PWROG
PWR Owners Group

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WCAP-16168-NP, Rev. 1
Project Number 694

U.S. Nuclear Regulatory Commission
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Subject: Pressurized Water Reactor Owners Group
Responses to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-16168-NP, Rev. 1 "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (TAC NO. MC9768) MUHP 5097/5098/5099 Task 2008/2059

References:

1. WOG Letter from Ted Schiffley to Document Control Desk, Request for Review and Approval of WCAP-16168-NP Rev. 1, entitled "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," dated January 2006, WOG-05-25, January 26, 2006.
2. Acceptance for Review of Westinghouse Owners Group (WOG) Topical Report WCAP-16168-NP, Rev. 1 "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (TAC NO. MC9768) MUHP 5097/5098/5099 Task 2008/2059, OG-06-311, September 22, 2006.
3. NRC emails from Sean E. Peters of NRR to Tom Laubham of PWROG dated March 9 and 12, 2007 "RAIs for WCAP-16168".

In January 2006, the WOG, now known as the Pressurized Water Reactor Owners Group (PWROG), submitted WCAP-16168-NP Rev. 1, entitled "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," for review and approval (Reference 1). In September 2006, the NRC accepted the topical report (Reference 2) and provided an informal Request for Additional Information (RAI) (Reference 3) on March 9 and 12, 2007.

Enclosure 1 to this letter provides the RAI responses to the questions received in Reference 3. Enclosure 2 is the marked-up WCAP.

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If you have any questions, please do not hesitate to contact me at (630) 657-3897, or if you require further information, please contact Mr. Jim Molkenthin of the PWR Owners Group Project Management Office at (860) 731-6727.

Regards,

Frederick P. "Ted" Schiffley, II, Chairman
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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION AND PWROG RESPONSES RE:

PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT (TR)

WCAP-16168-NP, REVISION 1, "RISK-INFORMED EXTENSION OF THE

REACTOR VESSEL IN-SERVICE INSPECTION INTERVAL"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

Materials

1. Section 3.2 of WCAP-16168-NP Revision 1, indicates that the pilot-plant studies included a probabilistic representation of the fatigue crack growth correlation for ferritic materials in water consistent with the previous and current models contained in Appendix A of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. The probabilistic representation was consistent with those used in the pc-PRAISE code and NRC-approved SRRA tool for piping risk informed inservice inspection. In Appendix A of the NRC staff safety evaluation (SE) on WCAP-14572, Revision 1, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," the staff concluded that the SRRA code addresses fatigue crack growth in an acceptable manner since it is consistent with the technical approach used by other state-of-the-art codes for probabilistic fracture mechanics. The staff noted that realistic predictions of failure probabilities require that the user define input parameters which accurately represent all sources of fatigue stress and the probability for preexisting fabrication cracks in welds.

a) The staff requests that the Westinghouse Owners Group (WOG) provide the transients and number of transients that were assumed in the analysis of the pilot-plant studies and explain why the proposed transients represent all the sources of fatigue stress.

> **Response:** For the Westinghouse and CE Nuclear Steam Supply System (NSSS) plant designs, all of the Reactor Coolant System (RCS) design basis transients were considered in the analysis. These RCS design basis transients (herein referred to as transients unless specifically noted) are identified in the plant final safety analysis reports. The PROBSBFD program, which was used to modify the FAVOR surface breaking flaw density input file (S.dat), requires the frequency (cycles per year) of the transient that produces the most crack growth and could represent the fatigue crack growth of all the transients in the reactor vessel design basis. Typically, fast transients with high temperature spikes produce high skin stresses which are of concern for initiation but do not provide sufficient energy to grow an existing crack. The ASME Code requires that the fatigue usage factor for all design basis transients be less than one. Therefore, provided a plant remains within it's design basis transient parameters and number of cycles, the design basis transients should not initiate a crack. Slow transients, where the thermal stresses are allowed to fully develop through the reactor vessel wall, such as heatup and cooldown, are of much more concern for

fatigue crack growth. For this reason the primary transient chosen to be evaluated with PROBSBFD for the Westinghouse, CE, and B&W pilot plants was the cooldown transient. While the fatigue cycle consists of heatup and cooldown, the cooldown portion is used to calculate the change in stress and stress intensity factor because it is the portion that results in tensile stresses on the inside of the vessel wall. For each pilot plant, the cooldown transient that was evaluated consisted of a 100°F/hour decrease in temperature from full operating temperature to ambient temperature. For the CE and B&W pilot plants this decrease in temperature was coincident with a decrease in pressure from normal operating pressure to atmospheric pressure. For the Westinghouse pilot plant, pressure was reduced at a rate of approximately 700 psi per hour starting at the time cooldown is initiated. These cooldown curves are consistent with design basis cooldown curves for the Westinghouse and CE pilot plant designs. B&W design basis data was not available so the B&W pilot plant cooldown transient was assumed to be comparable to the cooldown transient for the CE design.

After choosing the cooldown transient as the representative transient to be evaluated using the PROBSBFD code, it was necessary to determine a number of cooldown transients that would envelope the fatigue crack growth of all of the design basis transients. For the Westinghouse designs, previous fatigue crack growth analyses of flaws on the inside surface of the reactor vessel had shown that for all of the design basis transients, only the following design basis transients resulted in measurable crack growth:

- Heatup/Cooldown
- Pressure Tests
- Feedwater Cycling
- Inadvertent Depressurization

Heatup/Cooldown and Pressure Tests are a common contributor for all NSSS designs (Westinghouse, CE, and B&W). For the Westinghouse-NSSS designs, Feedwater Cycling and Inadvertant Depressurization may become dominant either as a result of the original plant specific design basis loading analysis or for uprating considerations. Table 1a provides a list of transients that were considered for the Westinghouse-NSSS design in addressing fatigue crack growth. A description of the four transients that were determined to contribute to crack growth is provided in Table 1b. Individual plant RCS design specifications provide additional detail on the transients.

Table 1a: Westinghouse NSSS Design Transients	
Transient	Transient
Plant Heatup/Cooldown-100F/hr	Small Loss of Coolant Accident
Pressure Test 3125 psia/2250 psia	Small Steam Break
Feedwater Cycling	Complete Loss of Flow
Inadvertent Depressurization	Feedwater Line Break
Unit Loading and Unloading Between 0 and 15% of Full Power	Reactor Coolant Pipe Break
Plant Loading/ Unloading at 5% of full power per minute	Large Steam Line Break
Step Load Increase/ Decrease of 10% of Full Power	Reactor Coolant Pump Locked Rotor
Large Step Load Decrease (with steam dump)	Control Rod Ejection

Table 1a: Westinghouse NSSS Design Transients	
Transient	Transient
Loop Out of Service, Normal Loop Shutdown/ Startup	Turbine Roll Test
Loss of Load	Refueling
Loss of Power	Boron Concentration Equalization
Loss of Flow	Excessive Feedwater Flow
Reactor Trip from full power	Inadvertent Auxiliary Spray
Inadvertant Startup of an Inactive Loop	Reduced Temperature Return
Control Rod Drop	Accumulator Injection Break
Inadvertant Safety Injection Actuation	Steady State Fluctuations

Table 1b: Westinghouse NSSS Design Transients Contributing to Crack Growth	
Transient	Description
Plant Heatup/Cooldown-100F/hr	Design heatup/cooldown transients are conservatively represented by continuous operations performed at a uniform temperature rate. The heatup considered going from ambient temperature and pressure condition to the no-load temperature and pressure condition. The cooldown considers going from the no-load temperature and pressure conditions to ambient temperature and pressure conditions.
Pressure Test 3125 psia/2250 psia	The pressure tests include both shop and field hydrostatic tests that occur as a result of component and system testing.
Feedwater Cycling	This transient addresses intermittent fluctuations in feedwater temperature that cause the reactor coolant average temperature to decrease to a lower value and then return to no-load conditions.
Inadvertent Depressurization	Several events can be postulated to occur during normal plant operation which will cause rapid depressurization of the reactor coolant system. Of these, the pressurizer safety valve actuation causes the most severe transient and is commonly used as an umbrella case to conservatively represent the impact on the system from any of the inadvertent depressurization events.

Existing analyses of these transients had been performed using a 10% through-wall initial flaw. Therefore, sensitivity studies were performed on the four contributing transients using the PROBSBFD Code with an initial flaw depth equivalent to the thickness of the cladding (then rounded up to the nearest whole percent of the wall thickness). The analysis showed that the only design basis transient that resulted in significant crack growth was the cooldown transient. The sensitivity study using the PROFSBFD indicated that the flaw growth contribution of the Feedwater Cycling and Inadvertant Depressurization transients was at least an order of magnitude less than the contribution from the heatup/cooldown transient. Pressure test transients were enveloped by the heatup/cooldown transient. To envelope the contribution of the Feedwater

Cycling and Inadvertent Depressurization transients and any partial cooldowns, 2 additional cooldown transients per year were conservatively added to the design basis of 5 cooldown cycles per year. Therefore, 7 cooldown cycles per year were evaluated with PROBSBFD to determine the surface breaking flaw density for the Westinghouse NSSS design pilot plant.

Previous fatigue crack growth studies were not available for the CE NSSS designs and therefore, all design basis transients were evaluated using the PROBSBFD code:

- Plant Heatup and Cooldown
- Plant Loading and Unloading at 5%/min
- 10% Step Load Increase and Decrease
- Reactor Trip, Loss of Flow, and Loss of Load
- Loss of Secondary Pressure
- Hydrostatic Test 3125 psia/2250 psia
- Safety Valve Relief

Table 2 provides a list of transients that were considered for the CE-NSSS design in addressing fatigue crack growth, including a description of the transients. Individual plant RCS design specifications provide additional detail on the transients.

Table 2: CE NSSS Design Transients

Transient	Description
Plant Heatup/Cooldown-100F/hr	Design heatup/cooldown transients are conservatively represented by continuous operations performed at a uniform temperature rate. The heatup considered going from ambient temperature and pressure condition to the no-load temperature and pressure condition. The cooldown considers going from the no-load temperature and pressure conditions to ambient temperature and pressure conditions.
Pressure Test 3125 psia/2250 psia	The pressure tests include both shop and field hydrostatic tests that occur as a result of component and system testing.
Plant Loading/ Unloading 5%/min	The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5 percent per minute between 15 percent load and full load. This is the maximum possible rate consistent with operation under automatic reactor control.
10% Step Load Increase/ Decrease	The $\pm$ 10% Step load increase/decrease is a transient which is assumed to be a change in turbine control value opening.
Reactor Trip, Loss of Flow, Loss of Load	These include reactor trips due to a number of circumstances over the life of the plant.
Loss of Secondary Pressure	A reactor trip will occur as a result of the loss of secondary side pressure.

Table 2: CE NSSS Design Transients	
Transient	Description
Safety Valve Relief	Several events can be postulated to occur during normal plant operation which will cause rapid depressurization of the reactor coolant system. Of these, the pressurizer safety valve actuation causes the most severe transient and is commonly used as an umbrella case to conservatively represent the impact on the system from any of the other transients.

Consistent with the Westinghouse design, the cooldown transient produced the largest amount of fatigue crack growth. The loss of secondary pressure transient also produced measurable growth. However, the 12 cooldowns per year was considered to be conservative in comparison to the actual number of cooldowns a plant might experience in a given year of operation. Therefore, to envelope the contribution of the loss of secondary pressure transient, only 1 additional cooldown transient was added to the design basis of 12 cooldowns per year, thus resulting in 13 cooldowns per year being evaluated with PROBSBFD to determine the surface breaking flaw density for the CE design pilot plant.

For the B&W design twelve cooldown transients a year were assumed and evaluated with PROBSBFD to determine the surface breaking flaw density. As stated in the WCAP, for a B&W plant to apply the interval extension of this WCAP, it would have to be demonstrated that the 12 cooldown transients per year envelope the fatigue crack growth from all of the design basis transients.

b) The staff requests that the WOG identify the initial flaw size, location and density assumed in the pilot plant fatigue crack growth analysis and the basis for the initial flaw size distribution and density. Identify and provide analyses of all inservice inspection results and destructive test results that were used to determine the initial flaw size, location and density assumed in the pilot plant fatigue crack growth analysis.

Response: In Revision 1 of WCAP-16168-NP, the initial flaw distributions, including the surface breaking flaw distribution used for the fatigue crack growth analysis, are discussed on pages 3-8 and 3-9 of section 3.2. The distributions for the three representative plants were all generated using the computer code VFLAW03 developed by PNLL as described in Revision 1 of NUREG/CR-6817, *A Generalized Procedure for Generating Flaw-Related Inputs for the FAVOR Code*, 2006.

The technical bases for the surface breaking flaw distributions are described in Section 8 of this report and the application of VFLAW03 Computer Code for surface-breaking flaws in single-pass cladding is described in Section 9.6 (pages 9.24 to 9.26). Figure 9.17 of this report provides the input surface breaking flaw distribution for the Oconee Unit 1 vessel that can be compared with the input to the PROBSBFD Computer Program in Sections K-1 and K-2 in Appendix K of the WCAP Report. Input variable 1 (FIFDepth) gives the fractional initial flaw depth as 0.03, which corresponds to the non-zero density in the row for N=3 (percent of wall thickness) in Figure 9.17.

Input variable 2 (IFlawDen) gives the initial flaw density as 0.0036589 flaws per square foot in Figure 9.17. PROBSBFD input variables 16 to 19 (Percent1-Percent4) gives the percentages for the 4 values of aspect ratio specified in input variables 8 to 11 (Aspect1-Aspect4) of 2, 6, 10 and 99 (infinite) as 67.450, 20.769, 3.9642 and 7.8166, respectively , which agrees with the values in Figure 9.17 of the NUREG/CR Report. The same input to VFLAW03 was used, except for plant-specific values of vessel wall thickness and cladding thickness, which were set equal to the bead size for a single-pass cladding, to generate the initial surface flaw distribution input to PROBSBFD for Beaver Valley Unit 1 in Sections C-1 and C-2 of Appendix C and for Palisades in Sections G-1 and G-2 of Appendix G in the WCAP Report. As indicated in the WCAP Report, the information for calculating cladding (surface breaking) flaws in Tables B-2 (page B-8), F-2 (page F-8) and J-2 (page J-8) is taken directly from Table 4-2 of the December 2002 Draft NUREG Report on the Technical Basis for Revision of the PTS Rule (ADAMS: ML030090626). Note that the citation for this reference within Appendices B, F and J in Revision 1 of WCAP-16168-NP will be changed from [7] to the correct reference number of [8].

c) The staff requests that the WOG identify the fatigue crack growth curves (crack growth versus change in stress intensity factor) used in the pilot-plant studies.

Response: The fatigue crack growth rate equations for ferritic materials, such as the vessel wall base metal, are taken from Section 4.2.2 of the *Theoretical and Users Manual for pc-PRAISE* (NUREG/CR-5864, July 1992). As noted in this report, these "equations provide a probabilistic representation of the fatigue growth relationship for ferritic materials in water contained in Appendix A of Section XI of the ASME Boiler and Pressure Vessel Code." Figure A-4300-2, Reference Fatigue Crack Growth Curves for Carbon and Low Alloy Ferritic Steels Exposed to Water Environments, from Appendix A to Section XI in the current edition of the ASME Boiler and Pressure Vessel Code, is also provided below for a graphical representation of these equations. It should be noted that the fatigue crack growth curves in Appendix A of Section XI of the ASME Boiler and Pressure Vessel Code have not changed since they were originally included in the 1978 Edition of Section XI. Furthermore, there are presently no known plans to revise the curves in the future.

$R \leq 0.25$

$$\frac{da}{dN} = \begin{cases} 1.02\times 10^{-12}\,\Delta K^{5.95} Q & \Delta K < 19 \\ 1.01\times 10^{-07}\,\Delta K^{1.95} Q & \Delta K \geq 19 \end{cases}$$

$$Q = \exp\left(-0.408 + 0.542 C_F\right)$$

$0.25 < R < 0.65$

$$\frac{da}{dN} = \begin{cases} f_1 \Delta K^{5.95} Q & \Delta K \leq f_3 \\ f_2 \Delta K^{1.95} Q & \Delta K > f_3 \end{cases}$$

$$
\begin{aligned}
f_1 &= 1.02 \times 10^{-12}(26.9R - 5.725\ ) \\
f_2 &= 1.01 \times 10^{-07}(3.75R + 0.06\ ) \\
f_3 &= (f_2 / f_1\ )^{1/4} \\
Q &= \exp[0.1025R - 0.433625 + (0.6875R + 0.370125\ )C_F\ ]
\end{aligned}
$$

$R \geq 0.65$

$$
\frac{da}{dN} = \begin{cases} 1.20 \times 10^{-11} \Delta K^{5.95} Q & \Delta K < 10 \\ 2.52 \times 10^{-07} \Delta K^{1.95} Q & \Delta K \geq 12 \end{cases}
$$

$$
Q = \exp(-0.367 + 0.817C_F\ )
$$

In the above equations, R is K_{min} / K_{max}, ΔK is K_{max} – K_{min} and C_F is normally distributed with a mean of 0 and standard deviation of one. The units on the applied stress intensity factor, K, are ksi-(inch)$^{0.5}$ and inches per cycle on the crack growth rate, da/dN. Note that the normally distributed random value of C_F ,which is used to calculate the uncertainty factor Q, is specified as input variable 14 (FCGR-UC) to the PROBSBFD Computer Program as first shown in Section C-1 in Appendix C of the WCAP Report.

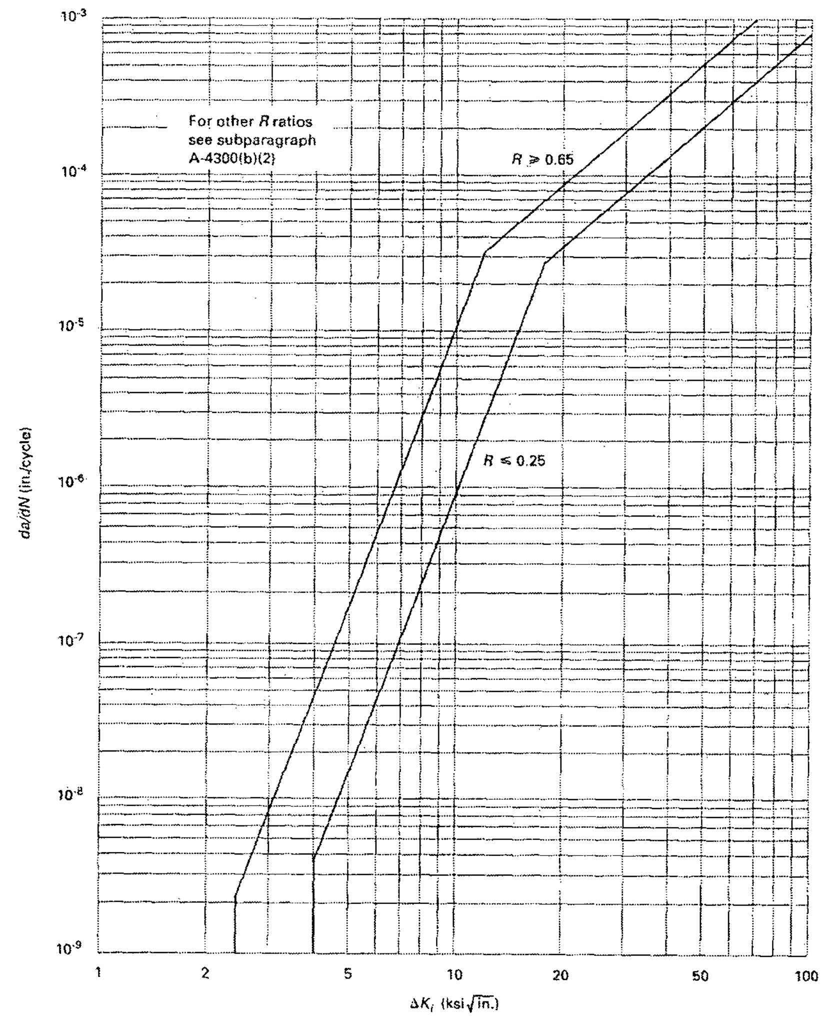
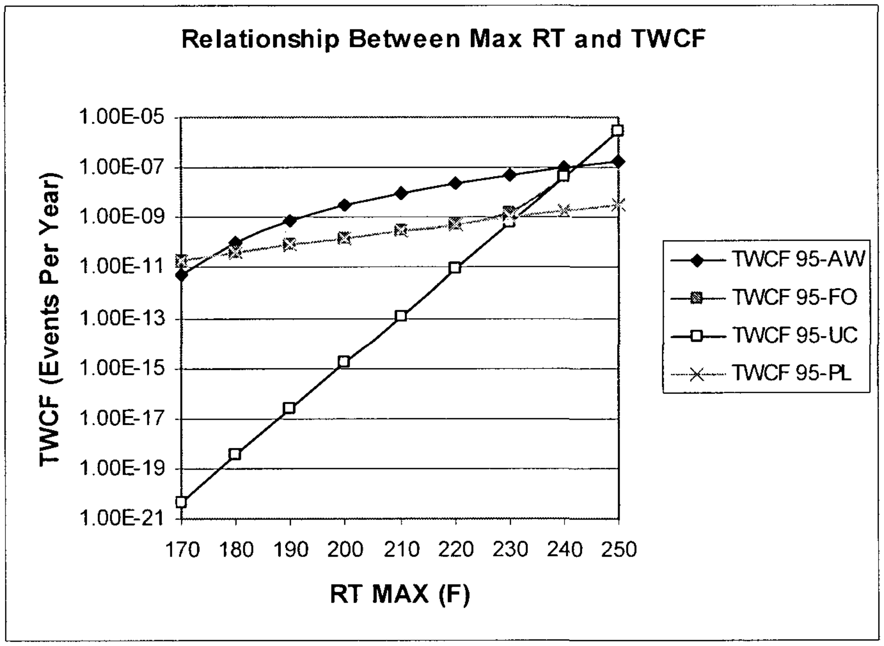


Figure A-4300-2, Reference Fatigue Crack Growth Curves for Carbon and Low Alloy Ferritic Steels Exposed to Water Environments
(From Appendix A of Section XI of the ASME BPV Code)

2. In Attachment 1 to the June 8, 2006 letter from the WOG, the WOG indicated that under-clad cracks in forgings are so shallow that chance for them initiating during a severe pressurized thermal shock (PTS) transient would be fairly small. Analyses (NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)") performed by the staff indicates that for severe PTS transients the through wall crack frequency (TWCF) for forgings with under-clad cracks are greater than those for axial welds with equivalent material reference temperature. How does the staff analysis impact the WOG analyses and the TWCF pilot-plant screening criteria in Appendix A of WCAP-16168-NP, Revision 1?

Response: The statement in the Staff's question that "the through wall crack frequency (TWCF) for forgings with under-clad cracks are greater than those for axial welds with equivalent material reference temperature" is only valid for reference temperatures greater than 240°F. Below this temperature, the TWCF of forgings is equivalent to that of plates. This is confirmed by plotting the TWCF correlations in NUREG-1874 for plates (PL), axial welds (AW), forgings (FO), and under-clad cracking (UC) on the same graph as shown below.



The correlations from NUREG-1874 that were used to produce this graph are as follows:

$TWCF_{95\text{-}AW} = \exp\{5.5198\ln(RT_{MAX\text{-}AW} - 616) - 40.542\}\beta$

$TWCF_{95\text{-}FO} = \exp\{23.737\ln(RT_{MAX\text{-}FO} - 300) - 162.38\}\beta + \eta\{1.3 \times 10^{-137}10^{0.185*RTMAX\text{-}FO}\}\beta$ (This correlation is for forgings with underclad cracking; $\eta=1$)

$TWCF_{95\text{-}UC} = \{1.3 \times 10^{-137}10^{0.185*RTMAX\text{-}FO}\}\beta$ (This is the underclad-cracking portion of the correlation above. Without this underclad cracking portion, the forging is equivalent to a plate)

$TWCF_{95\text{-}PL} = \exp\{23.737\ln(RT_{MAX\text{-}PL} - 300) - 162.38\}\beta$

For the graph, β was chosen to have a value of "1" corresponding to a reactor vessel beltline wall thickness of less than 9.5 inches per NUREG-1874. While, this selection is appropriate because most US PWRs have a reactor vessel beltline wall thickness less than 9.5", for thicker vessels the conclusions discussed in this response do not change. The graph was plotted using degrees Rankin. However, the X-axis was adjusted to display degrees Fahrenheit.

As shown in Table 3.4 of NUREG-1874, the highest $RT_{MAX\text{-}FO}$ value for the ring-forged plants in the domestic Pressurized Water Reactor (PWR) fleet is 187.3°F at 32 EFPY and 198.6°F at 48 EFPY. Therefore, it is unlikely that the $RT_{MAX\text{-}FO}$ value for any domestic PWR will ever exceed 240°F (even above 60 EFPY) and the TWCF value for forgings will remain below that for axial welds with equivalent reference temperatures. Therefore, the Staff analysis on the effects of under-clad cracking has no impact on the WCAP analyses when applied to the domestic PWR fleet. In the unlikely event that the $RT_{MAX\text{-}FO}$ value for a plant exceeds 240°F, this analysis and the 20-year inspection interval would not be applicable without further evaluation.

Since the TWCF correlations have been revised from those in NUREG-1806 and a correlation has been determined for forgings (even though it has no impact for domestic PWRs), the WCAP will be revised to reflect the changes to the correlations. The pilot plant TWCF values, which are presented in Appendices B, F, and J and used in Appendix A, will be revised using the updated correlations. Since all vessel forgings in domestic plants will not be affected by under-clad cracking, there is no need to determine whether the cladding was fabricated in accordance with Regulatory Guide 1.43 as directed in NUREG-1874.

3. Appendix A-1 and Appendix A-2 of WCAP-16168-NP, Revision 1 identifies that the TWCF is a critical parameter in determining whether the licensee's reactor vessel is bounded by the analyses performed for the pilot-plants.

a) Licensees requesting to extend the inservice inspection interval from 10 years to 20 years must provide the following information at the time of their request: (1) determine their plant-specific TWCF using the latest methodology approved by the staff for calculating the TWCF based on their plant-specific RT_{MAX} and NUREG-1874; (2) determine the ΔT_{30} values using the latest approved methodology documented in Regulatory Guide 1.99 or other NRC-approved methodology; and (3) provide all material properties that were used to determine the plant-specific TWCF (i.e. RT_{MAX-AW}[1], RT_{MAX-PL}[2], RT_{MAX-FO},[3] RT_{MAX-CW}[4], $RT_{NDT(u)}$, ΔT_{30} value for limiting materials in the beltline, maximum neutron fluence (φt_{FL}) for limiting materials in the beltline, cold leg temperature under normal operating conditions, neutron flux for limiting materials in the beltline, and wt-% phosphorus, wt-% manganese, wt-% nickel, wt-% copper for limiting materials in the beltline).

Response: Appendix A of the WCAP will be revised to require that the plant specific TWCF, RT_{MAX}, and ΔT_{30} values be calculated as stated above. Appendix A will also be revised to include all material properties required to determine the plant specific TWCF.

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[1]RT_{MAX-AW} characterizes the reactor pressure vessel's resistance to fracture initiating from flaws found along the axial weld fusion lines, and is evaluated for each axial weld fusion line.

[2]RT_{MAX-PL} characterizes the reactor pressure vessel's resistance to fracture initiating from flaws found in plates that are not associated with welds, and is evaluated for each plate.

[3]RT_{MAX-FO} characterizes the reactor pressure vessel's resistance to fracture initiating from flaws found in forgings that are not associated with welds, and is evaluated for each forging.

[4]RT_{MAX-CW} characterizes the reactor pressure vessel's resistance to fracture initiating from flaws found along the circumferential weld fusion lines, and is evaluated for each circumferential weld fusion line.

(b) Licensees that have received approval to extend the inservice inspection interval from 10 years to 20 years must provide within one year of completing its next beltline inservice inspection, the analysis and data requested in Section (d) of the voluntary PTS rule, 10 CFR 50.61(a).

Response: To address the NRC requirements for reporting and evaluation of inspection data, the following requirements will be added to Appendix A of the WCAP:

"All data on embedded flaws of concern with a through-wall extent (TWE) greater than 0.1 inch shall be provided to NRC within one year of completing the next vessel beltline inservice inspection per ASME Section XI, Appendix VIII, Supplement 4. For potential vessel failure due to PTS, embedded flaws of concern are axially oriented planar flaws in the vessel beltline within the inner 12.5% (1/8th) of the vessel wall thickness.

An assessment of the inservice inspection results relative to the flaw distributions used in the pilot plant analyses shall also be provided. This assessment shall be performed in accordance with the requirements of Section (d) in the final published version of the voluntary PTS rule, 10 CFR 50.61(a)."

The limitation on the minimum TWE is taken from Section 2.10.2.2 on Probability of Detection and Figure 2.8 in NUREG-1874. As noted, flaws with a smaller TWE were not included in the vessel samples used for inspection qualification via the Performance Demonstration Initiative.

Note that Section (d) of the voluntary PTS rule refers to Section (e)(2), which provides the requirements for measurement and evaluation of surface breaking flaws. For potential vessel failure due to PTS, surface breaking flaws of concern are those with flaw depths all the way through the cladding and into the base metal.

The definition of "embedded flaws of concern" was added to the WCAP insert to address the NRC concern that critical flaw conditions should be based on the flaw distribution, location and density that significantly contribute to the TWCF criteria for the pilot-plants. This concern was stated in the purpose of the technical basis document for the embedded flaw limitations on density and size for welds, plates, and forgings that were provided in Enclosure 1 of SECY-07-104 on June 25, 2007 (ADAMS: ML070570283). The technical basis document was provided by the NRC as an Attachment to an April 2007 NRC Memo, *Development of Flaw Size Distribution Tables for Draft Proposed Title 10 of the Code of Federal Regulations (10CFR) 50.61a,* ADAMS: ML070950392. The definition is based upon those flaws which contribute most to TWCF as determined by the results of the latest PTS risk calculations that are summarized in Section 3.3.1.3 of NUREG-1874.

4. Section 5 of WCAP-16168-NP, Revision 1 indicates ASME Code, Section XI, Category B-J welds, "Pressure Retaining Welds in Piping," at the reactor vessel nozzles may be inspected at 20 year frequency based on the analyses in the report. The staff does not consider the analysis performed in accordance with this WCAP applicable for piping. The staff in a letter dated December 15, 1998, reviewed WCAP-14572, Revision1, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping inservice Inspection Topical Report," approved a methodology for evaluating piping. The staff recommends that justification for increasing the inservice inspection interval from 10 to 20 years for piping be justified in accordance with WCAP-14572, Revision 1. Please revise WCAP-16168-NP accordingly.

Response: The PWROG will remove Category B-J welds from the applicability of the 10 to 20 year interval extension justified in WCAP-16168-NP, Revision 1.

5. WCAP-16168-NP, Revision 1 was written to justify increasing the inservice inspection interval from 10 to 20 years for ASME Code, Section XI, Category B-D welds, "Full Penetration Welded Nozzles in Vessels." Figures 3-1 and 3-2 indicate that the beltline welds have the lowest ratio of code allowable stress intensity values ($K_{I\ allowable}/K_{I\ applied}$). These figures do not include the full penetration nozzle to vessel welds. The staff requests that the WOG provide the ratio of code allowable stress intensity value for full penetration nozzle-to-vessel welds to demonstrate that the beltline welds are the limiting locations.

Response: The margin ratios of stress intensity values ($K_{I\ allowable}/K_{I\ applied}$) for the Category B-D, nozzle-to-vessel welds, are shown in the table below.

Nozzle	Flaw Orientation	Year	Margin Ratio ($K_{I\text{-}Allowable}/ K_{I\text{-}Applied}$)
Inlet	Axial	10	1.20
		20	1.17
		30	1.15
		40	1.12
	Circumferential	10	5.71
		20	5.71
		30	5.71
		40	5.71
Outlet	Axial	10	1.07
		20	1.04
		30	1.01
		40	0.98
	Circumferential	10	8.62
		20	8.62
		30	8.62
		40	8.62

The least limiting location in the reactor vessel beltline has an ASME Code allowable stress intensity factor to applied stress intensity factor margin ratio of 0.504. Since this is less than the most limiting nozzle-to-vessel weld location, these locations are not the most limiting region of the reactor vessel.

6. The Probabilistic Fracture Mechanics Computer Tool and Methodology portion of Section 3.2 of WCAP-16168-NP, Revision 1, indicates that the failure frequency and distribution for all flaws in the reactor vessels were calculated using the latest version (05.1) of the FAVOR code. This code has been

significantly revised by Oak Ridge National Laboratory. Provide an analysis that demonstrates the impact of using the latest version of FAVOR on the failure frequency and distributions documented in WCAP-16168-NP, Revision 1. Describe how the results from the latest version of FAVOR code would impact the conclusions in the WCAP.

Response: The bounding differences in through wall cracking frequency (TWCF) and large early release frequency (LERF) for different versions of the FAVOR Code are provided in the following table for the three pilot plants. The bounding differences in TWCF and LERF were calculated in the responses to RAIs 9 Part c and 12 Parts a and c. FAVOR Versions 02.4 and 03.1 were used for Revision 0 of WCAP-16168-NP, while FAVOR Version 05.1was used for Revision 1. The bounding differences in TWCF and LERF for FAVOR Version 06.1, which was used to calculate the values of TWCF for each plant in NUREG-1874, *Recommended Screening Limits for Pressurized Thermal Shock (PTS),* 2007, are provided in the response to RAI 8. As can be seen in the table below, the estimated bounding change in LERF due to different ISI intervals would still result in an insignificant change in risk (<1.0E-07/year) per the requirements of Regulatory Guide 1.174. Therefore, the risk-informed conclusions of WCAP-16168-NP, Revision 1 remain valid for all versions of the FAVOR Code that were used in the risk evaluations. This conclusion is also expected to remain valid for the next potential version of the FAVOR Code (possibly version 07.1) that contains the modified embrittlement trend curves that are proposed in the Voluntary PTS Rule (10CFR50.61a). This expectation is based upon a comparison of results in Tables 3.1 and C.1 in NUREG-1874. This comparison showed a maximum difference in embrittlement index (RT_{MAX}) of 5 °F and a maximum difference in TWCF of less than 20 percent at risk analysis conditions (as defined in Table C.1 of NUREG-1874) well beyond those shown in the following table.

Effects of FAVOR Versions on TWCF and LERF Bounding Differences			
Representative Plant Name	Beaver Valley Unit 1	Palisades	Oconee Unit 1
PTS Risk Analysis Condition	60 EFPY	60 EFPY	Ext-A
Bounding Differences from FAVOR Version 02.4 (Rev. 0 of WCAP)	3.44E-09	2.68E-08	N/A
Bounding Differences from FAVOR Version 03.1 (Rev. 0 of WCAP)	3.11E-09	2.14E-08	N/A
Bounding Differences from FAVOR Version 05.1 (Rev. 1 of WCAP)	2.49E-09	4.40E-09	7.96E-10
Bounding Differences from FAVOR Version 06.1 (response to RAI 8)	9.37E-10	1.81E-08	1.26E-08

7. On page 3-12, the Topical states that "The following effects also need to be considered along with the change in ISI interval: Extent of inspection (percent coverage), Probability of detection (POD) with flaw size, Repair criterion for removing flaws from service." Also on page 3-12, it states that "For the pilot plant evaluations, examinations were assumed to be conducted in accordance with Section XI Appendix VIII, so that Figure 4 could be used." Figure 4 is a graph of POD vs. flaw size. But, on page3-17 it states "For example, if the probability of detection for the first inspection was 90 percent, then the flaw density was effectively multiplied by10 percent for input to the next iteration." These calculations determine how many flaws and the sizes of those flaws that will be included in the "s.dat" file for surface-breaking flaws in the FAVOR code calculations. Because the Appendix VIII inspections are required for only welds and

a small portion of adjacent plate material, these inspections typically cover only a few percent of the vessel surface in the belt-line region. However, FAVOR typically models surface-breaking flaws as being randomly distributed across the entire inner surface of the vessel in the belt-line region. It is not clear from the topical how the effects on the density of surface-breaking flaws were modified to reflect the fraction of the surface area covered by the inspections. Please provide the percent coverage for each of the pilot plants. Please provide a clarification of the FAVOR calculations that explains how the percent coverage was incorporated. In particular, please be clear regarding assumptions about the presence and affects of inspections on surface-breaking flaws in the areas not subject to Appendix VIII inspections.

> **Response:** As discussed in Section 2.10.1 of NUREG-1874, the flaw models now used in version 06.1 of FAVOR do not directly consider the effects of in-service inspection. To evaluate the effects of fatigue crack growth and in-service inspection on any surface breaking flaws, the flaw input file to FAVOR must be modified to include these effects. However, Section 4.4 of the Theory and Implementation Manual for version 06.1 of the FAVOR Code states that the flaw information in the one input file (S.dat) for the 1000 surface breaking flaw distributions is applied in the same manner to cladding over welds and cladding over base metal (plates and forgings). Therefore, the effect of the very small inspection coverage in the base metal was NOT considered in the PTS risk analyses discussed in Revision 1 of WCAP-16168-NP. If it had been considered, then there would be absolutely zero difference in the TWCF due to inspection interval for the surface breaking flaws in the base metal that are never inspected. Although this effect is small, neglecting it is none the less conservative, because the actual differences in TWCF and LERF due to the change in inspection interval would be lower than those estimated in the WCAP Report. That is, the actual differences in TWCF and LERF would be even less statistically significant relative to zero, as discussed in the response to Part d) of RAI 12 because the effects of the uninspected base-metal flaws were included.

Risk

8. During an October 11, 2005, public meeting with the Nuclear Regulatory Commission (NRC) (summarized in ML052910148), the NRC staff and Westinghouse discussed the relationship between the proposed WCAP and the PTS rulemaking work. The NRC staff noted that Nuclear Reactor Regulation's (NRR's) comments regarding the pressurized thermal shock (PTS) technical basis may affect the results of the calculations in the WCAP-16168. The NRC staff also noted that if the Westinghouse Owners Group (WOG) submitted WCAP-16168 prior to the resolution of NRR's comments by RES, the WOG would be expected to address NRR's comments as they affect the WCAP-16168 calculations. A critical component of the justification of the requested inspection interval extension is a fracture mechanics evaluation of the reactor vessel. The PTS technical basis and the Topical use the FAVOR code to estimate the conditional probability of reactor vessel failure. The resolution of NRRs continuing review of the PTS rulemaking technical basis has caused the FAVOR code to be modified to correct deficiencies in the code. According to Reference 26 in WCAP-16168, FAVOR code version 05.1 was used in the analysis used in the Topical, the current version of the code used in the PTS technical basis is FAVOR 6.1. The changes made to version 05.1 resulted in substantially increased values of through wall cracking frequency (TWCF) for the pilot plants and significantly different correlations of TWCF to material reference temperatures. Both of these factors are important when licensees relate the Topical analyses to their plants. Please update the FAVOR computer code analyses using the latest version of the FAVOR code and make any corresponding changes to the analyses presented in WCAP-16168 (The NRC does not expect the FAVOR code to undergo additional changes before the technical basis for the PTS rulemaking is completed, but concludes

that the version of the FAVOR code ultimately found acceptable in the PTS technical basis will be the version that will be acceptable for reference by the WOG in this Topical).

Response: The pilot plant analyses and change in risk calculations have been updated using version 06.1 of the FAVOR Code to be consistent with NUREG-1874. The results of the analyses and change in risk calculations for the three pilot plants are presented in the table below.

TWCF and LERF Results (Events per Year)			
Case	**Beaver Valley Unit 1**	**Palisades**	**Oconee Unit 1**
10 year ISI only (Mean Value)/(Standard Error)	5.04E-09/2.54E-10	7.62E-08/4.08E-09	3.11E-08/2.55E-09
Upper Bound	5.55E-09	8.44E-08	3.62E-08
10 year Interval (Mean Value)/(Standard Error)	5.23E-09/3.12E-10	7.39E-08/3.80E-09	2.62E-08/1.28E-09
Lower Bound	4.61E-09	6.63E-08	2.36E-08
Bounding Change in Risk	9.37E-10	1.81E-08	1.26E-08

The WCAP will be revised to include the revised results. Appendices E, I, and M will also be revised to include the FAVPOST output from version 06.1.

9. Page 4-6 provides a description defined as a conservative/bounding acceptance criteria relating,

Change in CDF= Change in LERF=Increase in frequency of through wall crack growth<1E-7/yr.

Page 4-8 states that, "[f]or this evaluation, the CDF and LERF were calculated by

CDF=LERF=IE*CPF
where
CDF= Core damage frequency from a failure (events per year)
LERF = Large early release frequency from a failure (events per year)
IE = Initiating event frequency (events per year)
CPF = Conditional probability of reactor vessel failure.

a) Please precisely define CPF and fully describe the processes used to calculate the values. Is this the conditional probability of failure given a PTS event on the last day of the last operating year? Is this the average conditional probability of failure given a PTS event randomly occurring during the operating life of the plant? Or is this some other parameter?

Response: CPF is the distribution of the conditional probabilities of failure given that all postulated PTS events occur on the first day of full power operation following the refueling outage after the last operating year for the extended license of the plant. This is a conservative approach as discussed in the response to RAI 9. The calculation of CPF and TWCF for the two inspection intervals is summarized in the subsection entitled "Probabilistic Fracture Mechanics Computer Tool and Methodology" (pages 3-14 to 3-17) in Section 3.2 of the WCAP Report. The calculation of CPF by the FAVOR computer code is described in the following NRC Reports: 1) Sections 7.1 to 7.10 of NUREG-1806, *Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10CFR50.61): Summary Report*, 2006; 2) Section 4 on Crack Initiation and Section 5 on Through-Wall Cracking in NUREG-1807, *Probabilistic Fracture Mechanics – Models, Parameters, and Uncertainty Treatment Used in FAVOR Version 04.1*, 2007; 3) Sections 3 and 4 (Equations 1 to 132) of NUREG/CR-6854, *Fracture Analysis of Vessels*

– *Oak Ridge FAVOR, v04.1, Computer Code: Theory and Implementation of Algorithms, Methods and Correlations*, 2006 and 4) Section 2 and Appendix A of NUREG-1874, *Recommended Screening Limits for Pressurized Thermal Shock (PTS)*, 2007. Appendix A of NUREG-1874 describes the requested changes in going from FAVOR version 05.1, which was used in Revision 1 of WCAP-16168-NP, to FAVOR version 06.1, which was used to calculate the CPF and TWCF results in NUREG-1874.

IE is the distribution of frequencies for each postulated PTS transient (initiating event) that is combined with the CPF distribution to obtain the distribution of through-wall cracking frequency (TWCF) for that PTS transient. This combination of the CPF and IE distributions and summation of the TWCF distributions for all contributing PTS transients is performed in the FAVPOST Module of FAVOR as described in Section 2.7 of NUREG/CR-6855, *Fracture Analysis of Vessels – Oak Ridge FAVOR, v04.1, Computer Code: User's Guide*, 2006. Section 2.7 also provides the methodology and equations for calculating the statistical parameters for the total TWCF distributions that are provided in the FAVPOST output (Appendices E, I and M for the three pilot plants in the WCAP report, respectively). The total TWCF distribution becomes the LERF distribution because the conditional probability of large early release given vessel failure is taken as 1.0 as described in Section 10.5 of NUREG-1806.

The WCAP will be revised to show that:

$$\text{TWCF} = \text{LERF} = \text{CDF} = \sum_{i=1}^{N} IE_i * CPF_i$$

Where:
CDF= Core damage frequency from vessel failures due to all PTS events (events per year)
LERF = Large early release frequency from vessel failures due to all PTS events (events per year)
IE_i = Initiating event frequency (events per year) for a given PTS transient, i
CPF_i = Conditional probability of reactor vessel failure for a given PTS transient i, and
N = The total number of postulated PTS transients for a given plant.

b) Please describe and justify the operating life selected for a) above

Response: Because vessel failure during a postulated PTS event is more likely to occur with a higher degree of embrittlement, which increases with operating time due to the accumulated neutron fluence, an operating time that is realistic but not overly conservative was desired. Another consideration was trying to bound most of the plants of each nuclear steam supply system (NSSS) vendor's design using one of the operating conditions in the PTS Risk Study performed by the NRC (i.e. from Table 8.5 of NUREG-1806) through the end of the first license renewal period (60 years), as requested by the PWR Owners Group. Using the information in Table 9.5, Plant List for Generalization Study, in NUREG-1806, Beaver Valley 1 at 60 EFPY was judged to be bounding for embrittlement at all the plants with a Westinghouse NSSS design, including more embrittled Salem 1, at 60 calendar years. Likewise, Palisades at 60 EFPY was judged to be bounding for embrittlement at all the plants with a Combustion-Engineering NSSS design, including more embrittled Fort Calhoun, at 60 calendar years. However, Oconee 1 at 60 EFPY would not be bounding for embrittlement at all the plants with a Babcock & Wilcox NSSS design, specifically TMI-1, at 60 calendar years so the next higher extended condition A was used for this pilot plant. These EFPY conditions set the vessel accumulated fluence and material embrittlement levels. A maximum operating time of 80 calendar years was used instead of 60 years. This longer operating time is not only considered to be bounding but is also conservative for two reasons. First, the transients in the plant design duty cycle that could produce fatigue crack growth are specified using a given rate per calendar year as described in the response to Part a) of RAI 1. Therefore the fatigue crack growth would be about 33% higher due to the larger total number of fatigue transients. Second, the effects of in-service inspections at 60, 70 and 80 calendar years are added in the cases for ISI every 10 years (a 60% increase relative to the last inspection after 50 years). Both of these conservatisms would tend to maximize the differences in TWCF for ISI every 10 years relative to the cases for 10-year ISI only.

c) As indicated in the use of a "conditional probability of reactor vessel failure," reactor vessel failure only occurs when a demand (the PTS event) is placed on a vessel that has become susceptible to failure through the growth of cracks. Cracks grow over time and may become large enough to fail given a PTS event but remain hidden until revealed through a reactor vessel weld inspection or through a PTS event and subsequent failure. Without an event or an inspection, the CPF increases over time as the cracks grow throughout the interval. Normally, the risk from unrevealed faults during an inspection interval is estimated based on the random occurrence of the upset event during the interval combined with the likelihood of the unrevealed fault as it increases throughout the inspection interval. The risk associated with the extended interval is similarly estimated. The risk increase is the difference between these two risk estimates. Please provide this estimate of the change in risk associated with extending the inspection interval from 10 to 20 years, or justify that the estimate in the topical yields a bounding estimate of this value.

Response: The estimated change in large early release frequency associated with extending the inspection interval from 10 to 20 years is provided for the three pilot plants for each of the NSSS vendor designs in the last row of Table 4-1 on page 4-8 in Revision 1 of WCAP-16168-NP. All of these values are considered to be bounding estimates for the following reasons:

1) The values were calculated using the same methodology that was used in the PTS Risk Study performed by the NRC, which has 11 known conservatisms per items (a) through (k) on pages 12-11 to 12-12 in Section 12.4 of NUREG-1806, as compared to only 3 potential non-conservatisms per items (a) through (c) on page 12-12.
2) For most of the plants, the embrittlement at 60 EFPY is used to bound plants at the end of their first license extension (60 years). For the remaining plants, the embrittlement at the EFPY at extended condition A is used to bound plants at the end of their first license extension.
3) The number of design duty cycle transients that could produce fatigue crack growth is about 33% higher than the value for 60 years of operation.
4) Most plants are projected to not reach their design basis (40-year) number of transients after 60 years of operation (first license extension).
5) The effects of ISI are assumed to be cumulative, which is conservative because the greater the effectiveness of the ISI, the greater the difference in TWCF due to the change in inspection interval.
6) The number of in-service inspections that is credited in the cases for ISI every 10 years is 8 (60% higher) rather than the expected value of 5, since no credit is usually taken for any inspections after the extended operating license has expired (after 60 years of operation).

7) The cases for only one inspection after 10 years of operation are conservatively used to under-estimate the effects of in-service inspection every 20 years (See Figures 3-6 to 3-11 in the WCAP report).
8) Credit for the reduction in flaw density due to in-service inspections is conservatively applied to portions of the plates and forgings that are not even inspected, which would over-estimate the differences due to changes in the inspection interval per the response to RAI 18.
9) Upper 2-sigma bound values (~97.5%) on the mean TWCF are conservatively used to estimate the bounding differences instead of the FAVOR calculated mean values for the cases with ISI only after 10 years of operation (see response to RAI 12 part c).
10) Lower 2-sigma bound values (~2.5%) on the mean TWCF are conservatively used to estimate the bounding differences instead of the FAVOR calculated mean values for the cases with ISI every 10 years (see response to RAI 12 part c).
11) Using separate upper and lower 2-sigma bounds in items 9) and 10) is about 40% conservative relative to using an upper 2-sigma bound on the combined uncertainties in the difference in mean values of TWCF.

10. Page 4-8 states that, "the transient initiating frequency distributions were identified in the NRC PTS Risk Study [7] and are included in Appendices D, H, and L for the pilot plants. The Appendices include the (grouped) sequences but do not include the transient frequency distribution. Please provide the mean values of the transient frequency in the Appendices to provide the link to the PTS technical basis results that are used in the Topical.

Response: The PTS transients are the same as those in Appendix A of NUREG-1806. The "TH#" in Appendices D, H, and L corresponds to that in Appendix A of NUREG-1806. A column will be added to the tables in Appendices D, H, and L in the WCAP report to include the mean initiating event frequency from NUREG-1806 for each of the transients.

11. Extending the interval for inspection of reactor vessel welds will, to some extent, increase the likelihood that a PTS event will cause a reactor vessel failure. A reactor vessel failure will fail the reactor coolant fission product boundary, and may directly fail the reactor fuel fission product boundary. The discussion on page 4-9 and 4-10 about maintaining defense-in-depth emphasizes 1) the low likelihood of a PTS induced rupture, 2) that a "sampling of plants" inevitably undergo examinations in a given year so that unknown degradation mechanisms will not be ignored for 20 years, and 3) that all reactor coolant pressure boundary failures occurring to date have been identified though leakage.

a) The defense-in-depth evaluation is performed in parallel with the risk evaluation in the integrated decision making process. Please assess the proposed increase in inspection interval against each of the defense-in-depth elements listed on page 4-4 of WCAP-16168.

Response: Page 4-4 of WCAP-16168-NP, Revision 1, also states from Regulatory Guide 1.174 that:

"Defense-in-depth philosophy is not expected to change unless:

- A significant increase in the existing challenges to the integrity of the barriers occurs.
- The probability of failure of each barrier changes significantly.
- New or additional failure dependencies are introduced that increase the likelihood of failure

compared to the existing conditions.
- The overall redundancy and diversity in the barriers changes."

The extension in inspection interval will not result in any of the changes identified above. For this reason the defense in depth elements listed on page 4-4 will not be impacted. Additional assessment of the impact on each of the defense-in-depth elements from page 4-4 is provided in the following:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved:

  The proposed increase in inspection would not cause an increased reliance on any of the identified elements. Therefore, the interval increase would not change the existing balance among prevention of core damage, prevention of containment failure, and consequence mitigation.

- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided:

  The change in inspection interval does not change the robustness of the vessel design in any way. It is because of this robustness that the inspection interval can be doubled with no significant change in failure frequency.

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences to the system (e.g., no risk outliers):

  The proposed increase in inspection interval does not impact system redundancy, independence, or diversity in any way, since it is not changing the plant design or how it is operated.

- Defenses against potential common cause failures are preserved and the potential for introduction of new common cause failure mechanisms is assessed:

  The proposed increase in inspection interval does not impact any defenses against any common cause failures and there is no reason to expect the introduction of any new common cause failure mechanisms. This requirement applies to multiple active components. There is only one reactor vessel per plant and it is a passive component.

- Independence of barriers is not degraded (the barriers are identified as the fuel cladding, reactor coolant pressure boundary, and containment structure):

  The increase in inspection interval does change the relationship between the barriers in anyway and therefore does not degrade the independence of the barriers. The change in inspection interval does not change the robustness of the vessel design in any way. It is because of this robustness that the inspection interval can be doubled with no significant change in failure frequency.

- Defenses against human errors are preserved:

  The increase in the RV inspection interval does not impact any defenses against human errors in any way. The increase in the inspection interval reduces the frequency for which the lower internals need to be removed. Reducing this frequency reduces the possibility for human error and potentially damaging the core.

b) It is likely that all plants will request to extend the inspection interval from 10 to 20 years. Universal, or near universal, adoption of this option would, unless otherwise arranged, lead to a 10 year period were no reactor vessel weld inspections would be required. Please provide additional discussion specifying how a "sampling of plants" performing reactor vessel welds inspection over the next 10 years can be achieved.

> **Response:** On October 31, 2006, the PWROG submitted to the NRC Letter OG-06-356, "Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval'." This letter provides a plan of when inspections will be performed provided that Revision 1 of WCAP-16168-NP is approved and that each plant makes a plant specific request to the Staff to extend their interval. As discussed in the letter and as previously agreed upon by the Staff, the Staff will review plant specific requests to implement the 20 year interval against the PWROG Plan. Approval of the plant specific request is expected if the date requested in the plant specific request is within one-refueling cycle of that identified in the PWROG Plan.

12. The following repeats part of Table 4-1 in the Topical.

Table 4-1 (mean values) Large Early Release Frequencies			
	BV1	Palisades	OC1
10-Year ISI Only	5.04E-09	1.54E-08	2.06E-09
ISI Every 10 Years	4.10E-09	1.67E-08	2.18E-09

For Beaver Valley Unit 1(BV1), the ISI Every 10 years (4.10E-9) is less than the 10-year inservice inspection (ISI) Only (5.04E-9). This seems reasonable because the repetitive ISI provides opportunities to find and remove growing cracks before they can lead to vessel failure given a PTS event. However, for Palisades and Oconee Unit 1(OC1) (and in a number of individual bin frequencies in the Appendices) the situation is reversed. For example, for Palisades above, the ISI every 10 years (i.e., 1.67E-08) is greater than the 10-year ISI Only (1.54E-08). This appears to indicate that it is riskier to inspect than to not inspect, but it may demonstrate that the Monte Carlo calculations by the FAVOR code were not converged sufficiently to reduce the uncertainty of the mean values of the total TWCF to less than the effect of detecting and removing surface-breaking flaws found by inspections at ten-year intervals.

a) Please explain why the mean failure estimates are sometimes opposite of what is expected. The explanation should include a justification that this analysis is precise enough to support the change in risk estimates instead of further investigating the apparent discrepancy and developing results that no longer appear contradictory. If the answer to b) below removes this apparent discrepancy the answer to this question can be referred to b).

Response: The technical basis for the PWR Owners Group project to extend the vessel ISI interval was always based on the premise that surface breaking flaws would never be a significant contributor to vessel failure (through-wall cracking) frequency due to PTS transients. This was based upon the fact that the frequency of surface breaking flaws would have to be very small, since none had ever been discovered during either pre-service or in-service examinations and even if they did exist, their circumferential orientation due to the cladding welding process (see Section 9.6.1 of NUREG/CR-6817) would lead to arrest before through-wall fracture (see Figure 9.7 of NUREG-1806). The results reported in Revision 1 of WCAP-16168-NP just confirm this premise even when potential fatigue crack growth is explicitly considered. Because of the uncertainty in how accurately an insignificant (null) effect can be calculated by FAVOR using standard Monte-Carlo simulation methods, a conservative method of comparing upper and lower 2-sigma bounds was used as described in the response to Part c of this RAI.

b) The TWCF estimates are dominated by the more numerous embedded axial flaws, with little contribution from the surface-breaking circumferential flaws that are varied by the WCAP FAVOR analysis. The FAVOR code treats each flaw independently of every other flaw, and it is possible to calculate the effect of the TWCF contribution for only surface breaking flaws, without including any of the embedded flaws in the calculation. Such an evaluation would isolate the parameter of interest (the TWCF caused by surface flaws) and thereby eliminate the possibly dominant affect of the uncertainty on the quantitative results associated with the TWCF from embedded flaws. In order to appropriately evaluate the uncertainty of the TWCF contribution created by surface-breaking flaws as opposed to embedded flaws, please evaluate these flaws separately, so that probability distributions for the TWCF contribution of surface breaking flaws can be obtained and compared for the two inspection cases.

Response: For the reasons stated in the response to Part a of this RAI, there should be almost no contribution from surface-breaking circumferential flaws. However, with such few failures resulting from surface breaking flaws, there may be no way to obtain a converged solution using Monte-Carlo simulation because the accuracy is based upon the number of failures in the total number of vessel simulations. That is, to obtain convergence and acceptable accuracy, a significant number of failures are required for the specified number of simulations. Note that 70,000 vessel simulations were required for the results provided in the response to Part c of this RAI. Even for 500,000 simulations without any embedded flaws, the value of through-wall cracking frequency (TWCF) calculated by FAVOR was zero (no failures) for both ISI cases. The change in TWCF and the change in large early release frequency (LERF) would also be essentially zero, which would certainly be considered insignificant per the requirements of Regulatory Guide 1.174.

c) Please describe in detail how the mean upper bound and mean lower bound parameters (included in other entries in Table 4-1) are developed.

Response: The information in Table 4-1 is a summary of the vessel failure frequency results for each of the three pilot plants that are given in Tables 3-2, 3-3 and 3-4, respectively, in the WCAP. For the first plant (Beaver Valley Unit 1), the mean value and standard error for 10-Year ISI Only of 5.04E-09 and 4.83E-10, respectively, in Table 3-2 on page 3-18 were taken from the FAVPOST Output values of 5.0405E-09 and 4.8272E-10, respectively, on page E-5 in Section E-1 of Appendix E. As described in the first paragraph of Section 3.3 (page 3-18), "The Upper Bound Value was determined by adding 2 times the standard error as reported by FAVPOST to the mean value of the 10-Year ISI Only case." In Table 3-2, the Upper Bound Value of 6.01E-09 came from 5.0405E-09 + 2 x 4.8272E-10 = 6.00594E-09. The mean value and standard error for ISI Every 10 Years of 4.10E-09 and 2.89E-10, respectively, in Table 3-2 were taken from the FAVPOST Output values of 4.0995E-09 and 2.8934E-10, respectively, on page E-13 in Section E-2 of Appendix E. As described in the first paragraph of Section 3.3, "The Lower Bound Value was determined by subtracting 2 times the standard error as reported by FAVPOST from the mean value of the ISI Every10 Years case." In Table 3-2, the Lower Bound Value of 3.52E-09 came from 4.0995E-09 - 2 x 2.8934E-10 = 3.52082E-09. As described in the first paragraph of Section 3.3, "a change in failure frequency was conservatively calculated based on the difference between an Upper Bound and a Lower Bound." In Table 3-2, the Bounding Difference of 2.49E-09 came from 6.00594E-09 - 3.52082E-09 = 2.48512E-09. These same calculations were also performed for Palisades in Table 3-3 in Section 3-4 using the FAVPOST Output on pages I-6 and I-12 in Sections I-1 and I-2 of Appendix I and for Oconee Unit 1 in Table 3-4 in Section 3-5 using the FAVPOST Output on pages M-7 and M-14 in Sections M-1 and M-2 of Appendix M.

The following equations for the standard deviation and standard error are provided in Section 2.7 on FAVPOST Output in NUREG/CR-6855, *Fracture Analysis of Vessels – Oak Ridge FAVOR, v04.1, Computer Code: User's Guide*, 2006:

$$\text{Standard Deviation, } s = \sqrt{\frac{\sum_{i=1}^{n}(x_i - \bar{x})^2}{n-1}}$$

$$\text{Standard Error} = \sqrt{\frac{\sum_{i=1}^{n}(x_i - \bar{x})^2}{n(n-1)}}$$

In these equations, x_i is the value of TWCF for all PTS transients, which is calculated for each vessel simulation i, and the x with the bar over it is the mean value of TWCF for all n vessel simulations, which is typically greater than 60,000. Note that the standard error, which is a measure on the uncertainty on the mean value, is equal to the standard deviation, which is a measure of the uncertainty in all the simulated values of TWCF, divided by the square root of the number of simulations. The uncertainty on the mean value of TWCF is used per the guidance in Section 2.2.5.5, Comparisons with Acceptance Guidelines, in Revision 1 of Regulatory Guide 1.174. This section states: "Because of the way the acceptance guidelines were developed, the appropriate numerical measures to use in the initial comparison of the PRA results to the acceptance guidelines are mean values. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the input parameters and those model uncertainties explicitly represented in the model."

This same approach was used to calculate the upper bound, lower bound, and change in failure frequency for the FAVOR version 06.1 results presented in the response to RAI 8.

d) Page 3-18 states that; "[s]tatistically, the difference between the mean failure frequencies for the "ISI Every 10 Years" case and the "10-year ISI Only" case is insignificant." Please describe the statistical techniques used to develop this statement, e.g., was hypothesis testing about the two means performed? How does this observation support the use of the Topical methodology in demonstrating that the increase in the inspection interval from 10 to 20 years satisfies the risk-informed guidelines in RG 1.174.

> **Response:** The null hypothesis is that the risk difference for the two ISI cases is zero for the reasons stated in the responses to previous parts of RAI 12. For the difference in mean values to be statistically significant at the 99 percent confidence level, the T-statistic would require its value to be equal to or greater than 2.35 times the sample standard deviation. For the detailed BV1 example in part c above, the sample standard deviations would be the square root of the sum of the squares of the standard errors for the two ISI cases, $(4.8272^2 + 2.8934^2)^{0.5}$ x 1.0E-10 = 5.6279E-10. This value is 2.35 times the 99% confidence bound of 1.3226E-09. The actual difference in mean values is 5.0405E-09 - 4.0995E-09 = 0.9410E-09, which is therefore not statistically significant relative to zero at the 99% confidence level. Even if the results were reversed and the difference was - 0.9410E-09, it would still not be statistically significant relative to zero at the 99% confidence level.
>
> Section 4 in Revision 1 of WCAP-16168-NP, including the methodology to calculate the change in risk in Table 4-1 as described in the response to Part c of this RAI, clearly demonstrates that the increase in the inspection interval from 10 to 20 years satisfies the risk-informed guidelines in Regulatory Guide 1.174.

13. The Tables in Appendix A appear to illustrate the information that the WOG proposes will be contained in individual licensee relief requests. We note that the "Plant Specific Basis" proposed by WOG in the Tables in Appendix A refers to the "PTS Generalization Study," a document that was not submitted by the WOG as part of the Topical and is not being reviewed by the staff for use in relief request to extend the inspection interval of reactor vessel welds. We also note that there is a plant specific quantitative estimate of the "Through Wall Cracking Frequency" in the two examples that implies a plant specific calculation. In particular, please explain the value of 2.15E-12 events/year provided for the Wolf Creek example plant in Table 1 of Appendix A-1. Also, please explain the value of 4.67E-9 events/year provided for the pilot plant in the same table, which does not seem to match other pilot plant information elsewhere in the Topical. Please describe the analysis that the WOG proposes that licensee's will need to perform to support a plant specific relief request, and relate these analyses to the methodology and results in the Topical for which the WOG is requesting approval.

> **Response:** The "PTS Generalization Study" (ADAMS Accession number: ML042880482) is Reference 25 of WCAP-16168-NP, Revision 1. This study was performed as part of the NRC PTS Risk Re-evaluation as described in NUREG-1806. The purpose and conclusions of the Generalization Study are stated on pages 3-6 and 3-7 of the WCAP, respectively. The purpose is consistent with that stated in the first paragraph in Section 9.3 of NUREG-1806: "Our aim was to identify whether the design and operational features that are the key contributors to PTS risk (see Section 8.6) vary significantly enough in the larger population of PWRs to question the generality of our results." The overall conclusion is consistent with that stated in the last paragraph in Section 9.3.3 of NUREG-1806: "These combined observations support the overall conclusion that the TWCF estimates produced for the detailed analysis plants are sufficient to characterize (or bound) the TWCF estimates for the five generalization plants and, thus, by inference, PWRs in general." The Generalization Study was reviewed by the Staff as part of the PTS Risk Re-evaluation, and it

is the basis for the fleet-wide applicability of the proposed PTS rule in NUREG-1806 and NUREG-1874. Therefore, the Generalization Study was not submitted for review with WCAP-16168-NP, Revision 1.

The through-wall cracking frequency value of 2.15E-12 for Wolf Creek was calculated using the TWCF correlations in NUREG-1806 and the information in the table below. The beltline material properties in this table were taken from the NRC Reactor Vessel Integrity Database (RVID) and the fluence projections were taken from WCAP-16030, *Evaluation of Pressurized Thermal Shock for Wolf Creek*, May 2003.

Reactor Vessel Beltline Material Properties for Wolf Creek											
Major Material Region Description									Un-Irradiated $RT_{NDT(u)}$		Fluence @ 54 EFPY [10^{19} Neutron/cm^2, E>1 MeV]
#	ID	Component Type	Heat	Location	Flux Type / Base metal	Cu [wt%]	Ni [wt%]	P [wt%]	[°F]	Method	
1	R2508-3	Plate	C4935-2	Lower	A 533B	0.070	0.620	0.008	40.0	Plant Specific	3.51
2	R2508-1	Plate	B8759-2	Lower	A 533B	0.090	0.670	0.009	0.0	Plant Specific	3.51
3	R2508-2	Plate	C4840-2	Lower	A 533B	0.060	0.640	0.008	10.0	Plant Specific	3.51
4	R2005-2	Plate	NR61 783-1	Intermediate	A 533B	0.040	0.640	0.007	-20.0	Plant Specific	3.51
5	R2005-3	Plate	NR61 799-1	Intermediate	A 533B	0.050	0.630	0.007	-20.0	Plant Specific	3.51
6	R2005-1	Plate	NR61 836-1	Intermediate	A 533B	0.040	0.660	0.008	-20.0	Plant Specific	3.51
7	101-142A	Axial Weld	90146	Lower	Linde 0091	0.040	0.080	0.005	-50.0	Plant Specific	1.58
8	101-142B	Axial Weld	90146	Lower	Linde 0091	0.040	0.080	0.005	-50.0	Plant Specific	3.08
9	101-142C	Axial Weld	90146	Lower	Linde 0091	0.040	0.080	0.005	-50.0	Plant Specific	3.08
10	101-124A	Axial Weld	90146	Intermediate	Linde 0091	0.040	0.080	0.005	-50.0	Plant Specific	1.58
11	101-124B	Axial Weld	90146	Intermediate	Linde 0091	0.040	0.080	0.005	-50.0	Plant Specific	3.08
12	101-124C	Axial Weld	90146	Intermediate	Linde 0091	0.040	0.080	0.005	-50.0	Plant Specific	3.08
13	101-171	Circ. Weld	90146	Int/Lower	Linde 124	0.040	0.080	0.007	-50.0	Plant Specific	3.51

The pilot plant TWCF value of 4.67E-9 was also obtained using the TWCF correlations in NUREG-1806. This value does not match the values in the tables in the WCAP because the values in the tables were determined using the FAVOR Code rather than the TWCF correlation based upon maximum values of RT_{NDT} for the beltline components.

The pilot plant TWCF values were recalculated using the TWCF correlations in NUREG-1874 and are presented in the Table below. The WCAP will be revised to include these values.

	Beaver Valley Unit 1	**Palisades**	**Oconee Unit 1**
Condition	60 EFPY	60 EFPY	Ext-A
RT_{MAX-AW} (°F)	204	247	253
RT_{MAX-CW} (°F)	253	231	277
RT_{MAX-PL} (°F)	253	209	158
RT_{MAX-FO} (°F)	0	0	0
$TWCF_{95-AW}$	4.49E-09	1.57E-07	2.23E-07
$TWCF_{95-CW}$	7.54E-11	7.11E-12	5.72E-10
$TWCF_{95-PL}$	3.66E-09	2.52E-10	7.35E-12
$TWCF_{95-FO}$	0.00E+00	0.00E+00	0.00E+00
$TWCF_{95-TOTAL}$	1.76E-08	3.16E-07	4.42E-07

To implement the extended inservice inspection interval justified in the WCAP, a licensee would have to demonstrate that the pilot plant analyses are bounding. The criteria to be evaluated to determine whether the pilot plant analyses are bounding are identified in Table A-1 of Appendix A of the WCAP. These criteria were selected based on feedback from the Staff during meetings prior to the submittal of the WCAP for review.

Dominant PTS Transients in the NRC PTS Risk Study are applicable:
The transients evaluated in the WCAP pilot plant analyses were the PTS transients from the NRC PTS Risk Re-evaluation. For this criterion, it is necessary to demonstrate that these transients are applicable to a specific plant. At the time Revision 0 of the WCAP was issued, the Generalization Study had not yet been completed. Therefore, it would have been necessary for each plant to compare design features to determine if the pilot plant PTS transients were applicable to the specific plant. However, the Generalization Study has now been performed and the pilot plant PTS transients have been found to be representative of all the PWR plants in the domestic fleet. As stated in the last paragraph in Section 3.2.1 of NUREG-1874, this "study demonstrates that risk-significant PTS transients do not have any appreciable plant-specific differences within the population of PWRs currently operating in the United States." Therefore, plant specific analyses are no longer needed for this criterion.

Through Wall Cracking Frequency (TWCF):
The plant specific TWCF value determined using the correlations in NUREG-1874 must be lower than the pilot plant TWCF value calculated using the TWCF correlations in NUREG-1874. The TWCF is essentially a measure of the embrittlement of the reactor vessel and by demonstrating that the pilot plant has a higher TWCF value, the pilot plant change in risk calculation is bounding.

Frequency and Severity of Design Basis Transients:
It is necessary to demonstrate that the amount of fatigue crack growth considered in the pilot plant analyses is bounding for a specific plant. Since the amount of fatigue crack growth was calculated using the design basis transients, a comparison of design basis transients must be performed to ensure that the assumed number of heatup-coodown transients per year is also applicable to the specific plant.

Cladding Layers (Single/Multiple):
The pilot plant analyses were performed assuming a single layer of cladding because the probability of having a surface breaking flaw in multi-layer cladding is much less than that of single-layer cladding. Since the pilot plant analyses were performed with single-layer, all plants are bounded by this parameter and this criteria is documented strictly for informational purposes.

Table A-2 provides additional criteria relative to inspection. The purpose of the *Inspection Methodology*, *Number of Past Inspections*, and *Number of Indications Found* fields is discussed in the response to Part b of RAI 3. The purpose of the *Proposed inspection schedule for balance of plant life* field is for comparison to the inspection plan contained in PWROG letter OG-06-356, as discussed in the response to Part b of RAI 11.

14. The frequency of PTS challenges is a primary input to the change in risk estimates associated with extending the inspection interval for reactor vessel welds from 10 to 20 years. The Topical states that the transient frequency results developed for the PTS technical basis are used in the risk increase calculations in the Topical. Regulatory Guide 1.174 states that a probabilistic risk analysis used to support each risk-informed application should be technically adequate. Technically adequate is defined, at the highest level, as an analysis that is performed correctly, in a manner consistent with accepted practices, commensurate with the scope and level of detail required to support the requested change.

a) Please describe how the Topical proposes that individual licensees' will obtain or develop PTS transient frequency estimates to use in support of their request for relief.

Response: Individual licensees will not be required to obtain or develop PTS transient frequency estimates to use in support of their request for interval extension. As discussed in the response to RAI 13, it was originally intended that individual licensees would have to compare significant design features such as PORV capacity and RWST temperature to determine if the pilot plant PTS transients were applicable to their specific plant. However, since that time, the PTS Generalization Study has been completed as summarized on pages 3-6 and 3-7 and of the WCAP Report. As stated in the last paragraph in Section 3.2.1 of NUREG-1874, this "study demonstrates that risk-significant PTS transients do not have any appreciable plant-specific differences within the population of PWRs currently operating in the United States." Furthermore, the overall conclusion from this study is provided in the last paragraph in Section 9.3.3 of NUREG-1806: "the TWCF estimates produced for the detailed analysis plants are sufficient to characterize (or bound) the TWCF estimates for the five generalization plants and, thus, by inference, PWRs in general."

b) Given the response to a), please propose how the probabilistic risk assessment analyses that will be relied upon to support the relief requests will be demonstrated to be of sufficient technical adequacy so that there is confidence that the increases in core damage frequency or risk caused by the extension of the reactor vessel weld inspection interval from 10 to 20 years is small. One acceptable approach to assess technical adequacy is to assess the analysis against endorsed standard as described in RG 1.200.

Response: The conditional probabilities for core damage and large early release assumed in the WCAP and PTS Risk Re-evaluation are 100% for a through-wall crack in the vessel. Given this assumption, the conclusions of the PTS Generalization Study, and the response to Part a of this RAI, no probabilistic risk assessment analyses will be relied upon to support the requests for interval extension. Therefore, it will not be required that licensees demonstrate the technical adequacy of the PRA. This is consistent with the NRC proposed voluntary PTS Rule, which is not expected to require the plant PRA to satisfy R.G. 1.200 requirements.

15. When TWCF increases due to increases in neutron fluence and its resulting embrittlement, the fractional contributions to TWCF from different flaw types (e.g., surface-breaking vs. embedded, circumferential vs. axial, small vs. large) can change substantially. Is the WCAP analysis applicable to plants which have TWCF values substantially greater than the TWCF of the pilot plants in the WCAP? Is so, please provide an example to illustrate the application and specify any TWCF limit to the range of applicability.

Response: The pilot plants were chosen with the intent that there would be no domestic PWR plants with higher TWCF values. Since the time the pilot plants were chosen, analyses performed by NRC Research using plant data available in RVID have shown that there are several Westinghouse plants that could have TWCF values higher than those of the pilot plant by end of their operating license. For these plants, additional evaluation would be required to demonstrate that, even though the TWCF values are higher than the pilot plant values, the conclusions from the pilot plant analyses are still applicable. Furthermore, plants that have implemented the extended inspection interval will be required to reevaluate their TWCF value consistent with the response to RAI 16. In the event that a plant specific TWCF value exceeds the appropriate NSSS pilot plant value as a result of this reevaluation, additional evaluation would also be required. This discussion will be included as part of the clarification to be added to Appendix A as part of the response to RAI 13.

16. New industry experience or information may arise that indicates that the TWCF estimates may need to be reevaluated. For example, licensees that utilize relaxations available under the new PTS rulemaking (50.61a) may make changes to their plants that could increase the TWCF above the values in the WCAP pilot plants (due to increases in neutron fluence and its resulting embrittlement) which were intended to be bounding examples. Principle 5 of RG 1.174 is to provide a monitoring program to assure that parameters critical to the conclusion of acceptability remain at acceptable values during the life of the change to the license requirements. What type of monitoring and feedback process is proposed in the Topical that would call for a re-evaluation of the TWCF as appropriate to ensure that, over time, the validity of the analysis demonstrating an acceptable increase in risk is maintained?

Response: The PWROG proposes that for plants implementing the extended interval, TWCF be re-evaluated any time fluence is projected to increase by more than 10 percent, which is less than one standard deviation on the global fluence that is input to FAVOR. Fluence may be projected to increase as a result of core reloading, core loading pattern, power uprating, or when a surveillance capsule is removed from the reactor vessel and evaluated. This is consistent with the current approach for re-evaluating pressure-temperature limit curves and RT_{PTS} values. The WCAP will be revised to include this requirement.

17. Current analyses of crack stresses during plant operations indicate that embedded cracks will not grow with time. Because the staff's FAVOR analysis indicates that embedded axial cracks contribute nearly all of the TWCF, it follows that the assumption that embedded axial cracks do not grow with time is an important modeling assumption that contributes to the small risk increase estimated for extending the inspection interval. RG 1.174 recommends address important modeling assumptions by performing sensitivity studies or using qualitative arguments. Please discuss how sensitive the quantitative results of the change in risk analysis are to the assumption that embedded cracks will not grow?

Response: There is no sensitivity to growth of embedded flaws to subcritical crack growth relative to embrittled vessel failure due to postulated PTS transients. This lack of sensitivity is based upon the NRC evaluation described in Section 3.2, "Assumption of No Subcritical Crack Growth," of NUREG-1807, *Probabilistic Fracture Mechanics – Models, Parameters, and Uncertainty Treatment Used in FAVOR Version 04.1*, 2007. Note that this evaluation concluded that there is no significant subcritical crack growth of either surface breaking or embedded flaws due to stress corrosion cracking or fatigue. Because embedded flaws are not exposed to the primary coolant, their crack growth is substantially less than that for surface breaking flaws subjected to the same loading. However, because the high sensitivity of TWCF due to any potential increase in the size of the embedded flaws (April 2007 NRC Memo, *Development of Flaw Size Distribution Tables for Draft Proposed Title 10 of the Code of Federal Regulations (10CFR) 50.61a*, ADAMS: ML070950392), periodic inspection every 20 years is proposed to ensure no embedded flaw crack growth has occurred. This is being done even though the risk analyses in Revision 1 of WCAP-16168-NP show that no inspections are required except the initial one after 10 years of operation, to satisfy the acceptably small change in risk (LERF) criteria per Regulatory Guide 1.174.