unit 3).

5.0 RECENT LICENSE EXEMPTION REQUESTS

Virginia Electric and Power Company (Dominion) requested an exemption from the requirements of 10 CFR 50.61 and 10 CFR 50 Appendix G to revise the initial RT_{NDT} and associated σ_I values of the Linde 80 weld materials present in the beltline region of Surry Unit 1 and Unit 2 reactor pressure vessels using AREVA Topical Report BAW-2308, Revision 1-A [19]. This exemption request was accompanied by RT_{PTS} and ART calculations, which utilized the revised initial RT_{NDT} and associated σ_I values in BAW-2308, Revision 1-A. Attachment 2 of Reference 19 details the Surry exemption request, to which the Oconee license exemption request below was based. The U.S. Nuclear Regulatory Commission approved this license exemption request for the Surry Power Station Units 1 and 2 [20].

Florida Power and Light Company has requested an exemption from the requirements of 10 CFR 50.61 and 10 CFR 50 Appendix G for Turkey Point Units 3 and 4 [21]. The exemption request incorporates the methodology of BAW-2308, Revision 2-A.

FirstEnergy Nuclear Operating Company has requested an amendment of the operating license for the Davis-Besse Nuclear Power Station [22]. The amendment incorporates the methodology of BAW-2308, Revision 2-A for 10 CFR 50.61 and 10 CFR 50 Appendix G.

6.0 LICENSE EXEMPTION JUSTIFICATION

6.1 Introduction

In accordance with the provisions of 10 CFR 50.60(b) and 10 CFR 50.12, Duke Energy is submitting a request for exemption from certain requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Thermal Shock Events," and 10 CFR 50, Appendix G, "Fracture Toughness Requirements." The requested exemption would allow use of alternate initial RT_{NDT} (reference nil ductility temperature), as described in AREVA NP Topical Report BAW-2308, Revision 1-A and Revision 2-A, for determining the adjusted RT_{NDT} of the Linde 80 weld materials present in the beltline region of the Oconee Units 1, 2 and 3 reactor pressure vessels.

6.2 Background

10 CFR 50.61 (a)(5) and 10 CFR 50, Appendix G (II)(D)(i), require that the pre-service or unirradiated condition RT_{NDT} be evaluated according to the procedures in the ASME Code, Section III, Paragraph NB-2331, from Charpy V-notch impact tests and drop weight tests.

AREVA NP Topical Report BAW-2308, Rev. 2-A provides an NRC-approved alternate initial RT_{NDT} and associated σ_I values of the Linde 80 weld materials present in the beltline region of the reactor pressure vessels at Oconee Units 1, 2 and 3.

The following Condition and Limitation is stated in the NRC's Safety Evaluation for Topical Report BAW-2308, Rev. 1-A [17]:

> *"Any licensee who wants to utilize the methodology of TR* BAW-2308, *Revision 1 as outlined in items* (1) *through (3) above, must request an exemption, per* 10 *CFR* 50.12, *from the requirements of Appendix G to* 10 *CFR Part* 50 *and* 10 *CFR 50.61 to do so."*

In the above quotation, Condition and Limitation (1) pertains to NRC-accepted values of initial (unirradiated) reference temperature, IRT_{To}, and the corresponding uncertainty term, σ_I, for Linde 80 weld materials based on the Master Curve methodology using direct testing of fracture toughness in accordance with ASTM Standard Test Method E-1921.

Condition and Limitation (2) requires that a minimum chemistry factor of 167.0°F be applied when the methodology of Regulatory Guide 1.99, Revision 2, and 10 CFR 50.61 is used to assess the shift in nil-ductility transition temperature due to irradiation.

Condition and Limitation (3) requires that a value of σ_Δ = 28.0°F be used to determine the margin term, as defined in Topical Report BAW-2308, Revision 2-A, and Regulatory Guide 1.99, Revision 2.

6.3 Proposed Exemption

The exemption requested by Duke Energy addresses portions of the following regulations:

(1) Appendix G to 10 CFR Part 50, which sets forth fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the system may be subjected over its service lifetime;

(2) 10 CFR 50.61, which sets forth fracture toughness requirements for protection against pressurized thermal shock (PTS).

The exemption from Appendix G to 10 CFR 50 is to replace the required use of the existing Charpy V-notch and drop-weight-based methodology with the use of an alternate methodology that incorporates the use of fracture toughness test data for evaluating the integrity of the Linde 80 weld materials present in the Oconee Units 1, 2 and 3 RPV beltline regions. The alternate methodology employs direct fracture toughness testing per the Master Curve methodology based on use of ASTM Standard Method E 1921 (1997 and 2002 editions), and ASME Code Case N-629. The exemption is required since Appendix G to 10 CFR 50 requires that for the pre-service or unirradiated condition, RT_{NDT} be evaluated by Charpy V-notch impact tests and drop weight tests according to the procedures in the ASME Code, Paragraph NB-2331.

The exemption from 10 CFR 50.61 is to use an alternate methodology to allow the use of direct fracture toughness test data for evaluating the integrity of the Linde 80 weld materials present in the Oconee Units 1, 2 and 3 RPV beltline regions, based on the use of ASTM E 1921 (1997 and 2002 editions) and ASME Code Case N-629. The exemption is required because the methodology for evaluating RPV material fracture toughness in 10 CFR 50.61 requires that the pre-service or unirradiated condition be evaluated using Charpy V-notch impact tests and drop weight tests according to the procedures in the ASME Code, Paragraph NB-2331.

Additionally, the NRC's Safety Evaluation for Topical Report BAW-2308, Revision 1-A, concludes that an exemption is required to address issues related to 10 CFR 50.61 inasmuch as the methodology presented in Topical Report BAW-2308, Revision 1-A, as modified and approved by the NRC staff, represents a significant change to the methodology specified in 10 CFR 50.61 for determining the PTS reference temperature (RT_{PTS}) value for Linde 80 weld material. The changes in the methodology described in BAW-2308, Revision 1-A, with respect to the methodology per 10 CFR 50.61, include the requirements for use of a minimum chemistry factor of 167°F and a value of σ_Δ = 28.0°F for Linde 80 weld materials.

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that: 1) the exemption is authorized by law, 2) the exemption will not result in an undue risk to public health and safety, 3) the exemption is consistent with the common defense and security, and 4) special circumstances, as defined in 10 CFR 50.1 2(a)(2) are present. The requested exemption to allow the use of Topical Report BAW-2308, Revision 1-A and Revision 2-A (Revision 2-A is a supplement to Revision 1-A), as the basis for the Linde 80 weld material initial properties at Oconee Units 1, 2 and 3 satisfy these requirements as described below.

1. The requested exemption is authorized by law.

No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendix G when an exemption is granted by the Commission under 10 CFR 50.1 2.

In addition, 10 CFR 50.61 permits other methods for use in determining the initial material properties provided such methods are approved by the Director, Office of Nuclear Reactor Regulation.

2. The requested exemption does not present an undue risk to the public health and safety.

The proposed material initial properties basis described in Topical Report BAW-2308 Revision 2-A represents an NRC-approved methodology for establishing weld wire specific and generic IRT_{T_0} values for Linde 80 welds. Topical Report BA-2308, Revision 2-A, includes appropriate conservatisms to ensure that use of the proposed initial material properties basis does not increase the probability of occurrence or the consequences of an accident at Oconee Units 1, 2 and 3, and will not create the possibility for a new or different type of accident that could pose a risk to public health and safety.

The use of this proposed approach ensures that the intent of the requirements specified in 10 CFR 50 Appendix G and 10 CFR 50.61 are satisfied.

The requested exemption is consistent with the NRC staff requirements specified in the Safety Evaluation for the approved Topical Report BAW-2308, Revision 1-A and Revision 2-A; consequently, the exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security.

The requested exemption is specifically concerned with RPV material properties and is consistent with NRC staff requirements specified in the Safety Evaluation for approved Topical Report BAW-2308, Revision 2-A. Consequently, the requested exemption will not endanger the common defense and security.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.61 and 10 CFR 50 Appendix G.

Pursuant to 10 CFR 50.1 2(a)(2), the NRC will not consider granting an exemption to the regulations unless special circumstances are present. The requested exemption meets the special circumstances of paragraph 10 CFR 50.1 2(a)(2)(ii) since application of the methodology in BAW-2308, Revision 1-A and Revision 2-A, in this particular circumstance serves the underlying purpose of the regulations.

The underlying purpose of 10 CFR 50.61 and 10 CFR 50 Appendix G is to protect the integrity of the reactor coolant pressure boundary by ensuring that each reactor vessel material has adequate fracture toughness. Application of paragraph NB-2331 of ASME Section III in the determination of initial material properties was conservatively developed based on the level of knowledge existing in the early 1970s concerning RPV materials and the estimated effects of operation. Since the early 1970s, the level of knowledge concerning these topics has greatly expanded. This increased knowledge level permits relaxation of the ASME III NB-2331 requirements via application of Topical Report BAW-2308, Revision 2-A, while maintaining the underlying purpose of the ASME Code and NRC regulations to ensure an acceptable margin of safety is maintained.

This submittal presents the reactor vessel integrity assessments for Oconee Units 1, 2 and 3 utilizing the methodology of Topical Report BAW-2308, Revision 2-A for Linde 80 weld materials. The assessment documents the integrity of the RPV for Oconee Units 1, 2 and 3 relative to the requirements and underlying purpose of 10 CFR 50.61 and 10 CFR 50 Appendix G.

Therefore, the intent of 10 CFR 50.61 and 10 CFR 50 Appendix G will continue to be satisfied for the proposed change in reactor vessel material initial properties basis, thus justifying the exemption request. Issuance of an exemption from the criteria of these regulations to permit the use of Topical Report BAW-2308, Revision 2-A for Oconee Units 1, 2 and 3 will not compromise the safe operation of the reactors, and will ensure that RPV integrity is maintained.

7.0 REFERENCES

1. AREVA Document 32-9109428-001, "RTPTS Values for Oconee Units 1, 2 and 3 at 60 Calendar Years," April 2010.
2. AREVA Document 32-9108770-002, "ART Values for Oconee Units 1, 2 and 3 at 60 Calendar Years," June 2010.
3. AREVA Document 43-2308-004, "Initial RT_{NDT} of Linde 80 Weld Materials," (BAW-2308, Revision 2-A), March 2008.
4. Code of Federal Regulations, Title 10, "Domestic Licensing of Production and Utilization Facilities," Part 50.12, "Specific Exemptions," Effective Date: December 12, 1995.
5. Code of Federal Regulations, Title 10, "Domestic Licensing of Production and Utilization Facilities," Part 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock," Effective Date: August 28, 1996.
6. Code of Federal Regulations, Title 10, "Domestic Licensing of Production and Utilization Facilities," Part 50 Appendix G, "Fracture Toughness Requirements," Effective Date: December 19, 1995.
7. AREVA NP Document 43-2325-01, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," (BAW-2325, Revision 1), January 1999.
8. Letter from Duke Energy Corporation forwarding application for renewal of operating licenses for the Oconee Nuclear Station, Unit Nos. 1, 2, and 3, U. S. Nuclear Regulatory Commission, ACN: 9807200136, Fiche: A4344:001-A4347:255, July 6, 1998.
9. U. S. Nuclear Regulatory Commission, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Regulatory Guide 1.190, March 2001.
10. J. R. Worsham, et al., "Fluence and Uncertainty Methodologies," BAW-2241P-A, Revision 1, AREVA NP, Inc., Lynchburg, Virginia, April 2000.
11. AREVA NP Document 43-1543-04, "Master Integrated Reactor Vessel Surveillance Program," BAW-1543, Revision 4, February 1993.

12. AREVA Drawing 02-128706-09, "Core Flooding Nozzle," Oconee Unit 1.
13. BAW-1820, "Babcock & Wilcox Owner's Group 177-Fuel Assembly Reactor Vessel and Surveillance Program Materials Information," December 1984.
14. AREVA Drawing 02-128737-06, "Upper Shell Assembly," Oconee Unit 2.
15. AREVA Drawing 02-99319-04, "Upper Shell Forging," Oconee Units 2 and 3.
16. AREVA Drawing 02-149904-05, "Upper Shell Assembly," Oconee Unit 3.
17. AREVA NP Document 43-2308-002, "Initial RT_{NDT} of Linde 80 Weld Materials," (BAW-2308, Revision 1-A), August 2005.
18. U. S. Nuclear Regulatory Commission, "Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Revision 2, May 1988.
19. Letter to NRC, "Virginia Electric and Power Company Surry Power Station Units 1 and 2 Update to NRC Reactor Vessel Integrity Database and Exemption Request for Alternate Material Properties Basis Per 10 CFR 50.60(b)," ML061650080, June 2006.
20. Letter from NRC, "Surry Power Station, Unit Nos. 1 and 2, Exemption from the Requirements of 10 CFR Part 50, Appendix G and 10 CFR Part 50, Section 50.61," ML071160287, June 2007.
21. Letter to NRC, "Turkey Point, Units 3 and 4, Update to NRC Reactor Vessel Integrity Database and Exemption Request for Alternate Material Properties Bases Per 10 CFR 50.12 and 10 CFR 50.60 (b)," ML090920408, March 2009.
22. Letter to NRC, "Davis-Besse Nuclear Power Station, Unit No. 1 Docket No. 50-346, License No. NPF-3 License Amendment Request to Incorporate the Use of Alternate Methodologies for the Development of Reactor Pressure Vessel Pressure-Temperature Limit Curves, and Request for Exemption From Certain Requirements Contained in 10 CFR 50.61 and 10 CFR 50, Appendix G," ML091130228, April 2009.