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Enclosure 2

LTR-PAFM-10-123-NP, Revision 0
"Technical Justification to Support Alternative Visual Examination Intervals for Fort Calhoun Reactor Vessel Outlet Nozzle to Safe End Dissimilar Metal Welds"
(Non-Proprietary)

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Technical Justification to Support Alternative Visual Examination Intervals for Fort Calhoun Reactor Vessel Outlet Nozzle to Safe End Dissimilar Metal Welds

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1.0 Introduction

Service induced cracking of the nickel-base alloy components and weldments have been occurring more and more frequently in recent years, resulting in the need to repair and/or replace these components. Such cracking and leakage have been observed in the reactor vessel upper and bottom head penetration nozzles as well as the dissimilar metal butt welds of the pressurizer and reactor vessel nozzles. These Pressurized Water Reactor (PWR) power plant field experiences and the potential for Primary Water Stress Corrosion Cracking (PWSCC) require reassessment of the examination frequency as well as the overall examination strategy for nickel-base alloy components and weldments. Code Case N-722 (Reference 1) provides the visual inspection guidelines for the primary system piping dissimilar metal (DM) butt welds to augment the current regulatory requirements.

In accordance with Code Case N-722 guidelines, visual examinations are required for the dissimilar metal butt welds at the Reactor Vessel (RV) nozzles. Since these PWSCC susceptible welds are being exposed to high reactor coolant temperatures, visual inspection is required every refueling outage for the outlet nozzles. The results of a deterministic fracture mechanics analysis is presented in this document to justify deviating from the visual examination requirements of Code Case N-722 for the outlet nozzles. For the Fort Calhoun station, the reactor vessel nozzles are encased in a "sandbox" that must be removed to provide access to the outside surface of the nozzles and NRC Safety Evaluation Report (Reference 2) acknowledged that performing outside surface inspections on the reactor vessel nozzles would result in hardship or unusual difficulty without a compensating increase in quality or safety.

The objective of this letter report is to provide technical justification to deviate from the Code Case N-722 guidelines by not performing any visual examinations before the Spring 2014 Re-Fueling Outage (RFO) at the reactor vessel outlet nozzle DM welds since Spring 2014 RFO is the mitigation/re-inspection outage for these DM welds. Both pre-EPU (Extended Power Uprate) and post-EPU conditions will be considered in the development of the technical justification since EPU is scheduled to be implemented during the Fall 2012 RFO. Similar technical justification has been developed and used to support deviation from visual examinations for Indian Point Unit 2 reactor vessel outlet nozzle DM welds (Reference 3) since it has similar nozzle geometry and configuration as Fort Calhoun.

The following sections provide a discussion of the methodology, geometry, loading and results from the flaw tolerance analyses used to develop the technical justification.

2.0 Methodology

The key aspect of the technical justification for not performing any visual examinations before the Spring 2014 RFO is to demonstrate the structural integrity of the RV nozzle DM welds subjected to the PWSCC crack growth mechanism. In order to demonstrate structural integrity, it is essential to determine the maximum allowable flaw size at the

DM welds and the service life required for a hypothetical maximum undetected flaw to reach this maximum allowable flaw size, since no recordable indications were detected during the Fall 2009 RFO. The maximum allowable flaw size used in the structural integrity determination is the maximum allowable end-of-evaluation period flaw size determined in accordance with the ASME Section XI (Reference 4).

In order to determine the maximum allowable end-of-evaluation period flaw sizes and the crack tip stress intensity factors used for the PWSCC analysis, it is necessary to establish the stresses, crack geometry and the material properties at the locations of interest. The applicable loadings which must be considered consist of piping reaction loads acting at the dissimilar metal weld regions and the residual stresses which exist in the region of interest.

The piping loads at the reactor vessel outlet nozzle DM weld locations resulting from the Replacement Steam Generator program (Reference 5) are used as the pre-EPU loads. The post-EPU piping loads for the RV outlet nozzle DM welds are provided in Reference 6. In addition to the piping loads, the effects of welding residual stresses are also considered. For PWSCC, the crack growth model for the dissimilar metal weld material is based on that given in MRP-115 (Reference 7).

The nozzle geometry and piping loads used in the fracture mechanics analysis are shown in Section 3.0. A discussion of the welding residual stress distributions used for the dissimilar metal welds is provided in Section 4.0. The determination of end-of-evaluation period allowable flaw sizes is discussed in Section 5.0.

The service life required for a hypothetical flaw to reach the maximum end-of-evaluation period flaw size depends on the initial postulated flaw size and the PWSCC crack growth rate. The initial postulated flaw size is determined based on the minimum detectable flaw depth from the current equipment detection capabilities. Based on this initial postulated flaw size, PWSCC crack growth is calculated based on the normal operating temperatures and the crack tip stress intensity factors resulting from the normal operating steady state piping loads and welding residual stresses as discussed in Section 6.0. The total crack growth due to PWSCC based on the pre- and post-EPU conditions at the RV outlet nozzle DM weld locations is used to demonstrate that the postulated initial flaw size would not exceed the ASME maximum allowable end-of-evaluation period flaw size before the Spring 2014 RFO. Section 7.0 provides the initial flaw size determination and the crack growth curves developed to provide technical justification to deviate from the Code Case N-722 guidelines by not performing any visual examinations before the Spring 2014 RFO at the reactor vessel outlet nozzle DM welds.

3.0 Nozzle Geometry and Loads

The dissimilar metal weld geometry for the Fort Calhoun Reactor Vessel outlet nozzle is based on the nozzle detail drawings (Reference 8). The RV outlet nozzle geometry and normal operating temperatures are summarized in Table 3-1.

The piping reaction loads at the RV outlet nozzle DM weld locations for the pre-EPU condition are from Reference 5, which considered the impacts resulting from the Replacement Steam Generator program. The post-EPU piping loads are from Reference 6. The pre-EPU piping loads are summarized in Tables 3-2 for the outlet nozzles, while the post-EPU piping loads are presented in Table 3-3. These loads are used in determining the most limiting end-of-evaluation period flaw sizes and the PWSCC crack growth results.

Table 3-1
Fort Calhoun Reactor Vessel Nozzle Geometry and Normal Operating Temperatures

Dimension	Outlet Nozzle
Outside Diameter (in.)	38.25
Inside Diameter (in.)	32.375
Thickness (in.) [1]	2.938
Post-EPU Conditions: [2] Outlet Nozzle Fluid Temperature = 603°F Pre-EPU Conditions: [3] Outlet Nozzle Fluid Temperature = 591.8°F	

Notes:

1. Dimension at the thinnest section of the DM weld without considering the inner cladding.
2. Best estimate plant temperatures.
3. Actual plant operating temperatures over the last two cycles.

Table 3-2

Fort Calhoun Reactor Vessel Outlet Nozzle Pre-EPU Piping Loads

Loading	Forces (kips)	Moments (in-kips)		
	Fx (Axial)	Mx (Torsion)	My (Bending)	Mz (Bending)
Deadweight	-0.53	-0.96	-3.12	7660.7
Normal Operating Thermal	10.9	9.0	-1.7	35084.8
OBE (Operational Basis Earthquake)	145.8	1651.8	3701.6	4531.4
SSE (Safe Shutdown Earthquake)	329.2	2909.8	5834.3	9914.6
Max LOCA (Loss of Coolant Accident)	949.0	12.5	199.7	16069.8

Table 3-3

Fort Calhoun Reactor Vessel Outlet Nozzle Post-EPU Piping Loads

Loading	Forces (kips)	Moments (in-kips)		
	Fx (Axial)	Mx (Torsion)	My (Bending)	Mz (Bending)
Deadweight + Normal Operating Thermal	4.2	-7.4	-2.7	23930
OBE (Operational Basis Earthquake)	108.03	1152.96	2154.96	2144.88
SSE (Safe Shutdown Earthquake)	161.05	1857	3202.92	3089.28
Max LOCA (Loss of Coolant Accident)	331	2712	16533	16725

4.0 Dissimilar Metal Weld Residual Stress Distribution

There are no prior inside surface weld repairs at any of the reactor vessel nozzle DM welds based on a review of all the relevant weld fabrication records by Fort Calhoun plant personnel. Other than the weld repair history, there are other factors that are key contributors to the through-wall residual stress distribution such as the dissimilar metal weld configuration, nozzle geometry, length of the stainless steel safe end and post-weld heat treatment (PWHT). The Fort Calhoun reactor vessel outlet nozzle DM welds have been post-weld heat treated. The welding residual stresses used in the flaw tolerance analysis for the dissimilar metal weld region at Fort Calhoun is based on a review of the results from various residual stress parametric studies for the reactor vessel outlet nozzle DM welds with similar geometry and configuration as that in Fort Calhoun (References 9, 10 and 11). Even though there are no prior inside surface weld repairs at the DM welds, residual stress profiles with a 25% inside surface weld repairs are conservatively used in the flaw tolerance analysis to account for any differences between the plant specific DM weld configurations and the as-analyzed configurations. The best estimate residual stress profiles used in the PWSCC crack growth analysis for the Fort Calhoun reactor vessel outlet nozzle DM welds are shown in Figure 4-1.

a,c,e

Figure 4-1 Reactor Vessel Outlet Nozzle DM Weld Through-Wall Residual Stress Profiles with Post-Weld Heat Treatment and 25% Inside Surface Weld Repair

5.0 Maximum End-of-Evaluation Period Flaw Size Determination

In order to develop the technical justification for not performing visual examinations before Spring 2014 RFO, the first step is the determination of the maximum allowable end-of-evaluation period flaw sizes. The maximum allowable end-of-evaluation period flaw size is the size to which an indication is allowed to grow until the next inspection or evaluation period. This particular flaw size is determined based on the piping loads, geometry and the material properties of the component. The maximum end-of-evaluation flaw size is not directly calculated as part of the flaw evaluation process in the ASME Code Section XI (Reference 4). Instead, the failure mode and allowable flaw size are incorporated directly into the flaw evaluation technical basis, and therefore into the tables of "Allowable End-of-Evaluation Period Flaw Depth to Thickness Ratio," which are contained in paragraph IWB-3640. The evaluation guidelines and procedures are described in IWB-3640, and the allowable flaw size can be determined using the methodology described in ASME Section XI, Appendix C (Reference 4).

Rapid, nonductile failure is possible for ferritic materials at low temperatures, but is not applicable to the nickel-base alloy material. In nickel-base alloy material, the higher ductility leads to two possible modes of failure, plastic collapse or unstable ductile tearing. The second mechanism can occur when the applied J integral exceeds the J_{Ic} fracture toughness, and some stable tearing occurs prior to failure. If this mode of failure is dominant, then the load-carrying capacity is less than that predicted by the plastic collapse mechanism. The maximum end-of-evaluation period allowable flaw sizes of paragraph IWB-3640 for the high toughness materials are determined based on the assumption that plastic collapse would be achieved and would be the dominant mode of failure. However, due to the reduced toughness of the dissimilar metal welds, it is possible that crack extension and unstable ductile tearing could occur and be the dominant mode of failure. To account for this effect, penalty factors called "Z factor" were developed in ASME Code Section XI, which are to be multiplied by the loadings at these welds. In the current analysis for Fort Calhoun, Z factor based on Reference 12 is used in the analysis to provide a more representative approximation of the effects of the dissimilar metal welds. The use of Z factor in effect reduces the allowable end-of-evaluation period flaw sizes for flux welds and this has been incorporated directly into the evaluation performed in accordance with the procedure and acceptance criteria given in IWB-3640 and Appendix C of ASME Code Section XI. It should be noted that the maximum end-of-evaluation period allowable flaw sizes are limited to only 75% of the wall thickness in accordance with the requirements of ASME Section XI paragraph IWB-3640 (Reference 4).

The maximum end-of-evaluation period flaw sizes determined for both axial and circumferential flaws have incorporated the relevant material properties, pipe loadings and geometry. Loadings under normal, upset, emergency and faulted conditions are considered in conjunction with the applicable safety factors for the corresponding service

conditions required in the ASME Code Section XI. For circumferential flaws, axial stress due to the pressure, deadweight, thermal, seismic and LOCA loads are considered in the evaluation. As for the axial flaws, hoop stress resulting from pressure loading is used. The pre-EPU RV outlet nozzle piping loads (Tables 3-2) at the DM weld locations for Fort Calhoun are based on the Replacement Steam Generator program (Reference 5), while the post-EPU piping loads for the RV outlet nozzle (Table 3-3) are from Reference 6.

6.0 PWSCC Crack Growth Analysis

The crack growth analysis involves postulating a flaw at the dissimilar metal weld region in the nozzles of interest. The objective of this analysis is to determine the service life required for a postulated inside surface flaw before it propagates to a flaw size that exceeds the end-of-evaluation period allowable flaw depth as described in Section 5.0.

Crack growth due to PWSCC is calculated for both axial and circumferential flaws using the normal operating condition steady-state stresses. For axial flaws, the stresses included pressure and residual stresses, while for circumferential flaws, the stresses considered are pressure, 100% power normal thermal expansion, deadweight and residual stresses. The input required for the crack growth analysis is basically the information necessary to calculate the stress intensity factor (K_I), which depends on the geometry of the crack, its surrounding structure and the applied stresses. The geometry and loadings for the nozzles of interest are discussed in Section 3.0 and the applicable residual stresses used are discussed in Section 4.0. Once K_I is calculated, stress corrosion crack growth can be calculated using the applicable crack growth rate for the nickel-base alloy material (Alloy 182) from MRP-115 (Reference 7). For all inside surface flaws, the governing crack growth mechanism for the RV outlet nozzle is PWSCC.

Using the applicable stresses at the dissimilar metal welds, the crack tip stress intensity factors can be determined based on the stress intensity factor expressions from Reference 13. The through-wall stress distribution profile is represented by a 4th order polynomial:

$$\sigma\left(\frac{a}{t}\right) = \sigma_0 + \sigma_1\left(\frac{a}{t}\right) + \sigma_2\left(\frac{a}{t}\right)^2 + \sigma_3\left(\frac{a}{t}\right)^3 + \sigma_4\left(\frac{a}{t}\right)^4$$

where:

σ_0, σ_1, σ_2, σ_3, and σ_4 are the stress profile curve fitting coefficients,

a is the distance from the wall surface where the crack initiates;

t is the wall thickness; and

σ is the stress perpendicular to the plane of the crack.

The stress intensity factor calculations for semi-elliptical inside surface axial and circumferential flaws are performed. The influence coefficient at various points on the crack front can be obtained by using an interpolation method. The crack tip stress intensity factors can be expressed in the general form as follows:

$$K_I = \sqrt{\frac{\pi a}{Q}} \sum_{j=0}^{4} G_j\,(a/c, a/t, t/R, \Phi)\,\sigma_j \left(\frac{a}{t}\right)^j$$

where:

a: Crack Depth

c: Half Crack Length Along Surface

t: Thickness of Cylinder

R: Inside Radius

Φ: Angular Position of a Point on the Crack Front

G_j: G_j is influence coefficient for j^{th} stress distribution on crack surface (i.e., G_0, G_1, G_2, G_3, G_4).

Q: The shape factor of an elliptical crack, which is the square of the complete elliptical integral of the second kind or

$$\text{Shape Factor} = \left[\int_0^{\pi/2} (\cos^2\Phi + \frac{a^2}{c^2}\sin^2\Phi)^{1/2}\, d\Phi\right]^2$$. Q is approximated by:

$Q = 1 + 1.464(a/c)^{1.65}$ for $a/c \leq 1$ or $Q = 1 + 1.464(c/a)^{1.65}$ for $a/c > 1$.

Once the crack tip stress intensity factors are determined, PWSCC crack growth calculations can be performed using the crack growth rate below with the applicable normal operating temperature.

The PWSCC crack growth rate used in the crack growth analysis is based on the EPRI recommended crack growth curves for Alloy 182 material (Reference 7):

$$\frac{da}{dt} = \exp[-\frac{Q_g}{R}(\frac{1}{T} - \frac{1}{T_{ref}})]\,\alpha(K)^{\beta}$$

where:

$\frac{da}{dt}$	=	Crack growth rate in m/sec
Q_g	=	Thermal activation energy for crack growth = 130 kJ/mole (31.0 kcal/mole)
R	=	Universal gas constant = 8.314×10^{-3} kJ/mole-K (1.103×10^{-3} kcal/mole-°R)
T	=	Absolute operating temperature at the location of crack (K or °R)
T_{ref}	=	Absolute reference temperature used to normalize data = 598.15 K (1076.67°R)
α	=	Crack growth amplitude
	=	1.50×10^{-12} at 325°C (617°F)
β	=	Exponent = 1.6
K	=	Crack tip stress intensity factor (MPa $\sqrt{m}$)

The normal operating temperature used in the crack growth analysis for the pre-EPU condition is 591.8°F at the RV outlet nozzle, while the post-EPU temperature is 603°F.

7.0 Technical Justification for Not Performing Any Visual Examinations before the Spring 2014 RFO

In accordance with Code Case N-722 (Reference 1), visual examinations are required every outage for the outlet nozzles. Since no recordable indications were detected during the Fall 2009 RFO, technical justification can be developed to deviate from the visual examination requirements in Code Case N-722 by calculating the service life required for a hypothetical maximum undetectable flaw to reach the maximum end-of-evaluation period flaw size.

Inspection of the reactor vessel nozzle DM welds at Fort Calhoun Station was accomplished in 2003 and 2009 using state of the art PDI qualified examinations. The Eddy Current Testing (ECT) procedure used is designed to complement and supplement the Appendix VIII qualified Ultrasonic Testing (UT) procedure. The ECT data has proven to be extremely useful in the characterization of both surface connected flaws or fabrication related subsurface flaws. In its current form, the ECT procedure has been qualified for detection and length measurement of a target flaw size of 0.24 inch in length and 0.04 inch in depth (Reference 14). Since no indications were detected at the DM welds during the Fall 2009 outage inspection at the Fort Calhoun Station, an initial flaw depth of 0.125 inch and an initial flaw length of 0.25 inch based on an aspect ratio of 2, are conservative estimates for the hypothetical maximum undetected flaw size in the crack growth analysis. The use of an aspect ratio of 2 is consistent with the minimum aspect ratio specified in ASME Section XI Article IWA-3000. It is reasonable especially for axial flaws due to the DM weld configuration since any PWSCC axial flaw growth is limited to the width of the weld. The use of an aspect ratio of 2 is also consistent with the technical basis for axial flaw growth used to support plant startup for V. C. Summer Nuclear Plant (Reference 15). For circumferential flaws, a conservative aspect ratio of 10 is used in the crack growth analysis.

Based on the hypothetical maximum undetected flaw size, PWSCC crack growth is calculated using the normal operating temperatures and the crack tip stress intensity factors resulting from the normal operating steady state piping loads and the welding residual stresses. To account for both the pre-EPU and the post-EPU conditions, PWSCC crack growth for the outlet nozzle DM welds is first calculated for the initial 36 months, from Fall 2009 RFO leading up to the scheduled EPU implementation in Fall 2012 RFO, based on the pre-EPU loads shown in Table 3-2 and operating temperatures in Table 3-1. The resulting crack size at the end of the 36 months is then used as the initial flaw size for the next 18 months (Fall 2012 RFO to Spring 2014 RFO) to determine the PWSCC crack growth based on the EPU condition operating temperature (Table 3-1) and loads (Table 3-3). It should be noted that the time intervals between various refueling outages conservatively ignore the plant down time during the outage. The limiting flaw configuration is an axial flaw at the outlet nozzle DM welds and the corresponding crack growth curve is shown in Figure 7-1. This is limiting because the residual hoop stress shown in Figure 4-1 is higher than the residual axial stress at the DM weld. Figure 7-2 shows the crack growth curve for a circumferential flaw at the

outlet nozzle DM welds with an aspect ratio of 10 and demonstrated that the axial flaw is the limiting flaw orientation at the outlet nozzle.

Based on Figure 7-1, the total PWSCC growth due to the pre- and post-EPU conditions for the limiting undetected flaw configuration would not exceed the ASME maximum allowable end-of-evaluation period flaw size before the Spring 2014 RFO, which is 51 calendar months after the Fall 2009 RFO. The required service life is slightly over 60 EFPM as shown in Figure 7-1. The availability factor at Fort Calhoun Station over the past 5 years has been at 0.85 and this would be a good estimate for the next 4 years considering EPU implementation in 2012 and the historical outage performances. Using the availability factor of 0.85, the required service life of 60 EFPM is actually equivalent to about 70.6 calendar months, well past the Spring 2014 RFO. No fatigue crack growth calculation is necessary since crack growth due to the fatigue mechanism is negligible compared to PWSCC for the plant operation duration of interest. Since there were no recordable indications in the Fall 2009 RFO inspection, it is highly unlikely that any leakage would occur before the Spring 2014 RFO. As a result, degradation of the pressure boundary is not expected and the safety of the nuclear facility as well as the health and safety of the public is maintained. Therefore, it is technically justifiable to deviate from the Code Case N-722 guidelines by not performing any visual examinations before the Spring 2014 RFO at the reactor vessel outlet nozzle DM welds.

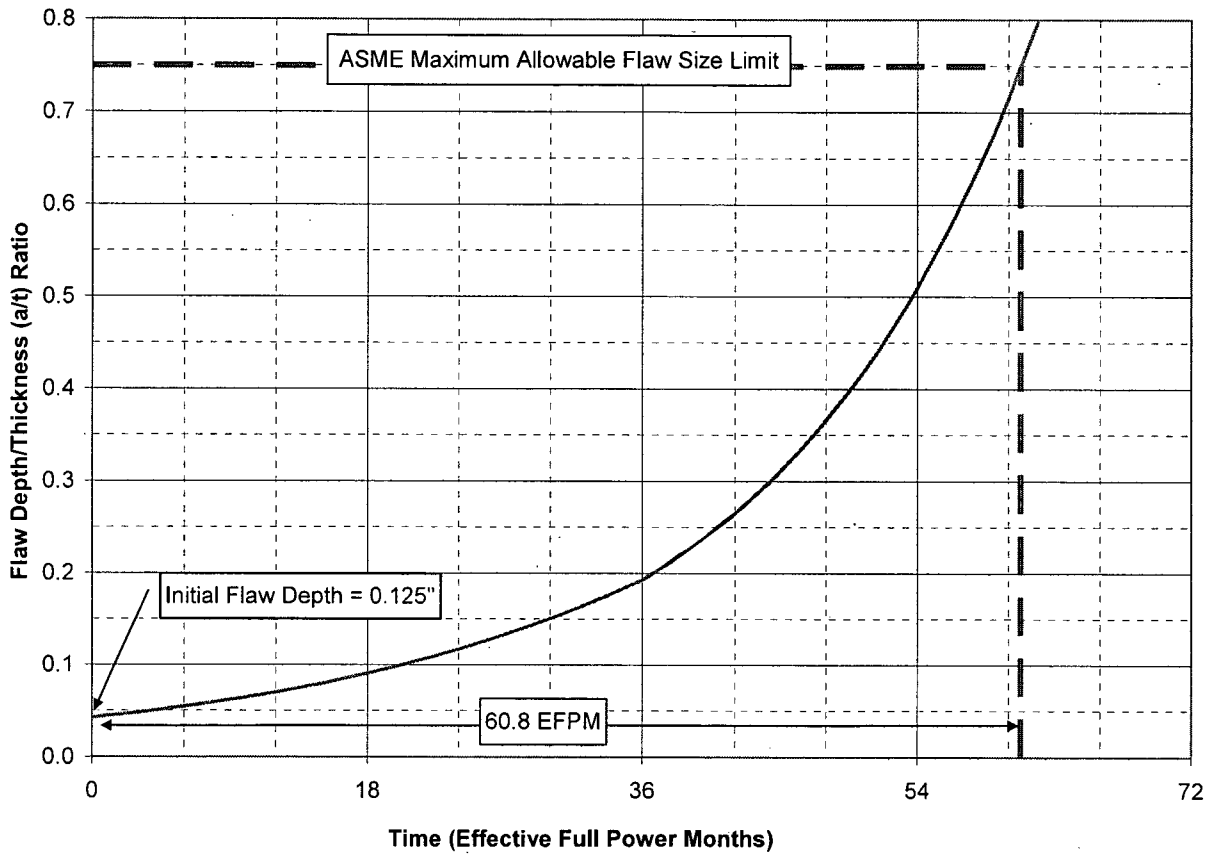

Figure 7-1 Crack Growth Curve for Outlet Nozzle Axial Flaw (DM weld), Aspect Ratio = 2, Initial Flaw Depth = 0.125"

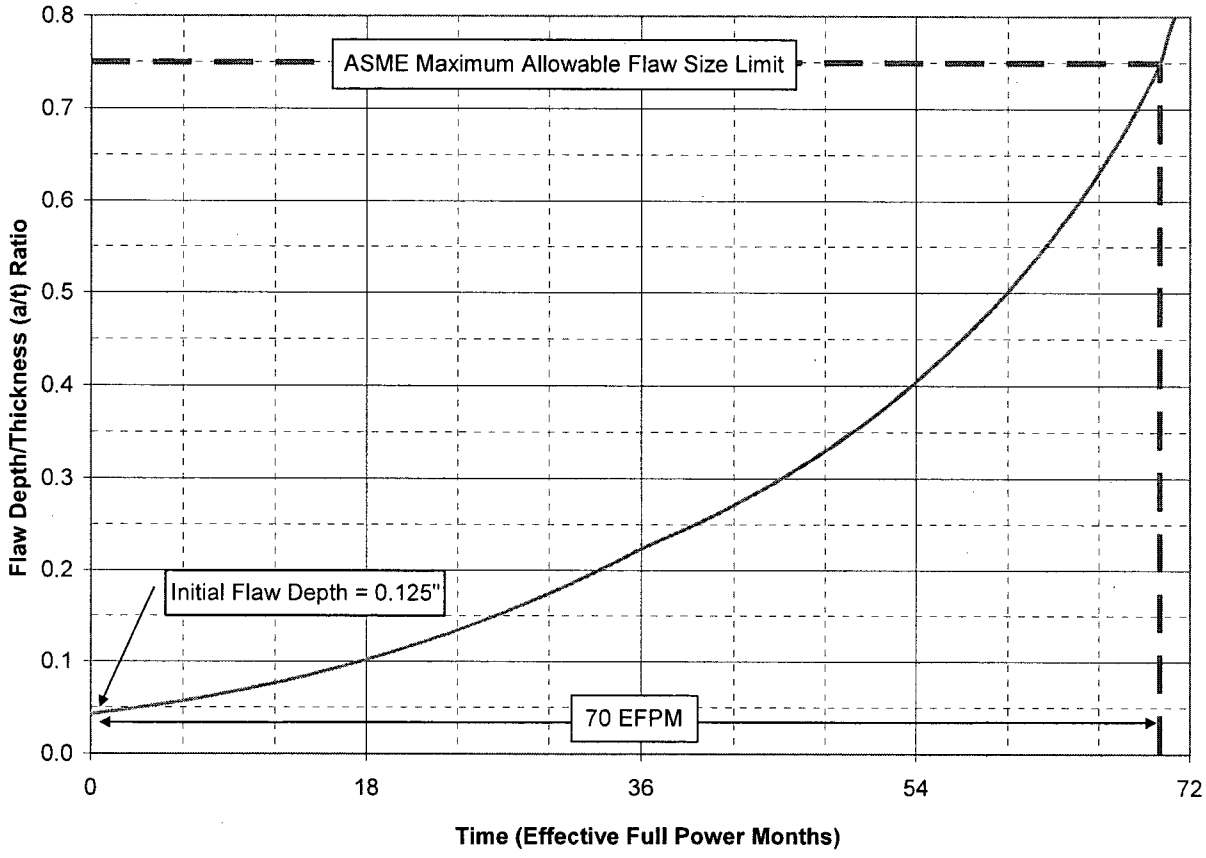

Figure 7-2 Crack Growth Curve for Outlet Nozzle Circumferential Flaw (DM weld), Aspect Ratio = 10, Initial Flaw Depth = 0.125"

8.0 Conclusions

The objective of this report is to provide technical justification to deviate from the Code Case N-722 visual examination requirements for the PWSCC susceptible dissimilar metal welds at the Reactor Vessel outlet nozzles. Based on the crack growth curve shown in Figure 7-1 for the limiting axial flaw configuration at the outlet nozzles and since there were no recordable indications during the Fall 2009 RFO inspection, any undetectable flaw in the dissimilar metal welds would not reach the maximum allowable end-of-evaluation period flaw size before the Spring 2014 RFO. It should be noted that a hypothetical 25% inside surface weld repair is conservatively assumed in the residual stress profiles used in the crack growth analysis even though fabrication records show no prior weld repairs were made at the reactor vessel outlet nozzle DM welds. As a result, it is highly unlikely that any undetected flaw would result in leakage causing degradation of the pressure boundary before the Spring 2014 RFO; thus, maintaining the safety of the nuclear facility as well as the health and safety of the public. It is therefore technically justifiable to deviate from the Code Case N-722 guidelines by not performing any visual examinations before the Spring 2014 RFO at the Fort Calhoun reactor vessel outlet nozzle DM welds.

9.0 References

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