PWROG
PWR Owners Group

Program Management Office
20 International Drive
Windsor, Connecticut 06095

March 31, 2008 WCAP-16168-NP, Rev 1

OG-08-107 Project Number 694

U. S. Nuclear Regulatory Commission
Document Control Desk and Chief Financial Officer
Washington, DC 20555-0001

Subject: PWR Owners Group
Comments on PWR Owners Group (PWROG) Report WCAP-16168-NP, Rev. 1 "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" Draft Safety Evaluation (TAC NO. MC9768) MUHP 5097/5098/5099 Task 2008/2059

References:

1. WOG Letter from Ted Schiffley to Document Control Desk, Request for Review and Approval of WCAP-16168-NP Rev. 1, entitled "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (PWROG Letter WOG-06-25), January 26, 2006.
2. Acceptance for Review of Westinghouse Owners Group (WOG) Topical Report WCAP-16168-NP, Rev. 1 "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (TAC NO. MC9768), September 19, 2006, MUHP 5097/5098/5099 Task 2008/2059, (PWROG Letter OG-06-311), dated September 22, 2006.
3. Responses to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-16168-NP, Rev. 1 "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (TAC NO. MC9768), MUHP 5097/5098/5099 Task 2008/2059 (PWROG Letter OG-07-455), dated October 16, 2007.
4. Draft Safety Evaluation for PWR Owners Group (PWROG) Report WCAP-16168-NP, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (TAC No. MC9768), MUHP 5097/5098/5099, Task 2008/2059, PA-MSC-0120 March 6, 2008, (PWROG Letter OG-08-79), dated March 7, 2008.

Westinghouse WCAP-16168-NP, Rev. 1 "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" was submitted by the PWROG for NRC review and approval on January 26, 2006 (Reference 1) and accepted by the NRC on September 19, 2006 (Reference 2). This was supported by responses to the NRC request for additional information on October 16, 2007 (Reference 3). On March 6, 2008, the NRC Staff issued the reference draft safety evaluation (Reference 4) for this report.

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The purpose of this letter is to transmit comments on the draft safety evaluation. The reviewers of the draft SER and its transmittal letter found them to be of high technical quality and well written. Therefore, we have only a few areas needing clarification or correction. A number of editorial changes are also provided, mostly in the area of the references in Section 6.0. The comments on the draft safety evaluation are provided in Attachment 1.

Per the NRC request in Reference 4, a marked-up copy of the draft SE showing proposed change is provided in Enclosure 1.

If you have any questions concerning this matter, please feel free to call Christine DiMuzio at 412-374-5680.

Sincerely yours,

Dennis E. Buschbaum, Chairman
PWR Owners Group

DEB:JPM:las

Attachment: (1) - Comments to Draft SE
Enclosure: (1) – Mark-up Copy of Draft SE

cc: PWROG Management Committee Participants in the RV ISI Program
PWROG Materials Subcommittee Participants in the RV ISI Program
PWROG PMO
G. Ament, W
C. Boggess, W
N. Palm, W
B. Bischoff, W
J. Andrachek, W
C. Brinkman, Westinghouse
H. Cruz, USNRC

PWROG Draft Comments on the NRC's Draft Safety Evaluation on WCAP-16168-NP, Rev. 1 "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval"

Table 1: PWROG Review Comments on the NRC's Draft Safety Evaluation on WCAP-16168-NP, Rev. 2 dated March 6, 2008

Comment Number	Page/Line	Comment
1	Page 3 Lines 19-21	It is stated "*This correlation took into consideration the contribution to TWCF 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, <u>forging,</u> axial weld, and circumferential welds.*" Note that the revised words are underlined.
2	Page 15 Line 41	The change in risk (9.43E-10/year) for Beaver Valley 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.
3	Page 17 Lines 11-18	The draft SER 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 12. It is requested that the SER 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, <u>or the final published version of 10 CFR 50.61a,</u> licensees will ensure..." Note that the revised words are underlined.
4	Page 18 Lines 27-34	The draft SER requires that the qualified vessel inspection results be evaluated per the existing requirements in Section (e) of 10 CFR50.61a in Enclosure 1 of SECY-07-0104, Reference 12. It is requested that the SER 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 12, <u>or the final published version of 10 CFR 50.61a</u>." and "...and analyses requested in Section (e) of the proposed rulemaking in SECY-07-0104, <u>or the final published version of 10 CFR 50.61a</u>, will be submitted..." Note that the revised words are underlined.
5	Page 18 Lines 41-44	It is stated "*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 <u>prior to implementing the extended interval</u>.* Note that the revised words are underlined.
6	Page 20 Lines 15-17	It is stated "*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 SER be revised to state "*Surface cracks that*

Table 1: PWROG Review Comments on the NRC's Draft Safety Evaluation on WCAP-16168-NP, Rev. 2 dated March 6, 2008		
Comment Number	**Page/Line**	**Comment**
		penetrate through the cladding and into the ferritic steel have not been observed in the beltline of operating PWR Reactors. PFM analyses indicate,...."
7	Page 21 Lines 21-28	The draft SER 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 12. It is requested that the SER 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 12, <u>or the final published version of 10 CFR 50.61a</u>." and "...and analyses requested in Section (e) of the proposed rulemaking in SECY-07-0104, <u>or the final published version of 10 CFR 50.61a</u>, will be submitted..." Note that the revised words are underlined.
8	Page 21 Lines 35-38	It is stated "*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 <u>prior to implementing the extended interval</u>.* Note that the revised words are underlined.
9	Page 23 Line 27	The date and ADAMS accession number for Revision 1 of Reference 11 are Oct. 31, 2003 and ML051790410, respectively.
10	Page 23 Line 35	ADAMS accession number ML012630333 for Reference 13 could not be found on ADAMS. ADAMS accession numbers ML042610469 and ML042610375 can be used for WCAP-14572 and Supplement 1 on the probabilistic SRRA tool, respectively. It is recommended that the SER be revised to include these accession numbers for Reference 13.
11	Page 23 Line 42	For version 06.1 of FAVOR, Reference 16, 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 SER.
12	Page 24 Line 11	For Reference 22, the ADAMS Accession No. is ML042880482. It is recommended that this accession number be added to the SER.
13	Page 24 Line 13	Reference 23 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 SER: <u>23. Letter Report, "Estimate of External Events Contribution to Pressurized Thermal Shock (PTS) Risk," October 1, 2004 (ADAMS Accession No. ML042880476).</u> Note that the recommended words are underlined.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 6, 2008

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: DRAFT SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG) TOPICAL REPORT (TR) WCAP-16168-NP, REVISION 2, "RISK- INFORMED EXTENSION OF THE REACTOR VESSEL IN-SERVICE INSPECTION INTERVAL" (TAC NO. MC9768)

Dear Mr. Bischoff:

By letter dated January 26, 2006, as supplemented by letter dated June 8, 2006, the PWROG submitted TR WCAP-16168-NP, Revision 1, to the U.S. Nuclear Regulatory Commission (NRC) staff for review. TR WCAP-16168-NP, Revision 2, and responses to the NRC staff's request for additional information (RAI) on TR WCAP-16168-NP, Revision 1, were submitted for NRC staff review by PWROG letter dated October 16, 2007. Enclosed for PWROG review and comment is a copy of the NRC staff's draft safety evaluation (SE) for the TR.

The NRC staff has accepted TR WCAP-16168-NP, Revision 2, based on the imposition of a condition related to the augmented evaluation of in-service inspection (ISI) results taken from Section (e) of the proposed Title 10 of the *Code of Federal Regulations* 50.61a, published in the Federal Register on October 3, 2007 (72 FR 56275). The NRC staff is in the process of reviewing public comments on the proposed rule and preparing the final rule. If the final 10 CFR 50.61a differs from the proposed 10 CFR 50.61a with regard to the augmented ISI evaluation requirements, the PWROG will be expected to review the requirements in the final 10 CFR 50.61a and determine whether a revision to the accepted TR WCAP-16168-NP, Revision 2, is required. The PWROG will be expected to notify the NRC staff, in writing, of the results of its determination within six months of the publication date of the final 10 CFR 50.61a. If, on this basis, a revision to the accepted TR WCAP-16168-NP, Revision 2, is required, the PWROG will be expected to submit the revised TR for NRC staff review within one year of the publication date of the final 10 CFR 50.61a.

Furthermore, licensees that chose to implement the proposed 10 CFR 50.61a must perform the ISI inspections required in Section (e) of the rule, and must submit the required information for review and approval to the Director, Office of Nuclear Reactor Regulation, in accordance with Section (c) of the rule, at least three years before the limiting RT_{PTS} value calculated under 10 CFR 50.61 is projected to exceed the PTS screening criteria in 10 CFR 50.61. Licensees 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 reactor vessel beltline welds.

Twenty working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The NRC staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the NRC staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes. If you have any questions, please contact Sean E. Peters at 301-415-1842.

Sincerely,

Stacey L. Rosenberg

Stacey L. Rosenberg, Chief
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Draft SE

cc w/encl: See next page

cc:
Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355
greshaja@westinghouse.com

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DRAFT 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

1.0 INTRODUCTION AND BACKGROUND

By letter dated January 26, 2006, as supplemented by letter dated June 8, 2006, the Westinghouse Owners Group (WOG), currently known as the Pressurized Water Reactor Owners Group (PWROG), submitted topical report WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (Reference 1 and Reference 2), for U.S. Nuclear Regulatory Commission (NRC) staff review. By letter dated October 16, 2007, the PWROG submitted responses to the NRC staff's request for additional information (RAI) on WCAP-16168-NP, Revision 1, and provided WCAP-16168-NP, Revision 2 (Reference 3), but did not expand its scope as originally submitted for NRC staff review.

In WCAP-16168-NP, Revision 2, (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) 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 recently-completed pressurized thermal shock (PTS) research program. The NRC's Office of Nuclear Regulatory Research (RES) completed this research program to update the PTS regulations. In an October 3, 2007, Federal Register Notice (72 FR 56275) (Reference 4), the NRC proposed to amend its regulations to provide updated fracture toughness requirements for protection against PTS events for PWR RVs. NUREG-1806, "'Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)" (the PTS Risk Study) (Reference 5 and Reference 6) and (2) NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)" (Reference 7), provided the technical basis for the rulemaking. These reports summarized and referenced several additional reports on the same topic.

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 Title 10 of the *Code of Federal Regulation* (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 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 [(SRP)] for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference 8). SRP Chapter 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 9). 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, Unit 1 (BV1), Palisades, and Oconee, 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 NRC 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 (Reference 7). This correlation took into consideration the contribution to TWCF from each of the most limiting plate, 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 be 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 can 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 PWROG letter OG-06-356, "Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP 16168-NP, Revision 1, "Risk Informed Extension of the Reactor Vessel In-Service Inspection Interval," MUHP 5097-99, Task 2059," dated October 31, 2006 (Reference 10).

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 safety evaluation (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 can 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 that is the technical basis for the proposed alternative fracture toughness requirements for pressurized thermal shock in 10 CFR 50.61a (Reference 4).

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 ($K_{Iapplied}$). The ratio of the maximum allowable stress intensity factor ($K_{I\,allowable}$), per the ASME Code, Section XI, Appendix A criteria, to $K_{Iapplied}$ 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 NRC 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.

Distributions and Uncertainties in Flaw Number and Size

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 NRC 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 NRC 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 11). 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 NRC PTS Risk Study, they are applicable for use in RV ISI interval extension analyses.

In Attachment 1 to the June 8, 2006 letter (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 fairly small. NUREG-1874 indicated that for severe PTS transients, the TWCF for forgings with underclad cracks can 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-FO} [1] is greater than 240 °F. Table 3.4 in NUREG-1874 indicated that the highest RT_{MAX-FO} for a PWR RV ring forging is 187.3 °F at 32 EFPY and 198.6 °F at 48 EFPY. Therefore, it is unlikely that the RT_{MAX-FO} value for any domestic PWR will ever exceed 240 °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-FO} values exceeding 240 °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 13), 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 NRC 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. Each 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, seven 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.

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1 RT_{MAX-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-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, in Enclosure 1 to the Proposed Rulemaking in SECY-07-0104 (Reference 12).

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 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 to be 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 result 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 the Theoretical and Users Manual for PC-PRAISE (Reference 14). 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 13 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 15). 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 16). The FAVOR Code, which was developed by Oak Ridge National Laboratory (ORNL) to perform PFM analyses for the NRC 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 NRC PTS Risk Study. A series of neutron transport calculations were performed for the NRC 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 17). The models incorporated pilot plant-specific geometry and operating data. The neutron fluence for energies greater than one million electron volts (E > 1MeV) 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 contribute 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 (FAVPFM), 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 NRC PTS Risk Study and is described in the FAVOR Code Theory Manual (Reference 16). The cladding stress used in the pilot plant studies was taken from the NRC PTS Risk Study. The cladding temperature dependence due to differential thermal expansion was based on a stress free temperature of 488 °F, which is consistent with that used in the NRC 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 NRC PTS Risk Studies. 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 NRC PTS Risk Study. As part of the NRC PTS Risk Study, PRA models were developed by the NRC staff for each of the three pilot plants using plant-specific information (References 18, 19, and 20). 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 ten 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 vessel failure (conservatively assumed to occur if a through-wall crack develops) given the thermal-hydraulic 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 ten years appropriately envelops the impact of the proposed change for any facility regardless of its inspections schedule and history.

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 ten 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 ten 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 ten

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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.

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.

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 ten 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 can 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% of the mean estimates would exceed the upper bound value and 2.5% 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 significant contributors to risk from RV failure,

- the bounding estimates from only one inspection versus an inspection every ten 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 NRC PTS Risk Study in its analysis. The TR conservatively assumed that core damage and large early release will 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 21). The analyses were described in detail in the plant-specific PRA reports (References 18, 19, and 20) 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% 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 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 (Reference 6). The objective of the peer review was to assess the adequacy and reasonableness of the technical basis to support the proposed revision of the 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 22). 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 72 FR 56275 (Reference 4), the NRC staff reported its conclusion that the TWCF 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 Reference 23 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 as a result 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 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 9.43E-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 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 effect 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 SE. The PWROG used 7 heat-up and cooldown cycles per year for Westinghouse-designed plants, 13 heat-up and cooldown cycles per year for CE-designed plants, and 12 heat-up and cooldown cycles per year for B&W-designed plants. The design basis for the Westinghouse plant was 5 cooldown cycles per year. Although it was determined that three 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 three 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 SE and must be monitored by determining whether the 95^{th} percentile $TWCF_{TOTAL}$ [4] for the plant requesting to implement the pilot plant study is less than the 95^{th} percentile $TWCF_{TOTAL}$ from the pilot plant study. The 95^{th}

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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.

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 SE. The PWROG utilized the flaw sizes and distributions in the NRC 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 the proposed 10 CFR 50.61a, Alternative fracture toughness requirements for protection against pressurized thermal shock, in Enclosure 1 to the proposed rulemaking in SECY-07-0104 described the allowable flaw distribution for embedded flaws and surface-breaking flaws that would be permitted for RVs that are at the PTS screening limits described in the proposed 10 CFR 50.61a. By monitoring flaw sizes in accordance with the criteria described in Section (e) of the proposed rulemaking in SECY-07-0104, licensees will ensure that their RVs do not have flaws that invalidate the results of the PWROG PFM analyses.

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 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 latest approved methodology documented in Regulatory Guide 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.

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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, Alternative fracture toughness requirements for protection against pressurized thermal shock, in Enclosure 1 to the Proposed Rulemaking in SECY-07-0104.

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-06-356, "Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP 16168-NP, Revision 1, "Risk Informed Extension of the Reactor Vessel In-Service Inspection Interval," MUHP 5097-99, Task 2059," dated October 31, 2006 (Reference 10).

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 be applicable.

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, the licensee must provide the information and analyses requested in Section (e) of the proposed 10 CFR 50.61a, <u>Alternative fracture toughness requirements for protection against pressurized thermal shock</u>, in Enclosure 1 to the proposed rulemaking in SECY-07-0104, Reference 12. 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 rulemaking in SECY-07-0104 will be submitted for NRC staff review and approval. The amendment to the license shall be submitted at the same time as the request for alternative.

Licensees that also implement the proposed 10 CFR 50.61a must perform the ISI inspections required in Section (e) of the rule and must submit the required information for review and approval to the Director, Office of Nuclear Reactor Regulation, in accordance with Section (c) of the rule, at least three years before the limiting RT_{PTS} value calculated under 10 CFR 50.61 is projected to exceed the PTS screening criteria in 10 CFR 50.61. 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.

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 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 SE, licensees must monitor the number of cycles of transients that could effect 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 proposed a new rulemaking (72 FR 56275) which would change the 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 proposed rule provided 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 and 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. 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 and approved by the NRC to extend any facility's ISI interval. 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. The amendment to the license shall be submitted at the same time as the request for an alternative. The request for an alternative will be for the remainder of the licensed period for the plant.

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 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.

The staff will not repeat its review of the matters described in WCAP-16168-NP, Revision 2, as modified by this SE, when the report appears as a reference 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 SE.

4.0 CONDITIONS AND LIMITATIONS

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 PWR Owners Group letter OG-06-356, "Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP 16168-NP, Revision 1, "Risk Informed Extension of the Reactor Vessel In-Service Inspection Interval," MUHP 5097-99, Task 2059," dated October 31, 2006 (Reference 10).

Within one year of completing each of the ASME Code Section XI Category B-A and B-D RV welds inspections required in the proposed ISI interval, the licensee must provide the information and analyses requested in Section (e) of the proposed 10 CFR 50.61a, Alternative fracture toughness requirements for protection against pressurized thermal shock, in Enclosure 1 to the proposed rulemaking in SECY-07-0104, Reference 12. 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 rulemaking in SECY-07-0104 will be submitted for NRC staff review and approval. The amendment to the license shall be submitted at the same time as the relief request.

Licensees that also implement the proposed 10 CFR 50.61a must perform the ISI inspections required in Section (e) of the rule and must submit the required information for review and approval to the Director, Office of Nuclear Reactor Regulation, in accordance with Section (c) of the rule, at least three years before the limiting RT_{PTS} value calculated under 10 CFR 50.61 is projected to exceed the PTS screening criteria in 10 CFR 50.61. Licensees also implementing Section (c) of the proposed 10CFR 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 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 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.

The NRC staff has accepted TR WCAP-16168-NP, Revision 2, based on the imposition of a condition related to the augmented evaluation of in-service inspection (ISI) results taken from Section (e) of the proposed Title 10 of the *Code of Federal Regulations* 50.61a, published in the Federal Register on October 3, 2007 (72 FR 56275). The NRC staff is in the process of reviewing public comments on the proposed rule and preparing the final rule. If the final 10 CFR 50.61a differs from the proposed 10 CFR 50.61a with regard to the augmented ISI evaluation requirements, the PWROG will be expected to review the requirements in the final 10 CFR 50.61a and determine whether a revision to the accepted TR WCAP-16168-NP, Revision 2, is required. The PWROG will be expected to notify the NRC staff, in writing, of the results of its determination within six months of the publication date of the final 10 CFR 50.61a. If, on this basis, a revision to the accepted TR WCAP-16168-NP, Revision 2, is required, the PWROG will be expected to submit the revised TR for NRC staff review within one year of the publication date of the final 10 CFR 50.61a.

5.0 CONCLUSION

The NRC staff has found that the methodology presented in WCAP-16168-NP, 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 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, Revision 2.

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. ML0729204120). Enclosure 2, WCAP-16168-NP, Revision 2, 'Risk-Informed Extension of Reactor vessel In-Service Inspection Interval', October 2007 (ADAMS Accession No. ML072920413).

4. Federal Register Notice, (72 FR 56275) "Alternative Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," October 3, 2007 (ADAMS Accession No. ML072780354)

5. 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)

6. 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)

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

8. 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)

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

10. PWR Owners Group letter OG-06-356, "Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP 16168-NP, Revision 1, "Risk Informed Extension of the Reactor Vessel In-Service Inspection Interval," MUHP 5097-99, Task 2059," dated October 31, 2006

11. NUREG/CR-6817, Revision 1, "A Generalized Procedure for Generating Flaw-Related Inputs for the FAVOR Code," March 1, 2004 (ADAMS Accession No. ML040830499)

12. SEC-07-0104, "Proposed Rulemaking-Alternate Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events," June 25, 2007, (ADAMS Accession No. ML070570525)

13. 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. ML012630327, ML012630349, and ML012630333)

14. Theoretical and Users Manual for PC-PRAISE, NUREG/CR-5864, July 1992

15. Electric Power Research Institute (EPRI) Nondestructive Examination (NDE) Center on the detection and sizing qualification of ISIs on the RV beltline welds

16. Fracture Analysis of Vessels - Oak Ridge (FAVOR) Code

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

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

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

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

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

22. Letter Report, “Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants,” December 14, 2004

Principle Contributors: Barry Elliott
Stephen Dinsmore

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