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Scott L. Batson  
Vice President  
Oconee Nuclear Station

**Duke Energy**  
ON01VP | 7800 Rochester Hwy  
Seneca, SC 29672

o: 864.873.3274  
f: 864.873.4208

Scott.Batson@duke-energy.com

10 CFR 50.90

September 10, 2013

ATTN: Document Control Desk  
U. S. Nuclear Regulatory Commission  
11555 Rockville Pike  
Rockville, MD 20852-2746

Subject: Duke Energy Carolinas, LLC  
Oconee Nuclear Station (ONS), Units 1, 2, and 3  
Docket Numbers 50-269, 50-270, and 50-287  
Renewed Operating License Numbers DPR-38, DPR-47, and DPR-55  
Response to Request for Additional Information (RAI) regarding License  
Amendment Request to Update Pressure-Temperature Limit Curves  
License Amendment Request (LAR) No. 2012-10, Supplement 1

On February 22, 2013, Duke Energy Carolinas, LLC (Duke Energy) submitted a License Amendment Request (LAR) requesting the Nuclear Regulatory Commission (NRC) approve new pressure-temperature (P-T) limit curves applicable to 54 effective full power years (EFPY) to replace the 33 EFPY P-T limit curves in Technical Specification (TS) 3.4.3 and approve changes to the operational requirements for unit heatup and cooldown in TS Tables 3.4.3-1 and 3.4.3-2. By letter dated June 21, 2013, NRC requested Duke Energy to provide additional information associated with the LAR. By electronic mail dated August 16, 2013, the NRC granted an extension to the due date for this response to September 10, 2013, to allow Duke Energy time to investigate an AREVA NP, Inc. (AREVA) notification made on August 16, 2013, associated with a potential deficiency in the current 33 EFPY and proposed 54 EFPY P-T limit curves for ONS Unit 3.

By letter dated August 16, 2013, AREVA notified Duke Energy that AREVA is evaluating issues under its Corrective Action Program (CAP) that indicate that the current P-T curves for ONS Unit 3 may be deficient. This deficiency was discovered during AREVA's preparation of responses to NRC RAIs on the proposed 54 EFPY P-T curves. During preparation of the RAI responses, AREVA reviewed the basis for the generic values to assess their applicability to the extended beltline reactor pressure vessel (RPV) components. As a result of this review, AREVA determined that existing generic values for initial reference temperature for nil-ductility transition ( $RT_{NDT}$ ) and copper content, while compliant with existing NRC guidance and regulations, may not have been appropriate. AREVA identified more limiting adjusted reference temperature (ART) input values using material data believed to be more appropriate for the ONS Unit 3 Lower Nozzle Belt (LNB) forging than the generic material data historically

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applied. Use of the more limiting ART data impacts the ONS Unit 3 P-T curves, specifically Figure 3.4.3-7 and Figure 3.4.3-8 in the ONS Technical Specifications.

As a near term conservative measure, AREVA recommended limiting heat-up and cool-down P-T curves by using the most limiting composite curve established by the Surge Line Limit Curve and the current ONS Unit 3 Low Temperature Overpressure Protection (LTOP) limit to ensure safe operation. This issue was entered into the Duke Energy CAP and appropriate conservative measures have been implemented until longer term corrective actions can be completed.

AREVA subsequently confirmed that this deficiency does not impact the RAI responses for the 54 EFPY P-T LAR since the RAIs are directed toward extended beltline materials. AREVA's assessment of extended beltline materials is not impacted by this issue. However, the ONS Unit 3 P-T Curves (33 EFPY and 54 EFPY) need to be revised. The 33 EFPY P-T curves were revised on August 23, 2013. AREVA is currently revising the 54 EFPY P-T curves. The use of the new 33 EFPY P-T curves is being controlled administratively until such time that the 54 EFPY P-T curves are approved and implemented at ONS. Duke Energy will submit a revision to the 54 EFPY P-T Curve LAR, which includes revised P-T curves for ONS Unit 3, by October 31, 2013.

Enclosures 1 and 2, which are non proprietary and proprietary respectively, provide the requested information. Enclosure 2 contains information that has been classified as proprietary by AREVA. The affidavit in Enclosure 3 sets forth AREVA basis on which the proprietary information may be withheld from public disclosure by the NRC pursuant to 10 CFR 2.390. A list of Regulatory Commitments is provided in Enclosure 4.

If there are any additional questions, please contact Boyd Shingleton, ONS Regulatory Affairs, at (864) 873-4716.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 10, 2013.

Sincerely,



Scott L. Batson  
Vice President  
Oconee Nuclear Station

Nuclear Regulatory Commission  
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bcc w/Enclosures:

D. E. Whitaker  
J. M. Shuping  
S. N. Severance  
L. S. Nichols  
T. L. Patterson  
E. L. Anderson  
M. E. Bailey  
R. D. Hart – CNS  
J. N. Robertson - MNS  
D. B. Alexander– NRI&IA  
D. R. Westcott - CR3  
L. J. Grzeck – BNP  
D. H. Corlett - HNP  
W.R. Hightower – RNP  
Regulatory Affairs Manager - ONS  
NSRB, EC05N  
ELL, ECO50  
File - T.S. Working  
ONS Document Management

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cc w/Enclosures:

**Mr. Victor McCree, Regional Administrator**  
**U. S. Nuclear Regulatory Commission - Region II**  
**Marquis One Tower**  
**245 Peachtree Center Ave., NE, Suite 1200**  
**Atlanta, GA 30303-1257**

**Mr. Richard Guzman, Senior Project Manager**  
**(by electronic mail only)**  
**Office of Nuclear Reactor Regulation**  
**U. S. Nuclear Regulatory Commission**  
**11555 Rockville Pike**  
**Mail Stop O-8G9A**  
**Rockville, MD 20852**

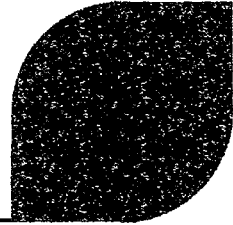
**Mr. Eddy Crowe**  
**Senior Resident Inspector**  
**Oconee Nuclear Site**

**Ms. Susan E. Jenkins, Manager**  
**Radioactive & Infectious Waste Management**  
**Division of Waste Management**  
**South Carolina Department of Health and Environmental Control**  
**2600 Bull St.**  
**Columbia, SC 29201**

**ENCLOSURE 1**

**NON PROPRIETARY VERSION**

**Duke Energy Response to  
NRC Request for Additional Information (RAI)**



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ANP-3127Q1NP

Revision 1

**Response to NRC Request for Additional  
Information Regarding License Amendment  
Request to Update Pressure-Temperature  
Limit Curves for Oconee Units 1, 2, and 3**

August 2013

AREVA NP Inc.

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Revision 1

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Request to Update Pressure-Temperature Limit Curves for Oconee Units 1, 2, and 3

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## Nature of Changes

Revision	Section(s) or Page(s)	Description and Justification
0	All	Initial Issue
1	Page v	Added RCPB and SER to Nomenclature
	2.1.1 & 2.1.2.1	Minor editorial changes
	2.1.2.1.4	Removed last sentence of first paragraph
	2.2.2	Classified the maximum 12.0 inch thickness of the nozzle belt forging as non-proprietary data

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## Nomenclature

<b>Acronym</b>	<b>Definition</b>
ART	Adjusted Reference Temperature
ASME	American Society of Mechanical Engineers
B&W	Babcock & Wilcox
EFPY	Effective Full Power Years
LST	Lowest Service Temperature
NRC	United States Nuclear Regulatory Commission
ONS (-1, -2, -3)	Oconee Nuclear Station (Unit 1, Unit 2, Unit 3)
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCPB	Reactor Coolant Pressure Boundary
RV	Reactor Vessel
RT <sub>NDT</sub>	Reference Temperature for Nil-Ductility Transition
SER	Safety Evaluation Report
TLAA	Time-Limited Aging Analysis

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## **ABSTRACT**

AREVA Document ANP-3127, Revision 1, "Oconee Nuclear Station Units 1, 2 & 3 Pressure-Temperature Limits at 54 EFPY," was prepared by AREVA for Duke Energy and subsequently submitted to the Nuclear Regulatory Commission (NRC) by Duke Energy. The NRC has issued the first set of Requests for Additional Information (RAIs) on this submittal, and this report provides the answers for generic RAI 1 and application-specific RAIs 2 through 8.

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## 1.0 INTRODUCTION AND SUMMARY

AREVA Document ANP-3127, Revision 1<sup>1</sup>, "Oconee Nuclear Station Units 1, 2 & 3 Pressure-Temperature Limits at 54 EFPY," was prepared by AREVA for Duke Energy and subsequently submitted to the NRC by Duke Energy<sup>2</sup>. The NRC has issued the first set of Requests for Additional Information (RAIs)<sup>3</sup> on this submittal, and this report provides the answers to those RAIs.

## 2.0 REQUESTS FOR ADDITIONAL INFORMATION (RAIs) AND RESPONSES

The NRC RAIs are reproduced from Reference 3 in Sections 2.1.1 through 2.8.1. The AREVA/Duke Energy responses are in Sections 2.1.2 through 2.8.2.

### 2.1 RAI-1

#### 2.1.1 Statement of RAI-1

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, Paragraph IV.A states that, "*The pressure-retaining components of the reactor coolant pressure boundary [RCPB] that are made of ferritic materials must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code [ASME Code, Section III], supplemented by the additional requirements set forth ... [in paragraph IV.A.2, "Pressure-Temperature (P-T) Limits and Minimum Temperature Requirements"].*" Therefore, 10 CFR Part 50, Appendix G requires that P-T limits be developed for the ferritic materials in the reactor vessel (RV) beltline (neutron fluence  $\geq 1 \times 10^{17}$  n/cm<sup>2</sup>, E > 1 MeV), as well as ferritic materials not in the RV beltline (neutron fluence <  $1 \times 10^{17}$  n/cm<sup>2</sup>, E > 1 MeV). Further, 10 CFR Part 50, Appendix G requires that all RCPB components must meet the American Society of Mechanical Engineers Code (ASME Code), Section III requirements. The relevant ASME Code, Section III requirement that will affect the P-T limits is the lowest service temperature requirement for all reactor coolant pressure boundary RCPB) components specified in Section III, NB-2332(b).

The P-T limit calculations for ferritic RCPB components that are not RV beltline shell materials may define P-T curves that are more limiting than those calculated for the RV beltline shell materials due to the following factors:

1. RV nozzles, penetrations, and other discontinuities have complex geometries that may exhibit significantly higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature ( $RT_{NDT}$ ) for these components is not as high as that of RV beltline shell materials that have simpler geometries.

2. Ferritic RCPB components that are not part of the RV may have initial  $RT_{NDT}$  values, which may define a more restrictive lowest operating temperature in the P-T limits than those for the RV beltline shell materials.

Consequently, please describe how the current P-T limit curves at 54 effective full power years (EFPY) for the Oconee units and the methodology used to develop these curves considered all RV materials (beltline and non-beltline) and the lowest service temperature of all ferritic RCPB materials, consistent with the requirements of 10 CFR Part 50, Appendix G in the proposed update. Your description shall include the following:

- Using a proposed composite heatup curve and a proposed cooldown curve as examples, point out the segments that were limited by closure head, outer nozzle, and beltline.
- Confirm availability of material data (initial  $RT_{NDT}$  and copper and nickel contents) for all non-beltline materials for all three unit RPVs and demonstrate that none of them will become limiting under the 54 EFPY fluence.
- Confirm that the lowest service temperatures (LSTs) for all Ferritic RCPB components that are not part of the RV have been established for all three Oconee units, and the lowest temperature of 60 °F in the proposed P-T limits are higher than these LSTs.

## **2.1.2 Response to RAI-1**

### **2.1.2.1 General Response to RAI-1**

The 54 EFPY P-T limits for Oconee Units 1, 2, and 3 were developed using the methods described in Topical Report BAW-10046A, Revision 2<sup>4</sup>. The P-T limit curves were developed in accordance with the requirements of 10 CFR 50, Appendix G<sup>5</sup>, utilizing the analytical methods of Topical Report BAW-10046A Revision 2, and ASME Code Section XI, Appendix G<sup>6</sup>.

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For Babcock & Wilcox nuclear steam supply systems, Topical Report BAW-10046A, Revision 2<sup>4</sup> describes methods for compliance with the requirements of 10 CFR 50 Appendix G, "Fracture Toughness Requirements". The safety evaluation report (SER) for this report, BAW-10046A, Revision 2<sup>4</sup> states "BAW 10046, Rev. 2 describes acceptable methods for the development of allowable pressure-temperature limits for normal operation and for test conditions to assure the prevention of non-ductile fracture. It may be referenced in future applications..."

The methodology used for B&W plants includes all the ferritic components in the Reactor Coolant System. Acceptable methods for the determination of P-T limits for the closure head, the reactor vessel outlet nozzle, and beltline region are documented. As stated in BAW-10046A, Revision 2, "These three regions are the only ones that, at different stages of the vessel's design life, regulate the pressure-temperature limitations of the RC system for normal operation and inservice pressure tests." The P-T limits determined for these components are determined and the resulting composite limiting curve is then determined utilizing the standard B&W methodology. Per BAW-10046, Revision 2, "The components for which the lowest service temperature must be defined include the RC loop piping and the control rod drive mechanism (the CRDM is an appurtenance to the reactor vessel). The lowest service temperature of these components is [ ] °F (based on  $RT_{NDT} + 100^{\circ}F$ ) for the piping and [ ] °F for the CRDM."

BAW-10046A, Revision 2 addresses all ferritic components of the beltline and non-beltline regions of the RCPB. The NRC has reviewed the methods in this Topical Report and approved this report by issuance of SER dated April 30, 1986.

## **2.1.2.1.1 Response to Item 1 of RAI-1**

One item that was not considered in BAW-10046A, Revision 2 was embrittlement from irradiation damage in the ferritic components adjacent to the components that have been traditionally considered as part of the RV beltline. As plants operate into the period of extended operation, these components adjacent to the RV beltline can

potentially accumulate fluence to an extent that was not previously considered. Of these components, the RV nozzles may eventually require some further evaluation. For Oconee Units 1, 2, and 3 at 54 EFPY, the fluence at the lower nozzle belt forging to outlet nozzle forging weld is less than  $1 \times 10^{17}$  n/cm<sup>2</sup> (E>1.0 MeV). Therefore, at 54 EFPY the attenuated fluence at the nozzle corner is even lower than this value. Since per NUREG-1801, Revision 2<sup>7</sup>, a time-limited aging analysis (TLAA) for neutron irradiation embrittlement is required for materials with a neutron fluence greater than  $1 \times 10^{17}$  n/cm<sup>2</sup> (E>1.0 MeV), embrittlement in this region does not need to be considered for the period of extended operation. Therefore, the original basis document, BAW-10046A, Revision 2 remains applicable for deriving P-T limits for the Oconee Unit 1, 2, and 3 nozzle regions at 54 EFPY.

#### **2.1.2.1.2 Response to Item 2 of RAI-1**

The methodology used to develop Oconee Units 1, 2, and 3 P-T limits at 54 EFPY used methodology that considered ferritic materials outside the reactor vessel (RV) beltline region having complex geometries that may exhibit higher stresses than those of the RV beltline shell regions. Components such as the RV outlet nozzles, the RV closure head, and the reactor coolant piping have reference temperatures (RT<sub>NDT</sub>'s) that are not as high as those for the RV beltline shell materials which have simpler geometries. These non-beltline components have been evaluated based on lowest service temperatures, as discussed in the response to the third bullet of RAI-1.

#### **2.1.2.1.3 Response to First Bullet of RAI-1**

The Oconee pressure-temperature limits were explicitly developed for the principal segments of the reactor vessel that are known to control reactor coolant system pressure: the reactor vessel closure head flange, the reactor vessel inlet, outlet, and core flood nozzles, the nozzle belt (or upper shell) region, the beltline region near the reactor core, the circumferential welds joining the lower and intermediate shell courses, and the axial seam welds in Unit 1. The governing P-T curve is the collection of controlling pressures considering all segments of the vessel. For the most part, data

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from the beltline region is the controlling segment due to the high adjusted reference temperatures of the beltline axial welds, as shown in the heatup and cooldown curves provided below for Oconee Unit 1. Only the portion of the heat up curve ranging from [ ] to [ ] °F is governed by the closure head flange region (as illustrated in Figure 2.1.2-1), while the entire cooldown curve of Figure 2.1.2-2 is controlled by the beltline axial welds. It is noted that the heatup curve between [ ] and [ ] °F is forced to be horizontal (flat) to avoid a negative slope when the reactor coolant pump starts at 170 °F.



**Figure 2.1.2-1 Composite Heatup P-T Curve for Oconee Unit 1 (Adjusted Figure 7-1 of ANP-3127)**

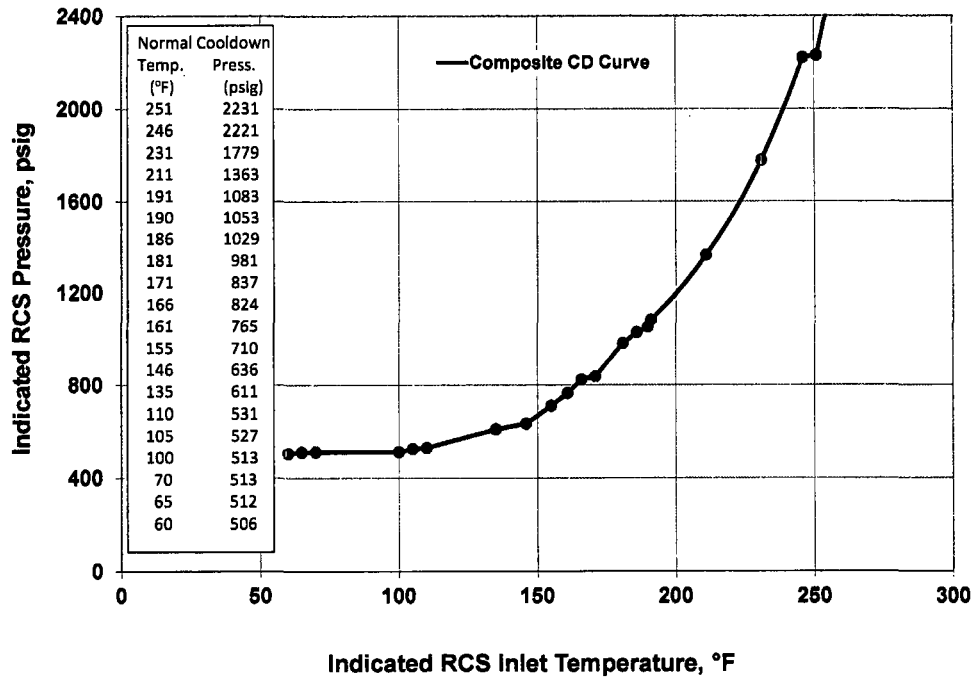


Figure 2.1.2-2 Composite Cooldown P-T Curve for Oconee Unit 1 (from Figure 7-2 of ANP-3127)

**2.1.2.1.4 Response to Second Bullet of RAI-1**

The material data (initial  $RT_{NDT}$  and copper and nickel contents) have been obtained for all three Oconee RPV non-beltline materials in the form of the original material CMTRs. Because of the vintage of the CMTRs, they contain limited material data, i.e., only nickel contents are reported. Due to the unavailability of measured initial  $RT_{NDT}$  values and copper contents for the Oconee RPV non-beltline materials, the use of generic mean values for these specific material properties could be applied.

Based on the projected fluence for the Oconee RPV non-beltline materials, components located above the lower nozzle belt forging to upper shell course circumferential weld (not including the lower nozzle belt forging) are projected to receive fluences less than  $10^{17}$  n/cm<sup>2</sup> (E > 1 MeV), i.e., these components are predicted to have fluences less than  $5.2 \times 10^{16}$  n/cm<sup>2</sup> (E > 1 MeV). These components are currently below the threshold requiring a TLAA for neutron irradiation embrittlement,  $1.0 \times 10^{17}$  n/cm<sup>2</sup>, as documented in NUREG-1801, Revision 2. Therefore for these components, the reference temperatures used as inputs to the development of the P-T curves for the Oconee units followed the methodology of BAW-10046A, Revision 2.

As for the RPV components in the lower portion of the vessel, the predicted 54 EPPY fluences at the lower shell forging to Dutchman forging welds are greater than  $10^{17}$  n/cm<sup>2</sup> (E > 1 MeV) for all three Oconee RPVs. Therefore, adjusted reference temperature values for the lower head components (transition ring forging, lower head, and their associated welds) were calculated in accordance with Regulatory Guide 1.99, Revision 2<sup>8</sup> using the lower shell forging to Dutchman forging weld fluences. The maximum adjusted reference temperature is conservatively calculated to be 115 °F (RPV lower head). Based on the predicted 54 EPPY adjusted reference temperatures, the P-T limits for portions of the vessel below the beltline region remain bounded by the closure head flange region of the vessel.

### 2.1.2.1.5 Response to Third Bullet of RAI-1

Per Article NB-3211, Sub-article (d)(2) of ASME Code Section III<sup>9</sup>, the lowest service temperature (LST) is the minimum fluid temperature retained by the component whenever pressure within the component exceeds 20% of the pre-operational system hydrostatic test pressure (location corrected values of 553 psig for Oconee 1 and 557 psig for Oconee 2 and 3). The ferritic RCPB components that are not part of the reactor vessels at Oconee Units 1, 2, and 3 are the carbon steel reactor coolant piping (minimum LST = [ ] °F) and the martensitic high-alloy chromium Type 403 modified steel motor tube (minimum LST = [ ] °F) of the CRDM as reported in BAW-10046, Revision 2. The P-T limits in Figure 2.1.2-1 for Oconee Unit 1 require that the reactor coolant temperature reach a temperature of 130 °F before the system pressure can exceed [ ] psig. Although [ ] °F is less than an RT<sub>NDT</sub>-based lowest service temperature of [ ] °F for the reactor coolant piping, it has been demonstrated by Appendix G analysis, as permitted by the alternate provision of Article NB-3211 in Sub-article (d)(1), that the allowable pressure in the reactor coolant piping is much higher than that of the closure head flange for the entire heatup transient. Figures 7-4 and 7-7 of ANP-3127, Revision 1 show that the system pressure at reactor coolant temperatures up to [ ] °F is less than the location corrected 20% preservice hydrostatic test pressure of [ ] psig for Oconee Units 2 and 3.

### 2.1.2.2 Conclusion for RAI-1

The 54 EFPY P-T limits for Oconee Units 1, 2, and 3 are consistent with the requirements of 10 CFR 50, Appendix G.

**2.2 RAI-2****2.2.1 Statement of RAI-2**

The fluence values at the one-quarter thickness (1/4T) and 3/4T of the RV in Table 3-1 of ANP-3127, Revision 1 (the ANP, Enclosure 2 of the February 22, 2013, submittal) appear to be not based on the same RV thickness for the same unit. Please confirm that you used Equation (3) in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to calculate the fluence at 1/4T and 3/4T of the RV. Please also provide the values that you used for "x" in Equation (3) in the ART calculations at 1/4T and 3/4T for the limiting materials shown in Table 3-1 and the RV radius that you used in the P-T limit calculations. Provide the same information for Units 2 and 3.

**2.2.2 Response to RAI-2**

The neutron fluence values at the 1/4T and 3/4T vessel wall locations were calculated in accordance with Equation 3 of Regulatory Guide 1.99, Revision 2<sup>8</sup>. For Oconee Units 1, 2 and 3 a minimum wall thickness of 8.44 inches was used for the reactor vessel beltline regions with a minimum cladding thickness of [ ] inches. The lower nozzle belt forgings vary in thickness from a minimum thickness of 8.44 inches to 12.0 inches with a minimum cladding thickness of [ ] inches. These wall thicknesses are depicted in Figure 2.2.2-1 and Figure 2.2.2-2. Based on the increase in thickness of the lower nozzle belt forgings, the depth into the wall measured from the inner (wetted) surface, "x", was calculated for the two (2) thicknesses:

**8.44 inch Wall Thickness**

$$\text{"x" @ } \frac{1}{4}\text{T} = [8.44 \cdot 0.25] + [ ] \text{ in.}$$

$$\text{"x" @ } \frac{3}{4}\text{T} = [8.44 \cdot 0.75] + [ ] \text{ in.}$$

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## 12.0 inch Wall Thickness

$$\text{"x" @ } \frac{1}{4}T = [12.0 \cdot 0.25] + [ \quad ] \text{ in.}$$

$$\text{"x" @ } \frac{3}{4}T = [12.0 \cdot 0.75] + [ \quad ] \text{ in.}$$

The  $\frac{1}{4}T$  and  $\frac{3}{4}T$  fluence values for the Oconee Units 1, 2 and 3 reactor vessel materials are calculated using these vessel wall depths and the neutron fluence at the inner wetted surface of the vessel; these values are those listed in Tables 3-1, Table 3-2 and Table 3-3 of ANP-3127, Revision 1.

The reactor vessel beltline inner radius used in the P-T limit calculations, 85- $\frac{1}{2}$  inches, is based on the vessel shell inner diameter equal to 171 inches. The nozzle belt inner radius used in the P-T limit calculations is [  $\quad$  ] inches.

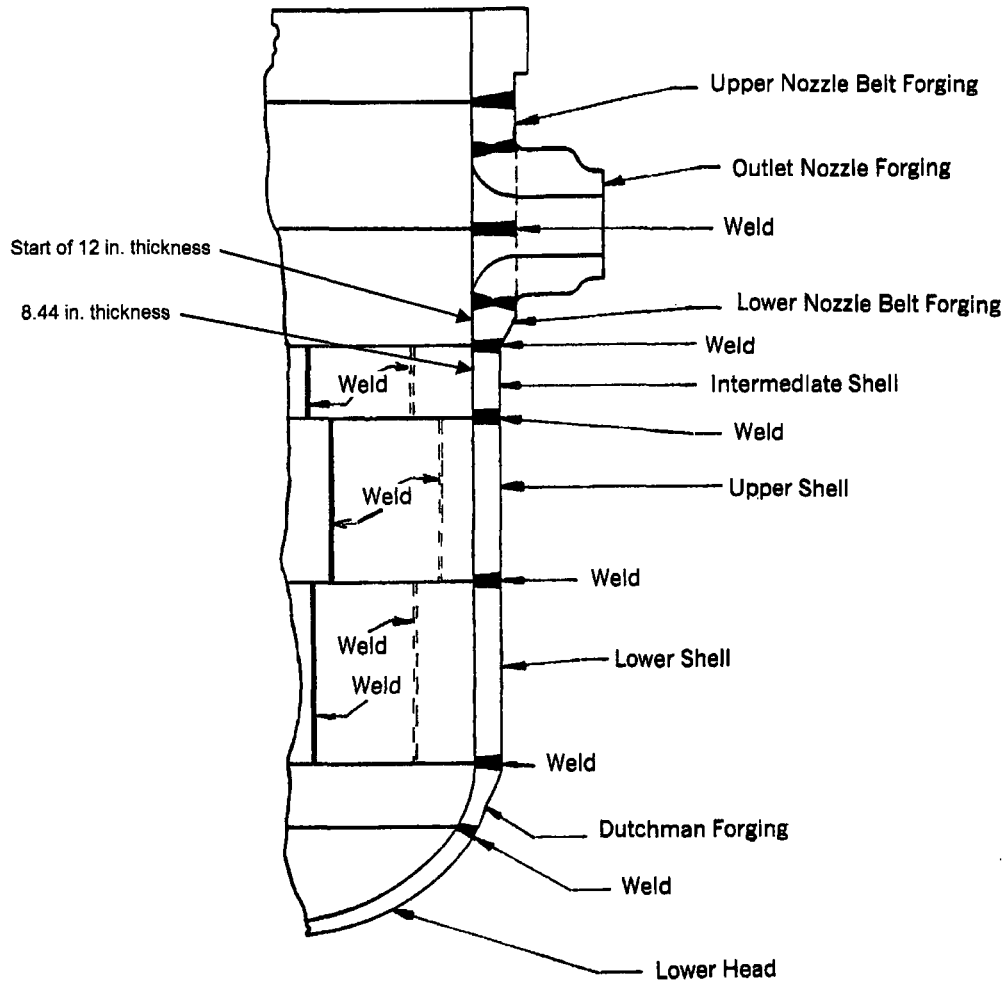


Figure 2.2.2-1 RPV Configuration for Oconee Unit 1

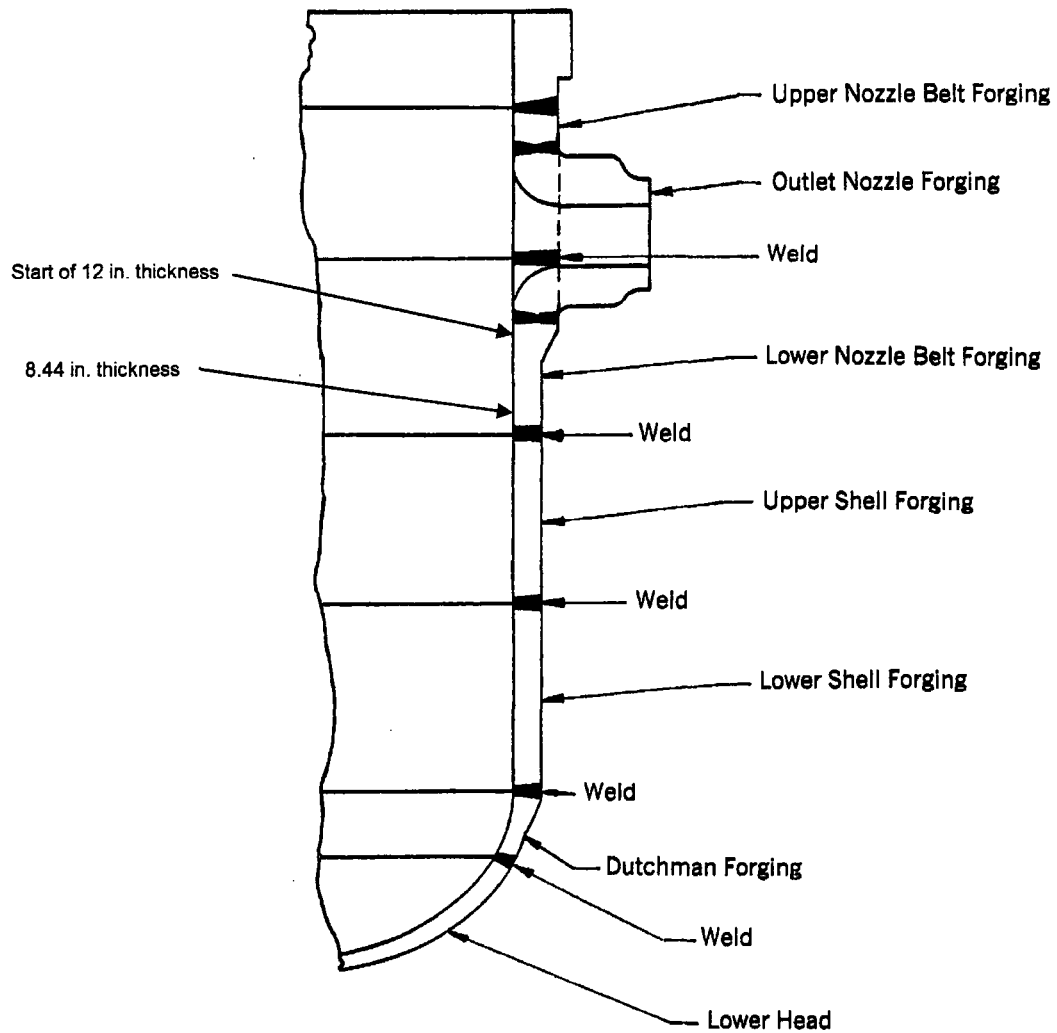


Figure 2.2.2-2 RPV Configuration for Oconee Units 2 and 3

## **2.3 RAI-3**

### **2.3.1 Statement of RAI-3**

Table 3-2 of the ANP indicated that the copper value of the lower nozzle belt to the upper shell circumferential weld of Heat 406L44 for Unit 2 is 0.27% and for the same type of weld of Heat 821T44 for Unit 3 is 0.24%, both are 0.01% lower than the corresponding values in the approved license renewal application. Since the accumulative effect of several small changes over the years can be significant, please provide justification for the revision.

### **2.3.2 Response to RAI-3**

The copper values for the approved license renewal application were based on best-estimate data available at the time the application was being prepared. These data were based on extensive chemical analyses performed on available as deposited weld metals fabricated with copper-plated filler wires and Linde 80 flux for use in reactor vessel embrittlement assessments. The sources for these weld metals included weldments in the form of nozzle belt forging dropouts, reactor vessel beltline region cutouts, surveillance program test blocks and test specimens, weld qualifications, and re-analysis of original weld qualification chemistry samples. On May 19-21, 1997 the NRC performed an inspection (Inspection Report No.: 99901300/97-01, dated January 28, 1998) at AREVA NP (formally Framatome Technologies, Inc.) to review records pertaining to the chemical composition of automatic submerged-arc welds in reactor vessels fabricated by Babcock & Wilcox (B&W). During the review process, the inspection identified additional data relevant to the determination of the best-estimate copper and nickel chemical contents for the high-copper Linde 80 weld metals.

Based on the additional data identified in the NRC inspection, the best-estimate copper and nickel chemical contents for the high-copper Linde 80 weld metals were updated. The updated best-estimate chemical compositions for the high-copper Linde 80 weld metals were determined by first establishing a mean value applicable for each particular material source (i.e., nozzle belt dropout, reactor vessel beltline region cutout,

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surveillance block/specimen, weld qualification, and weld qualification retest). These material source means were then used to calculate the mean values for the weld wire heat (e.g., mean-of-the-means).

The updated values based on the above chemical composition re-assessments were used in the adjusted reference temperature evaluations for the weld metals located in the Oconee Unit 1, 2 and 3 reactor vessels as reported in Table 3-1, Table 3-2, and Table 3-3 of ANP-3127, Revision 1.

## **2.4 RAI-4**

### **2.4.1 Statement of RAI-4**

Section 4.2 of the ANP states that, "A  $\frac{1}{4} t_{NB}$  ( $t_{NB}$  - the thickness at the nozzle belt) deep corner flaw is postulated on the inside surface of the reactor vessel inlet and outlet nozzles and core flood nozzle corner." This suggests that your nozzle methodology is more complete than that of BAW-10046, Revision 2, which performed analysis on outlet nozzles only, based on the belief that the outlet nozzle is the most limiting nozzle. Please confirm that actual calculations have been performed on inlet, outlet, and core flood nozzles to determine the most limiting P-T limits for nozzles in this license amendment request (LAR).

### **2.4.2 Response to RAI-4**

Although not required by BAW-10046, allowable pressures were calculated for the inlet, outlet, and core flood nozzles to ensure that the most limiting location was addressed in the non-beltline region of the vessel.

## **2.5 RAI-5**

### **2.5.1 Statement of RAI-5**

Section 4.4 of the ANP states, "The Pressure-Temperature limits derived for the reactor vessel head-to-flange conservatively bounds the minimum required temperature requirements as given in Table 1 of the Appendix G to 10 CFR Part 50." Please use a figure from Figure 7-1 to Figure 7-9 to support and explain this statement.

**2.5.2 Response to RAI-5**

P-T limits have been derived for the closure flange region of the Oconee reactor vessel head using stresses for the controlling transient condition (heatup stresses for a flaw on the outside surface in the crotch region of the vessel head) and the  $K_{IC}$  measure of fracture toughness. Based on Oconee specific analysis, the reactor coolant temperature must be at least 130 °F for pressures above 20% of the pre-service system hydrostatic test pressure, which conservatively satisfies the requirement of Item 2.b in Table 1 of Appendix G to 10 CFR Part 50 that the temperature must be at least 120 °F above the reference temperature of the closure head flange material, which is 0 °F for all three Oconee units. The pressure corrected (location adjusted to the pressure sensor) closure head limits are plotted in Figure 2.5.2-1 along with the required temperature for pressures above 20% of the preservice hydrostatic test pressure.



**Figure 2.5.2-1 Location Adjusted Closure Head P-T Limits**

## **2.6 RAI-6**

### **2.6.1 Statement of RAI-6**

Section 4.6 of the ANP indicated that both ramped and stepped transient definitions are modeled for normal operation heatup and cooldown. Please confirm whether the thermal stresses are based on (1) the stepped transient, which is likely to be limiting, or (2) the stepped and the ramped transients depending on time. If Case (2) applies, please provide a discussion of why the stepped transient may not be limiting.

### **2.6.2 Response to RAI-6**

Allowable pressures are determined for ramped transients at selected time increments during the transients, considering both the transient response at a selected time point as well as the steady state response to account for the possibility that the operator can hold a heatup or cooldown event at any point during the transient. Since it is not expected that the operator can follow a stepped transient nor can a step change in temperature actually be imposed on the reactor coolant, stepped transients are treated as a series of hold points, such that the allowable pressures for a stepped transient are calculated at the end of the step or hold period. Allowable pressures for the stepped transients, the ramped transients, and the steady state responses are compared and limiting values are selected to develop the P-T curves. In general, ramped transients control during heatup since the metal temperature at the flaw tip rises slower than it does for stepped transients. During cooldown, steady state conditions control at higher temperatures since steady state conditions are enforced by setting the flaw tip temperature equal to the fluid temperature at each time point in the transient, which results in a zero thermal stress intensity factor ( $K_{It} = 0$ ) while lowering the fracture toughness compared to the transient solution for either ramped or stepped transients. At lower temperatures, the transient through-wall temperature gradients are more fully developed and the allowable pressure is controlled by the thermal stress intensity factor. Under transient conditions, the ramped transient tends to produce slightly lower allowable pressures than the stepped transient as the cooldown rate decreases during the later stages of the cooldown transient.

## **2.7 RAI-7**

### **2.7.1 Statement of RAI-7**

Section 6 of the ANP discussed pressure corrections with the  $\Delta P$  listed in Table 6-1. Please confirm whether this  $\Delta P$  should be added to or subtracted from the calculated pressure values based on the ASME Code, Section XI, Appendix G methodology. Likewise, how do you adjust the calculated 1/4T metal temperature to the "indicated reactor coolant system (RCS) inlet temperature?"

### **2.7.2 Response to RAI-7**

- a) Based on the relative pressure heads between the pressure tap locations and the vessel locations where allowable pressures are calculated, the pressure corrections listed in Section 6 are subtracted from the calculated allowable pressures to obtain the location adjusted P-T limits.
- b) Allowable pressures are calculated as a function of the reactor coolant system fluid temperature (indicated RCS inlet temperature) based on the fracture toughness corresponding to the metal temperature at the 1/4T flaw depth. The transient thermal solution provides the necessary correlation between the fluid temperature and the 1/4T metal temperature.

## **2.8 RAI-8**

### **2.8.1 Statement of RAI-8**

Figures 7-1 and 7-2 of the ANP illustrate the P-T limits for heatup and cooldown, along with pressure temperature pairs of typical points along the P-T limit curves (on the left side of Figures 7-1 and 7-2). To assist the NRC staff verify the proposed heatup and cooldown curves, please use these pressure temperature pairs as examples and provide the corresponding thermal stress intensity factors ( $K_{ts}$ ).

## 2.8.2 Response to RAI-8

The Oconee-1 P-T limits for heatup and cooldown provided in Figures 7-1 and 7-2, respectively, of ANP-3127, Revision 1, are the controlling values considering flaw orientation (axial or circumferential), location (beltline, nozzle, or closure head flange), transient type (stepped or ramped), and initiation temperatures for pump starts/stops and decay heat removal. Pressure-temperature curves are also typically “smoothed” to eliminate anomalies such as non-monotonically increasing pressures with temperature. Values of  $K_{It}$  are presented by focusing on the normal heatup results for the ramped transient with reactor coolant pump start at [ ] °F and normal cooldown results for the ramped transient with decay heat removal initiation at [ ] °F. These two transients provide most of the data points in Figures 7-1 and 7-2. Figure 2.8.2-1 and Figure 2.8.2-2 below show the “raw” pressure-temperature curves for these two component transients (with pressure corrections for transducer location but without smoothing of discontinuities), along with the associated  $K_{It}$  thermal stress intensity factors. The figures also indicate which regions of the curves are controlled by transient or steady state conditions at the  $1/4t$  ( $1/4t$  thickness from the inside surface) or  $3/4t$  ( $1/4t$  thickness from the outside surface) flaw depths. It is noted that steady state results are obtained by setting  $K_{It}$  equal to zero. Furthermore,  $K_{It}$  is conservatively set equal to zero at any point during a transient where the thermal stress intensity factor is calculated to be a negative value. Digital data is also provided in Table 2.8.2-1 and Table 2.8.2-2 below.

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**Figure 2.8.2-1 Corrected P-T Curve for Oconee Unit 1 Ramped Normal Heatup  
with RCP Start at 100°F**

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**Figure 2.8.2-2 Corrected P-T Curve for Oconee Unit 1 Ramped Normal Cooldown  
with DHR at 190°F**

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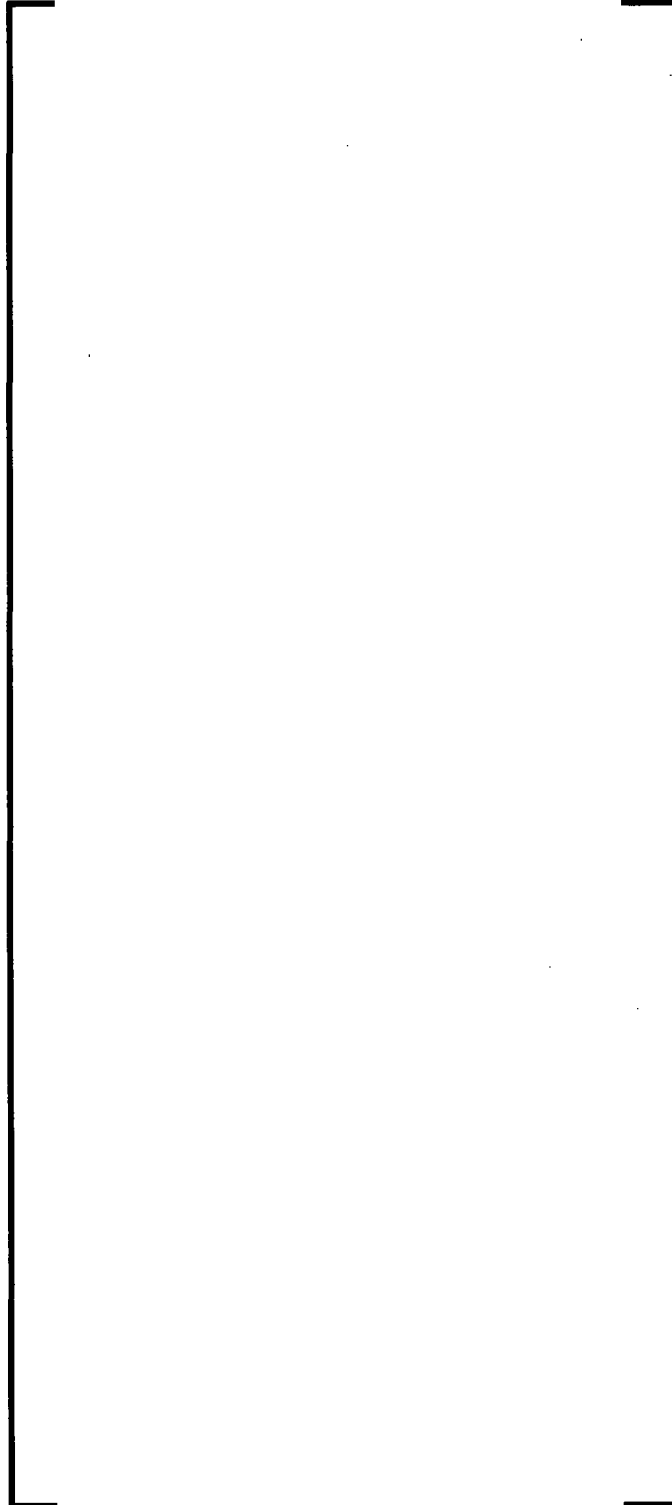
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**Table 2.8.2-1 Data Points for Figure 2.8.2-1**



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**Table 2.8.2-2 Data Points for Figure 2.8.2-2**



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## 3.0 REFERENCES

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- 1 ANP-3127, Revision 1, (AREVA Document ID 77-3127-001), "Oconee Nuclear Station Units 1, 2 & 3 Pressure-Temperature Limits at 54 EFPY," January, 2013
- 2 Letter, Duke Energy Carolinas, LLC, "Oconee Nuclear Station (ONS), Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287, License Amendment Request to Update Pressure-Temperature Limit Curves, License Amendment Request (LAR) No. 2012-10," NRC ADAMS Accession Number ML13058A059, February 22, 2013
- 3 Letter, John P. Boska (NRC) to Mr. Scott Batson (Duke Energy Carolinas), "Oconee Nuclear Station, Units 1, 2, and 3, Request for Additional Information Regarding Amendment Application Related to Reactor Coolant System Pressure and Temperature Limit Curves (TAC Nos. MF0763, MF0764, and MF0765)," NRC ADAMS Accession Number ML13165A147, June 21, 2013
- 4 AREVA NP Document BAW-10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G," June 1986
- 5 Code of Federal Regulations, Title 10, Part 50 – Domestic Licensing of Production and Utilization Facilities, Appendix G – Fracture Toughness Requirements, Federal Register Vol. 60, No. 243, January 31, 2008
- 6 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 1998 Edition with Addenda through 2000
- 7 NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," U.S. Nuclear Regulatory Commission, December 2010
- 8 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988
- 9 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NB, "Class 1 Components," 1998 Edition with Addenda through 2000