REVISED FINAL SAFETY EVALUATION BY

THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16168-NP-A, REVISION 2,

"RISK-INFORMED EXTENSION OF THE REACTOR VESSEL

IN-SERVICE INSPECTION INTERVAL"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION AND BACKGROUND

By letter dated January 26, 2006 (Reference 1), as supplemented by letter dated June 8, 2006 (Reference 2), the Westinghouse Owners Group (WOG), currently known as the Pressurized Water Reactor (PWR) Owners Group (PWROG), submitted Topical Report (TR) WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," for U.S. Nuclear Regulatory Commission (NRC) staff review. By letter dated October 16, 2007 (Reference 3), the PWROG submitted responses to the NRC staff's request for additional information (RAI) questions on WCAP-16168-NP, Revision 1, and provided WCAP-16168-NP, Revision 2, but did not expand its scope as originally submitted for NRC staff review. The NRC staff issued a final safety evaluation (SE) dated May 8, 2008 (Reference 4), "Final Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report WCAP-16168-NP, Revision 2," approving the TR for referencing.

In the accepted version of the TR, WCAP-16168-NP-A, Revision 2 (Reference 5) (hereafter referred to as the TR), the PWROG provided the technical and regulatory basis for decreasing the frequency of inspections by extending the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI inservice inspection (ISI) interval from the current 10 years to 20 years for ASME Code Section XI, Category B-A and B-D reactor vessel (RV) welds.

The TR described risk-informed pilot studies based, for the most part, on the results of the NRC's pressurized thermal shock (PTS) research program. The NRC's Office of Nuclear Regulatory Research completed this research program to update the PTS regulations. On January 4, 2010, by *Federal Register Notice* (75 FR 13, Reference 6), the NRC published alternate regulations to provide updated fracture toughness requirements for protection against PTS events for PWR RVs as Title 10, Section 50.61a, of the *Code of Federal Regulations* (10 CFR 50.61a). NUREG-1806, Volume 1, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report;" NUREG-1806, Volume 2, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Appendices" (the PTS Risk Study)

Enclosure

(References 7 and 8); and NUREG-1874, ‘‘Recommended Screening Limits for Pressurized Thermal Shock (PTS)’’ (Reference 9), provided the technical basis for the rulemaking. These reports summarized and referenced several additional reports on the same topic.

By letter dated December 1, 2009 (Reference 10), the PWROG communicated a need to revise the PWROG plan for implementation of the TR. Additionally, the letter requested clarification of the NRC staff’s expectations for applicants utilizing the TR. The PWROG further updated the implementation plan by letter dated July 12, 2010 (Reference 11). In response, the NRC staff decided to revise the SE to the TR, to provide clarification and include the latest NRC positions on applications for use of the TR. This revised final SE represents the update to the May 8, 2008, SE, addressing the PWROG concerns and revised implementation plans.

2.0 REGULATORY EVALUATION

ISI of ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). The regulation at 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The regulations require that ISI of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

The current requirements for the inspection of RV pressure retaining welds have been in effect since the 1989 Edition of ASME Code, Section XI. Article IWB-2000 of the ASME Code, Section XI establishes an inspection interval of 10 years. The TR proposed a methodology that can be used by individual licensees to demonstrate that extending the inspection interval on an interval-by-interval basis for their Category B-A pressure retaining RV welds and Category B-D full penetration RV nozzle welds from 10 to 20 years would provide an acceptable level of quality and safety.

The NRC staff based its review of the risk information on NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance” (Reference 12). SRP Section 19.2 directs the NRC staff to review each of the four elements suggested in Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Section 2 (Reference 13). These elements are: (1) Define the Proposed Changes, (2) Conduct Engineering Evaluations, (3) Develop Implementation and Monitoring Strategies, and (4) Document the Evaluations and Submit the Request.

The NRC staff also used further guidance in RG 1.174. RG 1.174 describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed licensing basis changes by considering engineering issues and applying risk insights.

One acceptable approach to making risk-informed decisions about the proposed change is to show that the proposed changes meet five key principles stated in RG 1.174, Section 2:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. When proposed changes result in an increase in core-damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
5. The impact of the proposed change should be monitored using performance measurement strategies.

RG 1.174 provides numerical risk acceptance guidelines that are helpful in determining whether or not the fourth key principle has been satisfied. These guidelines are not to be applied in an overly prescriptive manner; rather, they provide an indication, in numerical terms, of what is considered acceptable. The intent in comparing risk results with the risk acceptance guidelines is to demonstrate with reasonable assurance that the fourth key principle has been satisfied.

3.0 TECHNICAL EVALUATION

The objective of ISI is to identify conditions, such as flaw indications, that are precursors to leaks and ruptures and which violate pressure boundary integrity principles for plant safety. The TR includes a detailed analysis of the potential effects of extending the RV weld ISI interval for three pilot plants: Beaver Valley Power Station, Unit 1 (BV1), Palisades Nuclear Plant (Palisades), and Oconee Nuclear Station, Unit 1 (OC1). These three units include one unit from each of the PWR vendors and are the same plants that were evaluated in detail in the PTS Risk Study. The TR proposed a method that each licensee could use to apply the results from the three pilot plant applications to its plant.

The TR used the estimated through-wall cracking frequency (TWCF) as a measure of the risk of RV failure. The correlation for determining plant-specific TWCF was based on plant-specific data and can be found in NUREG-1874. This correlation took into consideration the contribution to TWCF from each of the most limiting plate, forging, axial weld, and circumferential welds. These individual TWCF contributions were then weighted based on pilot plant data and summed to determine a total RV TWCF.

3.1 Define the Proposed Change

The TR proposed to extend the inspection interval for ASME Code, Section XI, Category B-A and B-D RV welds from 10 years to a maximum of 20 years. The change will be accomplished through plant-specific requests for an alternative pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the alternative inspection interval provides an acceptable level of quality and safety.

The 20-year inspection interval is a maximum interval and the PWROG did not request, and the NRC staff does not endorse, that all RV inspections are discontinued for the 10 years following approval of this methodology (as would occur if every licensee were granted an extension from 10 to 20 years). In response to RAI 11b from Reference 3, the PWROG explained how a sampling of plants performing reactor inspections over the next 10 years could be achieved. In its request for an alternative, each licensee shall identify the years in which future inspections will be performed. The dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan provided to the NRC in Reference 11.

The inspection method, the acceptance criteria, and reporting requirements for inspection results that will modify from ASME Code requirements are discussed in Section 3.3 of this revised final SE.

3.2 Conduct Engineering Evaluations

According to the guidelines in RG 1.174 and SRP Chapter 19.2, the second element associated with a risk-informed application is an analysis of the proposed change using a combination of traditional engineering analysis with supporting insights from a risk assessment.

The objective of this study was to verify that a reduction in the frequency of volumetric examination of the RV full-penetration welds could be accomplished with an acceptably small change in risk. The methodology used to justify this reduction involved estimating the potential increase in risk caused by extending the RV inspection interval from 10 to 20 years. The increase in risk was evaluated against RG 1.174 criteria to determine if the values met the specified regulatory guidelines. The other key principles in RG 1.174 were also addressed in the evaluation. The intent was that licensees could then use the results of this bounding assessment to demonstrate that their RV and plant are bounded by the generic analysis, thereby justifying an extension of their plant-specific RV weld inspection interval.

The engineering evaluations in the TR were based on the NRC staff's PTS Risk Study and several associated papers (References 7, 10-11).

3.2.1 Engineering Evaluation

The ISI interval extension methodology was primarily based on a risk analysis, including a probabilistic fracture mechanics (PFM) analysis of the effect of different inspection intervals on the frequency of RV failure due to postulated PTS transients. RV failure is defined for the purposes of this study as through-wall cracking of the RV wall. The likelihood of RV failure was postulated to increase with increasing time of operation due to the growth of pre-existing fabrication flaws by fatigue in combination with a decrease in RV fracture resistance due to irradiation. Credible, postulated PTS transients that could potentially lead to RV failure were considered to occur at the worst time in the life of the plant (as defined by flaw size and level of RV embrittlement). The PFM methodology allowed for the consideration of distributions and uncertainties in flaw number and size, material properties, crack growth resulting from fatigue, accident transients, stresses, and the effectiveness of inspections. The PFM approach led to a conditional RV failure frequency due to a given loading condition and a prescribed inspection interval. The PFM analyses documented in the TR evaluated the impact of different inspection intervals on the three, previously identified pilot plants.

<u>Limiting Location for RV Failure</u>

To determine the limiting location in the RV, the PWROG evaluated the impact of flaws in each RV region. The PWROG used deterministic fracture mechanics analyses, which utilized a 10 percent through-wall flaw, assumed 40 effective full power years (EFPY) of embrittlement for the flaws in the RV beltline and included fatigue crack growth due to normal plant operating transients for all flaws. Each crack length was evaluated at the end of a 10-year interval to determine the maximum applied stress intensity factor ($K_{I\ applied}$). The ratio of the maximum allowable stress intensity factor ($K_{I\ allowable}$), per the ASME Code, Section XI, Appendix A criteria, to $K_{I\ applied}$ was used as a measure of the margins to failure. The lower the ratio of $K_{I\ allowable}/K_{I\ applied}$, the lower the margin to failure and the more limiting the location. Figures 3-1 and 3-2 in the TR indicated that the beltline welds have the lowest ratio of ASME Code allowable stress intensity values ($K_{I\ allowable}/K_{I\ applied}$). These figures do not include the full penetration nozzle-to-vessel welds. The NRC staff requested that the PWROG provide the ratio of ASME Code allowable stress intensity value for full penetration nozzle-to-vessel welds to demonstrate that the beltline welds were the limiting locations. In the response to RAI 5 from Reference 3, the PWROG provided the requested information. The PWROG analyses indicated that the beltline is more limiting than the full penetration nozzle-to-vessel welds.

The results from the PWROG deterministic analyses were consistent with assumptions utilized in the PTS Risk Study, which concluded that the limiting RV region was the beltline region. Since the RV beltline region has the lowest margin to failure, the NRC staff also concluded that the beltline region is the most limiting location and the beltline location can be used to determine the impact of different inspection intervals on the frequency of RV failure.

<u>Distributions and Uncertainties in Flaw Number and Size</u>

Section 3.2 of the TR indicated that surface-breaking and embedded flaws were used in the PFM analysis. Since embedded flaws do not grow significantly due to fatigue, they were not evaluated as part of the fatigue growth analysis. To simulate embedded flaws in welds and plates, the PWROG pilot plant studies for the RV ISI interval extension used the embedded flaw distribution for welds and plates from the PTS Risk Study.

Surface-breaking flaws were assumed to grow by fatigue as a result of normal operating conditions. A discussion of the initial size and distribution of the assumed surface-breaking flaws was provided by the PWROG in response to RAI 1 from Reference 3. The PWROG indicated that the initial size and distribution of the surface flaws were consistent with the size and distribution developed by Pacific Northwest National Laboratory (PNNL) for use in the PTS Risk Study. The initial size and distribution of surface-breaking flaws utilized the computer code VFLA W03, which was developed by PNNL and is described in NUREG/CR-6817, Revision 1, “A Generalized Procedure for Generating Flaw-Related Inputs for the FAVOR Code” (Reference 14). The initial surface-breaking flaw size and distribution were input into a fatigue crack growth and ISI analysis to determine a surface flaw density file after any ISI. Surface flaw density files were created to simulate two inspection routines. The first case simulated inspections performed on a 10-year interval as currently required by the ASME Code. The second case simulated a single inspection performed after the first 10 years of operation with no subsequent inspection. These surface-breaking flaw density files are then input into the PFM analysis as surface-breaking flaw density files. Since the characterization of embedded flaws in plates and welds and the initial surface-breaking flaw size for the fatigue analysis used distributions that were used in the PTS Risk Study, they are applicable for use in RV ISI interval extension analyses.

In Attachment 1 to Reference 2, the PWROG indicated that underclad cracks in forgings are so shallow that the probability of them growing through-wall during a severe PTS transient would be small. NUREG-1874 indicated that for severe PTS transients, the TWCF for forgings with underclad cracks could be greater than those for axial welds, plates, and forgings without underclad cracks. In its response to RAI 2 from Reference 3, the PWROG provided an analysis of the TWCF for axial welds, plates, forgings without underclad cracks, and forgings with underclad cracks. The analysis, which used correlations from NUREG-1874, indicated forgings with underclad cracks have a higher TWCF than welds, plates, and forgings without underclad cracks when the $RT_{MAX\text{-}FO}$ [1] is greater than 240 $^{\circ}$F.

Table 3.4 in NUREG-1874 indicated that the highest $RT_{MAX\text{-}FO}$ for a PWR RV ring forging is 187.3 $^{\circ}$F at 32 EFPY and 198.6 $^{\circ}$F at 48 EFPY. Therefore, it is unlikely that the $RT_{MAX\text{-}FO}$ value for any domestic PWR will ever exceed 240 $^{\circ}$F and the TWCF value for all such forgings will remain below that for axial welds with equivalent reference temperatures. The PWROG indicated that the analyses performed in the TR would not be applicable without further evaluation for RVs with $RT_{MAX\text{-}FO}$ values exceeding 240 $^{\circ}$F.

Fatigue Crack Growth Analysis

Section 3.2 of the TR indicated 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 ASME Code, Section XI, Appendix A. The probabilistic representation was consistent with those used in the pc-PRAISE computer code and NRC-approved structural reliability and risk assessment (SRRA) tool for piping risk-informed ISI. In Appendix A of the NRC staff SE on WCAP-14572, Revision 1, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report" (Reference 15), the NRC staff concluded that the SRRA tool addresses fatigue crack growth in an acceptable manner since it is consistent with the technical approach used by other state-of-the-art PFM computer codes. The NRC 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 defects in welds. As discussed in the preceding section of this SE, the size and distribution of preexisting surface-breaking fabrication flaws was consistent with the size and distribution developed by PNNL for use in the PTS Risk Study.

Design basis transients for the pilot plants were reviewed and the PWROG determined that the greatest contributor to fatigue crack growth for surface-breaking flaws initiating from the inside surface of the RV for the pilot plants is the RV heat-up and cool-down transient. This transient represents a full heat-up and cool-down cycle between atmospheric pressure at room temperature and full-system pressure at 100-percent power operating temperature. This transient envelopes many transients with smaller ranges of conditions. For the pilot plant evaluations, 7 heat-up and cool-down cycles per year were used for the Westinghouse-designed plant, BV1; 13 heat-up and cool-down cycles were used for the Combustion Engineering (CE)-designed plant, Palisades; and 12 heat-up and cool-down cycles were used for the Babcock and Wilcox (B&W)-designed plant, OC1, to bound all the design basis transients for the respective PWR plant designs in each fleet.

In response to RAI 1 from Reference 3, the PWROG provided a description of the analyses performed to determine whether the seven heat-up and cool-down cycles per year for Westinghouse plants and the 13 heat-up and cool-down cycles per year for CE plants bound all the design basis transients for the respective PWR Nuclear Steam Supply System (NSSS) designs in each fleet. For Westinghouse plants, previous fatigue crack growth analyses of flaws

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[1] $RT_{MAX\text{-}FO}$ means the material property, which characterizes the RV's resistance to fracture initiation from flaws in forgings that are not associated with welds in the forgings. $RT_{MAX\text{-}FO}$ value is calculated under the provisions of Sections (f) and (g) of 10 CFR 50.61a, "Alternative fracture toughness requirements for protection against pressurized thermal shock" (Reference 6).

on the inside surface of the RV had shown that only four transients result in measurable crack growth. Sensitivity studies for the four contributing transients were performed. These analyses indicated that the only design transient that resulted in significant crack growth was the cool-down transient. The design basis for the Westinghouse plant was based on five cool-down cycles per year. An additional two cycles per year were added to the analysis to envelope the contribution of the other three transients, which contributed to measurable fatigue crack growth.

Previous fatigue growth studies were not available for CE-designed plants. Therefore, all design transients were evaluated in the CE transient fatigue crack growth sensitivity study. This study indicated that the cool-down transient produced the largest amount of fatigue growth for a RV inside surface flaw. The loss of secondary pressure transient also produced measurable growth. Assuming 12 cool-down cycles per year was considered conservative in comparison to the actual number of cool-downs a plant might experience in a given year based on plant operating experience. One additional cool-down cycle was added to the analysis to envelope the contribution to fatigue crack growth of the loss of secondary pressure transient.

Based on the results of the fatigue crack growth sensitivity studies, the number of cool-down transients assumed for the Westinghouse and CE-designed pilot plants will envelope the fatigue crack growth from all Westinghouse and CE NSSS design transients. All RVs are inspected before operation providing confidence that there are no large flaws throughout the RV that have a high likelihood of failure given a PTS event. Only surface-breaking flaws are assumed to grow from fatigue crack growth.

Fatigue crack growth sensitivity studies were not performed to determine the effect of B&W design transients for fatigue crack growth in B&W designed plants. Therefore, any B&W plant licensee using the results of the TR to extend the RV ISI interval from 10 to 20 years, including the pilot plant, must demonstrate that the assumption of 12 heat-up/cool-down transients per year in the TR analysis bounds the fatigue crack growth for all design basis transients for that unit.

For the purpose of the pilot plant studies in the TR, an 80-year life for fatigue crack growth was used. This 80-year life envelopes plants seeking to obtain license extensions to 60 years and provides an additional margin of conservatism. This resulted in a total of 560 heat-up/cool-down transients for the Westinghouse-designed unit, 1040 heat-up/cool-down transients for the CE-designed unit, and 960 heat-up/cool-down transients for the B&W-designed unit. The PWROG indicated that most plants operational histories indicate that they will not reach this number of design transients by end of 80 years of operation. Hence, this calculation was performed as a bounding analysis based on actual plant operating histories.

In response to RAI 1 from Reference 3, the PWROG indicated that the fatigue crack growth rates that are used in the fatigue crack growth analysis are taken from Section 4.2.2 of NUREG/CR-5864, "Theoretical and Users Manual for pc-PRAISE" (Reference 16). 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 Code, provides 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 Code have not changed since they were originally included in the 1978 Edition of Section XI. Since the crack growth rate code used in the PWROG analysis was taken directly from a code that was previously reviewed and approved by the NRC staff in Reference 15 and is based on the ASME Code crack growth rate curves, the crack growth rate code used in the

PWROG analysis is acceptable.

Effectiveness of ISI

To determine the impact of different inspection intervals on the frequency of RV failure, the effectiveness of the ISI must be considered. The PWROG considered the impact of the probability of detection (POD) of flaws when ultrasonic inspection is performed on the RV welds and adjacent base metal. The basis for the POD used in the pilot plant studies for the RV ISI interval extension was taken from studies performed at the Electric Power Research Institute (EPRI) Nondestructive Examination (NDE) Center on the detection and sizing qualification of ISIs of the RV beltline welds (Reference 17). Figure 3-4 in the TR illustrates the POD as a function of flaw size. The POD ranges from 0.5 for very small flaws up to 0.9 and greater for flaws with through-wall depths greater than 0.25 inches.

For the pilot plant evaluations, ultrasonic examinations were assumed to be conducted in accordance with ASME Code, Section XI, Appendix VIII. Flaws that were detected were assumed to be repaired with the repaired area returned to a flaw-free condition. If the quality of inspection is not as good as assumed or the quality of the repair is less than 100 percent, then the result would be fewer flaws found and fewer flaws removed during repair, resulting in less difference in risk from one inspection interval to another. The POD values used in the analysis were relatively high and, therefore, the pilot plant studies conservatively calculated a larger potential difference in risk by maximizing the benefits of inspection.

Material Fracture Toughness and Neutron Embrittlement

The RV material properties for each of the pilot plant studies used plant-specific properties that are identified in Appendices B, F, and J in the TR. These material properties are input to the Fracture Analysis of Vessels – Oak Ridge (FAVOR) Code (Reference 18). The FAVOR code, which was developed by Oak Ridge National Laboratory (ORNL) to perform PFM analyses for the PTS Risk Studies, includes fracture toughness models which are based on extended databases of empirically obtained plane strain fracture toughness (K_{Ic}) and crack arrest fracture toughness (K_{Ia}) data points and include the effects of statistical bias for direct measurement of fracture toughness.

The input to the FAVOR code includes plant-specific neutron fluence maps for each of the pilot plants. For the pilot plant evaluations in the TR, the input neutron fluence distributions were taken directly from the PTS Risk Study. A series of neutron transport calculations were performed for the PTS Risk Study to determine the neutron fluence on the inner wall of the pilot plant RVs. The modeling procedures were based on the guidance contained in RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” (Reference 19). The models incorporated pilot plant-specific geometry and operating data. The neutron fluence for energies greater than one million electron volts (E > 1 MeV) was calculated as a function of the azimuthal and axial location on the inner wall of the RV. The neutron fluence was extrapolated from the current state point to various EFPY of operation assuming a linear extrapolation of the most recent operating cycles.

The neutron fluence values used in the RV ISI interval extension evaluations were for 60 EFPY for BV1 and Palisades and were for 500 EFPY for OC1. 500 EFPY were used for OC1 rather than 60 EFPY to envelope license extension consideration and because it is recognized that OC1 is not the most radiation sensitive RV in the B&W fleet. The use of 500 EFPY for OC1 should bound the embrittlement of the most highly embrittled RV in the B&W fleet.

Accident Transients

PTS events are viewed as providing the greatest challenge to PWR RV structural integrity. If a RV had an existing flaw of critical size and certain PTS transients were to occur, this flaw could rapidly propagate through the RV wall, resulting in a through-wall crack and challenging the integrity of the RV. The PTS Risk Study utilized plant-specific probabilistic risk assessment (PRA) models to determine the possible sequences, which could result in a PTS event for each of the pilot plants. Due to the large number of sequences which were identified, it was necessary to group (i.e., bin) sequences with like characteristics into representative transients (PTS transients) that are analyzed using thermal-hydraulic (TH) codes.
TH analyses were performed for each PTS transient to develop time histories of temperature, pressure, and heat transfer coefficients. These histories were then input into the FAVOR code where they were used during the calculation of the conditional probability of RV failure for each PTS transient. From this analysis, it was determined that only a portion of the PTS transients contributes to the total risk of RV failure, while the remaining transients have an insignificant or zero contribution. The transients, which were identified to be contributors to PTS risk, were then used for the PFM analysis in the PTS study and for the pilot plant studies in the TR.

Stresses Resulting from PTS Transients, Cladding, and Welding

For each PTS transient, deterministic calculations were performed to produce a load definition input file that includes time-dependent, through-wall temperature profiles, through-wall circumferential and axial stress profiles, and stress intensity factors for a range of axially and circumferentially-oriented embedded and inner surface-breaking flaw geometries. This load definition file was input into the FAVOR code to produce the conditional probability of failure (CPF) (i.e., the conditional probability of a through-wall crack) for each PTS transient. These probabilities estimated by the FAVOR code (complete with uncertainties) are conditional in the sense that, within the FAVOR code probabilistic fracture mechanics module, the TH transients are assumed to occur.

In addition to the stress resulting from PTS transients, the PWROG analysis included the impact of cladding and residual stresses on the probability of failure. The pilot plant studies for RV ISI interval extension used a residual weld stress distribution through the wall that was taken from the PTS Risk Study and is described in the FAVOR Code Theory Manual (Reference 18). The cladding stress used in the pilot plant studies was taken from the PTS Risk Study. The cladding temperature dependence due to differential thermal expansion was based on a stress free temperature of 488 ^{o}F, which is consistent with that used in the PTS Risk Study.

Staff Evaluation of Engineering Considerations in PFM Analysis

The material fracture toughness, neutron embrittlement, distribution and uncertainties in embedded and surface-breaking flaws, accident transients, frequency of transients, and stress resulting from PTS transients, cladding, and welding used in the PWROG ISI interval extension study are acceptable because the values and methodologies were derived from the PTS Risk Study. The fatigue crack growth analysis used in the PWROG ISI interval extension study is acceptable because it was performed using a code approved by the NRC and has considered all sources of fatigue stress and the probability for preexisting fabrication flaws. The effectiveness of ISI has been adequately determined because it used data from studies performed at the EPRI NDE Center on the detection and sizing qualification of ISIs of RV beltline welds. Based on the above conclusions, the NRC staff considers that the PWROG has adequately considered the engineering variables in determining the risk of RV failure in its ISI interval extension study.

The PWROG has identified two items that must be further evaluated. They are:

1) Licensees for B&W plants using the results of TR WCAP-16168-NP, Revision 2, to extend the RV ISI interval from 10 to 20 years (including the pilot plant) must demonstrate that the assumption of 12 heat-up/cool-down transients per year in the TR analysis bounds the fatigue crack growth for all design basis transients for that unit.

2) RVs with RT_{MAX-FO} values exceeding 240 °F require further evaluation because the analyses performed in TR WCAP-16168-NP, Revision 2, are not applicable.

3.2.2 Probabilistic Risk Assessment

PTS events were viewed as providing the greatest challenge to PWR RV structural integrity and, therefore, the PRA had to estimate the frequency and severity of PTS transients. PTS transients are not normally modeled in PRAs and the analyses of the pilot plants in the TR used the PTS transients and frequencies from the PTS Risk Study. As part of the PTS Risk Study, PRA models were developed by the NRC staff for each of the three pilot plants using plant-specific information (References 20-22). These three units included one unit from each of the PWR vendors. These PRA models included an event tree analysis that defined the sequences of events that are likely to produce a PTS challenge to RV structural integrity for each of the pilot plants. As discussed above, individual event tree sequences with like characteristics were binned into representative PTS transients.

The results of the PRA in the PTS Risk Study included descriptions of each PTS transient from which the TH characteristics of each transient can be developed, and estimates of the frequency with which each transient was expected to occur. The final transient frequency estimates were distributions (histograms) which represented the combined frequency, including uncertainties, of all the event tree sequences incorporated into each bin. Appendices D, H, and L in the TR briefly described the failures and the mean estimated frequency for each bin for each of the three pilot plants.

The transient frequencies were input into the FAVPOST module, the final module in the FAVOR code. This module combined the conditional initiation and through-wall cracking probabilities through a matrix multiplication with the frequency histograms for each PTS transient provided by the PRA analyses.

3.2.2.1 Estimating the Risk Associated with Extending the RV Weld Inspection Interval from 10 to 20 Years

The likelihood of RV failure was postulated to increase with increasing time of operation due to the growth of pre-existing fabrication flaws by fatigue in combination with a decrease in RV toughness due to irradiation. The PFM approach in the TR simulated the growth of flaws over time and the repair of flaws that are detected during a periodic ISI. The largest cracks were expected to exist at the end of the plant's operating life because, even with periodic inspection, flaws may be missed during an inspection. These flaws would remain in service and grow until eventually detected by ISI, causing RV failure during a PTS event, or the end of plant life is reached. The end of operating life is also the time when the RV will be most embrittled and most subject to failure for any size crack.

Therefore, instead of assuming that PTS transients can occur randomly during the operating life, the PWROG's response to RAI 9 from Reference 3 explained that the TR conservatively estimated the CPF for each PTS transient by applying the PTS loadings to the material properties and the distribution of flaws sizes expected to exist on the first day of full power operation following the refueling outage after the last operating year of the extended license of the plant. The NRC staff concurred that this process approximates the greatest CPF expected to exist during the life of the plant. The PTS transients' frequencies were not expected to change over the plant life so the product of these frequencies with the maximum CPF is acceptable because it results in a bounding estimate for the TWCF and associated increase in risk.

The current inspection interval is 10 years and the base case scenario for the change in risk analysis is one inspection every 10 years. Rather than evaluate each plants' specific inspection cycle, the TR bounded the impact of extending the interval by estimating the risk increase as the difference between the base case risk (assuming that the RV was inspected every 10 years) and the risk assuming that a plant only had one inspection after the first 10 years and then was never inspected again for the remaining life of the plant. Plant life was assumed to be 80 years, for both the base case (every 10-year inspection) and the bounding case (only one inspection). The NRC staff concurred that this evaluation is applicable to all plants and the change in risk estimated for this scenario will bound the change expected by extending the 10-year interval to a 20-year interval.

The TR assumed that a through-wall crack will lead to core damage and that core damage will lead to a large early release. The RG 1.174 guideline addressing an acceptable increase in large early release frequency (LERF) is the smallest guideline value. Requiring that the TWCF is less than the LERF guideline ensured that both the core damage frequency (CDF) and LERF guidelines are met. The equation in FAVPOST that was used to estimate risk with and without periodic inspection for plant j is:

$LERF_j = CDF_j = TWCF_j = \Sigma\ IE_{ji} * CPF_{ji}$

where,

IE_{ji} is the initiating event frequency (events per year) for each of the i representative PTS transients for plant j developed during the PTS Risk Study. The PTS Risk Study developed full distributions for the frequency of each PTS transient bin and the TR used the full distribution.[2] IE_{ji} does not change when the inspection period changes.

CPF_{ji} is the conditional probability of RV failure (conservatively assumed to occur if a through-wall crack develops) given the TH characteristics of each of the i representative PTS transients for plant j. As described above, the RV material properties and the distribution of flaw sizes are those expected to exist at the end of plant j's operating life. The distribution of flaw sizes is the parameter that changes when the inspection period changes and, therefore, CPF_{ji} changes when the inspection period changes.

The NRC staff concurs that the PRA models of PTS transient frequency, the IE_{ji} and CPF_{ji} parameters, and the above equation appropriately capture the significant contributors to risk from RV failure and, therefore, fulfill the RG 1.174 guidance that the analysis is capable of modeling the impact of the proposed change. The NRC staff also concurs that the bounding estimates from only one inspection versus an inspection every 10 years appropriately envelops the impact of the proposed change for any facility regardless of its inspections schedule and history.

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[2] Appendices D, H, and L include only the mean frequency estimates from the PTS transient bins, but the calculations illustrated in Appendices E, I, and M are performed using the full initiating event frequency distributions.

ISI is directed toward identifying surface-breaking and embedded flaws that have grown large enough to require repair. In the response to RAI 12a from Reference 3, the PWROG noted that the frequency of surface-breaking flaws should be very small because none had ever been discovered during either pre-service or in-service examinations of beltline welds. With few such flaws, few failures were observed from the simulations even when fatigue crack growth was included. With few failures, it was difficult to obtain a converged solution using Monte Carlo simulation in the FAVOR code because its precision is based upon the number of failures in the total number of simulations. In order to obtain a converged solution, the dominant contribution to TWCF from embedded flaws was included[3] in the simulations. The result of including the dominant contribution from embedded flaws in the simulation was that direct comparison of the mean TWCF with only one inspection and the mean TWCF with inspections every 10 years did not produce a stable metric. This is illustrated by, for example, the results in Table 4-1 in the TR which reported that the estimated TWCF for BV1 with only one inspection (5.04E-9/year) was smaller than the TWCF with one inspection every 10 years (5.23E-9/year) although the more frequent inspection program should result in a smaller TWCF.

In the response to RAI 12b from Reference 3, the PWROG reported on a sensitivity study that was performed by running the Monte Carlo simulation without the embedded flaws. The PWROG reported that the number of FAVOR simulations was increased from 70,000 to 500,000 but that no failures were obtained for both the only one inspection and the inspection every
10 years simulations. The PWROG noted that excluding embedded flaws results in a zero TWCF for both inspection intervals and, therefore, a zero increase in TWCF given the proposed interval extension.

Because of the uncertainty in how accurately an insignificant (null) effect can be calculated using standard Monte Carlo simulation, the PWROG included embedded flaws and estimated the change in risk by subtracting the lower bound mean estimate for one inspection every 10 years from the upper bound mean estimate for only one inspection. The PWROG argued that this difference represents the maximum statistically calculated value for the potential change in risk at a number of RV simulations for which the Monte Carlo statistical analysis has reached a stable solution. In its response to RAI 12c from Reference 3, the PWROG described the derivation of the standard error on the mean, which was used to calculate the upper, and lower bound estimates. The standard error is a statistical estimate reflecting how much sampling fluctuation was observed which could be used to estimate confidence intervals about the mean estimate. The PWROG chose to use two times the standard error to develop its confidence bounds. Therefore, if repetitive simulations (each with 70,000 trials) were performed, it is expect that in only 2.5 percent of the mean estimates would exceed the upper bound value and 2.5 percent would be less than the lower bound value.

The NRC staff concluded that the analyses described in the TR provided a reasonable or bounding estimate of the increase in risk associated with extending the inspection interval for RV welds from 10 to 20 years. As discussed above, the NRC staff based this conclusion on:

- the PRA models of PTS transient frequency, the IE_{ji} and CPF_{ji} parameters, and the equation used to calculate the risk from PTS events appropriately capturing the

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[3] The NRC staff concluded during the PTS Risk Study, that embedded flaws do not grow over time and therefore their contribution to TWCF is driven by the initial flaw distribution and is unaffected by the ISI interval.

significant contributors to risk from RV failure,

- the bounding estimates from only one inspection versus an inspection every 10 years appropriately modeling the impact of the proposed change for any facility regardless of its RV inspections schedule and history,

- the TWCF from surface-breaking flaws being so small that the Monte Carlo estimation techniques in the FAVOR code do not converge to a stable solution indicating that the TWCF from surface-breaking flaws is small regardless of the inspection program interval, and

- the subtraction of the lower bound mean estimate for one inspection every ten years from the upper bound mean estimate for only one inspection being consistent with the guidance in RG 1.174 that the difference in the means (in this case confidence estimates on the means) is the risk metric that should be compared with the acceptance guidelines.

3.2.2.2 Evaluation of PRA Technical Adequacy

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 proposed change. The PWROG used the PTS transient frequencies developed in the PTS Risk Study in its analysis. The TR conservatively assumed that core damage and large early release would inevitably follow a PTS transient that results in a through wall crack. Therefore, there is no PRA event and sequence modeling needed beyond the determination of the PTS transient frequencies.

The NRC staff developed plant-specific PRA analyses to estimate the PTS transient frequencies for each of the three pilot plants using a process described in detail in NUREG/CR-6859, "PRA Procedures and Uncertainty for PTS Analysis" (Reference 23). The analyses were described in detail in the plant-specific PRA reports and summarized in Chapter 5 of the PTS Risk Study. The process included a review of the PRA analyses performed during the 1980s in support of the first PTS rule and a search of licensee event reports for the years 1980 through 2000 to gain an understanding of the frequency and severity of observed overcooling events. The PRA analyses used realistic input values and models and an explicit treatment of uncertainties. Best estimate equipment failure values were used throughout based on generic nuclear industry data or, in cases where it was available, on plant-specific data. Parameters related to human performance were based on review of plant-specific procedures and training, observation of plant personnel responding to PTS-related sequences on their simulator, and performance data from actual plant operations. The scope of the study covered all event sequences in the range from zero power hot stand-by up to 100 percent power.

As discussed in the individual pilot plants' PRA reports, all analyses were conducted through plant visits and by numerous interactions (vocal, written, and e-mail exchanges) with each licensee as the analysis evolved. During a first site visit, the PTS study team collected information. After preliminary results were completed, reviews were performed both by licensee and by NRC project staff during a second site visit. The OC1 and BV1 models used system level fault trees and system level failure data. The Palisades model used detailed system level fault trees from the licensee's PRA. Formal reviews were carried out for OC1 and BV1. Palisades' models were developed by the licensee and reviewed by the NRC staff.

A final peer review was carried out by a panel of six experts to provide an independent review of the technical basis developed for the PTS Rulemaking. The results of this peer review are documented in Reference 8. The objective of the peer review was to assess the adequacy and reasonableness of the technical basis to support the alternate PTS rule. The peer reviewers focused on different parts of the PTS analysis. Comments related to the PRA aspects generally concluded that the work was well founded and reasonable and no serious weaknesses were identified.

Based on the PTS Risk Study's detailed review of past studies and operating experience, extensive interactions between the analysis team and the plant personnel at all units, and the opportunity for the same team to benefit from the multiple plant study insights while performing all the analyses, the NRC has confidence that the PTS transient frequency results from the PRA analyses in the PTS Risk Study are sufficiently well developed to be able to demonstrate that the change in risk estimates as developed in the TR does not exceed the acceptance guidelines in RG 1.174.

3.2.2.3 Generic Applicability and External Events

During the development of the PTS Risk Study, the NRC staff investigated the applicability of the results from the three pilot plants to the operating fleet of PWRs. These three units included one unit from each of the three PWR vendors. This investigation examined plant design and operational characteristics of five additional plants as described in Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," (Reference 24). The overall approach was to compare potentially important design and operational features (as related to PTS) of the other PWRs to the same features of the pilot plants to determine the extent these features are similar or different.

In Reference 6, the NRC staff reported its conclusion that the results from the PTS Risk Study can be applied to the entire fleet of operating PWRs. This conclusion was based on an understanding of characteristics of the dominant transients that drive their risk significance. The generic evaluation revealed no design, operational, training, or procedural factors that could credibly increase the severity of these transients or the frequency of their occurrence in the general PWR population above the severity/frequency characteristics of the three plants that were modeled in detail. As applied to the analyses included in the TR, this conclusion indicated that the PTS transient frequencies and TH characteristics used to estimate the change in risk are dependent only on the reactor vendor and are generally applicable to all PWRs from that vendor.

The detailed plant-specific PRAs in the PTS Risk Study evaluated the contribution of internal initiating events to TWCF. The study group also evaluated the potential contribution of external initiating events to PTS risk as described in Letter Report, "Estimate of External Events Contribution to Pressurized Thermal Shock (PTS) Risk" (Reference 25) and summarized in Section 9.4 of the PTS Risk Study. The external events included in the evaluation were fires, floods, high winds and tornados, and seismic events. This analysis was structured by identifying three broad types of overcooling scenarios and making conservative judgments with regard to the type and frequency of external events that could directly contribute to causing each overcooling scenario. The conservative judgments were directed toward bounding the PTS TWCF contributions attributable to external events for the worst situation that might arise at virtually any plant. The study's results indicated that the bounding total external event TWCF is approximately 2E-8/year, quantitatively comparable to the highest internal events contribution of 2E-8/year. The study concluded that there was considerable assurance that the external event contribution to the overall TWCF because of external event initiated PTS events is at least no greater than the highest best estimate contribution from internal events.

Based on the results of the PTS Generalization Study, the NRC staff has concluded that the PTS transient characteristics (both frequency and TH characteristics) are generically applicable for all similar plants (i.e., plants from the same vendor) in the fleet. Based on the results of the external events analyses, the NRC staff has also concluded that the contribution of external events to the change in risk has been adequately evaluated and that the contribution to risk from external events is equal or less than the contribution for internal events.

3.2.2.4.1 Comparison with RG 1.174 Acceptance Guidelines

The results of the change in risk analyses were summarized in Table 4-1 in the TR where the bounding increases in risk were reported as 9.37E-10/year, 1.81E-8/year, and 1.26E-8/year for BV1 (Westinghouse-designed plant), Palisades (CE-designed plant), and OC1 (B&W-designed plant), respectively. These increases are well below the guideline for a very small increase in LERF of 1E-7/year in RG 1.174.

The TR only incorporated the internal events PTS sequence frequency results from the PTS rulemaking into its change in risk analysis. The largest increase in LERF was estimated as 1.8E-8/year for the Palisades plant. The NRC staff's evaluation of external event contributions to PTS risk determined that the total PTS risk would, at most, double compared to the risk from internal events when the risk from external events are included. Since the total risk for the base case and the only one inspection case would both double, the total change in risk would also double. The NRC staff concluded that the greatest change in risk associated with extending the inspection interval at any PWR using the methods and guidelines described in the TR and endorsed in this revised final SE is less than 5E-8/year. The NRC staff finds that this increase is small and consistent with the intent of the Commission's safety goals.

3.3 Implementation and Monitoring

The third element in the RG 1.174 approach is to develop an implementation and monitoring program to ensure that no adverse safety degradation occurs because of the proposed changes. Therefore, an implementation and monitoring plan should be developed to ensure that the engineering evaluation conducted to examine the impact of the proposed changes continues to be valid after the change has been implemented. This will ensure that the conclusions that have been drawn from the evaluation remain valid.

RV integrity depends upon licensees ensuring that the critical elements of the PFM analysis described in the TR are valid. Licensees must monitor the number of cycles of transients that could affect the fatigue crack growth analysis, the change in fracture toughness of the limiting RV material due to exposure to radiation, and the flaw distribution in the RV welds and adjacent base metal.

The number of transient cycles that were utilized in the fatigue crack growth analysis was discussed in Section 3.2.1 of this revised final SE. The PWROG used 7 heat-up and cool-down cycles per year for Westinghouse-designed plants, 13 heat-up and cool-down cycles per year for CE-designed plants, and 12 heat-up and cool-down cycles per year for B&W-designed plants. The design basis for the Westinghouse plant was 5 cool-down cycles per year. Although it was determined that 3 other transients did not significantly contribute to fatigue crack growth in RV welds, an additional 2 cycles were conservatively added to envelope the

contribution of these
3 transients. Since the PWROG fatigue crack growth analysis for Westinghouse NSSS designed plants determined that the only design basis transient that resulted in significant crack growth was the cool-down transient, it is the only design basis transient that needs to be monitored. Since the PWROG fatigue crack growth analysis of CE NSSS designed plants determined that the amount of crack growth from 13 cool-down transients bounds the expected crack growth from both cool-down and loss of secondary pressure transients, CE plants should monitor the number of cool-down transients. Fatigue crack growth sensitivity studies were not performed to determine the effect of B&W design transient for fatigue crack growth in B&W designed plants. Therefore, any B&W plant using the results of the TR to extend the RV ISI interval from 10 to 20 years (including the pilot plants), must determine the design basis transients that contribute to significant crack growth in RV welds. These transients must be monitored by the licensee.

Material fracture toughness was discussed in Section 3.2.1 of this revised final SE and must be monitored by determining whether the 95th percentile $TWCF_{TOTAL}$[4] for the plant requesting to implement the pilot plant study is less than the 95th percentile $TWCF_{TOTAL}$ from the pilot plant study. The 95th percentile $TWCF_{TOTAL}$ was calculated based on the material property indexing parameter $RT_{MAX-X.}$[5] Appendix A in the TR identifies the 95th percentile $TWCF_{TOTAL}$ from the pilot plant studies for BV1, Palisades, and OC1. The 95th percentile $TWCF_{TOTAL}$ value calculated for BV1 at 60 EFPY was 1.76E-08 events per year. The 95th percentile $TWCF_{TOTAL}$ value calculated for Palisades at 60 EFPY was 3.16E-07 events per year. The 95th percentile $TWCF_{TOTAL}$ value calculated for OC1 at 500 EFPY was 4.42E-07 events per year.

The flaw distributions used in the PWROG PFM analyses are described in Section 3.2.1 of this revised final SE. The PWROG utilized the flaw sizes and distributions in the PTS Risk Study to simulate embedded flaws in welds, forgings, and plates and to simulate the initial size and distribution of surface-breaking flaws. Section (e) of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock," in Enclosure 1 of Reference 6 describes the allowable flaw distribution for embedded flaws and surface-breaking flaws that would be permitted for RVs that are at the PTS screening limits in 10 CFR 50.61a. By monitoring flaw sizes in accordance with the criteria described in Section (e) of 10 CFR 50.61a, licensees will ensure that their RVs do not have flaws that invalidate the results of the PWROG PFM analyses. For the first interval extension application, the applicant will not use Section (e) of the 10 CFR 50.61a rule, however, for subsequent interval extensions the applicant will be held to Section (e) of the 10 CFR 50.61a rule.

The NRC staff concludes that the implementation and monitoring described above will ensure that the conclusions that have been drawn from the evaluation remain valid.

3.4 Submit Proposed Change

The fourth and final element in RG 1.174 approach is the development and submittal of the

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[4] The 95 percentile $TWCF_{TOTAL}$ is the sum of the 95 percentile TWCF for all beltline materials. It is calculated in accordance with NUREG-1874.

[5] RT_{MAX-X} values are determined for each beltline material. RT_{MAX-X} is a material property, which characterizes the RVs resistance to fracture initiating from flaws in welds, plates, and forgings. The method of determining RT_{MAX-X} is described in Sections (f) and (g) of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock," in Reference 6.

proposed change to the NRC. Since the 10-year ISI interval is required by Section XI, IWB-2412, as codified in 10 CFR 50.55a, a relief for an alternative, in accordance 10 CFR 50.55a(a)(3)(i), must be submitted and approved by the NRC to extend the ISI interval. Licensees that submit a request for an alternative based on the TR need to submit the following plant-specific information:

1) Licensees must demonstrate that the embrittlement of their RV is within the envelope used in the supporting analyses. Licensees must provide the 95th percentile $TWCF_{TOTAL}$ and its supporting material properties at the end of the period in which the relief is requested to extend the inspection interval from 10 to 20 years. The 95th percentile $TWCF_{TOTAL}$ must be calculated using the methodology in NUREG-1874. The RT_{MAX-X} and the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level, ΔT_{30}, must be calculated using the methodology documented in the latest revision of RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," or other NRC-approved methodology. The PWROG response to RAI 3 from Reference 3 and Appendix A in the TR identifies the information that is to be submitted.

2) Licensees must report whether the frequency of the limiting design basis transients during prior plant operation are less than the frequency of the design basis transients identified in the PWROG fatigue analysis that are considered to significantly contribute to fatigue crack growth.

3) Licensees must report the results of prior ISI of RV welds and the proposed schedule for the next 20-year ISI interval. The 20-year inspection interval is a maximum interval. In its request for an alternative, each licensee shall identify the years in which future inspections will be performed. The dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan provided to the NRC in PWROG letter OG-10-238 (Reference 11).

4) Licensees with B&W plants must (a) verify that the fatigue crack growth of 12 heat-up/cool-down transients per year that was used in the PWROG fatigue analysis bound the fatigue crack growth for all of its design basis transients and (b) identify the design bases transients that contribute to significant fatigue crack growth.

5) Licensees with RVs having forgings that are susceptible to underclad cracking and with RT_{MAX-FO} values exceeding 240 °F must submit a plant-specific evaluation to extend the inspection interval for ASME Code, Section XI, Category B-A and B-D RV welds from 10 to a maximum of 20 years because the analyses performed in the TR are not applicable.

6) Licensees seeking second or additional interval extensions shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a.

3.5 Conformance to RG 1.174

In addition to the four-element approach discussed above, RG 1.174 states that risk-informed plant changes are expected to meet a set of key principles. This section summarizes these principles and the NRC staff findings related to the conformance of the TR methodology with these principles.

Principle 1 states that the proposed change must meet the current regulations unless it is explicitly related to a requested exemption or rule change. ISI of ASME Code Class 1, 2, and 3

components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). This risk-informed application requires a request for an alternative under CFR 50.55a(a)(3)(i) which meets the current regulations and, therefore, satisfies Principle 1.
Principle 2 states that the proposed change shall be consistent with the defense-in-depth philosophy. In the response to RAI 11a from Reference 3, the PWROG argued that the proposed change is consistent with the defense-in-depth philosophy because there is no change in RV design and no change in the robustness of the RV or other systems at the plant. The NRC staff believes that ISI is an integral part of defense-in-depth and extending the interval may change the robustness of the RV, albeit very slightly. However, the extension of the inspection interval is accompanied by various evaluations and a monitoring program and the NRC staff concludes that, in total, the proposed ISI program provides reasonable assurance that RV integrity will be maintained consistent with the philosophy of defense-in-depth. Therefore, Principle 2 is met.

Principle 3 states that the proposed change shall maintain sufficient safety margins. Section 12 in PTS Risk Study concluded that the calculations demonstrate that PTS events are associated with an extremely small risk of RV failure, suggesting the existence of considerable safety margin. Section 4.3 in the TR clarified that no safety analysis margins are changed and, aside from extending the inspection interval, no portions of the current inspection requirements are eliminated. The NRC staff concurred that the proposed change maintains sufficient safety margins because the change simply extends the inspection interval and does not change, for example, the acceptance criteria used to determine whether any identified flaws are acceptable or need to be repaired. Therefore, Principle 3 is met.

Principle 4 states that when proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goals. The NRC staff concluded that the greatest increase in LERF associated with extending the inspection interval at any PWR using the methods and guidelines described in the TR and endorsed in this revised final SE is less than 5E-8/year. The NRC staff found that this increase is small and consistent with the intent of the Commission's Safety Goals. Therefore, Principle 4 is met.

Principle 5 states that the impact of the proposed change should be monitored using performance measurement strategies. As described in Section 3.3 of this revised final SE, licensees must monitor the number of cycles of transients that could affect the fatigue crack growth analysis, the fracture toughness of the limiting RV material, and the flaw distribution in the RV welds and adjacent base metal. The NRC staff found that the planned monitoring program provides confidence that no adverse safety degradation will occur because of the proposed changes and that the engineering evaluation conducted to examine the impact of the proposed changes will continue to be valid after the change has been implemented. Therefore, Principle 5 is met.

3.6 NRC Staff Findings

The NRC recently completed its rulemaking (resulting in the creation of 10 CFR 50.61a) which provided alternate regulations regarding the requirements for protection against PTS events. In support of this rulemaking, the NRC staff concluded that the risk of through-wall cracking caused by PTS events is much lower than previously estimated. The rule provides new PTS screening criteria that are selected based on an evaluation that indicated that, after applying these new, relaxed criteria, the risk of through-wall cracking due to a PTS event at any PWR would be less than 1E-6/year. Most PWRs are not expected to need the new screening criteria and, therefore, would have a TWCF less than, or substantially less than, 1E-6/year.

The analysis developed to support this TR uses mostly the same inputs and models used in the PTS Risk Study. The PTS Risk Study concluded that embedded flaws do not grow and, therefore, after the first inspection, periodic ISIs do not affect the risk from embedded cracks. Surface cracks that penetrate through the cladding and into the ferritic alloy steel were not part of the PTS Risk Study because these types of flaws have not been observed in the beltline of operating PWR reactors. PFM analyses indicate, however, that surface cracks can grow over time when subject to fatigue. The TR has analyzed the growth of postulated surface cracks because extending the RV inspection interval could increase the risk of RV failure from such cracks. The NRC staff has concluded that the TR has appropriately postulated and modeled the potential change in risk that could be caused by fatigue crack growth over the life of operating facilities.

Based on the results of the PTS Generalization Study, the NRC staff has concluded that the PTS transient characteristics (both frequency and TH characteristics) are generically applicable for plants from the same reactor vendor. RV embrittlement is, however, RV material, operating history, and age specific. Therefore, the NRC staff found that, while the PTS transient work need not be repeated by each plant seeking to extend its interval, the analyses and monitoring to demonstrate that the RV embrittlement is within the envelope used in the supporting analyses must be performed by each plant as described.

The NRC staff found that licensees implementing the ISI interval extension program documented in the TR and endorsed in the SE will have a program that meets the five key principles stated in RG 1.174 and, therefore, the proposed alternatives would provide an acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i).

Based on the above conclusions, the ASME Code, Section XI, ISI interval for examination categories B-A and B-D welds in PWR RVs can be extended from 10 years to a maximum of 20 years on an interval-by-interval basis. Since the 10 year ISI interval is required by Section XI, IWB-2412, as codified in 10 CFR 50.55a, a request for an alternative, in accordance 10 CFR 50.55a(g)(6)(i), must be submitted on a per-interval basis and approved by the NRC to extend any facility's ISI interval.

Should the licensee seek to extend additional intervals after the first extended interval the licensee must provide the information and analyses requested in Section (e) of 10 CFR 50.61a.

The methodology in the TR is applicable to all operating PWR plants by confirming the applicability of the parameters in Appendix A of the TR on a plant-specific basis. Licensees must submit a request for an alternative that contains all the information in Section 3.4 of this revised final SE. However, since the analysis documented in the TR used plant-specific data for BV1, Palisades, and OC1, these plants need not confirm the applicability of the parameters in Appendix A of the TR for the current license term. An expanded explanation of NRC staff expectations concerning such submittals is contained in Appendix B to this revised final SE.

The NRC staff will not repeat its review of the matters described in WCAP-16168-NP-A, Revision 2, as modified by this revised final SE, when the report appears as the basis in a request for an alternative, except to ensure that the material presented applies to the specific plant involved and the licensee has submitted all the information requested in Section 3.4 of this revised final SE.

4.0 CONDITIONS AND LIMITATIONS

The 20-year inspection interval is a maximum interval and will be granted on an interval-by-interval basis. In its request for an alternative, each licensee shall identify the years in which the future inspection will be performed. The date provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan provided to the NRC in PWROG letter OG-10-238 (Reference 11).

The methodology in the TR is applicable to all operating PWR plants by confirming the applicability of the parameters in Appendix A of the TR on a plant-specific basis. Licensees must submit a request for an alternative that contains all the information in Section 3.4 of this revised final SE. However, since the analysis documented in the TR used plant-specific data for BV1, Palisades, and OC1, these plants need not confirm the applicability of the parameters in Appendix A of the TR for the current license term. A more detailed explanation of NRC staff expectations is contained in Appendix B of this SE.

Licensees seeking to extend additional intervals after the first extended interval shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a.

5.0 CONCLUSION

The NRC staff has found that the methodology presented in WCAP-16168-NP-A, Revision 2, in concert with the guidance provided by RG 1.174, is acceptable for referencing in license amendment requests for PWR plants in accordance with the limitations and conditions in Section 4.0 of this revised final SE. The NRC staff will consider extending the RV weld inspection interval beyond 10 years based on plant-specific requests for an alternative that reference WCAP-16168-NP-A, Revision 2, on an interval-by-interval basis.

6.0 REFERENCES

1. Letter from F. P. Schiffley, Westinghouse Owners Group, “Transmittal of WCAP-16168-NP, Revision 1, ‘Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval’, MUHP-5097/5098/5099, Tasks 2008/2059,” January 26, 2006 (ADAMS Accession No. ML060330504).

2. Letter from F. P. Schiffley, PWR Owners Group, “Evaluation of NRC Questions on the Technical Bases for Revision of the PTS Rule Relative to Their Effects on the Risk Results in WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," June 8, 2006 (ADAMS Accession No. ML0616004311).

3. Letter from F. P. Schiffley, PWR Owners Group, “Responses to the NRC Request for Additional Information (RAI) on PWR Owners’ Group (PWROG) WCAP-16168-NP, Revision 1, ‘Risk-Informed Extension of Reactor vessel In-Service Inspection Interval’, MUHP-5097/5098/5099, Tasks 2008/2059,” October 16, 2007, and Enclosure 1, RAI responses (ADAMS Accession No. ML072920412). Enclosure 2, WCAP-16168-NP, Revision 2, ‘Risk-Informed Extension of Reactor vessel In-Service Inspection Interval’, October 2007 (ADAMS Accession No. ML072920413).

4. U.S. NRC, “Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report WCAP-16168-NP, Revision 2, “Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval” Pressurized Water Reactor Owners Group Project No. 694,” May 8, 2008 (ADAMS Accession No. ML081060045).

5. Letter from D.E. Buschbaum, PWR Owners Group, “Transmittal of NRC Approved Topical Report WCAP-16168-NP-A, Rev. 2, “Risk-Informed Extension of Reactor Vessel

In-Service Inspection Interval" (TAC No. MC9768 (MUHP 5097/5098/5099 Task 2008/2059, PA MSC-0120)," June 13, 2008 (ADAMS Accession No. ML082820046).

6. *Federal Register* Notice, (75 FR 13) "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," January 3, 2010 (ADAMS Accession No. ML092710550).

7. NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report," August 2007 (ADAMS Accession Nos. ML072830076 and ML072830081).

8. NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Appendices," August 2007 (ADAMS Accession No. ML07282069).

9. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)", 2007 (ADAMS Accession No. ML070860156).

10. Letter from D.E. Buschbaum, PWR Owners Group, OG-09-454 "Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Rev. 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." PA-MSC-0120," December 1, 2009 (ADAMS Accession No. ML093370133).

11. Letter from M.L. Arey, PWR Owners Group, OG-10-238 "Revision to the Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Rev. 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." PA-MSC-0120," July 12, 2010 (ADAMS Accession No. ML11153A033).

12. U.S. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (ADAMS Accession No. ML071700658).

13. U.S. NRC, Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002 (ADAMS Accession No. ML023240437).

14. NUREG/CR-6817, Revision 1, "A Generalized Procedure for Generating Flaw-Related Inputs for the FAVOR Code," October 31, 2003 (ADAMS Accession No. ML051790410).

15. WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," February 1999 (ADAMS Accession Nos. ML042610469 and ML042610375).

16. NUREG/CR-5864, "Theoretical and Users Manual for pc-PRAISE," July 1992 (ML082210247).

17. Electric Power Research Institute, "BWRVIP-108: BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," October 2002 (ADAMS Accession No. ML023360494).

18. Draft NUREG/CR (ORNL/TM-2007/0030), "Fracture Analysis of Vessels – Oak Ridge FAVOR v06.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlation," 2007 (ADAMS Accession No. ML063350491).

19. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (ADAMS Accession No. ML010890301).

20. Letter Report, "Beaver Valley Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA)," March 3, 2005 (ADAMS Accession No. ML042880454).

21. Letter Report, "Palisades Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA)," March 3, 2005 (ADAMS Accession No. ML042880473).

22. Letter Report, "Oconee Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA)," March 3, 2005 (ADAMS Accession No. ML042880452).

23. NUREG/CR-6859, "PRA Procedures and Uncertainty for PTS Analysis," October 6, 2004 (ADAMS Accession No. ML061580379).

24. Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004 (ADAMS Accession No. ML042880482).

25. Letter Report, "Estimate of External Events Contribution to Pressurized Thermal Shock (PTS) Risk," October 1, 2004 (ADAMS Accession No. ML042880476)

Attachments: Appendix A: Resolution of PWROG Comments on Draft SE
Appendix B: NRC Expanded Discussion of WCAP-16168-NP-A, Revision 2, Submittal Expectations
Appendix C: Resolution of PWROG Comments on Revised Draft SE for Topical Report WCAP-16168-NP-A, Revision 2

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Date: July 26, 2011

APPENDIX A

REVISED RESOLUTION OF PRESSURIZED WATER REACTOR OWNERS GROUP COMMENTS ON DRAFT SAFETY EVALUATION FOR TOPICAL REPORT WCAP-16168-NP, REVISION 2, "RISK-INFORMED EXTENSION OF THEREACTOR VESSEL IN-SERVICE INSPECTION INTERVAL"
(TAC NOs. MC9768 AND ME3495)

By letter dated March 31, 2008 (Agencywide Documents Access and Management System Accession No. ML080930300), the Pressurized Water Reactor Owners Group provided 13 comments on the draft safety evaluation (SE) for Topical Report WCAP-16168-NP, Revision 2. The following are the U.S. Nuclear Regulatory Commission (NRC) staff's resolution of these comments. To ensure consistency when discussing the final and proposed rule within the SE, the NRC staff has made one additional change, noted at Number 14. Certain responses have been updated as part of the revision of the WCAP-16168-NP-A SE.

1. Page 3, Lines 19-21

PWROG Comment:

It is stated in the draft SE that: "This correlation took into consideration the contribution to TWCF [through-wall cracking frequency] from each of the most limiting plate, axial weld, and circumferential welds." This correlation also took into consideration forgings. Therefore, the following change is suggested: "This correlation took into consideration the contribution to TWCF from each of the most limiting plate, forging, axial weld, and circumferential welds."

NRC Response:

The NRC staff agrees with this change.

2. Page 15, Line 41

PWROG Comment:

The change in risk (9.43E-10/year) for Beaver Valley Power Station Unit 1 (BV1) should be revised to 9.37E-10/year to be consistent with the value documented in WCAP-16168-NP, Revision 2, and the response to Request for Additional Information question number 8.

NRC Response:

The NRC staff agrees with this change.

3. Page 17, Lines 11-18

PWROG Comment:

The draft SE requires that the qualified vessel inspection results be evaluated per the existing requirements in Section (e) of 10 CFR 50.61a in Enclosure 1 of SECY-07-0104, Reference 16. It is requested that the SE be revised to state that the requirements of Section (e) in

Enclosure 1 of SECY-07-0104 should only be used until the applicable requirements in the final version of 10 CFR 50.61a are published in the *Federal Register*. The following revision is recommended, "By monitoring flaw sizes in accordance with the criteria described in Section (e) of the proposed rulemaking in SECY-07-0104, or the final published version of 10 CFR 50.61a, licensees will ensure…."

NRC Response:

Subsequent to the original resolution of this comment the 10 CFR 50.61a was published and finalized. The NRC staff now holds the position that licensees seeking to extend additional intervals after the first extended interval shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a.

4. Page 18, Lines 27-34

PWROG Comment:

The draft SE requires that the qualified vessel inspection results be evaluated per the existing requirements in Section (e) of 10 CFR 50.61a in Enclosure 1 of SECY-07-01 04, Reference 16. It is requested that the SE be revised to state that the requirements of Section (e) in Enclosure 1 of SECY-07-0104 should only be used until the applicable requirements in the final version of 10 CFR 50.61a are published in the *Federal Register*. The following revisions are recommended, "...in Enclosure 1 to the proposed rulemaking in SECY-07-01 04, Reference 16, or the final published version of 10 CFR 50.61a." and "…and analyses requested in Section (e) of the proposed rulemaking in SECY-07-0104, or the final published version of 10 CFR 50.61a, will be submitted…."

NRC Response:

Subsequent to the original resolution of this comment the 10 CFR 50.61a was published and finalized. The NRC staff now holds the position that licensees seeking to extend additional intervals after the first extended interval shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a.

5. Page 18, Lines 41-44

PWROG Comment:

It is stated in the draft SE that: "Licensees also implementing Section (c) of the proposed 10 CFR 50.61a must perform the inspections and analyses required by Section (e) of the proposed 10 CFR 50.61a and may not defer the ISI inspection of the RV beltline welds." The following revision is recommended: "Licensees also implementing Section (c) of the proposed 10 CFR 50.61a must perform the inspections and analyses required by Section (e) of the proposed 10 CFR 50.61a prior to implementing the extended interval."

NRC Response:

Subsequent to the original resolution of this comment the 10 CFR 50.61a was published and finalized. The NRC staff now holds the position that licensees seeking to extend additional intervals after the first extended interval shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a. As a result, the language discussed in this comment is no longer applicable.

6. Page 20, Lines 15-17

PWROG Comment:

It is stated in the draft SE that: "Surface cracks that penetrate through the cladding ....were not part of the PTS Risk Study." However, Oconee Unit 1 included these surface cracks in the PTS risk analyses of NUREG-1806 and NUREG-1874, even though they did not contribute to the TWCF. It is suggested that the SE be revised to state, "Surface cracks that penetrate through the cladding and into the ferritic steel have not been observed in the beltline of operating PWR Reactors. PFM analyses indicate…."

NRC Response:

The NRC staff does not agree with the change. Surface defects through the clad were included in the PTS study. However, surface defects though the clad that penetrate into the ferritic steel were not included in the PTS study. Therefore, the SE will not be revised with the suggested wording.

7. Page 21, Lines 21-28

PWROG Comment:

The draft SE requires that the qualified vessel inspection results be evaluated per the existing requirements in Section (e) of 10 CFR 50.61a in Enclosure 1 of SECY-07-0104, Reference 16. It is requested that the SE be revised to state that the requirements of Section (e) in Enclosure 1 of SECY-07-0104 should only be used until the applicable requirements in the final version of 10 CFR 50.61a are published in the *Federal Register*. The following revisions are recommended, "...in Enclosure 1 to the proposed rulemaking in SECY-07-0104, Reference 16, or the final published version of 10 CFR 50.61a." and "…and analyses requested in Section (e) of the proposed rulemaking in SECY-07-0104, or the final published version of 10 CFR 50.61 a, wil be submitted…."

NRC Response:

Subsequent to the original resolution of this comment the 10 CFR 50.61a was published and finalized. The NRC staff now holds the position that licensees seeking to extend additional intervals after the first extended interval shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a. As a result, the language discussed in this comment is no longer applicable.

8. Page 21, Lines 35-38

PWROG Comment:

It is stated in the draft SE that: "Licensees also implementing Section (c) of the proposed 10 CFR 50.61a must perform the inspections and analyses required by Section (e) of the proposed 10 CFR 50.61a and may not defer the ISI inspection of the RV beltline welds." The following revision is recommended, "Licensees also implementing Section (c) of the proposed 10 CFR 50.61a must perform the inspections and analyses required by Section (e) of the proposed 10 CFR 50.61a prior to implementing the extended interval."

NRC Response:

Subsequent to the original resolution of this comment the 10 CFR 50.61a was published and finalized. The NRC staff now holds the position that licensees seeking to extend additional intervals after the first extended interval shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a. As a result, the language discussed in this comment is no longer applicable.

9. Page 23, Line 27

PWROG Comment:

The date and Agencywide Documents Access and Management System (ADAMS) Accession number for Revision 1 of Reference 15 are October 31, 2003, and ML051790410, respectively.

NRC Response:

The NRC staff agrees with this change.

10. Page 23, Line 35

PWROG Comment:

ADAMS Accession number ML012630333 for Reference 15 could not be found on ADAMS. ADAMS Accession numbers ML042610469 and ML042610375 can be used for WCAP-14572 and Supplement 1 on the probabilistic structural reliability and risk assessment tool, respectively. It is recommended that the SE be revised to include these accession numbers for Reference 15.

NRC Response:

The NRC staff agrees with this change.

11. Page 23, Line 42

PWROG Comment:

For version 06.1 of FAVOR, Reference 20, the WCAP Technical Report used letter ORNL/TM-2007/0030, which is the same as “Williams 07” in NUREG-1874. It is recommended that this reference for FAVOR be used in the SE.

NRC Response:

The NRC staff agrees with this change.

12. Page 24, Line 11

PWROG Comment:

For Reference 26, the ADAMS Accession Number is ML042880482. It is recommended that this accession number be added to the SE.

NRC Response:

The NRC staff agrees with this change.

13. Page 24, Line 13

PWROG Comment:

Reference 27 is cited in Section 3.2.2.3 (Page 15, Line 18) but not included in the list of references in Section 5.0. The following text is suggested for addition to the SE: "23. Letter Report, "Estimate of External Events Contribution to Pressurized Thermal Shock (PTS) Risk," October 1, 2004 (ADAMS Accession No. ML042880476)."

NRC Response:

The NRC staff agrees with this change.

14. Page 20, Lines 41-44

NRC Comment:

To ensure consistency when discussing the final and proposed rule within the SE, the NRC staff has made one additional change to the SE. The NRC staff has modified the following sentence: "In addition, licensees that do not implement the proposed 10 CFR 50.61a must amend their licenses to require that the information and analyses requested in Section (e) of the proposed 10 CFR 50.61a will be submitted for NRC staff review and approval."

NRC Response:

Subsequent to the original resolution of this comment the 10 CFR 50.61a was published and finalized. The NRC staff now holds the position that licensees seeking to extend additional intervals after the first extended interval shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a. As a result, the language discussed in this comment is no longer applicable.

APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION (NRC) EXPANDED DISCUSSION OF WCAP-16168-NP-A, REVISION 2, SUBMITTAL EXPECTATIONS

By letter dated December 1, 2009, the Pressurized Water Reactor Owners Group (PWROG) requested that the U.S. Nuclear Regulatory Commission (NRC) staff clarify their position regarding Safety Evaluation (SE) implementation requirements to submit a license amendment request.

Firstly, the NRC staff will grant inservice inspection (ISI) interval extensions for the subject components on an interval-by-interval basis, i.e., only a facility's current ISI interval will be extended for up to 20 years. Licensees will have to submit subsequent requested alternatives, for NRC review and approval, to extend each following ISI interval from 10 years to 20 years, as needed. Each subsequent interval ISI interval will require a separate Title 10 *Code of Federal Regulations* (10 CFR) 50.55a-based application.

Licensees seeking to extend additional intervals after the first extended interval shall, within one year of completing each of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*, Section XI, Category B-A and B-D RV weld inspections required in the proposed ISI interval, provide the information and analyses requested in Section (e) of 10 CFR 50.61a.

The NRC staff has prepared the following synopsis of staff expectations for submittals as of the publication of this SE. This is a synopsis of the requirements listed in the revised final SE for Topical Report (TR) WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082820046); the December 1, 2009, PWROG letter OG-09-454, "Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, "Risk Informed Extension of the Reactor Vessel In-Service Inspection Interval"" (ADAMS Accession No. ML093370133); and the July 12, 2010, PWROG letter OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval"" (ADAMS Accession No. ML11153A033). The staff considers this synopsis as a clarification guide to the content of an acceptable application; not as a replacement to the use of the SE and the TR.

The primary expectation is that the applicant will establish that its plant operates and exists within the analysis space used in the TR. To establish this fact, Tables A-1 through A-3 of Appendix A of the TR should be filled out completely and submitted. Appendix A of the TR contains substantial explanation as well as examples of how Tables A-1 through A-3 are to be constructed and filled out and it is expected that the applicant will follow the guidance of the TR in constructing their submittal where it does not conflict with the SE to the TR.

Additionally the following caveats must be addressed:

1. The dates identified in the request for alternative should be within plus or minus one refueling cycle of the dates identified in the implementation plan provided to the NRC in letter OG-10-238. Any deviations from the implementation plan should be discussed in detail in the request for alternative ISI interval. The maximum interval for proposed ISI is 20 years.

2. The request for alternative ISI interval can use any NRC-approved method to calculate ΔT_{30} and RT_{MAX-X}. However, if the request uses the NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," methodology to calculate ΔT_{30}, then the request should include the analysis described in paragraph (6) of subsection (f) to the alternate pressurized thermal shock rule 10 CFR 50.61a. The analysis should be done for all of the materials in the beltline area with at least three surveillance data points.

3. If the subject plant is a Babcock & Wilcox plant:

   - Licensees must verify that the fatigue crack growth of 12 heat-up/cool-down transients per year bound the fatigue crack growth for all of its design basis transients
   - Licensees must identify the design basis transients that contribute to significant fatigue crack growth

4. If the subject plant has reactor vessel forgings that are susceptible to underclad cracking with RT_{MAX-FO} values exceeding 240 °F, then the WCAP analyses are not applicable. The licensee must submit a plant-specific evaluation for any extension to the 10-year inspection interval for ASME Code, Section XI, Category B-A and B-D RPV welds.

5. Licensees seeking additional interval extension shall within one year of completing each of the ASME Code, Section XI, Category B-A and B-D RV weld inspections required in the proposed ISI interval, provide the information and analyses requested in Section (e) of 10 CFR 50.61a

The staff expects that all information provided in the submittals be verifiable and/or previously accepted by the NRC. Examples of this include neutron fluences and weld chemistries.

APPENDIX C

RESOLUTION OF COMMENTS BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON REVISED DRAFT SAFETY EVALUATION FOR TOPICAL REPORT WCAP-16168-NP-A, REVISION 2, "RISK-INFORMED EXTENSION OF THE REACTOR VESSEL IN-SERVICE INSPECTION INTERVAL"

This attachment provides the U.S. Nuclear Regulatory Commission (NRC) staff's review and disposition of the comments made by the Pressurized Water Reactor Owners Group (PWROG) on the revised draft safety evaluation (SE) for Topical Report (TR) WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (Agencywide Documents and Management System Accession (ADAMS) Nos. ML102790090 and ML110530428). The PWROG provided its comments in a letter dated April 14, 2011, entitled "Comments on Revised Draft Safety Evaluation for PWR Owners Group (PWROG) Report WCAP-16168-NP, Rev. 1, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (TAC No. ME3495), PA-RMSC-0695" (ADAMS Accession No. ML111530379).

PWROG Comment #1:

The revised SER [safety evaluation report] refers to PWROG letter OG-09-454 which was submitted to the NRC on December 1, 2009. An updated implementation plan was sent to the NRC in PWROG letter OG-10-238 on July 12, 2010. However, PWROG letter OG-10-238 is not referenced anywhere in the revised SER.

NRC Response:

The NRC staff acknowledges that an updated implementation plan was sent. The staff has no contentions with the updated implementation plan and has revised the appropriate references to the plan in the SE to reference the updated implementation plan OG-10-238 instead of the older OG-09-454.

PWROG Comment #2:

Item #6 has been added to Section 3.4 of the SER: "Submit Proposed Change." This new item indicates "Licensees seeking second or additional interval extensions shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a." In all plant-specific ISI interval extensions that have been approved to date, when ASME Section XI Appendix VIII data was available, these analyses have been performed and the information has been provided in plant relief requests. It is anticipated that the vast majority of future plant submittals will be for plants that have performed an Appendix VIII qualified examination. Therefore, in many cases, the information is available such that these analyses can be performed for the first interval extension and not just the second or additional interval extension. The PWROG suggests that item #6 be revised to require that the information and analyses requested in Section (e) of 10 CFR 50.61a be provided in the first relief request if an inservice inspection that meets the requirements of ASME Section XI, Appendix VIII has previously been performed. It is suggested that this change also be made in the revised SER to the last paragraph of Section 4.0, the third paragraph of Appendix B, and item #5 of Appendix B.

ATTACHMENT 3

NRC Response:

The NRC staff does not agree with this change. The NRC staff considers its original position adequate: those licensees seeking to extend additional intervals after the first extended interval shall provide the information and analysis requested in Section (e) of 10 CFR 50.61a. Earlier submittals would produce burden with little compensatory advantage.

PWROG Comment #3:

The original SER to WCAP-16168-NP contained a requirement for licensees to submit a license amendment request (LAR) coincident with their relief request to implement the extended ISI interval. The purpose of this LAR was to require that plants not implementing 10 CFR 50.61a submit the information and analyses required by Section (e) of 10 CFR 50.61a to the NRC for review and approval within one year of completing the ISI of their Section XI Category B-A and B-D welds. The original intent of the LAR was to eliminate the need to resubmit relief requests every 20 years. Shortly after the issuance of the original SER, the NRC set a precedent by (1) requesting that Entergy [Entergy Operations, Inc.] withdraw its LAR for Waterford 3 [Waterford Stream Electric Station, Unit 3] and (2) indicating that relief requests would be required for each 20-year interval. Based on this precedent, it was requested by the PWROG in letter OG-09-454 that the LAR be removed in the revised SER. However, a similar requirement to submit a LAR remains in the third paragraph and item #5 of Appendix B of the revised SER. This LAR requirement is ambiguous and unnecessary. For the following reasons, this requirement should be removed entirely.

a. Item #6 of Section 3.4 and Section 4.0 of the revised SER already require that the information and analyses required by Section (e) of 10 CFR 50.61a be provided in the plant relief requests for plants seeking second or additional interval extensions. Therefore, for these plants, the LAR requirement would result in the same information being provided to the NRC for review and approval twice.

b. The effect of the LAR requirement seems to be that the results of the ISI evaluation required by 10 CFR 50.61a would be provided to the [NRC] Staff within one year of the exam being performed, as opposed to ~9 years later in the subsequent relief request. It is understood that the NRC has an interest in obtaining the ISI data as soon as possible in order to better establish the credibility of the ISI interval extension and the 10 CFR 50.61a. However, the PWROG believes that this information can be obtained by the NRC through other means without placing this requirement on utilities. One such means would be through the PWROG providing the data when available, as suggested in comment #2.

c. The LAR requirement indicates that the "…information and analyses…be submitted for NRC staff review and approval." If a licensee has not submitted a relief request at the time this information is provided to the [NRC] Staff, it is not clear what the [NRC] Staff intends to approve.

The revised SER does not provide an expectation as to when the LAR is intended to be submitted.

NRC Response:

The NRC staff agrees with the spirit of this comment and has removed the text referring to the LAR requirement. It is not the staff's intent to require an LAR, as it is now the staff's position that licensees seeking to extend additional intervals after the first extended interval shall provide the information and analysis requested in Section (e) of 10 CFR 50.61a.