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Project No. 694

WCAP 16168-NP-A, Rev 2

June 13, 2008

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U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: PWR Owners Group

<u>Transmittal of NRC Approved Topical Report WCAP-16168-NP-A, Rev. 2 "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (TAC NO. MC9768) (MUHP 5097/5098/5099 Task 2008/2059, PA MSC-0120)</u>

Reference: 1. Letter, H. Nieh (NRC) to G. Bischoff (PWROG), "Final 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), dated May 8, 2008.

The purpose of this letter is to transmit four (4) non-proprietary copies WCAP-16168-NP-A, Revision 2, for NRC files. WCAP-16168-NP-A, Revision 2, contains the staff's Safety Evaluation. This transmittal completes action on topical report WCAP-16168-NP-A, Revision 2; thus, the PWROG requests that TAC No. MC9768 be closed.

For technical questions regarding the enclosed report, please contact the program technical lead Cheryl Boggess (W) at (412) 374-4692 or Nathan Palm (W) at (724) 722-6016.

D048
Rec'd DCD 10/7/08 JRR
Add: Holly Cruz
E-RIDS

If you have any additional questions or comments on the enclosed report, feel free to contact Jim Molkenthin in the PWROG office at (860) 731-6727.

Sincerely,

J. Molkenthin Approving for D. Buschbaum

Dennis E. Buschbaum, Chairman
PWR Owners Group

DEB:JPM:las

Enclosure: (1) – PWROG Report WCAP-16168-NP-A, Revision 2

cc: PWROG Management Committee Participants in RV ISI Program
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Westinghouse Non-Proprietary Class 3

WCAP-16168-NP-A
Revision 2

June 2008

Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval

Westinghouse

WCAP-16168-NP-A
Revision 2

Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval

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June 2008

Approved: ***Electronically Approved****
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Approved: ***Electronically Approved****
Gordon C. Bischoff
PWR Owners Group

This work was performed for the PWR Owners Group under PWROG Project MUHP-5097, MUHP-5098, MUHP-5099, PWROG Project Authorization MSC-0119, MSC-0120 and CEOG Task 2008, 2059.

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WASHINGTON, D.C. 20555-0001

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MAY 14 2008
PWROG
Project Office

May 8, 2008

Mr. Gordon Bischoff, Manager
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SUBJECT: FINAL 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. 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. By letter dated March 6, 2008, an NRC draft safety evaluation (SE) regarding our approval of TR WCAP-16168-NP, Revision 2, was provided for your review and comments. By letter dated March 31, 2008, the PWROG commented on the draft SE. The NRC staff's disposition of PWROG's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR WCAP-16168-NP, Revision 2, is acceptable for referencing in licensing applications for Westinghouse, Combustion Engineering, and Babcock and Wilcox designed pressurized water reactors, for which an operating license was issued under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 prior to the date of this letter, to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of 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 10 CFR 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 choose to implement 10 CFR 50.61a must perform the ISI 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

G. Bischoff - 2 -

exceed the PTS screening criteria in 10 CFR 50.61. Licensees implementing Section (c) of 10 CFR 50.61a must perform the inspections and analyses required by Section (e) of 10 CFR 50.61a prior to implementing the extended interval.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that PWROG publish the accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed final SE after the title page. Also, it must contain historical review information, including NRC requests for additional information and your responses. The accepted version shall include an "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, the PWROG and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Final SE

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

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16168-NP, REVISION 2, "RISK-INFORMED EXTENSION OF THE

REACTOR VESSEL IN-SERVICE INSPECTION INTERVAL"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

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

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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.

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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, forging, axial weld, and circumferential welds. These individual TWCF contributions were then weighted based on pilot plant data and summed to determine a total RV TWCF.

3.1 Define the Proposed Change

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

The 20 year inspection interval is a maximum interval and the PWROG did not request, and the NRC staff does not endorse, that all RV inspections 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).

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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

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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

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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).

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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

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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.

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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,

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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

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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,"

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(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 9.37E-10/year, 1.81E-8/year, and 1.26E-8/year for BV1 (Westinghouse-designed plant), Palisades (CE-designed plant), and OC1 (B&W-designed plant), respectively. These increases are well below the guideline for a very small increase in LERF of 1E-7/year in RG 1.174.

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

3.3 Implementation and Monitoring

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

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

The number of transient cycles that were utilized in the fatigue crack growth analysis was discussed in Section 3.2.1 of this 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 95th percentile $TWCF_{TOTAL}$[4] for the plant requesting to implement the pilot plant study is less than the 95th percentile $TWCF_{TOTAL}$ from the pilot plant study. The 95th

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 final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a) 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.

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.

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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 final 10 CFR 50.61a (or Alternative fracture toughness requirements for protection against pressurized thermal shock, in Enclosure 1 to the proposed rulemaking in SECY-07-0104, Reference 12, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a). Licensees that do not implement 10 CFR 50.61a must amend their licenses to require that the information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 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 alternative.

Licensees that implement 10 CFR 50.61a must perform the ISIs 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 10 CFR 50.61a must perform the inspections and analyses required by Section (e) of 10 CFR 50.61a prior to implementing the extended interval.

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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

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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 10 CFR 50.61a must amend their licenses to require that the information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 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 NRC 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 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).

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 final 10 CFR 50.61a (or Alternative fracture toughness requirements for protection against pressurized thermal shock, in Enclosure 1 to the proposed rulemaking in SECY-07-0104, Reference 12, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a). Licensees that do not implement 10 CFR 50.61a must amend their licenses to require that the information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 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 alternative.

Licensees that implement 10 CFR 50.61a must perform the ISIs 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 10 CFR 50.61a must perform the inspections and analyses required by Section (e) of 10 CFR 50.61a prior to implementing the extended interval.

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

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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).

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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," October 31, 2003 (ADAMS Accession No. ML051790410)

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. ML042610469 and ML042610375)

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. Letter Report, Oak Ridge National Laboratories/TM-2007/0030, "Fracture Analysis of Vessels"(FAVOR Code, Version 06.1)

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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 (ADAMS Accession No. ML042880482)

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

Attachment: Resolution of PWROG Comments on Draft SE

Principle Contributors: Barry Elliott
Stephen Dinsmore

Date: May 8, 2008

RESOLUTION OF PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

COMMENTS ON DRAFT SAFETY EVALUATION (SE) FOR TOPICAL REPORT (TR)

WCAP-16168-NP, REVISION 2, "RISK-INFORMED EXTENSION OF THE

REACTOR VESSEL IN-SERVICE INSPECTION INTERVAL"

(TAC NO. MC9768)

By letter dated March 31, 2008, the PWROG provided thirteen comments on the draft SE for TR WCAP-16168-NP, Revision 2. The following are the NRC staff's resolution of these comments. To ensure consistency when discussing the final and proposed rule within the SE, the NRC staff has made one additional change, noted at Number 14.

1. Page 3, Lines 19-21

PWROG Comment:

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

NRC Response:

The NRC staff agrees with this change.

2. Page 15, Line 41

PWROG Comment:

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

NRC Response:

The NRC staff agrees with this change.

3. Page 17, Lines 11-18

PWROG Comment:

The draft SE requires that the qualified vessel inspection results be evaluated per the existing requirements in Section (e) of 10 CFR 50.61a in Enclosure 1 of SECY-07-0104, Reference 12. It is requested that the SE be revised to state that the requirements of Section (e) in Enclosure 1 of SECY -07 -0104 should only be used until the applicable requirements in the final

ATTACHMENT

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version of 10 CFR 50.61a are published in the Federal Register. The following revision is recommended, "By monitoring flaw sizes in accordance with the criteria described in Section (e) of the proposed rulemaking in SECY-07-0104, or the final published version of 10 CFR 50.61a, licensees will ensure...."

NRC Response:

While the NRC staff agrees with the intent of the requested change, the NRC staff does not agree the revised wording accomplishes the intent. The NRC staff has made the following change: "By monitoring flaw sizes in accordance with the criteria described in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a) licensees will ensure that their RVs do not have flaws that invalidate the results of the PWROG PFM analyses."

4. Page 18, Lines 27-34

PWROG Comment:

The draft SE requires that the qualified vessel inspection results be evaluated per the existing requirements in Section (e) of 10 CFR 50.61a in Enclosure 1 of SECY-07-01 04, Reference 12. It is requested that the SE be revised to state that the requirements of Section (e) in Enclosure 1 of SECY-07-0104 should only be used until the applicable requirements in the final version of 10 CFR 50.61a are published in the Federal Register. The following revisions are recommended, "...in Enclosure 1 to the proposed rulemaking in SECY-07-01 04, Reference 12, or the final published version of 10 CFR 50.61 a." and "...and analyses requested in Section (e) of the proposed rulemaking in SECY-07-0104, or the final published version of 10 CFR 50.61a, will be submitted...."

NRC Response:

While the NRC staff agrees with the intent of the requested change, the NRC staff does not agree the revised wording accomplishes the intent. The NRC staff has made the following change: "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 final 10 CFR 50.61a (or Alternative fracture toughness requirements for protection against pressurized thermal shock, in Enclosure 1 to the proposed rulemaking in SECY-07-0104, Reference 12, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a). Licensees that do not implement 10 CFR 50.61a must amend their licenses to require that the information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 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 alternative."

5. Page 18, Lines 41-44

PWROG Comment:

It is stated in the draft SE that: "Licensees also implementing Section (c) of the proposed

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

NRC Response:

The NRC staff has made the following change: "Licensees that implement 10 CFR 50.61a must perform the ISIs 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 10 CFR 50.61a must perform the inspections and analyses required by Section (e) of 10 CFR 50.61a prior to implementing the extended interval."

6. Page 20, Lines 15-17

PWROG Comment:

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

NRC Response:

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

7. Page 21, Lines 21-28

PWROG Comment:

The draft SE requires that the qualified vessel inspection results be evaluated per the existing requirements in Section (e) of 10 CFR 50.61a in Enclosure 1 of SECY-07-0104, Reference 12. It is requested that the SE be revised to state that the requirements of Section (e) in Enclosure 1 of SECY-07-0104 should only be used until the applicable requirements in the final version of 10 CFR 50.61a are published in the Federal Register. The following revisions are recommended, "...in Enclosure 1 to the proposed rulemaking in SECY-07-0104, Reference 12, or the final published version of 10 CFR 50.61a." and "...and analyses requested in Section (e)

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of the proposed rulemaking in SECY-07-0104, or the final published version of 10 CFR 50.61 a, wil be submitted...."

NRC Response:

While the NRC staff agrees with the intent of the requested change, the NRC staff does not agree the revised wording accomplishes the intent. The NRC staff has made the following change: "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 final 10 CFR 50.61a (or Alternative fracture toughness requirements for protection against pressurized thermal shock, in Enclosure 1 to the proposed rulemaking in SECY-07-0104, Reference 12, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a). Licensees that do not implement 10 CFR 50.61a must amend their licenses to require that the information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 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 alternative."

8. Page 21, Lines 35-38

PWROG Comment:

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

NRC Response:

The NRC staff has made the following change: "Licensees that implement 10 CFR 50.61a must perform the ISIs 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 10 CFR 50.61a must perform the inspections and analyses required by Section (e) of 10 CFR 50.61a prior to implementing the extended interval."

9. Page 23, Line 27

PWROG Comment:

The date and Agencywide Documents Access and Management System (ADAMS) Accession number for Revision 1 of Reference 11 are October 31, 2003, and ML051790410, respectively.

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NRC Response:

The NRC staff agrees with this change.

10. Page 23, Line 35

PWROG Comment:

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 structural reliability and risk assessment tool, respectively. It is recommended that the SE be revised to include these accession numbers for Reference 13.

NRC Response:

The NRC staff agrees with this change.

11. Page 23, Line 42

PWROG Comment:

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 SE.

NRC Response:

The NRC staff agrees with this change.

12. Page 24, Line 11

PWROG Comment:

For Reference 22, the ADAMS Accession Number is ML042880482. It is recommended that this accession number be added to the SE.

NRC Response:

The NRC staff agrees with this change.

13. Page 24, Line 13

PWROG Comment:

Reference 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 SE: "23. Letter Report, "Estimate of External Events Contribution to Pressurized Thermal Shock (PTS) Risk," October 1, 2004 (ADAMS Accession No. ML042880476)"

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NRC Response:

The NRC staff agrees with this change.

14. Page 20, Lines 41-44

NRC Comment:

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

NRC Response:

The NRC staff has made the following change: "In addition, licensees that do not implement 10 CFR 50.61a must amend their licenses to require that the information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a) will be submitted for NRC staff review and approval."

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PWR Owners Group
Member Participation* for PWROG Project MUHP-5097, MUHP-5098, MUHP-5099, and CEOG Task 2059.

Utility Member	Plant Site(s)	Participant	
		Yes	No
AmerenUE	Callaway (W)	X	
American Electric Power	D.C. Cook 1&2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)	X	
Constellation Energy Group	Calvert Cliffs 1 & 2 (CE)	X	
Constellation Energy Group	Ginna (W)	X	
Dominion Connecticut	Millstone 2 (CE)	X	
Dominion Connecticut	Millstone 3 (W)	X	
Dominion Kewaunee	Kewaunee (W)	X	
Dominion VA	North Anna 1 & 2, Surry 1 & 2 (W)	X	
Duke Energy	Catawba 1 & 2, McGuire 1 & 2 (W), Oconee 1, 2, 3 (B&W)	X	
Entergy	Palisades (CE)	X	
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)	X	
Entergy Operations South	Arkansas 2, Waterford 3 (CE), Arkansas 1 (B&W)	X	
Exelon Generation Co. LLC	Braidwood 1 & 2, Byron 1 & 2 (W), TMI 1 (B&W)	X	
FirstEnergy Nuclear Operating Co	Beaver Valley 1 & 2 (W), Davis-Besse (B&W)	X	
Florida Power & Light Group	St. Lucie 1 & 2 (CE)		X
Florida Power & Light Group	Turkey Point 3 & 4, Seabrook (W)	X	
Florida Power & Light Group	Pt. Beach 1&2 (W)	X	
Luminant Power	Comanche Peak 1 & 2 (W)	X	
Nuclear Management Company	Prairie Island 1&2	X	
Omaha Public Power District	Fort Calhoun (CE)		X
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)	X	
Progress Energy	Robinson 2, Shearon Harris (W), Crystal River 3 (B&W)	X	
PSEG - Nuclear	Salem 1 & 2 (W)	X	

Utility Member	Plant Site(s)	Participant	
		Yes	No
Southern California Edison	SONGS 2 & 3 (CE)	X	
South Carolina Electric & Gas	V.C. Summer (W)	X	
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)	X	
Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2 (W)	X	
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	X	

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

PWR Owners Group
International Member Participation* for PWROG Project MUHP-5097, MUHP-5098, MUHP-5099, and CEOG Task 2059.

Utility Member	Plant Site(s)	Participant	
		Yes	No
British Energy	Sizewell B	X	
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3	X	
Hokkaido	Tomari 1 & 2 (MHI)	X	
Japan Atomic Power Company	Tsuruga 2 (MHI)	X	
Kansai Electric Co., LTD	Mihama 1, 2 & 3, Ohi 1, 2, 3 & 4, Takahama 1, 2, 3 &4 (W & MHI)	X	
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 Yonggwang 1 & 2 (W)	X	
Korea Hydro & Nuclear Power Corp.	Yonggwang 3, 4, 5 & 6 Ulchin 3, 4 , 5 & 6(CE)	X	
Kyushu	Genkai 1, 2, 3 & 4, Sendai 1 & 2 (MHI)	X	
Nuklearna Electrarna KRSKO	Krsko (W)	X	
Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)	X	
Ringhals AB	Ringhals 2, 3 & 4 (W)	X	
Shikoku	Ikata 1, 2 & 3 (MHI)	X	
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2 (W)	X	
Taiwan Power Co.	Maanshan 1 & 2 (W)	X	
Electricite de France	54 Units	X	

* **This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending documents to participants not listed above.**

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LIST OF ACRONYMS AND ABBREVIATIONS

ADV	Atmospheric dump valve
AFW	Auxiliary feedwater
ART	Adjusted reference temperature
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
BV1	Beaver Valley Unit 1
CCDP	Conditional core damage probability
CDF	Core damage frequency
CE	Combustion Engineering
ECT	Eddy current examination
EFPY	Effective full-power year
EOL	End of life
EPRI	Electric Power Research Institute
FENOC	FirstEnergy Nuclear Operating Company
FCG	Fatigue crack growth
FP	Failure probability
FSAR	Final Safety Analysis Report
GQA	Graded quality assurance
HPI	High-pressure injection
HUCD	Heat-up and cool-down transient
HZP	Hot-zero power
IEF	Initiating event frequency
IGSCC	Intergranular stress corrosion cracking
ID	Inner diameter
ISI	In-service inspection
IST	In-service testing
LBLOCA	Large-break loss-of-coolant accident
LERF	Large early release frequency
LOCA	Loss-of-coolant accident
MBLOCA	Medium-break loss-of-coolant accident
MSIV	Main steam isolation valve
MSLB	Main steam line break
MT	Magnetic particle examination
NDE	Non-destructive examination
NMC	Nuclear Management Company
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OC1	Oconee Unit 1
OD	Outer diameter
ORNL	Oak Ridge National Laboratory
PFM	Probabilistic fracture mechanics
PNNL	Pacific Northwest National Laboratory
POD	Probability of detection

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

PRA	Probabilistic risk assessment
PT	Liquid penetrant examination
PTS	Pressurized thermal shock
PVRUF	Pressurized Vessel Research User Facility
PWR	Pressurized water reactor
PWROG	PWR Owners Group
QA	Quality Assurance
RAI	NRC Request for Additional Information
RCP	Reactor coolant pump
RCS	Reactor Coolant System
RG	NRC Regulatory Guide
RI-ISI	Risk-informed ISI
RPV	Reactor pressure vessel
RT_{NDT}	Reference nil-ductility transition temperature
RV	Reactor vessel
RV ISI	Reactor Vessel In-service Inspection
RVID	Reactor vessel integrity database
SBLOCA	Small-break loss-of-coolant accident
SER	NRC Safety Evaluation Report
SG	Steam generator
SRP	Standard Review Plan
SRRA	Structural Reliability and Risk Assessment
SRV	Safety and relief valve
SSC	Structures, systems, and components
TH	Thermal hydraulics
TWCF	Through Wall Cracking Frequency
UT	Ultrasonic examination
VT	Visual examination

EXECUTIVE SUMMARY

The current requirements for the inspection of reactor vessel pressure-containing welds have been in effect since the 1989 Edition of *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code*, Section XI, as supplemented by Nuclear Regulatory Commission (NRC) Regulatory Guide 1.150. The manner in which these examinations are conducted has recently been augmented by Appendix VIII of Section XI, 1996 Addenda, as implemented by the NRC in an amendment to 10CFR50.55a effective November 22, 1999. The industry has expended significant cost and man-rem exposure that have shown no service-induced flaws in the reactor vessel (RV) for ASME Section XI Category B-A or B-D RV welds.

The objective of the methodology discussed in this report is to provide the technical basis for decreasing the frequency of inspection by extending the Section XI Inspection interval from the current 10 years to 20 years for ASME Section XI Category B-A and B-D RV nozzle welds. Specific pilot studies have been performed on the Westinghouse, Combustion Engineering, and Babcock and Wilcox reactor vessel and NSSS designs. The results show that the change in risk associated with eliminating all inspections after the initial 10-year in-service inspection satisfies the guidelines specified in Regulatory Guide 1.174 for an acceptable change in risk for large early release frequency (LERF).

This conclusion is applicable to all Westinghouse, Combustion Engineering, and Babcock and Wilcox reactor vessel designs given that the applicable individual plant parameters are bounded by the critical parameters identified in Appendix A.

1 INTRODUCTION

The current requirements for the inspection of reactor vessel (RV) pressure containing welds have been in effect since the 1989 Edition of *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code*, Section XI [1], as supplemented by the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.150 [2]. The manner in which these examinations are conducted has been augmented by Appendix VIII of Section XI, 1996 Addenda, as implemented by NRC in an amendment to 10CFR50.55a effective November 22, 1999 [3]. The industry has expended significant cost and man-rem exposure by performing the required examinations that have shown no service-induced flaws in the RV for ASME Section XI Category B-A or B-D RV nozzle welds. The current code criteria for the selection of examination areas and the frequency of examinations is not be an effective way to expend inspection resources.

The objective of this study was to verify that a reduction in 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 an evaluation of the change in risk associated with extending the 10-year in-service inspection (ISI) interval for three pilot plant bounding cases based on the calculated difference in the frequency of RV failure. RV failure was defined for this study to be the extension of a crack all the way through the RV wall. The difference in frequency of RV failure was evaluated using RG 1.174 [4] to determine if the values met the specified regulatory guidelines. 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 a plant-specific extension in the RV weld inspection interval.

This study followed the approach specified in ASME Code Case N-691 [5], which provides guidelines for using risk-informed insights to increase the inspection interval for pressurized water reactor (PWR) vessel welds.

2 BACKGROUND

The original objective of the ASME B&PV Code, Section XI [1] ISI program was to assess the condition of pressure-containing components in nuclear power plants to ensure continued safe operation. If non-destructive examination (NDE) found indications that exceeded the allowable standards, examinations were extended to additional welds in components in the same examination category. If NDE found indications that exceeded the acceptance standards in those welds, then the examinations were extended further to similar welds in similar components, etc.

With respect to the method defined in this report, 100 percent of the present examination areas will be retained. The methodology is limited to justification of a reduction in the frequency of examination, i.e., increasing the time interval between inspections.

The original examination interval of 10 years was based on "wear-out" rate experience in the pre-nuclear utility and petrochemical process industries. As with some other Section XI ISI requirements, with no indications being found in the vessel welds under evaluation in this report, these inspections are decreasing in value with increasing industry experience to rely upon. The U.S. NRC has granted a number of exemptions to inspections for other areas and components (e.g., piping [6], reactor coolant pump motor flywheels [7], etc.) based on experience and man-rem reductions. This has been attributed to the combined design, fabrication, examination, and Quality Assurance (QA) rigor of the nuclear codes, and more careful control of plant operating parameters by the utilities.

A critical component of the justification of the interval extension is a fracture mechanics evaluation of the reactor vessel, which shows that flaws, if they do exist, would not grow to a critical size if the inspection interval is increased to more than 10 years. This can be demonstrated by selecting critical areas of the reactor vessel for the evaluation such as, the beltline, flange, and outlet nozzle regions. These locations are known to be areas of primary concern and are currently considered in ASME Section III, Appendix G [1] evaluations for protection against nonductile failure of the reactor vessel. As part of this study, a deterministic fracture mechanics evaluation of limiting locations in a typical geometry for a RV identified that the beltline region was the critical location with respect to the potential for growth of fatigue cracks. Fatigue crack growth is recognized as the primary degradation mechanism in the carbon and low alloy steel components in PWR Nuclear Steam Supply System (NSSS), that could contribute to any potential growth of existing flaws in the component base materials and weld metals.

Fatigue can be defined as repeated exposure to cyclic loading resulting from a variety of operating conditions or events (e.g., heatups, cooldowns, reactor trips). Design basis documents provided descriptions of the conditions that would contribute to cyclic fatigue. This information was used to identify and define the frequency of occurrence for each of the events that was considered when determining the potential for fatigue crack growth.

A technical consideration critical to success was the application of risk-informed assessment techniques to substantiate the deterministic fracture mechanics flaw growth evaluation. Risk assessment techniques provided a means to quantify and calculate cumulative results from contributing mechanisms and uncertainties associated with the critical parameters. A probabilistic fracture mechanics (PFM) methodology was used to consider the distributions and uncertainties in flaw numbers, flaw sizes, fluence, material properties, crack growth rate, stresses, and the effectiveness of inspections. The PFM methodology was also used to calculate the change in the frequency of RV failure due to a change in inspection interval. This change in RV failure frequency was used to evaluate the viability of such an inspection interval change. Recognized guidelines for evaluating the change in failure frequencies are provided in RG 1.174 [4] and the NRC risk assessment developed in conjunction with the current pressurized thermal shock (PTS) evaluations [8, 9].

Significant work is on-going in the nuclear industry to investigate the impacts from PTS or "off-normal" plant transients that may be outside the current design basis. These transients are commonly understood to present the most severe challenge to RV structural integrity. The NRC effort to address PTS has identified FirstEnergy Nuclear Operating Company's (FENOC's) Beaver Valley Unit 1 (BV1), Nuclear Management Company's (NMC's) Palisades, and Duke Energy's Oconee Unit 1 (OC1) as the representative plants based on geometry and embrittlement for the Westinghouse, Combustion Engineering (CE), and Babcock and Wilcox (B&W) PWR designs. These are the primary PWR manufacturers in the U.S. and were evaluated by the NRC and Oak Ridge National Laboratory (ORNL) as part of the NRC PTS Risk Study [8, 9].

This report summarizes the results from an evaluation of the extension of the inspection of ASME Section XI [1] Examination Category B-A and B-D welds in the reactor pressure vessel (RPV) from the current requirement of every 10 years to an extension of 20 years. It demonstrates that for the pilot plant reactor vessel geometry and fabrication history, any potential change in risk when the inspection interval is extended meets the change in risk evaluation guidelines defined in RG 1.174 [4]. The evaluation documented in this report considers FENOC's BV1 as the Westinghouse pilot plant. NMC's Palisades Plant and Duke Energy's OC1 are the respective Combustion Engineering (CE) and Babcock and Wilcox (B&W) pilot plants for this evaluation. To apply the results of this report to non-pilot plants, it must be shown, using the tables contained in Appendix A that the pilot plant evaluations for the respective design bound the non-pilot plant.

The following paragraphs address the current Section XI ISI requirements for PWR RV welds under consideration for the proposed extension. The following topics are included:

1. Reactor Vessel In-Service Inspection (RV ISI)
2. Location-specific ISI data from participating plants
3. The man-rem exposure and other costs of RV weld inspection
4. Generic RV weld experience at various plants
5. Development of the ISI interval extension methodology
6. Pilot plants
7. Safety impact

2.1 REACTOR VESSEL IN-SERVICE INSPECTION

Since its beginning, ASME B&PV Code, Section XI [1] has required inspections of weld areas of reactor vessels and other pressure-containing nuclear system components. The selection of inspection locations was based on areas known to have high-service factors and additional areas to provide a representative sampling for the condition of pressure-containing nuclear system components. While weld and adjoining areas were specified, it was recognized that the volumetric examination of the weld and adjoining base material would result in a significant degree of examination of the base metal.

Examination Volumes

Initially, for longitudinal and circumferential welds in a reactor vessel shell, Section XI required examination of 10 percent of the length of longitudinal welds, and 5 percent of the length of circumferential welds. Welds receiving exposure in excess of specified neutron fluence would require an inspection of 50 percent of the length. The 1977 Edition of Section XI increased the examination of RV welds from 5 or 10 percent of the length to 100 percent, with all welds examined in the first 10-year interval. Subsequent intervals required 100 percent examination of specified circumferential and longitudinal welds. The 1989 Edition of Section XI [1] extended the examination to include all welds.

There has been no report of structural failure or leakage from any full-penetration weld being addressed in this report in a PWR RV shell, globally. In volumetric examinations of these welds in ISIs performed in accordance with the requirements of Section XI (and RG 1.150 [2]), flaws identified in the original construction have been detected and were acceptable under Section XI requirements. These flaws have been monitored and to date, no growth has been identified. There has been no evidence of in-service flaw initiation in these welds.

Examination Approaches

The preceding discussion of RV welds addresses the Category B-A, RV seam welds of Table IWB-2500-1 of Section XI. Category B-D, RV nozzle welds and nozzle inner radius are also included in this evaluation.

The ultrasonic examinations (UTs) of these RV welds, as of the 1996 Addenda of Section XI, were conducted in accordance with Appendix I, I-2110. This Addenda requires Appendix VIII inspections for:

- Shell and head welds excluding flange welds
- Nozzle-to-vessel welds
- Nozzle inside radius region

Precedent for Change

There have been a number of revisions (often by ASME Code Case) to the Section XI ISI program that have eliminated or reduced the extent of examinations and tests based on successful operating experience and analytical evaluation. Examples of ASME Code Cases applicable to the RV and its piping connections include:

N-481 [10] Associated with cast austenitic pump casings. This was the first example of substituting an analysis plus a visual examination (VT) for a volumetric examination, for a Class 1 component.

N-560 [11] Permits a reduction in the examination of Class 1 Category B-J piping welds from 25 to 10 percent, provided a specified risk-importance ranking selection process is followed. This was a substantive reduction of an established Class 1 examination.

N-577 [12] N-578 [13]	Provide requirements for risk-informed ISI of Class 1, 2, and 3 piping. The cases provide different methods to achieve the same objective. This was the first use of the plant probabilistic risk assessment (PRA). Both methods have received extensive implementation in the U.S. and in several other countries in Europe and Asia.
N-613 [14]	Reduces the examination volume of Category B-D nozzle welds in adjacent material from 1/2 shell thickness to 1/2 inch. This permits a significant reduction in qualification and scanning time.
N-552 [15]	Permits computational modeling for the qualification of nozzle inner radius examination techniques, in lieu of qualification on a multitude of configurations.
N-610 [16]	Permits a K_{IR} curve in Appendix G, in lieu of a K_{IA} curve. Indirectly, this is beneficial to the pressure-temperature limit curve during plant startup.

Not all of the changes in Section XI, due to operating considerations, have led to a relaxation in inspection or evaluation requirements.

Over the past 10 years, there have also been a number of changes (often by code case) to the Section XI ISI program that have increased the extent of examinations and tests based on operating experience and analytical evaluation. The following examples of ASME Code Cases are limited to those applicable to the RV and its piping connections.

N-409 [17]	Introduced procedure and personnel qualification requirements for UT of intergranular stress corrosion cracking (IGSCC) in austenitic piping welds, a precursor to Appendix VIII, UT performance demonstration requirements.
N-512 [18]	Provided requirements for the assessment of RVs with low upper shelf Charpy impact energy levels.
N-557 [19]	Introduced requirements for in-place dry annealing of a PWR RV.

2.2 LOCATION-SPECIFIC ISI DATA FROM PARTICIPATING PLANTS

While it is known that the number of flaws found in RPV welds is very small, it is important to relate their number to the number of welds that have been examined over the past 30 years with no evidence of the development of service-induced flaws.

To develop location-specific ISI data from nuclear plants, ISI data on the RV weld categories noted above were gathered in a survey [20]. This information focused on service-induced flaws. It did not address the detection of original fabrication flaws, unless the flaws had grown due to service conditions. The response to this survey is summarized in Table 2-1.

Table 2-1 Summary of Survey Results on RV ISI Findings [20]							
No. of Plants	**Total Years of Service Prior to Survey**	**ASME Weld Category / Item**	**No. of Welds in Category**	**Welds with No Flaws**	**Welds with Flaws**	**Means of Detection[1]**	**Cause of Flaw/Failure**
14	301	**B-A**					
		Shell, B1.10	112	112	0		
		Head. B1.20	105	105	0		
		Shell-to-flange. B1.30	16	16	0		One plant reported 3 indications that may be just scratches.
		Head-to-flange, B1.40	16	16	0		One plant reported 3 indications that may be just scratches.
		B-D					
		Nozzle-to-shell, B3.90	102	102	0		
		Nozzle inside radius B3.100	102	102	0		
		B-F					
		Dissimilar metal, B.5.10	84	84	0		
		B5.30	32	32	0		
		B-K					
		Welded attach, B10.10	4	4	0		
		B-N					

Table 2-1 Summary of Survey Results on ISI Findings [20] (cont.)

No. of Plants	Total Years of Service Prior to Survey	ASME Weld Category	No. of Welds in Category	Welds with No Flaws	Welds with Flaws	Means of Detection[1]	Cause of Flaw/Failure
		Vessel interior, B13.10	34	34	0		
		Interior attach.-beltline, B13.50	6	6	0		
		Other interior attach., B13.60	53	53	0	VT-3, UT, ECT	One plant reported crack arrest holes drilled in core barrel.
		Core support struct., B13.70	41	5	0		

Note 1: VT = Visual Inspection, UT = Ultrasonic Inspection, ECT = Eddy Current Inspection

2.3 EXPOSURE AND COST REDUCTION

Data was gathered on CE and Westinghouse plants related to the cost of a typical RV ISI outage, as well as the cost of the exposure affecting the involved personnel [20]. The objective of this effort was to investigate the exposure and financial aspects of the RV ISI. The results of the survey were tabulated based on the probability of a life extension program (60 years), and the potential savings were calculated with regards to a proposed extension of the RV ISI interval to 20 years. The radiation exposure cost is contingent on the utility and is typically $15,000 to $20,000 per man-rem. A summary of the results is presented in Table 2-2.

Table 2-2 Savings on the Proposed Extension of RV ISI Interval from 10-Years to 20-Years (Per Plant) [20]

Probability of 20-Year Life Extension (%)		0%	50%	100%
Cost of Typical RV ISI Outage, $	min	506,410	759,615	1,012,820
	max	7,680,000	9,600,000	11,520,000
	average	3,878,521	5,391,656	7,115,317
Dose of Exposure, Man-rems	min	0.2	0.4	0.6
	max	6.5	9.75	13.0
	average	1.66	2.32	2.98
Cost of Dose of Exposure, $	min	2,492	4,984	7,476
	max	65,000	97,500	130,000
	average	20,611	28,856	37,101

As shown in Table 2-2, the savings associated with even the most conservative assumption, i.e., no life extension program (40 years) for any of the surveyed plants, are significant. The extension of the RV ISI interval to 20 years will save every unit an average of $3,878,521 for the cost of the outage, and 1.66 man-rems of exposure.

The saving values associated with the less conservative assumption of the guaranteed life extension program (60 years) for any of the surveyed plants are considerably higher. The extension of the RV ISI interval to 20 years will save every unit an average of $7,115,317 for the cost of outage, and 2.98 man-rems of exposure. The critical path outage time for RV inspections is approximately 3 ½ days. While this data was gathered for Westinghouse and CE designed plants, the savings for B&W designed plants are expected to be similar.

2.4 GENERIC REACTOR VESSEL WELD EXPERIENCE AT VARIOUS PLANTS

Section XI ISI requirements developed in the early 1970s were based on the detection of fatigue cracking in primary welds. This has not been substantiated by subsequent operating experience. Fatigue cracking in primary welds has not been a problem. Random sampling for the assessment of condition of pressure-containing components has not been effective; when leakage and other deterioration have been identified, it has been by examinations other than the Section XI ISI NDE.

Primary system failures/leakage have almost always been associated with dissimilar metal welds or control rod drive, bottom mounted instrumentation, or vent connections of the RV and its head. The latter connections are all partial penetration welds. They were not included in the survey, since the current effort does not propose to recommend changes to their present ISI interval requirements. Their examinations are not contingent on the removal of the reactor internals and the use of the RV inspection tool. Category B-F dissimilar metal welds, Category B-K welded attachments, and Category B-N interior attachment and support welds were not included in the inspection interval extension.

In many plants, the most highly stressed reactor vessel weld is the weld between the closure head flange and the dome. There have been no reports of degradation of this joint. This joint ranks quite low in its contribution to cumulative risk determined through typical PFM methods. Calculations [21] have shown that flaw growth due to fatigue would be extremely small, so that even pre-existing flaws that clearly exceed the acceptance standards would not be subject to measurable growth.

2.5 DEVELOPMENT OF ISI INTERVAL EXTENSION METHODOLOGY

The ISI interval extension methodology is primarily based on a risk analysis, including a PFM analysis of the effect of different inspection intervals on the frequency of reactor vessel failure due to postulated PTS transients. Reactor vessel failure is defined for the purposes of this study as the point which a crack has extended all the way through the RV wall. The likelihood of reactor vessel failure is 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 reactor vessel toughness due to irradiation. Credible, postulated PTS transients that could potentially lead to reactor vessel failure must be considered to occur at the worst time in the life of the plant. The PFM methodology allows the consideration of distributions and uncertainties in flaw number and size, fluence, material properties, crack growth rate, stresses, and the effectiveness of inspections. The PFM approach leads to a conditional reactor vessel failure frequency due to a given loading condition and a prescribed inspection interval. All locations of interest in the reactor vessel can be addressed in a similar way or, as in the case of this study, a bounding approach can be used to minimize the areas receiving a detailed evaluation.

A feasibility study was performed [20] that showed that this fracture mechanics and risk methodology can be used to calculate the change in the frequency of reactor vessel failure due to a change in inspection interval and to evaluate the acceptability of the associated change in risk. The impact on plant safety from the change in risk presented in this study was based on the standards for risk-informed assessment as defined by RG 1.174 [4].

3 PILOT PLANT SUMMARY

The risk evaluations summarized in this report utilized the same pilot plants as used in the NRC PTS Risk Re-evaluation effort [8, 9]. The NRC effort to address PTS risk identified FirstEnergy Nuclear Operating Company's (FENOC's) Beaver Valley Unit 1 (BV1), Nuclear Management Company's (NMC's) Palisades, and Duke Energy's Oconee Unit 1 (OC1) as the pilot plants. These pilot plant applications also used fleet-specific design transient data for the Combustion Engineering (CE) and Westinghouse designs. A typical generic heatup/cooldown transient was used for the Babcock & Wilcox (B&W) study. A study was also performed to determine the bounding location from among the applicable weld locations on a typical PWR reactor vessel. The results of all of these investigations are included in the following sections.

3.1 BOUNDING LOCATION

The focus of the evaluations for reactor vessel inspection interval extension was on the beltline of the RV. To confirm that the beltline location represented the bounding location for the reactor vessel, all locations currently required for examination in the reactor pressure vessel (RPV) needed to be identified and considered. The beltline weld locations were found to be the bounding locations primarily due to irradiation induced change in the fracture toughness. This was consistent with the location assumptions used to support the NRC PTS Risk Study [8, 9]. Table 3-1 summarizes the current ISI requirements for RPV inspection as identified in Table IWB-2500-1 of the ASME B&PV Code, Section XI [1]. While this table identifies all welds with Section XI inspection requirements, this report only addresses the ISI interval extension of the Category B-A and B-D welds.

Table 3-1	ASME Section XI [1] ISI Requirements for RPVs (ASME Section XI, Table IWB-2500-1)		
	Item No.	**RPV Location**	**Examination Requirement**
		Pressure Retaining Welds in Reactor Vessel	
B-A	B1.10	Shell Welds	Volumetric
B-A	B1.11	Circumferential	Volumetric
B-A	B1.12	Longitudinal	Volumetric
B-A	B1.20	Head Welds	Volumetric
B-A	B1.21	Circumferential	Volumetric
B-A	B1.22	Meridional	Volumetric
B-A	B1.30	Shell-to-Flange Weld	Volumetric
B-A	B1.40	Head-to-Flange Weld	Surface and Volumetric
B-A	B1.50	Repair Welds	Volumetric
B-A	B1.51	Beltline Region	Volumetric
		Full Penetration Welded Nozzles in Vessels	
B-D	B3.90	RPV Nozzle-to-Vessel Welds	Volumetric
B-D	B3.100	RPV Nozzle Inside Radius Section	Volumetric
		Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles	
B-F	B5.10	RPV Nozzle-to-Safe End Butt Welds, NPS 4 or Larger	Surface and Volumetric
B-F	B5.20	RPV Nozzle-to-Safe End Butt Welds, Less Than NPS 4	Surface
B-F	B5.30	RPV Nozzle-to-Safe End Socket Welds	Surface
		Pressure Retaining Welds in Piping	
B-J	B9.10	NPS 4 or Larger	Surface and Volumetric
B-J	B9.11	Circumferential Welds	Surface and Volumetric
		Welded Attachments for Vessels, Piping, Pumps and Valves	
B-K	B10.10	Welded Attachments	Surface
		Interior of Reactor Vessel	
B-N-1	B13.10	Vessel Interior	Visual, VT-3
		Welded Core Support Structures and Interior Attachments to Reactor Vessels	
B-N-2	B13.50	Interior Attachments within Beltline Region	Visual, VT-1
B-N-2	B13.60	Interior Attachments Beyond Beltline Region	Visual, VT-3
		Removable Core Support Structures	
B-N-3	B13.70	Core Support Structure	Visual, VT-3

To confirm that the beltline was the limiting location, an assessment was performed using deterministic fracture mechanics that considered the following:

- Existence of 10-percent through-wall initial flaw
- In-service fatigue crack growth of the flaw due to normal plant operating transients
- 40 EFPY embrittlement throughout plant life
- Peak reactor vessel ID fluence assumed regardless of flaw depth, i.e., maximum embrittlement
- Design basis heat-up and cool-down transients
  - 500 cycles/40 years for CE NSSS
  - 200 cycles/40 years for Westinghouse NSSS
- 7 Weld Locations
  - Closure Head to Flange
  - Upper Shell to Flange
  - Lower Shell Transition
  - Bottom Head to Shell
  - Beltline
  - Inlet Nozzle to Safe End
  - Outlet Nozzle to Safe End

The study evaluated the effect of various ISI intervals by comparing the change in margins on ASME Code allowable flaw sizes for the respective locations. This approach was preceded by considering 3 iterative steps:

1. Select the first inspection interval, I1, based on the growth of the assumed initial flaw to a fraction of the tolerable flaw size.
2. Perform the inspection. If no defects larger than the assumed flaw size are found, the second inspection interval, I2, is the same as the first.
3. Continue subsequent inspections until actual flaws are detected that require repair or augmented inspections.

The results of the study are summarized in Figures 3-1 and 3-2. Inspection intervals were based on 10-, 20-, 30-, or 40-year inspection intervals over a 40-year plant life. Each reactor vessel location was evaluated by calculating the amount of crack extension that would occur due to fatigue crack growth over a 10-year period of operation. Each crack length was then evaluated for the maximum applied K_I from a transient. The ratio of the maximum allowable K_I, per the ASME Section XI [1] Appendix A criteria, to the maximum K_I applied, was used as a measure of the margin a flaw in a given location has to the acceptance criteria. Note that in Figure 3-1 the margins on the acceptance standard are greater than 1, except for the beltline region axial and circumferential flaws. This indicates that all of the flaw sizes in other locations are acceptable with varying degrees of margin. The margin less than one for the beltline locations is an indication that the assumed initial flaw size of 10-percent throughwall was greater than the acceptable flaw size. The other feature to note in Figures 3-1 and 3-2 is that, for each subsequent 10-year period that was evaluated, there was an insignificant change in the degree of margin for all of the locations. This observation was simply a reflection of the fact that the increments of fatigue crack growth of the flaws were so small that the applied K_I values were not changing. Therefore, the ratios of the applied to allowable K_I did not change. Though not shown in figures 3-1 and 3-2, the reactor vessel nozzle to shell weld was also evaluated and found to be have a margin greater than that of the reactor vessel beltline axial and circumferential welds.

These results confirmed that the beltline was the limiting location and that the change in fatigue crack growth increment for RPV flaws was insignificant relative to the inspection interval. While a specific number of design basis heat-up and cool-down transients was not analyzed for B&W designs in this bounding location assessment, it is reasonable to expect that the conclusions of this assessment would also be applicable to B&W plants due to similarities in the RV and NSSS designs.

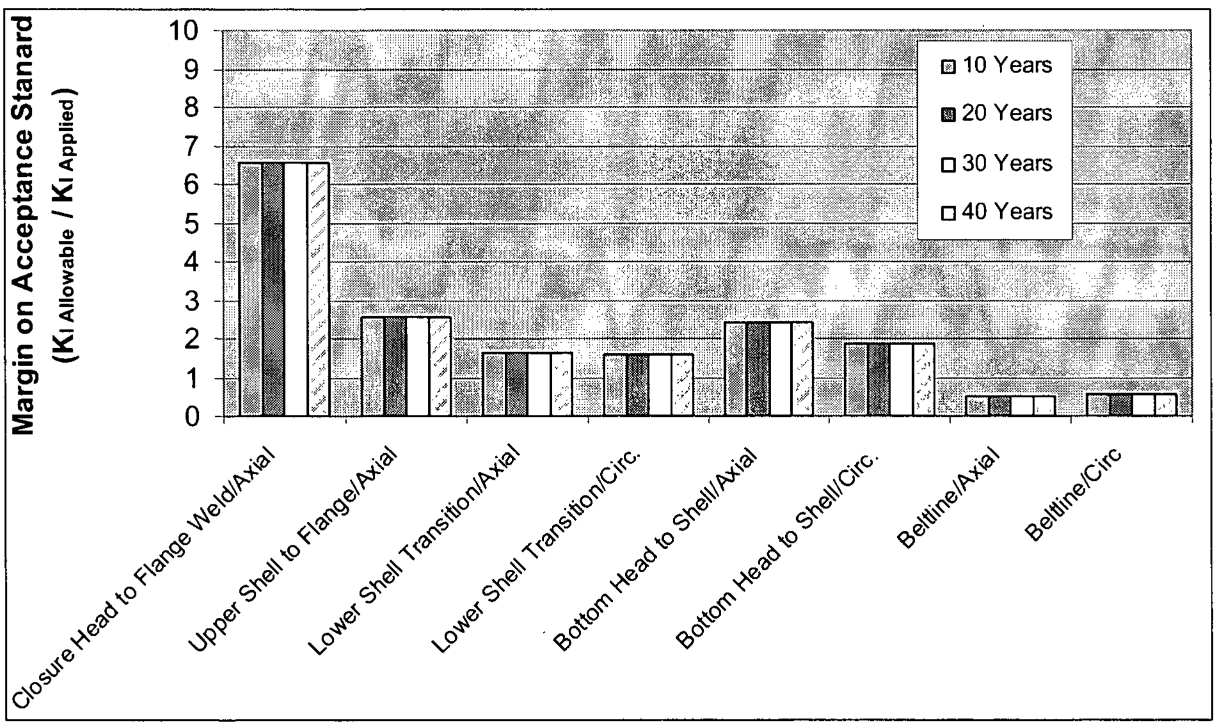

Figure 3-1 Comparison to Acceptance Criteria – Minimum Margins Code Allowable

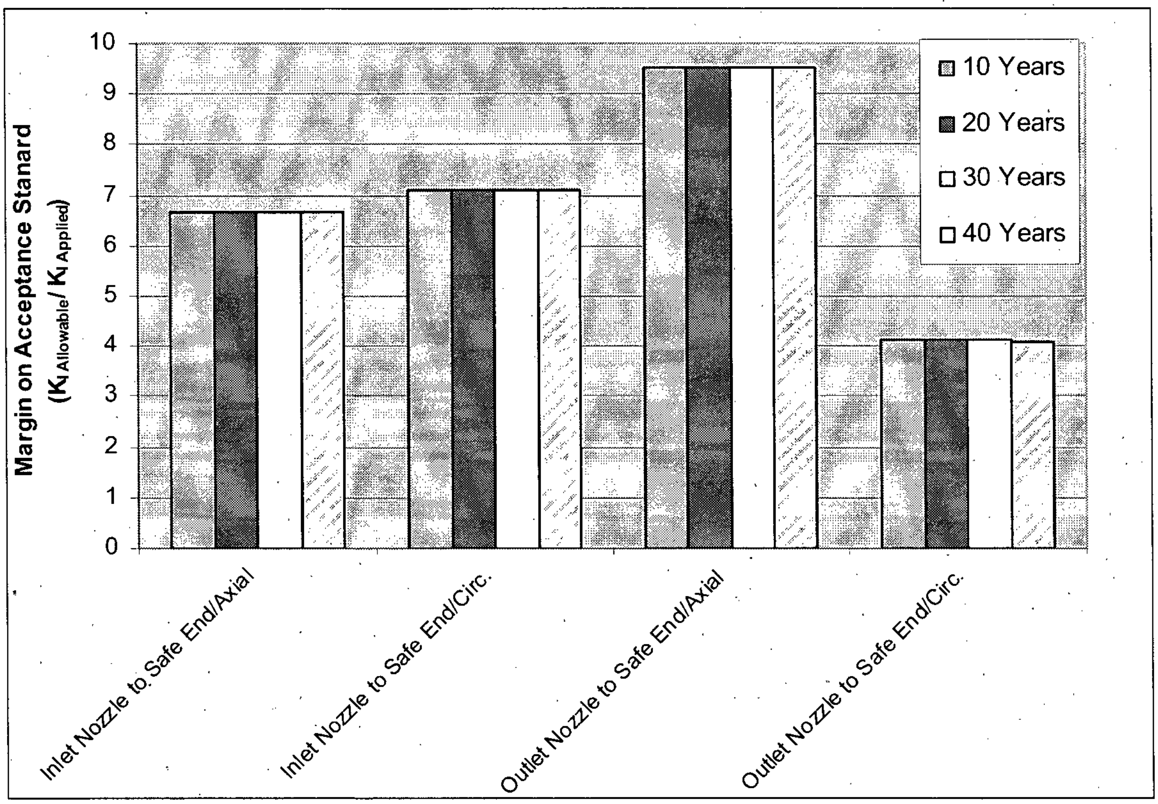

Figure 3-2 Comparison to Acceptance Criteria – Minimum Margins Code Allowable

3.2 BASIS FOR RISK DETERMINATION

As indicated in ASME Code Case N-691 [5], the application of risk-informed insights from PFM and risk analyses can be used to justify an increase from 10 to 20 years in the requirements of Section XI, IWB-2412 for the inspection interval for the examination of Category B-A and B-D welds in PWR reactor vessels. The guidelines in Regulatory Guide 1.174 provide the basis for an acceptable change in risk resulting from an extension in inspection interval. As the basis for determining the change in risk, the inputs to the RV PFM and risk analyses included the following:

Accident Transients and Frequency

ASME Code Case N-691 [5] states that it is necessary to define a complete set of accident transients that can be postulated to realistically result in RV failure and their frequencies of occurrence. As previously mentioned, PTS events are viewed as providing the greatest challenge to PWR RPV structural integrity. For this reason, the pilot plant applications in this report used the PTS transients and frequencies from the NRC PTS Risk Study [8, 9]. As part of the NRC study, probabilistic risk assessment (PRA) models were developed for each of the pilot plants using plant specific information [22, 23, 24]. These PRA models included an event-tree analysis that defined both the sequences of events that are likely to produce a PTS challenge to RPV structural integrity and the frequency with which such events can be expected to occur. The typical sequence of concern was cool-down and depressurization due to the initiating event, followed by repressurization due to high-pressure safety injection or charging. Historically, a small-break loss-of-coolant accident (SBLOCA) with low decay heat has been the sequence identified as a major contributor to PTS risk. However, other events considered included a large break in the main steam line upstream of the main steam isolation valves, a double-ended main steam line break (MSLB) upstream of the main steam isolation valves (MSIVs), small steam line break downstream of the MSIVs, and excessive feedwater flow, all with the reactor coolant pump (RCP) shutdown and multiple failures of the operator to take remedial action.

The PTS Risk Study utilized the plant specific 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 that could later be analyzed using thermal-hydraulic codes. This resulted in 178 binned sequences for OC1, 118 for BV1, and 65 for Palisades. Thermal-hydraulic analyses were performed for each of these bins (i.e., representative transients) to develop time histories of temperature, pressure, and heat transfer coefficients [25]. These histories were then input into the PFM analysis to determine conditional probability of reactor vessel failure for each transient. From this analysis, it was determined that only a portion of the transients contribute to the total risk of RPV failure, while the remainder 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 this report. Consistent with the PTS Risk Study, 61 transients were analyzed for BV1, 30 for Palisades, and 55 for OC1 in this study on the impact of extending the RV ISI interval. Details of the transients are provided in Appendix D for BV1, Appendix H for Palisades, and Appendix L for OC1.

As part of the NRC PTS Risk Reevaluation Program, a study was performed to determine the applicability of the pilot plant detailed analyses to the remainder of the domestic PWR fleet. This "Generalization" Study [26] examined the results from the three detailed pilot plant studies (BV1, Palisades, and OC1) and identified a set of plant design and operational features considered to be important in determining whether or not certain types of overcooling scenarios are significant contributors to PTS. These features were then analyzed for five additional plants and compared to the features of the pilot plants. These five plants included the following:

- Salem Unit 1 (Westinghouse 4-loop plant comparable to Beaver Valley Unit 1)
- TMI Unit 1 (B&W plant comparable to Oconee Unit 1)
- Fort Calhoun (CE plant comparable to Palisades)
- Diablo Canyon (Westinghouse 4-loop plant comparable to Beaver Valley Unit 1)
- Sequoyah Unit 1 (Westinghouse 4-loop plant comparable to Beaver Valley Unit 1)

They were chosen for the generalization study on the basis of:

- having a high reference temperature metric (RT_{PTS}), which reflects their potential sensitivity to PTS,
- further demonstrating the applicability of the pilot plant analyses to the remainder of the fleet for the nuclear steam supply system (NSSS) vendors, and
- including plants having different limiting materials (i.e., welds, plates, and forgings).

It was determined in the generalization study that there were no differences in plant features that from a PRA, thermal hydraulic, and PFM standpoint would be expected to cause significant differences in the through wall cracking frequencies due to the postulated PTS scenarios. It was further concluded through the generalization study that the pilot plant results at a comparable embrittlement level could be applied to the remainder of the domestic PWR fleet.

Operational Transients and Cycles

ASME Code Case N-691 [5] states that the operational transients that contribute to fatigue crack growth and the number of cycles occurring each year must be identified. Typically, the start-up (heat-up) and shut-down (cool-down) events are the dominant loading conditions as seen in ASME Code Section XI, Non-Mandatory Appendix A [1] calculations for fatigue crack growth of an existing flaw.

For the purpose of the pilot plant studies in this report, 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. The design basis transients for the pilot plants were reviewed and it was determined that the greatest contributor to fatigue crack growth for the pilot plants is heat-up and cool-down. 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, and thus envelopes many transients with a smaller range of conditions. For the pilot plant evaluations, 7 heat-up and cool-down cycles per year were used for Westinghouse plants (BV1) and 13 cycles were used for CE plants (Palisades) to bound all the design basis transients for the respective PWR plant designs in each fleet. Based upon available information, 12 cycles were used for Babcock and Wilcox plants. For any B&W plant using the results of this WCAP to extend the reactor vessel ISI interval from 10 to 20 years, including the pilot plant (OC1), the fatigue crack growth for 12 heatup/cooldown transients per year will have to be verified to bound the fatigue crack growth for all design basis transients.

It is important to note that most plants' operational histories indicate that they will not reach this number of design transients by end of life (EOL) (80 years). However, this calculation was performed as a bounding analysis and the number of design transients was used rather than the number of operational transients so that plants with operational histories different than those of the pilot plants would be enveloped.

Initial Flaw Distribution

ASME Code Case N-691 [5] requires credible flaw distributions for a PWR reactor vessel. Significant work by Pacific Northwest National Laboratory (PNNL) and the NRC was performed to more completely specify the initial flaw size distributions and their densities for input into the NRC PTS Risk Study [8, 9]. This work focused on making detailed destructive and non-destructive measurements of fabrication flaws in nuclear grade RPV welds and plates. Whenever possible, this experimental evidence was used exclusively or given the greatest "weight" in establishing the flaw distributions. In cases where experimental evidence was not sufficient, physical models and expert opinion were used to supplement the experimental evidence in establishing the flaw distributions. For the NRC PTS Risk Study, flaw distributions were developed for embedded flaws in welds, plates (includes forgings), and inner surface breaking flaws.

The weld flaw distribution was based on the highest densities of the Shoreham reactor vessel and the largest sizes of the PVRUF vessel. The embedded flaws are distributed evenly through the thickness of the weld. Flaws are postulated only in the same orientation as the weld. The flaw distribution represents a blended combination of weld types with 2% of the welds assumed to be repair welds, which have the largest flaw sizes.

Empirical evidence to support a plate flaw distribution is much more limited than that for welds. For this reason, the density for flaws of depths less than 6mm is 10% of that for weld flaws, while the density for flaws of depth above 6mm is 2.5% of that for weld flaws. Half of the simulated flaws are assumed to be axially oriented while the other half are assumed to be circumferentially oriented.

For weld and plate flaws, the pilot plant studies for the RV ISI interval extension study used the flaw distributions from the NRC PTS Risk Study directly. These densities are input into the FAVOR Code PFM analyses as flaw density files, P.dat (plate-embedded flaws) and W.dat (weld-embedded flaws). This is discussed further in the "PFM Computer Tool and Methodology" section.

The inner-diameter of the RPV is clad with a thin layer of stainless steel. Lack of inter-run fusion can occur between adjacent weld beads, resulting in circumferentially oriented cracks (the cladding in the RV is deposited circumferentially). However, none of the cracks discovered in the PNNL studies had broken through the cladding layer on the inside surface of the RV. Therefore, for the NRC PTS Risk Study [8, 9], the BV1 and Palisades evaluations used multi-pass cladding with no surface breaking flaws. Multi-layer cladding is assumed to have no surface breaking flaws due to the small likelihood of two flaws aligning in two different weld layers. The OC1 pilot evaluation used an assumed surface flaw completely through the cladding with a density of 1/1000th of the embedded flaws through the vessel wall.

For this investigation on the impact of extending the RV ISI interval it is important to consider the effects of fatigue crack growth. Due to the fact that embedded flaws do not grow significantly due to fatigue, for the pilot plant studies, the presence of surface breaking flaws with an initial flaw depth equal to the cladding thickness was postulated. Therefore, for the pilot plant evaluations to bound all the plants of the same design, single-pass cladding was conservatively assumed. The initial flaw size and distribution was input into a fatigue crack growth and ISI analysis to determine a surface flaw density file after any inspections (ISI). Surface flaw density files were created two simulate two cases. 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 file S.dat. The methodology for determining the flaw depth and density included in this file is described in the section on PFM and Computer Tool Methodology. Cladding details for the pilot plants are identified in Appendices B, F, and J.

Fluence Distribution

ASME Code Case N-691 [5] requires that the fluence distribution versus operating time, both axial and azimuthal, be based on plant-specific or bounding data for the current operating time and extrapolated as applicable to the end of the current 40 year license or for license renewal to 60 years.

For the pilot plant evaluations in this report, the input fluence distributions were taken directly from the NRC PTS Risk Study [8, 9]. For the NRC PTS Risk Study a series of neutron transport calculations were performed to determine the neutron fluence on the inner-wall of the pilot plant RPVs. The modeling procedures were based on the guidance contained in NRC Reg. Guide 1.190[27]. The models incorporated pilot plant specific geometry and operating data. The fluence for E>1MeV was calculated as a function of the azimuthal and axial location in the inner reactor vessel wall. The fluence was extrapolated from the current state point to various effective full-power years (EFPYs) assuming a linear extrapolation of the most recent operating cycles.

The fluences used in the RV ISI interval extension evaluations were for 60 EFPY for BV1 and Palisades and for fluences at 500 EFPY for OC1 to envelope license extension. 500 EFPY were used for OC1 rather than 60 EFPY because it is recognized that it is not the most embrittled 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 when evaluated against the parameters identified in Appendix A. Representative fluence maps for BV1, Palisades, and OC1 at 32 EFPY, can be found in Appendices B, F, and J, respectively. While the magnitude of the fluence on these maps correspond to 32 EFPY rather than the 60 EFPY and 500 EFPY used in the pilot plant evaluations, the contour of the fluence relative to the reactor vessel weld layout still applies.

Material Fracture Toughness

ASME Code Case N-691 [5] states that the material fracture toughness of the limiting beltline plates and weld materials need to be based on the following plant-specific data:

- Physical and mechanical properties of the base metal, clad, and welds (e.g., copper and nickel content) and their uncertainties.
- Initial reference nil-ductility transition temperature (RT_{NDT}), including uncertainty
- ΔRT_{NDT} due to radiation embrittlement ,versus time and depth, including uncertainty
- Fracture toughness versus time and depth, including uncertainty

These reactor vessel material properties for the BV1, Palisades, and OC1 pilot plants evaluated in this report are identified in Appendices B, F, and J, respectively.

Embrittlement due to irradiation in RPV steels occurs due to matrix hardening and age hardening [8, 9]. Based on the physical insights into these hardening mechanisms a relationship between material composition, irradiation-condition variables, and measurable quantities such as yield strength increase, Charpy-transition-temperature shift, and toughness-transition-temperature shift was established for the NRC PTS Risk Study [8, 9]. Furthermore, a quantitative relationship was developed from the database of Charpy shift values generated in domestic reactor surveillance programs. The Eason and Wright irradiation shift model was developed by fitting this data. This model is used in the FAVOR Code [28] for the NRC PTS Risk Study and the RV ISI interval extension pilot plant studies to calculate the shift and irradiated reference temperature as a function of time.

The results of the significant work at ORNL, the NRC, and within industry to more completely specify the distribution on fracture toughness and its uncertainty for the NRC PTS Risk Study [8, 9] are included in the FAVOR Code which is used for the pilot plant studies for RV ISI interval extension. The FAVOR Code includes fracture toughness models which are based on extended databases of empirically obtained K_{Ic} and K_{Ia} data points and include the effects of the statistical bias for direct measurement of fracture toughness (Master Curve Method). Furthermore, the FAVOR Code [28] uses the latest correlation on irradiated upper shelf fracture toughness.

It should be noted that along with the inspection of a weld, there is a specified amount of base metal inspected. In the FAVOR Code evaluation, if a flaw is placed within a weld that is adjacent to a more highly embrittled plate, the flaw is assigned the embrittlement characteristics of the plate rather than the weld and is assumed to fracture and propagate in the direction of the plate.

The NRC has proposed that through wall cracking frequency (TWCF) can be correlated to the embrittlement index (reference temperature) of the reactor vessel components. The correlation for determining plant specific TWCF based on the plant specific data mentioned can be found in Reference 9. This correlation takes into consideration the contribution to TWCF for each of the most limiting plate, axial weld, and circumferential welds. These individual TWCF contributions are then weighted based on experimental pilot plant data and summed to determine a total reactor vessel TWCF. For application to other plant reactor vessels, the plant specific TWCF must be equal to or less than the values used for the applicable pilot plants evaluated in this report (see Appendix A) at 60 EFPY.

Crack Growth Rate Correlation

ASME Code Case N-691 [5] requires that the basic physical models for fatigue crack growth due to operational transients (e.g., heat-ups, cool-downs, normal plant operating changes, and reactor trips) including the effects of uncertainties, be used for the PFM analysis. Also used are the basic physical models for crack growth during these transient events (i.e., the change in applied stress intensity and the corresponding change in flaw size) for the surface breaking flaws and their uncertainties.

The pilot-plant studies in this report included a probabilistic representation of the fatigue crack growth correlation for ferritic materials in water that was consistent with the previous and current models contained in Appendix A of the ASME Code, Section XI [1]. These correlations represented the behavior of the ferritic reactor vessel materials for all domestic PWRs. This probabilistic representation was consistent with that used by the NRC-supported pc-PRAISE code [29] and the NRC-approved SRRA tool for piping-risk informed ISI [30].

Cladding and Residual Stresses

ASME Code Case N-691 [5] requires that the residual stress distribution in welds and the cladding stress and its temperature dependence due to differential thermal expansion be considered. For the pilot plant studies for RV ISI interval extension, the residual stress distribution through the wall was taken from the NRC PTS Risk Study [8, 9] and is described in the FAVOR Code Theory Manual [28]. This distribution is shown in Figure 3-3. The stress profile was determined for the NRC PTS Risk Study thorugh experiments in which a radial slot was cut in a longitudinal weld in a shell segment from an actual RPV and the deformation of the slot was measured after cutting. Finite element analysis was used to determine the residual stress profile from the measured deformations. 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 [8, 9].

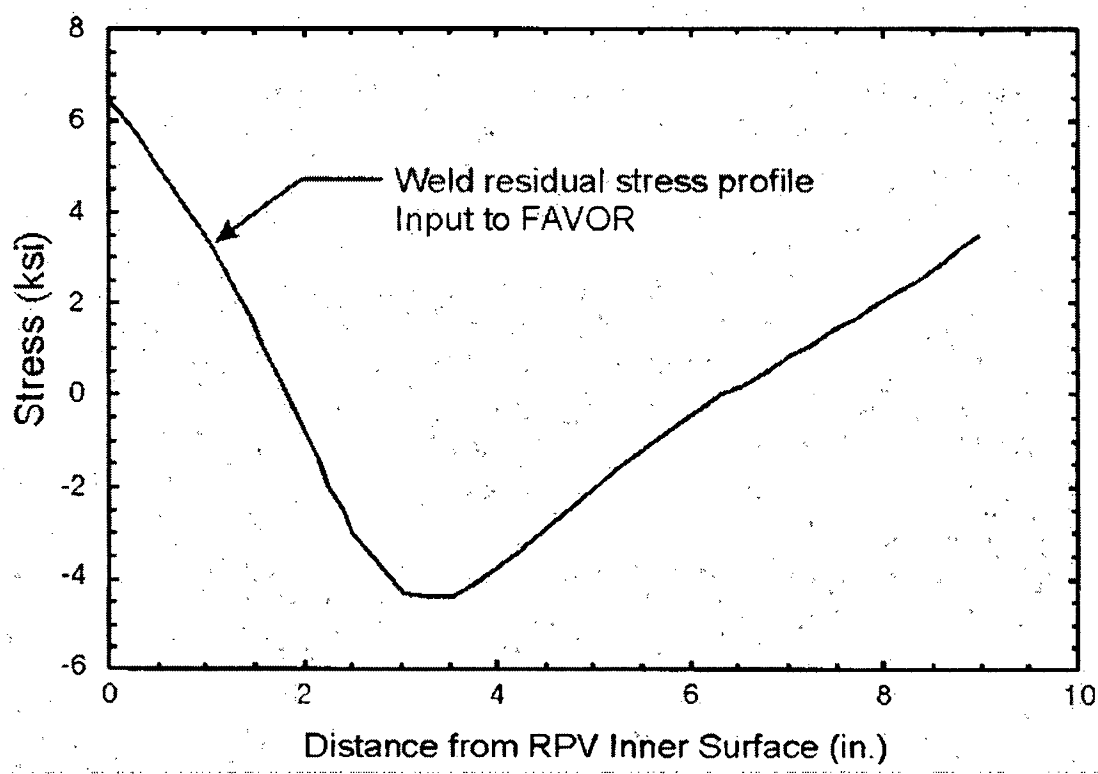

Figure 3-3 Weld Stress Profile

Effectiveness of ISI

The essential requirement for an effective volumetric examination in ASME Code Case N-691 [5] is that it be conducted in accordance with Section XI Appendix VIII [1] or RG 1.150 [2].

The following effects also need to be considered along with the change in ISI interval:

- Extent of inspection (percent coverage)
- Probability of detection (POD) with flaw size
- Repair criterion for removing flaws from service

The POD should correlate to the respective examination method for the RV weld of interest.

The basis for the probability of flaw detection used in the pilot plant studies for the RV ISI interval extension was taken from studies performed at the EPRI NDE Center on the detection and sizing qualification of ISIs on the RV beltline welds [31]. Figure 3-4 shows the probability of detection with respect to flaw size used in the pilot studies in this report.

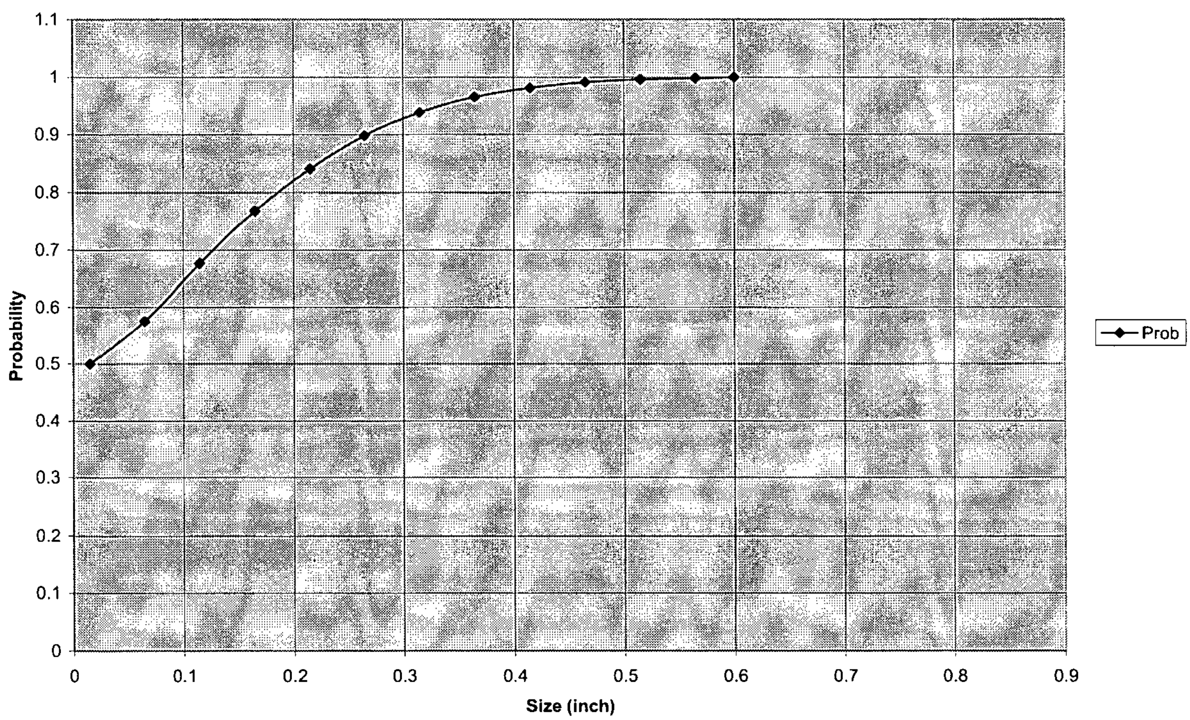

Figure 3-4 ISI Detection Probability

For the pilot plant evaluations, examinations were assumed to be conducted in accordance with Section XI Appendix VIII [1], so that Figure 3-4 could be used. 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 (e.g. ISI per Regulatory Guide 1.150) 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. Therefore, the pilot plant studies conservatively calculated a larger potential difference in risk by maximizing the benefits of inspection.

Impact of Other ASME Code Cases on RPV Inspection

While no ASME Code Cases have been found that directly overlap the actions included in ASME Code Case N-691 [5], there are related ASME Code Cases and "problem areas" that may affect implementation of the Code Case. ASME Code Cases that concern reactor vessel inspections but do not affect the applicability of the Code Case are identified in the following:

ASME Code Case N-697 [32] addresses Examination Requirements for PWR Control Rod Drive and In-Core Instrumentation Housing Welds. It adds requirements for examination of in-core instrumentation housing welds greater than 2" Nominal Pipe Size to Examination Category B-O. If these UT or surface examinations of the housing weld inner surface were conducted from inside the RPV, they could result in examination intervals incompatible with effective implementation of N-691 [5]. However, these welds are not inspected from inside the RPV and, therefore, there is no impact.

A top priority in Section XI is to work with the Material Reliability Program Alloy 600 Issue Task Group to identify and incorporate changes needed in the examination of affected partial penetration and dissimilar metal welds. This could result in incompatible examination intervals for Examination Category B-F welds to reactor vessel nozzles, and dissimilar metal welds in Examination Category B-J not covered by Category B-F. A possible approach for some plants, where access permits, would be to examine these welds from the pipe outer diameter (OD) at alternate 10-year intervals, and from the inner diameter (ID) during the Case N-691 [5] examinations.

ASME Code Case N-700 [33] addresses Examination Category B-K, surface examination of welded attachments. It permits examination of a single welded reactor vessel attachment each inspection interval.

ASME Code Case N-648-1 [34] permits a VT-1 visual examination of a reactor vessel nozzle inner radius in lieu of a volumetric examination. Applicability of this Code Case would not be affected by the increased examination interval.

ASME Code Case N-624 [35] provides for modification of the sequence of successive examinations. The increased examination interval would be applicable.

ASME Code Case N-623 [36] permits deferral to the end of the interval of shell-to-flange and head-to-flange welds of a reactor vessel. The methodology of Case N-691 [5] would not be affected by application of this Code Case.

ASME Code Case N-615 [37] permits ultrasonic examination as a surface examination method for Category B-F and B-J piping welds of 4" Nominal Pipe Size and larger. It would be compatible with the increased examination interval.

ASME Code Case N-613-1 [38] reduces the nozzle weld examination volume of Examination Category B-D. It would be compatible with the increased examination interval.

ASME Code Case N-598 [39] provides alternatives to the required percentages of examinations each inspection period. ASME Code Case N-691 [5] would increase the length of the inspection period but would not affect the percentage requirements.

Probabilistic Fracture Mechanics Computer Tool and Methodology

For the pilot-plant applications of the PFM methodology, the failure frequency distributions for all postulated flaws in the RV were calculated using the latest version (06.1) of the FAVOR code [28]. The **F**racture **A**nalysis of **V**essels – **O**ak **R**idge (FAVOR) computer program was developed as part of the NRC PTS Risk Study [8, 9]. It is a program that performs a probabilistic analysis of a nuclear reactor pressure vessel when subjected to events in which the reactor pressure vessel wall is exposed to time-varying thermal-hydraulic boundary conditions.

To run the FAVOR code, 3 modules (FAVLOAD, FAVPFM and FAVPOST) and various input files were required as shown in Figure 3-5. In the NRC PTS Risk Study [8, 9], the effects of fatigue crack growth and ISI were not considered. However, to perform the risk evaluation for changing the inspection interval from 10 to 20 years, these effects were quantified. Program PROBSBFD (Probabilistic Surface Breaking Flaw Density) was developed to include these effects by modifying the surface-breaking flaw input file to FAVOR (S.dat) as shown in Figure 3-5.

The first module in FAVOR is the load module, FAVLOAD, where the thermal-hydraulic time histories are input for the dominant PTS transients. For each PTS transient, deterministic calculations are performed to produce a load-definition input file for FAVPFM (FAVPFS is also used in this analysis). These load-definition files include 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 (both infinite and finite-length).

The FAVPFS module in Figure 3-5 is a modification of the FAVPFM module, which is the second module contained in the FAVOR code that was used in the NRC PTS risk study. The modification allows FAVPFS to have a 4 times finer depth distribution for surface breaking flaws in S.dat. The FAVPFS FAVOR module uses the input flaw distributions (e.g., S.dat, W.dat, and P.dat), the loads for the PTS events from the FAVLOAD module and fluence/chemistry input data at 60 EFPY (effective full-power years) to calculate the initiation and failure probabilities for each PTS transient.

The FAVPOST post-processor is the third module in FAVOR. It combines the distributions of initiating frequencies for the dominant PTS transients with the results of the PFM analysis (performed with the FAVPFS module) to generate probability distributions for the frequencies of reactor vessel crack initiation and reactor vessel failure. This module also generates statistical information on these distributions and the distributions for the conditional probabilities of reactor vessel crack initiation and failure for each PTS transient included in the risk analysis.

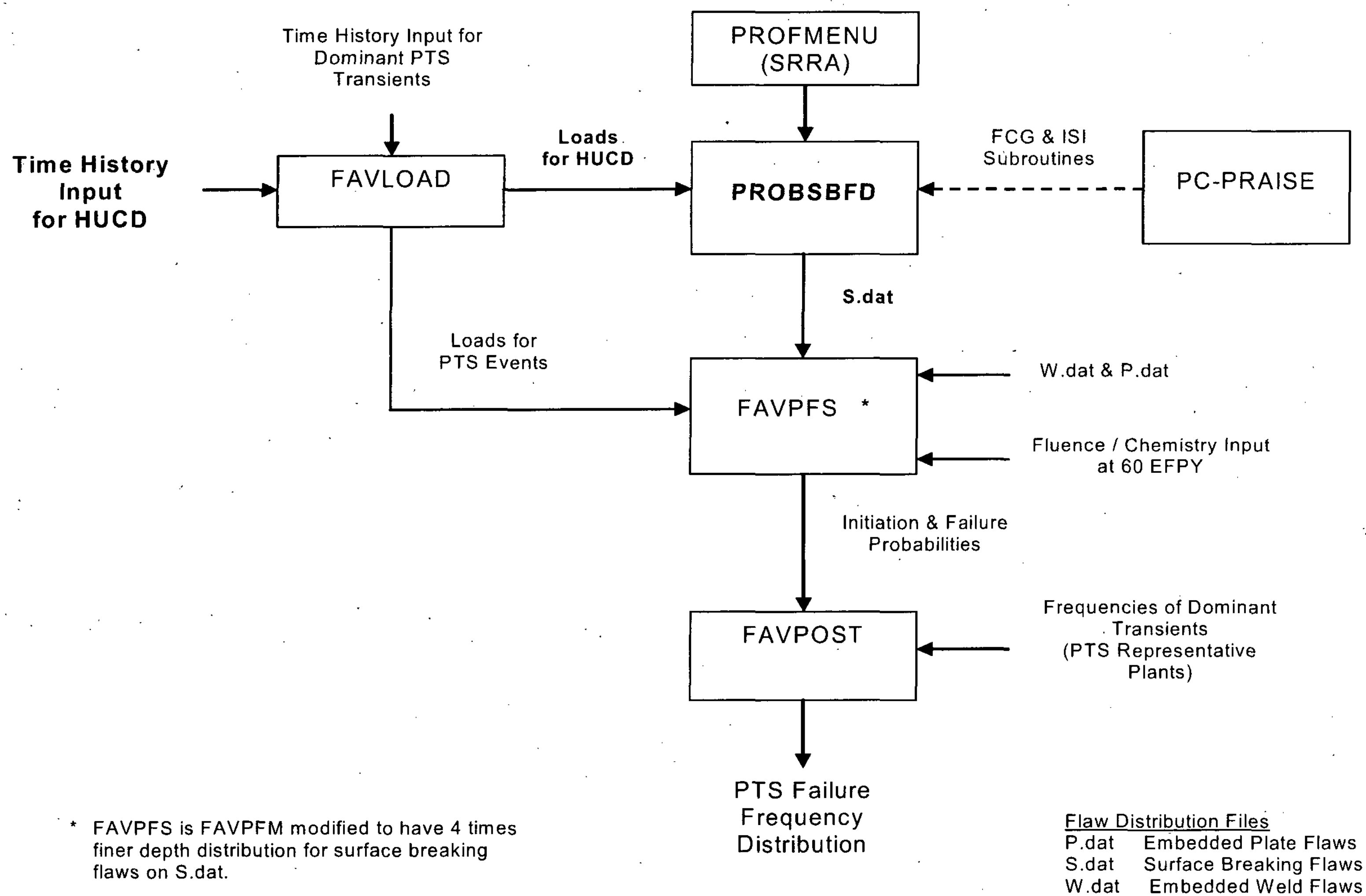

Figure 3-5 Software and Data Flow for Pilot Plant Analyses

The PROBSBFD code was specifically developed for the RV ISI interval extension project and verified in accordance with the Westinghouse Quality Assurance requirements. This program utilizes the Westinghouse Structural Reliability and Risk Assessment (SRRA) library program, which provides standard input and output, including probabilistic analysis capabilities (e.g., random number generation and importance sampling). PROBSBFD was used to develop 1000 random surface breaking flaw distributions that fed into the FAVPFS module via an input file (S.dat is the default name). The loads were determined using the FAVLOAD module, for the input with time histories of temperature, pressure, and heat transfer characteristics for the operational transients (e.g., heat-up and cool-down) that could grow the initial flaws by means of fatigue. The applied stress intensity factor (K) at various times and various depths through the reactor vessel wall were taken directly from the FAVLOAD output file and input into PROBSBFD (FAVLOADS.dat for PROBSBFD).

The beneficial effects of ISI were modeled in the same way as in the NRC's probabilistic analysis code pc-PRAISE [29] and the SRRA Code [30] used with the PWROG/ASME piping risk-informed in-service inspection (RI-ISI) program. Specifically, only the flaws not detected during an ISI exam, at 10 years for example, remained. For example, if the probability of detection for the first inspection was 90 percent, then the flaw density was effectively multiplied by 10 percent for input to the next iteration. The effects of subsequent inspections, where the probability of detection was increased because the flaw was bigger (see Figure 3-4), could be either cumulative or independent.

For each of the 1000 simulations performed by PROBSBFD, the initial flaw depth and density were defined. Four aspect ratios, 2, 6, 10, and infinite, were considered. For each time-step and flaw-aspect ratio, the effects of ISI, the stress intensity factors, and the random crack growth were calculated. After all the time steps were completed, the distribution of flaw densities by depth and aspect ratio were written to a surface-breaking, flaw-distribution input file for FAVPFS, which was in the same format as the default S.dat file (see Figure 3-5).

3.3 RESULTS FOR THE WESTINGHOUSE PILOT PLANT: BV1

Reactor vessel failure frequencies were calculated for BV1 for two cases corresponding to the two surface flaw density files discussed in the section on "Initial Flaw Distribution". These cases were referred to as "ISI Every 10 Years" and "10-year ISI Only". As the names imply, the "ISI Every 10 Years" case simulates the current ASME Code required inspections while the "10-year ISI Only" case simulates a discontinuation of inspections after the first 10-year ISI. Statistically, the difference between the mean failure frequencies for the "ISI Every 10 Years" case and the "10-year ISI Only" case is insignificant. This is due to the fact that the difference between the mean values is less than the standard error for each of the cases. However, to calculate a change in risk for comparison to regulatory guidelines, a change in failure frequency was conservatively calculated based on the difference between an "Upper Bound" and a "Lower Bound." The Lower Bound was determined by subtracting 2 times the standard error as reported by FAVPOST from the mean value of the "ISI Every 10 Years" case. The Upper Bound was determined by adding 2 times the standard error as reported by FAVPOST to the mean value of the "10-Year ISI Only" case.

Elimination of ISI after the first 10-year ISI for the BV1 RPV results in a difference in failure (through-wall flaw) frequency of less than 1E-09. A summary of the results of the evaluation are included in Table 3-2. The results reflect the maximum statistically calculated value for the potential change in risk at a number of reactor vessel simulations at which the Monte Carlo statistical analysis has reached a stable solution. The difference between the Upper Bound and Lower Bound represents the bounding difference between the 10-year inspection interval currently applicable under ASME criteria and elimination of all future inspections following an inspection within the first 10 years of operation.

This change in failure frequency is acceptable per the regulatory guidance discussed in Section 4.1. Transient input was based on design basis transients and the transients used in the NRC PTS Risk Study [8, 9]. The input data included consideration of postulated life extension to 60 EFPY. The FAVPOST outputs for the cases presented in Table 3-2 are presented in Appendix E.

Table 3-2 BV1 Reactor Vessel Failure Frequency Results

10-Year ISI Only (Mean Value / Standard Error)	5.04E-09 / 2.54E-10
Upper Bound Value	5.55E-09
ISI Every 10 Years (Mean Value / Standard Error)	5.23E-09 / 3.12E-10
Lower Bound Value	4.61E-09
Bounding Difference in Risk	9.4E-10

The mean effects of fatigue crack growth and ISI on the surface breaking flaw density for 1000 simulations are shown in Figures 3-6 and 3-7. These figures plot the flaw density as a function of the flaw depth for the cases of one initial 10-year ISI, a 10-year ISI interval, and a 20-year ISI interval. These plots display the results for the 10-to-1 and infinite aspect ratio sizes. The PROBSBFD outputs used to generate these plots are included in Appendix C. The crack growth and density reduction due to ISI would both be reduced for the flaw length-to-depth aspect ratios of 2-to-1 and 6-to-1 also considered in the pilot plant study.

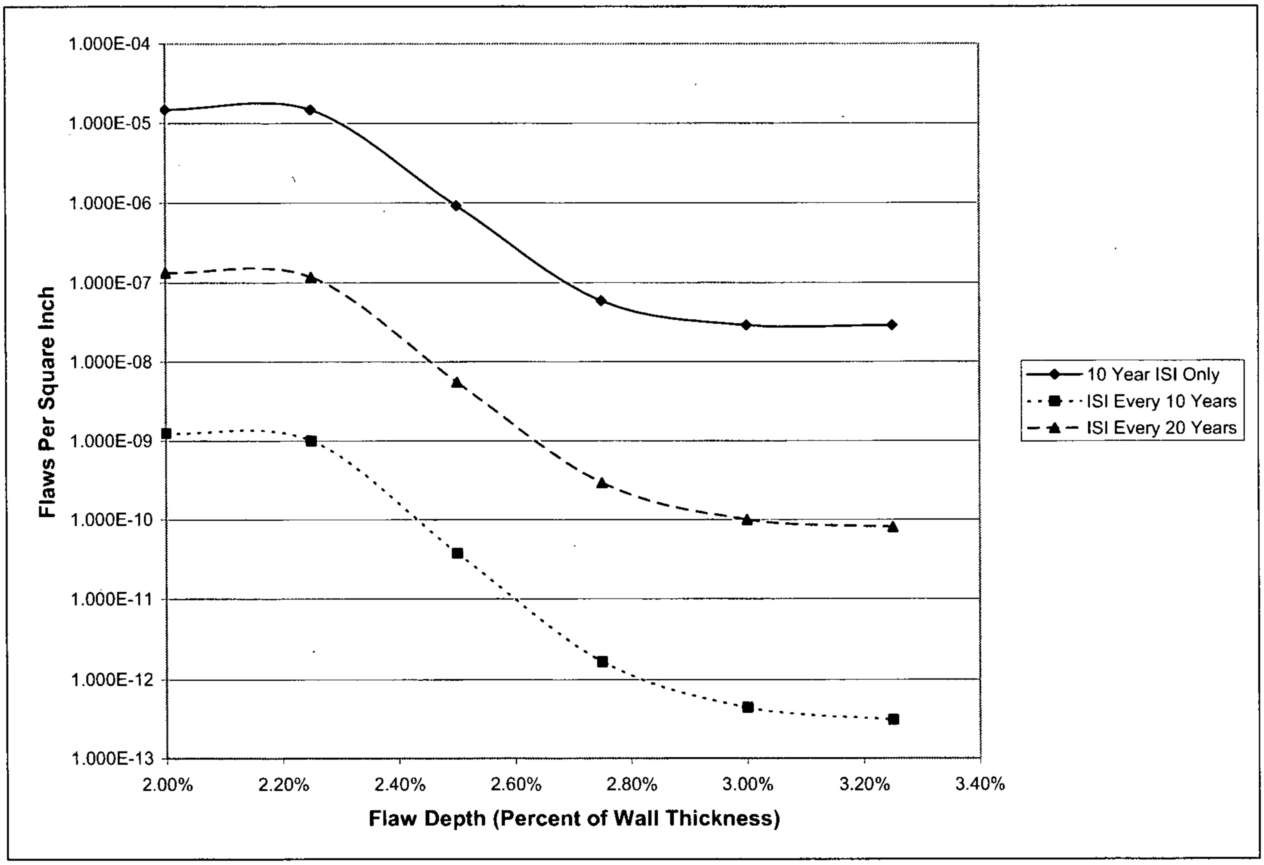

Figure 3-6 Growth of Flaws with an Aspect Ratio of 10 for BV1

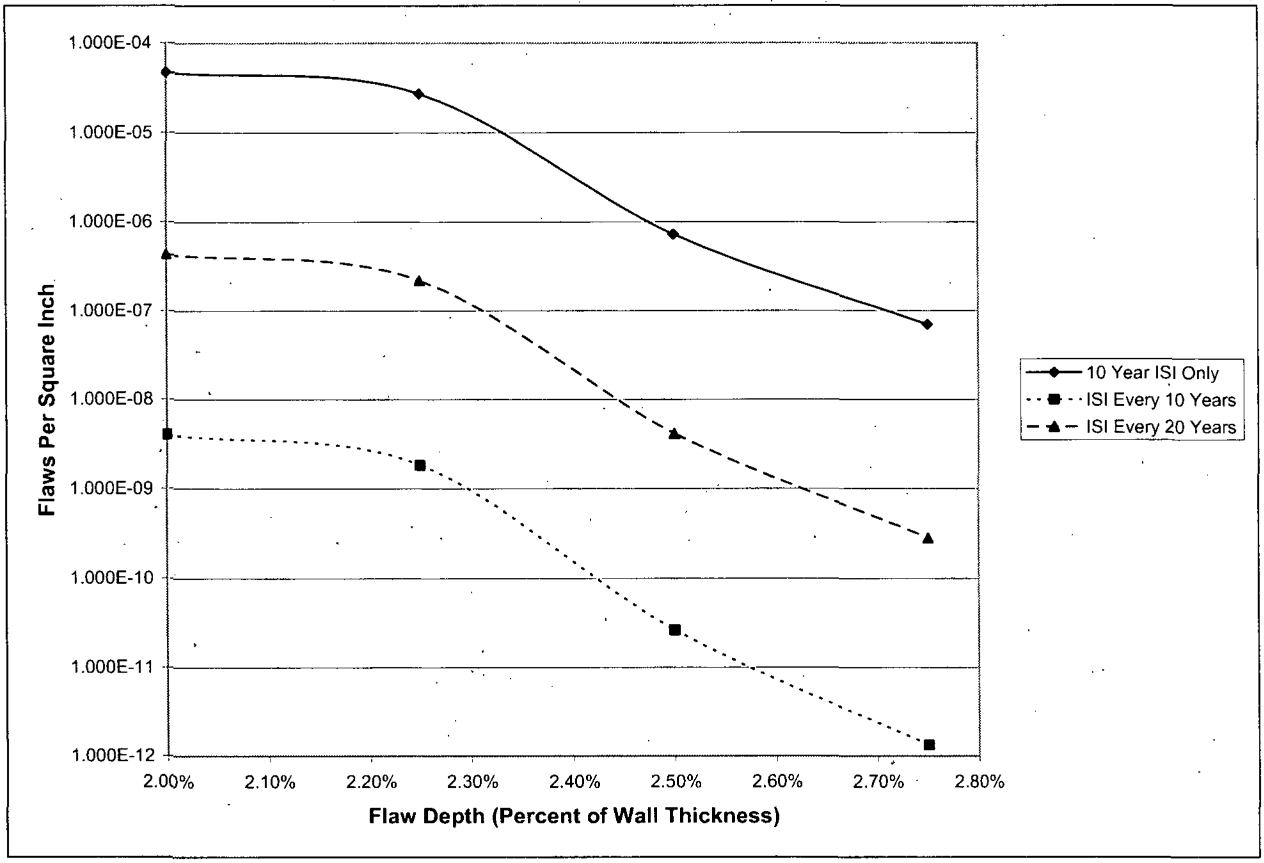

Figure 3-7 Growth of Flaws with an Infinite Aspect Ratio for BV1

3.4 RESULTS FOR THE COMBUSTION ENGINEERING PILOT PLANT: PALISADES

Reactor vessel failure frequencies were calculated for Palisades for two cases corresponding to the two surface flaw density files discussed in the section on "Initial Flaw Distribution". These cases were referred to as "ISI Every 10 Years" and "10-year ISI Only". As the names imply, the "ISI Every 10 Years" case simulates the current ASME Code required inspections while the "10-year ISI Only" case simulates a discontinuation of inspections after the first 10-year ISI. While the failure frequency for the "ISI Every 10 Years" case is higher than the "10-Year ISI Only" case, statistically, the difference between the mean failure frequencies for the "ISI Every 10 Years" case and the "10-year ISI Only" case is insignificant. This is due to the fact that the difference between the mean values is less than the standard error for each of the cases. However, to calculate a change in risk for comparison to regulatory guidelines, a bounding change in failure frequency was calculated based on the difference between an "Upper Bound" and a "Lower Bound." The Lower Bound was determined by subtracting 2 times the standard error as reported by FAVPOST from the mean value of the "ISI Every 10 Years" case. The Upper Bound was determined by adding 2 times the standard error as reported by FAVPOST to the mean value of the "10-Year ISI Only" case.

Elimination of ISI after the first 10-year ISI for the Palisades RPV results in a bounding difference in failure (through-wall flaw) frequency of less than 1.81E-08. A summary of the results of the evaluation are included in Table 3-3. The results reflect the maximum statistically calculated value for the potential change in risk at a number of reactor vessel simulations at which the Monte Carlo statistical analysis has reached a stable solution. The difference between the Upper Bound and Lower Bound represents the bounding difference between the 10-year inspection interval currently applicable under ASME criteria and elimination of all future inspections following an inspection within the first 10 years of operation.

This change in failure frequency is acceptable per the regulatory guidance discussed in Section 4.1. Transient input was based on design basis transients and the transients used in the NRC PTS Risk Study [8, 9]. The input data included consideration of postulated life extension to 60 EFPY. The FAVPOST outputs for the cases presented in Table 3-3 are presented in Appendix I.

Table 3-3 Palisades Reactor Vessel Failure Frequency Results	
10-Year ISI Only (Mean Value / Standard Error)	7.62E-08/4.08E-09
Upper Bound Value	8.44E-08
ISI Every 10 Years (Mean Value / Standard Error)	7.39E-08/3.80E-09
Lower Bound Value	6.63E-08
Bounding Difference in Risk	1.81E-08

The mean effects of fatigue crack growth and ISI on the surface breaking flaw density for 1000 simulations are shown in Figures 3-8 and 3-9. These figures plot the flaw density as a function of the flaw depth for the cases of 1 initial 10-year ISI, a 10-year ISI interval, and a 20-year ISI interval. These plots display the results for the of 10-to-1 and infinite aspect ratio sizes. The PROBSBFD outputs used to generate these plots are included in Appendix G. The crack growth and density reduction due to ISI would both be reduced for the flaw length-to-depth aspect ratios of 2-to-1 and 6-to-1 also considered in the pilot plant study.

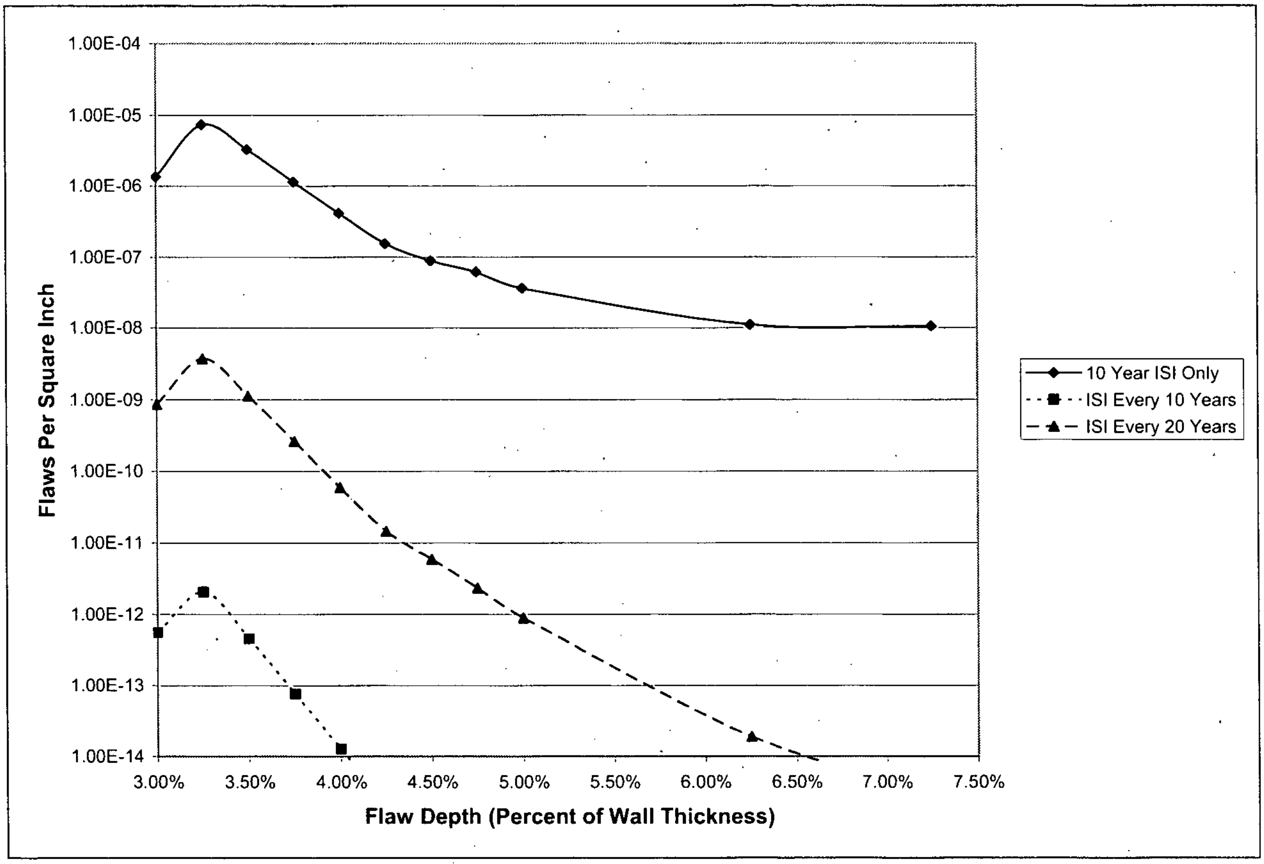

Figure 3-8 Growth of Flaws with an Aspect Ratio of 10 for Palisades

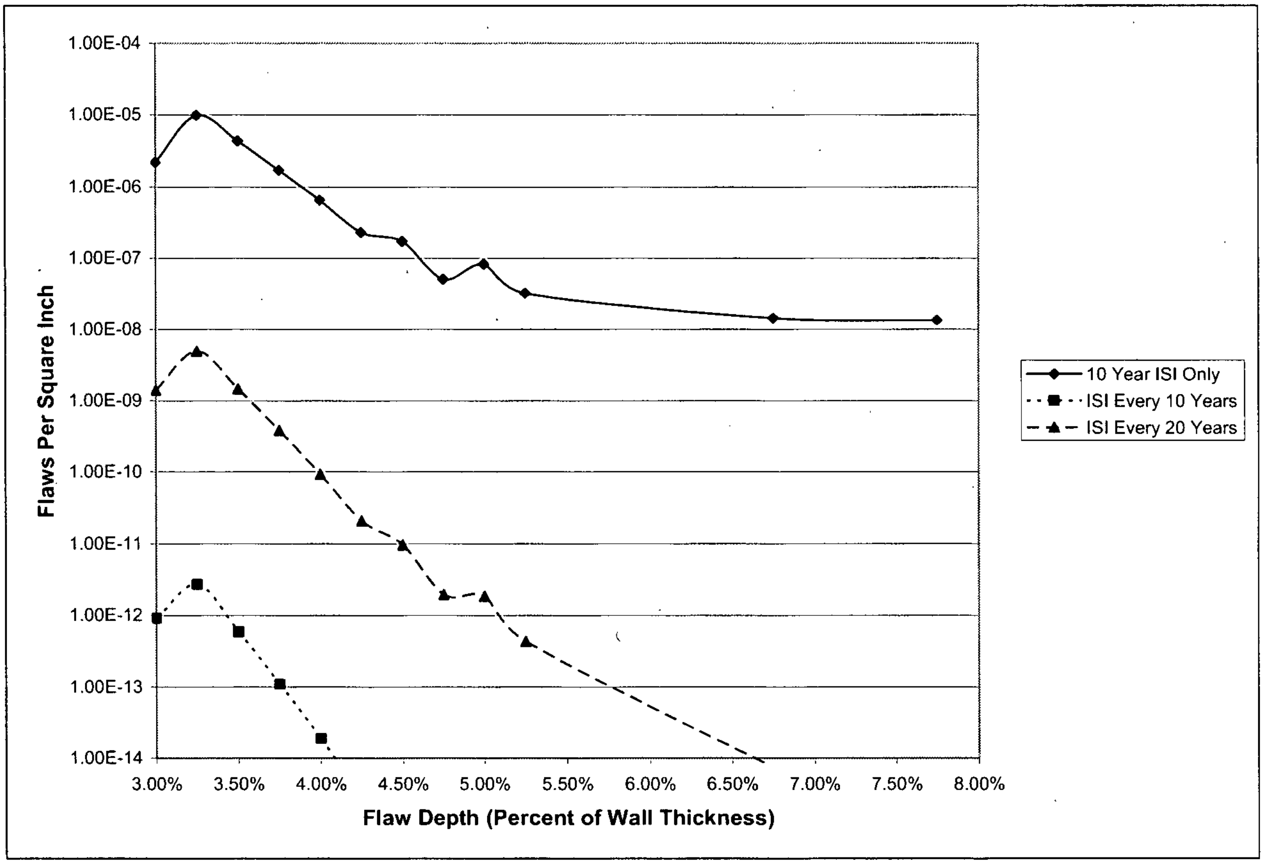

Figure 3-9 Growth of Flaws with an Infinite Aspect Ratio for Palisades

3.5 RESULTS FOR THE BABCOCK AND WILCOX PILOT PLANT: OC1

Reactor vessel failure frequencies were calculated for OC1 for two cases corresponding to the two surface flaw density files discussed in the section on "Initial Flaw Distribution". These cases were referred to as "ISI Every 10 Years" and "10-year ISI Only". As the names imply, the "ISI Every 10 Years" case simulates the current ASME Code required inspections while the "10-year ISI Only" case simulates a discontinuation of inspections after the first 10-year ISI. While the failure frequency for the "ISI Every 10 Years" case is higher than the "10-Year ISI Only" case, statistically, the difference between the mean failure frequencies for the "ISI Every 10 Years" case and the "10-year ISI Only" case is insignificant. This is due to the fact that the difference between the mean values is less than the standard error for each of the cases. However, to calculate a change in risk for comparison to regulatory guidelines, a bounding change in failure frequency was calculated based on the difference between an "Upper Bound" and a "Lower Bound." The Lower Bound was determined by subtracting 2 times the standard error as reported by FAVPOST from the mean value of the "ISI Every 10 Years" case. The Upper Bound was determined by adding 2 times the standard error as reported by FAVPOST to the mean value of the "10-Year ISI Only" case.

Elimination of ISI after the first 10-year ISI for the OC1 RPV results in a difference in failure (through-wall flaw) frequency of 1.26E-08. A summary of the results of the evaluation are included in Table 3-4. The results reflect the maximum statistically calculated value for the potential change in risk at a number of reactor vessel simulations at which the Monte Carlo statistical analysis has reached a stable solution. The difference between the Upper Bound and Lower Bound represents the bounding difference between the 10-year inspection interval currently applicable under ASME criteria and elimination of all future inspections following an inspection within the first 10 years of operation.

This change in failure frequency is acceptable per the regulatory guidance discussed in Section 4.1. Transient input was based on design basis transients and the transients used in the NRC PTS Risk Study [8, 9]. The input data included consideration of postulated life extension to 60 EFPY. The FAVPOST outputs for the cases presented in Table 3-4 are presented in Appendix M.

Table 3-4 OC1 Reactor Vessel Failure Frequency Results

10-Year ISI Only (Mean Value / Standard Error)	3.11E-08/2.55E-09
Upper Bound Value	3.62E-08
ISI Every 10 Years (Mean Value / Standard Error)	2.62E-08/1.28E-09
Lower Bound Value	2.36E-08
Bounding Difference in Risk	1.26E-08

The mean effects of fatigue crack growth and ISI on the surface breaking flaw density for 1000 simulations are shown in Figures 3-10 and 3-11. These figures plot the flaw density as a function of the flaw depth for the cases of 1 initial 10-year ISI, a 10-year ISI interval, and a 20-year ISI interval. These plots display the results for the 10-to-1 and infinite aspect ratio sizes. The PROBSBFD outputs used to generate these plots are included in Appendix K. The crack growth and density reduction due to ISI would both be reduced for the flaw length-to-depth aspect ratios of 2-to-1 and 6-to-1 also considered in the pilot plant study.

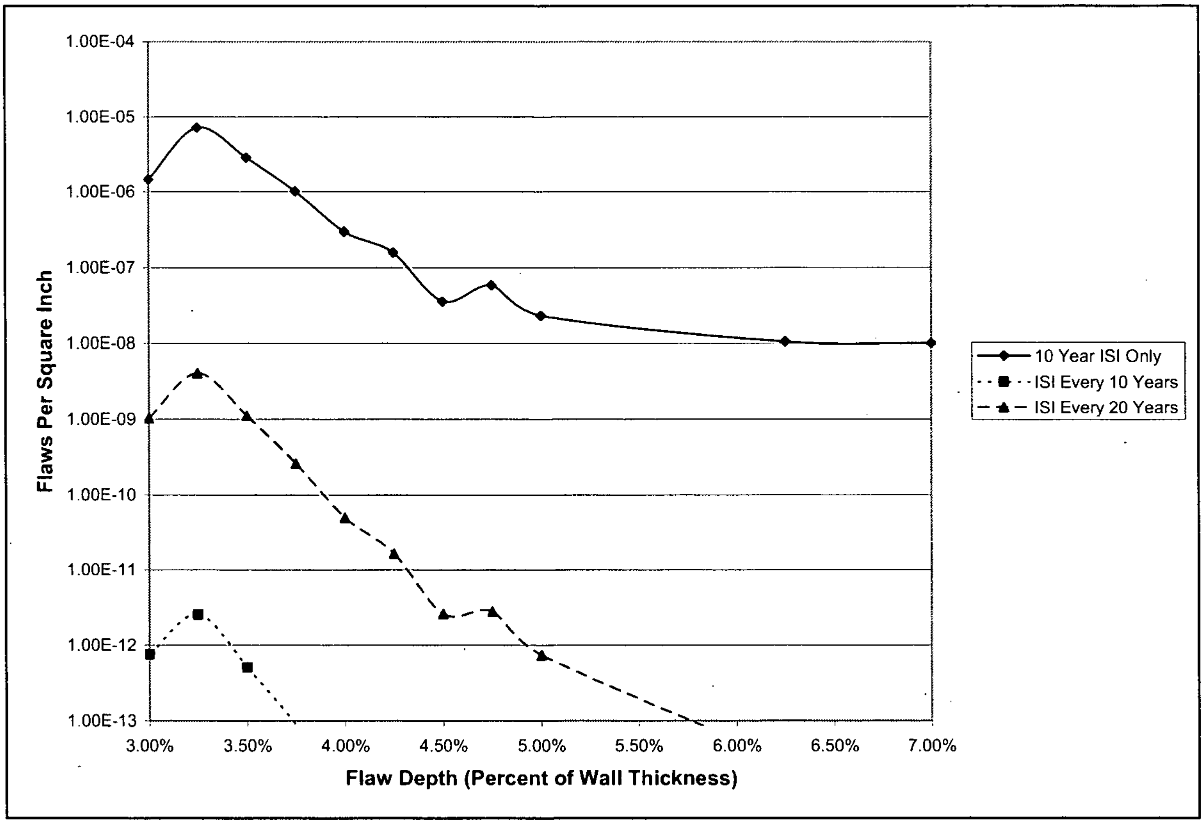

Figure 3-10 Growth of Flaws with an Aspect Ratio of 10 for OC1

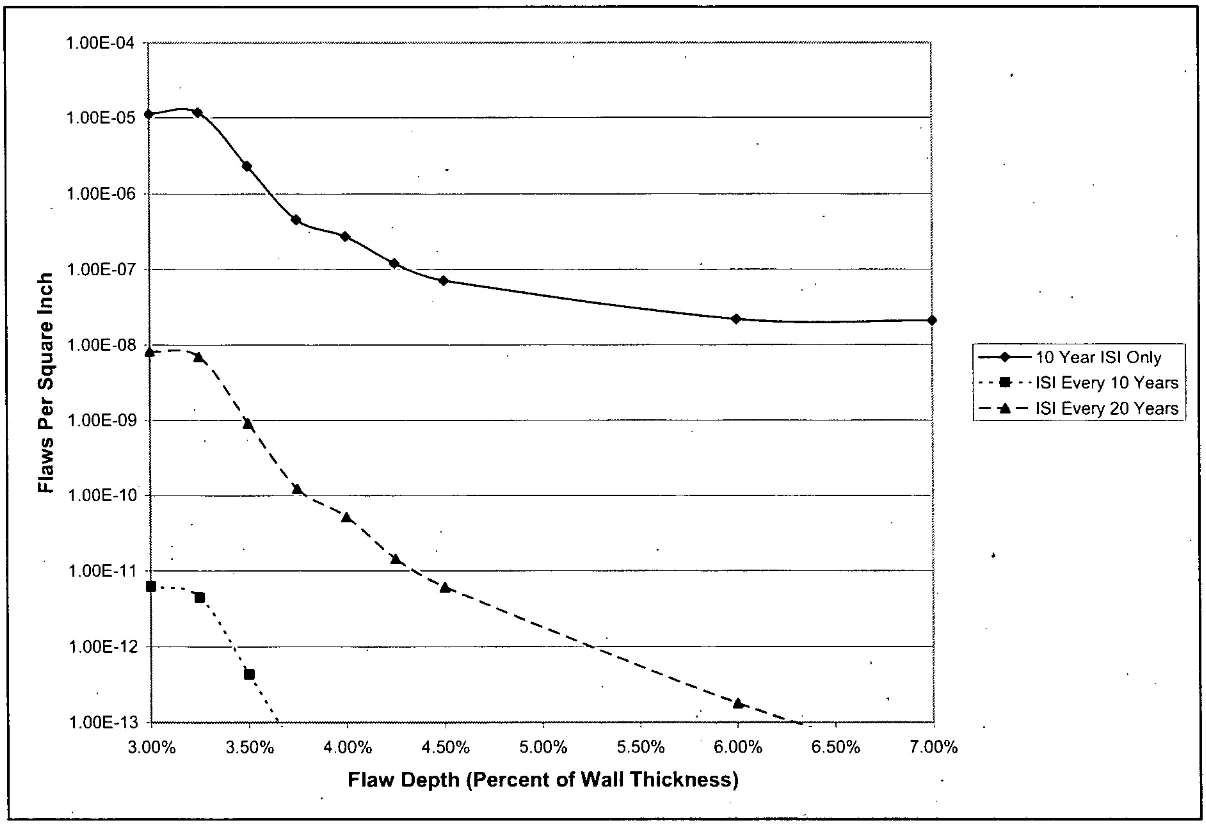

Figure 3-11 Growth of Flaws with an Infinite Aspect Ratio for OC1

4 RISK ASSESSMENT

The quantitative risk assessment discussed below shows that extending the inspection interval from 10 to a maximum of 20 years has an acceptably small impact on risk (core damage frequency [CDF] and large early release frequency [LERF]), i.e., that it is within the bounds of RG 1.174 [4]. A discussion on the requirements of RG 1.174 is included.

4.1 RISK-INFORMED REGULATORY GUIDE 1.174 METHODOLOGY

The NRC has developed a risk-informed regulatory framework. The NRC definition of risk-informed regulation is: "insights derived from probabilistic risk assessments are used in combination with deterministic system and engineering analysis to focus licensee and regulatory attention on issues commensurate with their importance to safety."

The NRC issued RG 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis* [4]. In addition, the NRC issued application-specific RGs and Standard Review Plans (SRPs):

- RG-1.175 [40] and SRP Chapter 3.9.7, related to in-service testing (IST) programs
- RG-1.176 [41], related to Graded Quality Assurance (GQA) programs
- RG-1.177 [42] and SRP Chapter 16.1, related to Technical Specifications
- RG-1.178 [44] and SRP-3.9.8, related to ISI of piping programs

These RG and SRP chapters provide guidance in their respective application-specific subject areas to reactor licensees and the NRC staff regarding the submittal and review of risk-informed proposals that would change the licensing basis for a power reactor facility.

Regulatory Guide 1.174 Basic Steps

The approach described in RG 1.174 was used in each of the application-specific RGs/SRPs, and has 4 basic steps as shown in Figure 4-1. The four basic steps are discussed below.

<u>Step 1: Define the Proposed Change</u>

This element includes identifying:

1. Those aspects of the plant's licensing bases that may be affected by the change.

2. All systems, structures, and components (SSCs), procedures, and activities that are covered by the change and consider the original reasons for inclusion of each program requirement.

3. Any engineering studies, methods, codes, applicable plant-specific and industry data and operational experience, PRA findings, and research and analysis results relevant to the proposed change.

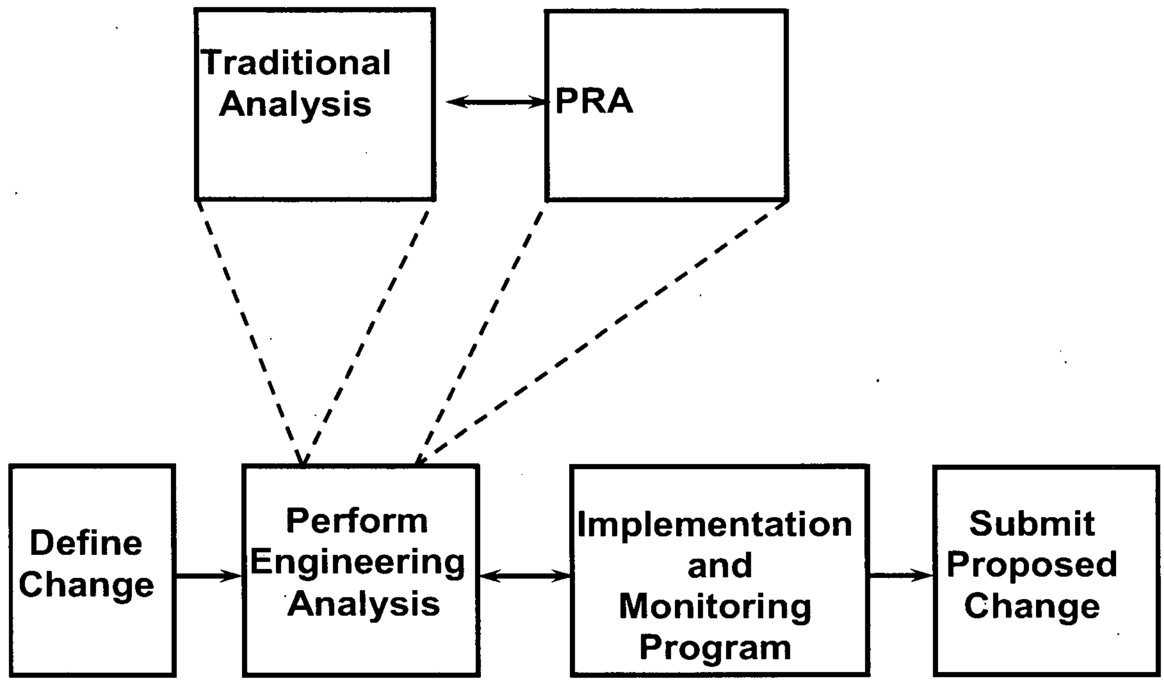

Figure 4-1 Basic Steps in (Principal Elements of) Risk-Informed, Plant-Specific Decision Making (from NRC RG 1.174)

Step 2: Perform Engineering Analysis

This element includes performing the evaluation to show that the fundamental safety principles on which the plant design was based are not compromised (defense-in-depth attributes are maintained) and that sufficient safety margins are maintained. The engineering analysis includes both traditional deterministic analysis and probabilistic risk assessment (PRA). The evaluation of risk impact should also assess the expected change in CDF and LERF, including a treatment of uncertainties. The results from the traditional analysis and the PRA must be considered in an integrated manner when making a decision.

Step 3: Define Implementation and Monitoring Program

This element's goal is to assess SSC performance under the proposed change by establishing performance monitoring strategies to confirm assumptions and analyses that were conducted to justify the change. This is to ensure that no unexpected adverse safety degradation occurs because of the changes. Decisions concerning implementation of changes should be made in light of the uncertainty associated with the results of the evaluation. A monitoring program should have measurable parameters, objective criteria, and parameters that provide an early indication of problems before becoming a safety concern. In addition, the monitoring program should include a cause determination and corrective action plan.

Step 4: Submit Proposed Change

This element includes:

1. Carefully reviewing the proposed change in order to determine the appropriate form of the change request.

2. Assuring that information required by the relevant regulation(s) in support of the request is developed.

3. Preparing and submitting the request in accordance with relevant procedural requirements.

Regulatory Guide 1.174 Fundamental Safety Principles

Five fundamental safety principles are described that each application for a change must meet. These are shown in Figure 4-2, and are discussed below.

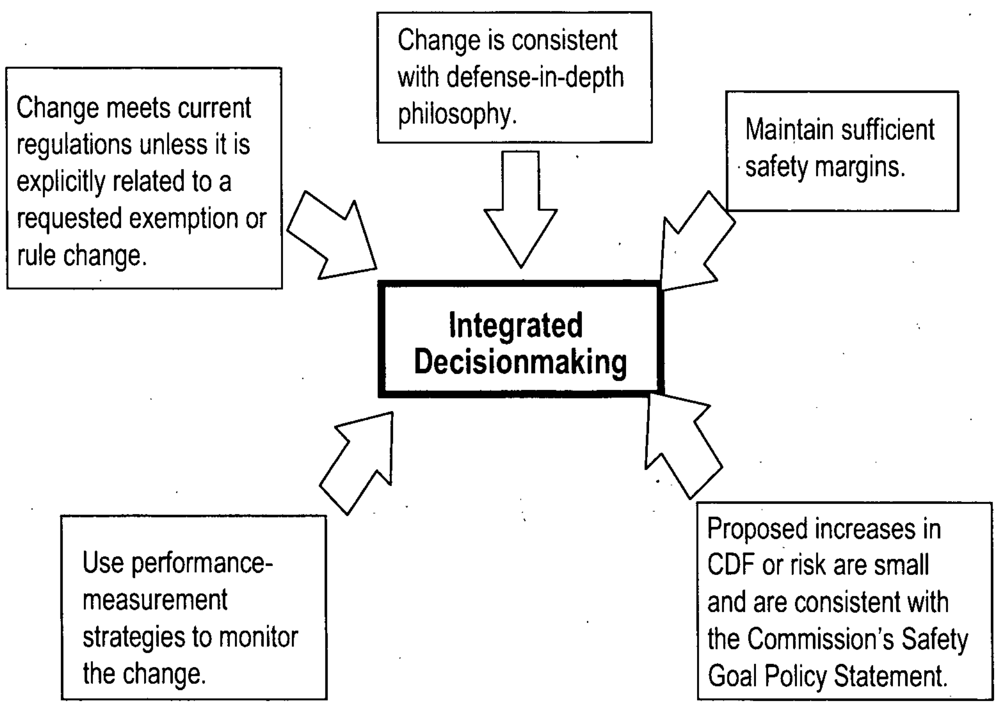

Figure 4-2 Principles of Risk-Informed Regulation (from NRC RG 1.174)

Principle 1: Change meets current regulations unless it is explicitly related to a requested exemption or rule change.

The proposed change is evaluated against the current regulations (including the general design criteria) to either identify where changes are proposed to the current regulations (e.g., Technical Specification, license conditions, and FSAR), or where additional information may be required to meet the current regulations.

Principle 2: Change is consistent with defense-in-depth philosophy.

Defense-in-depth has traditionally been applied in reactor design and operation to provide a multiple means to accomplish safety functions and prevent the release of radioactive material. As defined in RG 1.174 [4], defense-in-depth is maintained by assuring that:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences to the system (e.g., no risk outliers).
- Defenses against potential common cause failures are preserved and the potential for introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded (the barriers are identified as the fuel cladding, reactor coolant pressure boundary, and containment structure).
- Defenses against human errors are preserved.

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

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

Principle 3: Maintain sufficient safety margins.

Safety margins must also be maintained. As described in RG 1.174, sufficient safety margins are maintained by assuring that:

- Codes and standards, or alternatives proposed for use by the NRC, are met.
- Safety analysis acceptance criteria in the licensing basis (e.g., FSARs, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

Principle 4: Proposed increases in CDF or risk are small and are consistent with the Commission's Safety Goal Policy Statement.

To evaluate the proposed change with regard to a possible increase in risk, the risk assessment should be of sufficient quality to evaluate the change. The expected change in CDF and LERF are evaluated to address this principle. An assessment of the uncertainties associated with the evaluation is conducted. Additional qualitative assessments are also performed.

There are two acceptance guidelines, one for CDF and one for LERF, both of which should be used.

The guidelines for CDF are:

- If the application can be clearly shown to result in a decrease in CDF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to CDF.

- When the calculated increase in CDF is very small, which is taken as being less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF.

- When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year.

- Applications that result in increases to CDF above 10^{-5} per reactor year would not normally be considered.

The guidelines for LERF are:

- If the application can be clearly shown to result in a decrease in LERF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to LERF.

- When the calculated increase in LERF is very small, which is taken as being less than 10^{-7} per reactor year, the change will be considered regardless of whether there is a calculation of the total LERF.

- When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year.

- Applications that result in increases to LERF above 10^{-6} per reactor year would not normally be considered.

These guidelines are intended to provide assurance that proposed increases in CDF and LERF are small and are consistent with the intent of the Commission's Safety Goal Policy Statement.

Principle 5: Use performance-measurement strategies to monitor the change.

Performance-based implementation and monitoring strategies are also addressed as part of the key elements of the evaluation as described previously.

Risk-Acceptance Criteria for Analysis

For the purposes of this bounding analysis of the risk impact of the proposed change in RV inspection frequency, the following criteria are applied with respect to Principle 4 (small change in risk):

- Change in CDF < 1×10^{-6} per reactor year
- Change in LERF < 1×10^{-7} per reactor year

These values are selected so that the proposed change may be later considered on a plant-specific basis regardless of the plant's baseline CDF and LERF.

To conservatively simplify these acceptance criteria, it will be assumed that through-wall crack growth is equivalent to reactor vessel failure, and that reactor vessel failure results in both core damage and a large early release. It is also conservatively assumed that the conditional probability of a large early release given core damage is 1.0 (See Section 4.3).

Therefore, the simplified conservative/bounding acceptance criterion becomes:

$$\text{Change in CDF} = \text{Change in LERF} = \begin{array}{c}\text{Increase in frequency of}\\ \text{through-wall crack}\\ \text{growth due to increase in}\\ \text{inspection interval}\end{array} < \begin{array}{c}1 \times 10^{-7}\text{ per}\\ \text{reactor year}\end{array}$$

4.2 FAILURE MODES AND EFFECTS

Failure Modes

The failure mode of concern was thermal fatigue crack growth due typical plant operation. The growth of an existing undetected fabrication flaw in the RV base metal, cladding, or weld metal was assumed to reach a critical size that would lead to reactor vessel through-wall fracture if a PTS-type transient would occur.

Failure Effects

A through-wall flaw failure of the RV was assumed to result in core damage and a large early release.

4.3 CORE DAMAGE RISK EVALUATION

The objective of the risk assessment was to evaluate the core damage risk from the extension of the examination of the RV relative to other plant risk contributors through a qualitative and quantitative evaluation.

NRC RG 1.174 [4] provided the basis for this evaluation as well as the acceptance guidelines to make a change to the current licensing basis.

Risk was defined as the combination of likelihood of an event and severity of consequences of an event. Therefore, the following two questions were addressed:

- What was the likelihood of the event?
- What would the consequences be?

The following sections describe the likelihood and postulated consequences. The likelihood and consequences were then combined in the risk calculation and the results of the evaluation are presented in this report.

What is the Likelihood of the Event?

The likelihood of the event was addressed by identifying the plant transients or operational events that might lead to failure of the RV, and estimating the frequency of these events.

What are the Consequences?

The consequences were defined in terms of the CDF and LERF risk metrics.

For this evaluation, the conditional core damage probability given the failure of the RV was assumed to be 1.0 (no credit for safety system actuation to mitigate the consequences of the failure). Since this was intended as a bounding assessment, it was also conservatively assumed that the conditional probability of a large early release given core damage for this scenario is 1.0 (i.e., no credit for consequence mitigation via the containment and related systems). Note that this was a simplifying assumption, and a specific mechanism for LERF was not implied or defined here.

Risk Calculation

For this evaluation, the CDF and LERF were calculated by:

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

where:

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

The transient initiating frequency distributions were identified in the NRC PTS Risk Study [8, 9] and are included in Appendices D, H, and L for the pilot plants. The probability of failure was calculated by the FAVPFS module of FAVOR. The FAVPOST module of FAVOR combined the transient initiating frequency distribution with the reactor vessel conditional failure probability distribution to determine a reactor vessel failure frequency distribution for each transient. From these failure frequency distributions, FAVPOST determined a mean reactor vessel failure frequency. In addition to this mean failure frequency a standard error was reported. To account for uncertainties, Upper and Lower Bounds are determined. The Upper Bound was determined by adding 2 times the standard error from the "10-Year ISI-Only" case. The Lower Bound was determined by subtracting 2 times the standard error from the "ISI Every 10 Years" case. The change in reactor vessel failure frequency was determined by subtracting the Lower Bound from the Upper Bound. The mean reactor vessel failure frequencies, Upper and Lower Bounds, and change in failure frequency are given in Sections 3.2 and 3.3. As previously stated, reactor vessel failure results in core damage which results in large early release. Therefore, the large early release frequencies were equal to the reactor vessel failure frequencies. The large early release frequencies, Upper and Lower Bounds, and change in large early release frequency are summarized in Table 4-1, based on FAVOR 06.1 evaluations.

Table 4-1 Large Early Release Frequencies

	BV1 (per year)	Palisades (per year)	OC1 (per year)
10-Year ISI Only	5.04E-09	7.62E-08	3.11E-08
Upper Bound	5.55E-09	8.44E-08	3.62E-08
ISI Every 10 Years	5.23E-09	7.39E-08	2.62E-08
Lower Bound	4.61E-09	6.63E-08	2.36E-08
Bounding Change in Large Early Release Frequency	9.37E-10	1.81E-08	1.26E-08

Risk Results and Conclusions

The analysis described above demonstrates that changes in CDF and LERF do not exceed the NRC's RG-1.174 [4] acceptance guidelines for a small change in CDF and LERF ($<10^{-6}$ per year for CDF, $<10^{7}$ per year for LERF).

As part of this evaluation, the key principles identified in RG-1.174 were reviewed and the responses based on the evaluation are provided in Table 4-2.

This evaluation concluded that extension of the RV in-service examination from 10 to 20 years would not be expected to result in an unacceptable increase in risk. Given this outcome, and the fact that other key principles listed in RG-1.174 continue to be met, the proposed change in inspection interval from 10 to 20 years is acceptable.

Table 4-2 Evaluation with Respect to Regulatory Guide 1.174 [4] Key Principles

Key Principles	Evaluation Response
Change meets current regulations unless it is explicitly related to a requested exemption or rule change.	Change to current RG 1.150 [2] requirements is proposed.
Change is consistent with defense-in-depth philosophy.	Potential for failure of the RV is acceptably small during normal or accident conditions, and does not threaten plant barriers. See the discussion below for additional information on defense in depth.
Maintain sufficient safety margins.	No safety analysis margins are changed.
Proposed increases in CDF or risk are small and are consistent with the Commission's Safety Goal Policy Statement.	Proposed increase in risk is estimated to be acceptably small.
Use performance-measurement strategies to monitor the change.	NDE examinations still conducted, but on less frequent basis not to exceed 20 years. Other indications of potential degradation of RV are available (e.g., foreign experience and periodic testing with visual examinations)

Defense-in-Depth

While the results presented in this report demonstrate that the contribution of eliminating future inspections after the initial 10 year ISI meets prescribed regulatory criteria for assessing risk, the proposed course of action is to extend the inspection interval requirements from 10 to 20 years while not eliminating any portion of the current inspection requirements. This provides additional margin for defense-in-depth and contributes directly toward maintaining plant safety.

Extending the RV ISI interval does not imply that generic degradation mechanisms will be ignored for 20 years. (With the number of PWR nuclear power plants in operation in the U.S. and globally, a sampling of plants inevitably undergo examinations in a given year.) This provides for early detection of any potential emerging generic degradation mechanisms, and would permit the industry to react with more frequent examinations if needed.

In addition, it must be recognized that all reactor coolant pressure boundary failures occurring to date have been identified as a result of leakage, and were discovered by visual examination. The proposed RV ISI interval extension does not alter the visual examination interval. The reactor vessel would undergo, as a minimum, the Section XI Examination Category B-P pressure tests and visual examinations conducted at the end of each refueling before plant start-up, as well as leak tests with visual examinations that precede each start-up following maintenance or repair activities.

Page 4-4 identifies from Regulatory Guide 1.174 that:

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

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

The extension in inspection interval will not result in any of the changes identified above. Also identified on page 4-4 are six elements for maintaining defense in depth. Due to the fact that the interval extension will not result in any of the changes identified above, it is expected that the defense in depth elements listed on page 4-4 will not be impacted. Additional assessment of the impact on each of the defense-in-depth elements from page 4-4 is provided below:

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

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

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

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

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

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

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

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

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

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

- Defenses against human errors are preserved:

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

5 CONCLUSIONS

Based on the results of this analysis, it is concluded that:

1. The beltline is the most limiting region for the evaluation of risk.

2. RV inspections performed to date have not detected any significant flaws.

3. Crack extension due to fatigue crack growth during service is small.

4. The man-rem exposure can be reduced by extending the inspection interval.

5. The failure frequencies for PWR RVs due to the dominant PTS transients are well below 10^{-7} per year.

6. The change in risk meets the RG 1.174 [4] acceptance guidelines for a small change in LERF.

7. The increase in the RV ISI interval from 10 to 20 years satisfies all the RG 1.174 criteria, including other considerations, such as defense-in-depth.

Based on the above conclusions, the ASME Section XI [1] 10-year inspection interval for examination categories B-A and B-D welds in PWR RVs can be extended to 20 years. In-service inspection intervals of 20 years for FENOC's Beaver Valley Unit 1, NMC's Palisades, and Duke Energy's Oconee Unit 1 are acceptable for implementation. The methodology in WCAP-16168-NP Revision 1 is applicable to plants other than the pilot plants by confirming the applicability of the parameters in Appendix A on a plant specific basis. Since the 10 year inspection interval is required by Section XI, IWB-2412, as codified in 10 CFR 50.55a, an exemption request must be submitted and approved by the NRC to extend the inspection interval to 20 years, unless 10 CFR 50.55a is amended to incorporate ASME Code Case N-691.

6 REFERENCES

1. *ASME Boiler and Pressure Vessel Code,* Section XI, 1989 Edition with the 1989 Addenda up to and including the 2004 Edition with the 2005 Addenda, American Society of Mechanical Engineers, New York.

2. NRC Regulatory Guide 1.150, Revision 1, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations*, February 1983 (ADAMS Accession # ML003739996).

3. 10CFR50.55a, *Codes and Standards*, 36 FR 11424, June 12, 1971.

4. NRC Regulatory Guide 1.174, Revision 1, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, November 2002 (ADAMS Accession #ML023240437).

5. *ASME Boiler and Pressure Vessel Code*, Code Case N-691, "Application of Risk-Informed Insights to Increase the Inspection Interval for Pressurized Water Reactor Vessels," Section XI, Division 1, 2003.

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33. *ASME Boiler and Pressure Vessel Code,* Code Case N-700, "Alternative Rules for Selection of Classes 1, 2, and 3 Vessel Welded Attachments for Examination," Section XI, Division 1, 2003.

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35. *ASME Boiler and Pressure Vessel Code,* Code Case N-624,"Successive Inspections," Section XI, Division 1, 1998.

36. *ASME Boiler and Pressure Vessel Code,* Code Case N-623, "Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel," Section XI, Division 1, 1998.

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APPENDIX A
PLANT SPECIFIC APPLICATION

WCAP-16168-NP Revision 2 describes the methodology used to demonstrate the feasibility of extending the reactor vessel inspection interval required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, as supplemented by Nuclear Regulatory Commission (NRC) Regulatory Guide 1.150. This methodology was used to perform risk analysis for pilot plants representing the Westinghouse and Combustion Engineering designs. It is an extension of work done as part of the NRC PTS Risk Study. Table A-1 identifies critical parameters to be used to determine if the pilot plant evaluations documented in this report bound a plant specific application. If the plant-specific parameter is not bounded by the pilot plant analysis, additional evaluations or sensitivity studies may be required to support the use of the pilot plant risk studies. Additional information relative to plant specific reactor vessel inspection is to be provided in Table A-2. Information required to calculate the plant specific through wall cracking frequency is to be provided in Table A-3. Additional information and requirements are provided following each table. Examples of plant specific use of these tables for Wolf Creek and Waterford 3 are contained in Appendices A-1 and A-2 respectively.

Table A-1 Critical Parameters for the Application of the Bounding Analysis

Parameter	Pilot Plant Basis	Plant Specific Basis	Additional Evaluation Required? (Y/N)
Dominant PTS Transients in the NRC PTS Risk Study are applicable			
Through Wall Cracking Frequency (TWCF)			
Frequency and Severity of Design Basis Transients			
Cladding Layers (Single/Multiple)			

For each of the four parameters in Table A-1, the licensee must identify the pilot plant basis and the plant specific basis. If the plant specific basis is not bounded by the pilot plant basis, additional evaluation is required.

<u>Dominant PTS Transients in the NRC PTS Risk Study are applicable</u>

The transients evaluated in the WCAP pilot plant analyses were the PTS transients from the NRC PTS Risk Re-evaluation (NUREG-1806 or NUREG-1874). For this parameter, it is necessary to demonstrate that the PTS transients used in the pilot plant analyses are applicable for a specific plant. As stated in the last paragraph in Section 3.2.1 of NUREG-1874, the "PTS Generalization Study demonstrates that risk-significant PTS transients do not have any appreciable plant-specific differences within the population of PWRs currently operating in the United States." Based on this statement, plant specific analyses are not needed for this criterion. A licensee may enter the "PTS Generalization Study" in Table A-1 as the basis for the applicability of the pilot plant PTS transients to their specific plant.

Through Wall Cracking Frequency (TWCF)

Each licensee shall calculate their plant specific $TWCF_{95\text{-}TOTAL}$ value using the correlations in NUREG-1874. The calculated plant specific $TWCF_{95\text{-}TOTAL}$ value must be lower than the applicable pilot plant $TWCF_{95\text{-}TOTAL}$ value calculated using the correlations in NUREG-1874. The TWCF is essentially a measure of the embrittlement of the reactor vessel components weighted by their contribution to PTS failure. By demonstrating that the pilot plant has a higher $TWCF_{95\text{-}TOTAL}$ value it follows that the pilot plant change in risk calculation is bounding of that for the specific plant.

The pilot plant $TWCF_{95\text{-}TOTAL}$ values calculated using the NUREG-1874 correlations depend upon the applicable pilot plant design:

Westinghouse:	Beaver Valley Unit 1:	1.76E-08 Events per year
CE:	Palisades:	3.16E-07 Events per year
B&W:	Oconee Unit 1:	4.42E-07 Events per year

The applicable correlations from NUREG-1874 are as follows:

$$TWCF_{95\text{-}TOTAL} = \alpha_{AW}TWCF_{95\text{-}AW} + \alpha_{PL}TWCF_{95\text{-}PL} + \alpha_{CW}TWCF_{95\text{-}CW} + \alpha_{FO}TWCF_{95\text{-}FO}$$

Where, α is determined as follows:

If $RT_{MAX\text{-}xx} \leq 625°R$, then $\alpha = 2.5$
If $625R < RT_{MAX\text{-}xx} < 875°R$ then $\alpha = 2.5 - (1.5/250)(RT_{MAX\text{-}xx} - 625)$
If $RT_{MAX\text{-}xx} \geq 875°R$, then $\alpha = 1$

and $TWCF_{95\text{-}XX}$ values are calculated as follows:

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

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

$$TWCF_{95\text{-}CW} = \exp\{9.1363*\ln(RT_{MAX\text{-}CW} - 616) - 65.066\}*\beta$$

$$TWCF_{95\text{-}FO} = \exp\{23.737*\ln(RT_{MAX\text{-}FO} - 300) - 162.38\}*\beta + \ldots$$
$$+\eta*\{1.3 \times 10^{-137}*10^{0.185*RTMAX\text{-}FO}\}*\beta$$

η is equal to "0" for ring forged vessels fabricated compliant with Regulatory Guide 1.43 and equal to "1" for ring forged vessels not fabricated compliant with Regulatory Guide 1.43.

β is determined as follows irrespective of the set of TWCF formulas used.

If $T_{WALL} \leq 9\frac{1}{2}$ -in, then β=1
If $9\frac{1}{2} < T_{WALL} < 11\frac{1}{2}$ -in, then $\beta = 1+8(T_{WALL}- 9\frac{1}{2})$
If $T_{WALL} \geq 11\frac{1}{2}$ -in, then $\beta = 17$

Where T_{WALL} is the thickness of the RPV wall (inches), including the cladding.

$RT_{MAX\text{-}XX}$ values are calculated in degrees Rankin (°R) as follows:

$$RT_{MAX\text{-}AW} \equiv \underset{i=1}{\overset{n_{AWFL}}{MAX}}\left[MAX_{AWFL(i)} \left\{ \begin{aligned} &\left(RT_{NDT(u)}^{adj-aw(i)} + \Delta T_{30}^{adj-aw(i)}\left(\phi t_{FL}\right)\right), \\ &\left(RT_{NDT(u)}^{adj-pl(i)} + \Delta T_{30}^{adj-pl(i)}\left(\phi t_{FL}\right)\right) \end{aligned} \right\} \right]$$

Where:

n_{AWFL}	is the number of axial weld fusion lines in the beltline region of the vessel,
i	is a counter that ranges from 1 to n_{AWFL}.
ϕt_{FL}	is the maximum fluence occurring on the vessel ID along a particular axial weld fusion line,
$RT_{NDT(u)}^{adj-aw(i)}$	is the unirradiated RT_{NDT} of the weld adjacent to the ith axial weld fusion line,
$RT_{NDT(u)}^{adj-pl(i)}$	is the unirradiated RT_{NDT} of the plate adjacent to the ith axial weld fusion line
$\Delta T_{30}^{adj-aw(i)}$	is the shift in the Charpy V-Notch 30-foot-pound (ft-lb) energy produced by irradiation to ϕt_{FL} of the weld adjacent to the ith axial weld fusion line, and
$\Delta T_{30}^{adj-pl(i)}$	is the shift in the Charpy V-Notch 30-foot-pound (ft-lb) energy produced by irradiation due to ϕt_{FL} of the plate adjacent to the ith axial weld fusion line.

$$RT_{MAX\text{-}PL} \equiv \underset{i=1}{\overset{n_{PL}}{MAX}}\left[RT_{NDT(u)}^{PL(i)} + \Delta T_{30}^{PL(i)}\left(\phi t_{MAX}^{PL(i)}\right)\right]$$

Where:

n_{PL}	is the number of plates in the beltline region of the vessel,
i	is a counter that ranges from 1 to n_{PL},
$\phi t_{MAX}^{PL(i)}$	is the maximum fluence occurring over the vessel ID occupied by a particular plate,
$RT_{NDT(u)}^{PL(i)}$	is the unirradiated RT_{NDT} of a particular plate, and
$\Delta T_{30}^{PL(i)}$	is the shift in the Charpy V-Notch 30-foot-pound (ft-lb) energy produced by irradiation to $\phi t_{MAX}^{PL(i)}$ of a particular plate.

$$RT_{MAX\text{-}FO} \equiv \underset{i=1}{\overset{n_{FO}}{MAX}}\left[RT_{NDT(u)}^{FO(i)} + \Delta T_{30}^{FO(i)}\left(\phi t_{MAX}^{FO(i)}\right)\right]$$

Where:

n_{FO}	is the number of forgings in the beltline region of the vessel,
i	is a counter that ranges from 1 to n_{FO}
$\phi t_{MAX}^{FO(i)}$	is the maximum fluence occurring over the vessel ID occupied by a particular forging,
$RT_{NDT(u)}^{FO(i)}$	is the unirradiated RT_{NDT} of a particular forging, and
$\Delta T_{30}^{FO(i)}$	is the shift in the Charpy V-Notch 30-foot-pound (ft-lb) energy produced by irradiation to $\phi t_{MAX}^{FO(i)}$ of a particular forging.

$$RT_{MAX-CW} \equiv \underset{i=1}{\overset{n_{CWFL}}{MAX}} \left[MAX_{CWFL(i)} \begin{Bmatrix} \left(RT_{NDT(u)}^{adj-cw(i)} + \Delta T_{30}^{adj-cw(i)}(\phi t_{FL})\right), \\ \left(RT_{NDT(u)}^{adj-pl(i)} + \Delta T_{30}^{adj-pl(i)}(\phi t_{FL})\right), \\ \left(RT_{NDT(u)}^{adj-fo(i)} + \Delta T_{30}^{adj-fo(i)}(\phi t_{FL})\right) \end{Bmatrix} \right]$$

Where:

n_{CWFL}	is the number of circumferential weld fusion lines in the beltline region of the vessel,
i	is a counter that ranges from 1 to n_{CWFL},
ϕt_{FL}	is the maximum fluence occurring on the vessel ID along a particular circumferential weld fusion line,
$RT_{NDT(u)}^{adj-cw(i)}$	is the unirradiated RT_{NDT} of the weld adjacent to the i^{th} circumferential weld fusion line,
$RT_{NDT(u)}^{adj-pl(i)}$	is the unirradiated RT_{NDT} of the plate adjacent to the i^{th} circumferential weld fusion line (if there is no adjacent plate this term is ignored),
$RT_{NDT(u)}^{adj-fo(i)}$	is the unirradiated RT_{NDT} of the forging adjacent to the i^{th} circumferential weld fusion line (if there is no adjacent forging this term is ignored),
$\Delta T_{30}^{adj-cw(i)}$	is the shift in the Charpy V-Notch 30-foot-pound (ft-lb) energy produced by irradiation due to ϕt_{FL} of the weld adjacent to the i^{th} circumferential weld fusion line,
$\Delta T_{30}^{adj-pl(i)}$	is the shift in the Charpy V-Notch 30-foot-pound (ft-lb) energy produced by irradiation to ϕt_{FL} of the plate adjacent to the i^{th} axial weld fusion line (if there is no adjacent plate this term is ignored), and
$\Delta T_{30}^{adj-fo(i)}$	is the shift in the Charpy V-Notch 30-foot-pound (ft-lb) energy produced by irradiation to ϕt_{FL} of the forging adjacent to the i^{th} axial weld fusion line (if there is no adjacent forging this term is ignored).

The ΔT_{30} shift shall be determined using the latest approved methodology in Regulatory Guide 1.99 or other NRC-approved methodology (Equations in Section 3.5.2 of NUREG-1874 were used to calculate pilot plant values). All material properties used to determine the $TWCF_{95-TOTAL}$ value shall be documented in Table A-3.

The plant specific $TWCF_{95-TOTAL}$ value shall be re-evaluated any time fluence is re-projected to increase as a result of core reloading, core configuration, power uprating, or when a surveillance capsule is pulled from the reactor vessel. In the case that the calculated plant specific $TWCF_{95-TOTAL}$ value exceeds the pilot plant value (at any time the evaluation is performed), additional evaluation shall be performed to demonstrate that the change-in-risk associated with the extension in the inservice inspection is acceptable.

Frequency and Severity of Design Basis Transients

It is necessary to demonstrate that the amount of fatigue crack growth considered in the pilot plant analyses is bounding for a specific plant. Since the amount of fatigue crack growth used in the pilot plant analyses was calculated based on the design basis transients, a comparison of design basis transients shall be performed to ensure that the assumed number of heatup-coodown transients per year is also applicable for the specific plant. The pilot plant basis depends on the applicable design:

Westinghouse:	7 Cooldowns per year
CE:	13 Cooldowns per year
B&W:	12 Cooldowns per year (assumed)

For CE and Westinghouse designs, if the specific plant has operated within its design basis, the amount of fatigue crack growth in the pilot plant will be bounding. If a plant has operated outside of its design basis, an additional evaluation is required to demonstrate that the pilot plant fatigue crack growth is still bounding. For B&W plants, an evaluation must be performed to show that the fatigue crack growth used in the pilot plant analysis is bounding of that which may occur at the specific plant. If the plant specific fatigue crack growth is projected to be greater than pilot plant fatigue crack growth considered, additional evaluation is required to demonstrate that the change-in-risk associated with the extension in the inservice inspection is acceptable.

Cladding Layers

The pilot plant analyses were performed assuming a single layer of cladding because the probability of having a surface breaking flaw in multi-layer cladding is much less than that of single-layer cladding. The licensee shall identify whether their reactor vessel was fabricated with single or multi-layer cladding. Since the pilot plant analyses were performed with single-layer, all plants are bounded by this parameter.

Table A-2 Additional Information Pertaining to the Reactor Vessel Inspection

Inspection methodology:	
Number of past inspections:	
Number of indications found:	
Proposed inspection schedule for balance of plant life:	

Table A-2 is to be completed with plant specific reactor vessel inservice inspection data to meet the requirements stated as follows.

Inspection Methodology

The licensee shall identify the methodology used for the most recent inservice inspection performed on the ASME Category B-A and B-D welds that are included in this evaluation. Typically the methodology used will be either Regulatory Guide 1.150 or ASME Section XI Appendix VIII.

Number of Past Inspections

The licensee shall identify the number of past inspections that that have been performed on the ASME Category B-A and B-D welds that are included in this evaluation.

Number of Indications Found

The licensee shall identify the number of flaws of concern (as defined in part c of the requirements above) found during the most recent inservice inspection.

Proposed Inspection Schedule for Balance of Plant Life

The licensee shall identify the years in which future inservice inspections will be performed. The dates identified 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. If the licensee identifies dates in Table A-2 that are not within plus or minus one refueling cycle of the dates in the plan, the licensee will be required to have additional discussion with the NRC Staff.

After implementation of the extended interval the following requirement must be met:

> *All data on embedded flaws of concern with a through-wall extent (TWE) greater than 0.1 inch shall be provided to NRC within one year of completing the next vessel beltline inservice inspection per ASME Section XI, Appendix VIII, Supplement 4. For potential vessel failure due to PTS, embedded flaws of concern are axially oriented planar flaws in the vessel beltline within the inner 12.5% ($1/8^{th}$) of the vessel wall thickness.*
>
> *An assessment of the inservice inspection results relative to the flaw distributions used in the pilot plant analyses shall also be provided. This assessment shall be performed in accordance with the requirements of Section (d) in the final published version of the voluntary PTS rule, 10 CFR 50.61(a).*

Table A-3 Details of TWCF Calculation

Inputs								
Reactor Coolant System Temperature, T_{RCS}[°F]:							T_{wall} [inches]:	
#	Region/Component Description	Material	Cu [wt%]	Ni [wt%]	P [wt%]	Mn [wt%]	Un-Irradiated $RT_{NDT(u)}$ [°F]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]

Outputs						
Methodology Used to Calculate ΔT_{30}:						
	Region # (From Above)	$RT_{MAX\text{-}XX}$ [R]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]	ϕ (flux)	ΔT_{30} [°F]	$TWCF_{95\text{-}XX}$
Limiting Axial Weld - AW						
Limiting Plate - PL						
Forging - FO						
Circumferential Weld - CW						
$TWCF_{95\text{-}TOTAL}$ ($\alpha_{AW}TWCF_{95\text{-}AW} + \alpha_{PL}TWCF_{95\text{-}PL} + \alpha_{CW}TWCF_{95\text{-}CW} + \alpha_{FO}TWCF_{95\text{-}FO}$):						

The information used to calculate the through wall cracking frequency (TWCF) shall be included in Table A-3. The fields are defined in the TWCF correlation definition for Table A-1. Additional rows should be added to the input section for each beltline region/component where region/component is a particular weld, plate, or forging. Refer to Appendices A-1 and A-2 for examples of how Table A-3 are to be completed.

APPENDIX A-1
WOLF CREEK PLANT IMPLEMENTATION EXAMPLE

Table 1	Critical Parameters for Application of Bounding Analysis		
Parameter	**Pilot Plant Basis**	**Plant Specific Basis**	**Additional Evaluation Required? (Y/N)**
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are applicable	NRC PTS Risk Study (Reference 8)	PTS Generalization Study (Reference 26)	No
Through Wall Cracking Frequency	1.76-08 Events per year	2.51E-13 Events per year	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldowns per year	Bounded by 7 heatup/cooldowns per year	No
Cladding Layers (Single/Multiple)	Single	Single (assumed)	No

Table 2	Additional Information Pertaining to Reactor Vessel Inspection
Inspection methodology:	Past inspections have been performed to Regulatory Guide 1.150. Inspections performed during RF13 and RF 14 were also performed to ASME Section XI Appendix VIII.
Number of past inspections:	- Category B-A welds (reactor vessel): 2 inspections, RF8 – Spring 1996 and RF14 – Spring 2005 with the exception of weld RV-101-121 which was also inspected in RF2 – Spring 1987 and RF10 – Spring 1999 - Category B-A welds (closure head): 2 inspections, Interval 1 examinations in RF1 – Fall 1986, RF4 – Spring 1990, and RF6 – Spring 1993. Interval 2 examinations were performed in RF9 – Fall 1997, RF11 – Fall 2000, and RF13 – Fall 2003. 2 welds were examined each outage. - Category B-D welds (outlet nozzles): 3 inspections RF3 – Fall 1988, RF8 – Spring 1996, RF14 – Spring 2005 - Category B-D welds (inlet nozzles): 2 inspections, RF8 – Spring 1996, RF14 – Spring 2005
Number of indications found:	Zero reportable indications have been found to date. Any recordable indications have been acceptable per ASME Section XI IWB-3500. No flaws of concern were detected.
Proposed inspection schedule for balance of plant life:	Third inservice inspection currently scheduled for 2015. The third inservice inspection is proposed to be performed in 2025. The fourth inservice inspection interval is proposed to be performed in 2045.

Table 3 **Details of TWCF Calculation**								
Inputs								
Reactor Coolant System Temperature, T_{RCS}[°F]:			550	T_{wall} [inches]:				8.62
#	Region/Component Description	Material	Cu [wt%]	Ni [wt%]	P [wt%]	Mn [wt%]	Un-Irradiated $RT_{NDT(u)}$ [°F]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]
1	Lower Shell Plate	A 533B	0.070	0.620	0.008	1.35	40.0	3.90
2	Lower Shell Plate	A 533B	0.090	0.670	0.009	1.35	0.0	3.90
3	Lower Shell Plate	A 533B	0.060	0.640	0.008	1.35	10.0	3.90
4	Intermediate Shell Plate	A 533B	0.040	0.640	0.007	1.35	-20.0	3.90
5	Intermediate Shell Plate	A 533B	0.050	0.630	0.007	1.35	-20.0	3.90
6	Intermediate Shell Plate	A 533B	0.040	0.660	0.008	1.35	-20.0	3.90
7	Lower Shell Axial Weld	Linde 0091	0.040	0.080	0.005	1.35	-50.0	1.76
8	Lower Shell Axial Weld	Linde 0091	0.040	0.080	0.005	1.35	-50.0	3.42
9	Lower Shell Axial Weld	Linde 0091	0.040	0.080	0.005	1.35	-50.0	3.42
10	Inter. Shell Axial Weld	Linde 0091	0.040	0.080	0.005	1.35	-50.0	1.76
11	Inter. Shell Axial Weld	Linde 0091	0.040	0.080	0.005	1.35	-50.0	3.42
12	Inter. Shell Axial Weld	Linde 0091	0.040	0.080	0.005	1.35	-50.0	3.42
13	Inter. – Lower Circ Weld	Linde 124	0.040	0.080	0.007	1.35	-50.0	3.90

Outputs						
Methodology Used to Calculate ΔT_{30}:			NUREG-1874			
	Region # (From Above)	$RT_{MAX\text{-}XX}$ [R]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]	ϕ (flux)	ΔT_{30} [°F]	$TWCF_{95\text{-}XX}$
Limiting Axial Weld - AW	1	561.96	3.42	1.81E+10	62.27	2.47E-18
Limiting Plate - PL	1	565.05	3.90	2.06E+10	65.36	1.00E-13
Forging - FO	N/A	N/A	N/A	N/A	N/A	N/A
Circumferential Weld - CW	1	565.05	3.90	2.06E+10	65.36	5.52E-29
$TWCF_{95\text{-}TOTAL}$ ($\alpha_{AW}TWCF_{95\text{-}AW} + \alpha_{PL}TWCF_{95\text{-}PL} + \alpha_{CW}TWCF_{95\text{-}CW} + \alpha_{FO}TWCF_{95\text{-}FO}$):						2.51E-13

APPENDIX A-2
WATERFORD 3 PLANT IMPLEMENTATION EXAMPLE

Table 1 Critical Parameters for Application of Bounding Analysis

Parameter	Pilot Plant Basis	Plant Specific Basis	Additional Evaluation Required? (Y/N)
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are applicable	NRC PTS Risk Re-Evaluation (Reference 8)	PTS Generalization Study (Reference 26)	No
Through Wall Cracking Frequency	3.16E-07 Events per year	2.87E-14 Events per year	No
Frequency and Severity of Design Basis Transients	13 heatup/cooldowns per year	Bounded by 13 heatup/cooldowns per year	No
Cladding Layers (Single/Multiple)	Single	Single	No

Table 2 Additional Information Pertaining to Reactor Vessel Inspection

Inspection methodology:	Past inspections have been performed to Regulatory Guide 1.150
Number of past inspections:	- Category B-A welds (reactor vessel): 1 inspection – 1995, with the exception of weld 01-020 which was also inspected in 1988. - Category B-A welds (closure head): 4 inspections with 3 welds inspected 1986, 3 welds inspected 1989, 1 weld inspected 1994, 3 welds inspected 2000 - Category B-D welds (outlet nozzles): 2 inspections – 1988 and 1995, with the exception of weld 01-021 which was also inspected in 1989. - Category B-D welds (inlet nozzles): 1 inspection – 1995
Number of indications found:	Zero reportable indications have been found to date. Any recordable indications have been acceptable per ASME Section XI IWB-3500. No flaws of concern have been detected.
Proposed inspection schedule for balance of plant life:	Second inservice inspection currently scheduled for Spring 2008. The second inservice inspection is proposed to be performed in 2015. The third inservice inspection is proposed to be performed in 2035.

Table 3 Details of TWCF Calculation

Inputs								
Reactor Coolant System Temperature, T_{RCS}[°F]:			553	T_{wall} [inches]:				8.62
#	Region/Component Description	Material	Cu [wt%]	Ni [wt%]	P [wt%]	Mn [wt%]	Un-Irradiated $RT_{NDT(u)}$ [°F]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]
1	Lower Shell Plate	A 533B	0.030	0.580	0.005	1.35	22.0	4.49
2	Lower Shell Plate	A 533B	0.030	0.620	0.006	1.35	-15.0	4.49
3	Lower Shell Plate	A 533B	0.030	0.620	0.007	1.35	-10.0	4.49
4	Intermediate Shell Plate	A 533B	0.020	0.700	0.007	1.35	-42.0	4.49
5	Intermediate Shell Plate	A 533B	0.020	0.710	0.004	1.35	-30.0	4.49
6	Intermediate Shell Plate	A 533B	0.020	0.670	0.006	1.35	-50.0	4.49
7	Lower Shell Axial Weld	Linde 0091	0.030	0.200	0.007	1.35	-80.0	4.49
8	Lower Shell Axial Weld	Linde 0091	0.030	0.200	0.007	1.35	-80.0	4.49
9	Lower Shell Axial Weld	Linde 0091	0.030	0.200	0.007	1.35	-80.0	4.49
10	Inter. Shell Axial Weld	E 8018	0.020	0.960	0.010	1.35	-60.0	4.50
11	Inter. Shell Axial Weld	E 8018	0.020	0.960	0.010	1.35	-60.0	4.50
12	Inter. Shell Axial Weld	E 8018	0.020	0.960	0.010	1.35	-60.0	4.50
13	Inter. – Lower Circ. Weld	Linde 0091	0.050	0.160	0.008	1.35	-70.0	4.49

Outputs						
Methodology Used to Calculate ΔT_{30}:			NUREG-1874			
	Region # (From Above)	RT_{MAX-XX} [R]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]	ϕ (flux)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Axial Weld - AW	1	541.91	4.49	2.37E10	57.93	2.47E-18
Limiting Plate - PL	1	541.91	4.49	2.37E10	57.93	5.52E-29
Forging - FO	N/A	N/A	N/A	N/A	N/A	N/A
Circumferential Weld - CW	1	541.91	4.49	2.37E10	57.93	1.15E-14
$TWCF_{95-TOTAL}$ ($\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW} + \alpha_{FO}TWCF_{95-FO}$):						2.87E-14

APPENDIX B
INPUTS FOR THE BEAVER VALLEY UNIT 1 PILOT PLANT EVALUATION

A summary of the NDE inspection history based on Regulatory Guide 1.150 and pertinent input data for BV1 is as follows:

1. Number of ISIs performed (relative to initial pre-service and 10–year interval inspections) for full penetration Category B-A and B-D reactor vessel welds assuming all of the candidate welds were inspected: 2 (covering all welds of the specified categories).

2. The inspections performed covered: A total of 34 items. 15 Category B-A items had coverage of <90%. 1 Category B-A item had coverage > 90% but <100%. 6 Category B-A items had coverage of 100%. 6 Category B-D items had coverage of 90% and 6 had coverage of 100%.

3. Number of indications found during the most recent inservice inspection: 42
   This number includes consideration of the following additional information.

   a. Indications found that were reportable: 0
   b. Indications found that were within acceptable limits: 42
   c. Indications/anomalies currently being monitored: 0

4. Full penetration relief requests for the RV were submitted and accepted by the NRC for 15 items.

5. Fluence distribution at inside surface of RV beltline until end of life (EOL): see Figure B-1 taken from the NRC PTS Risk Study [44], Figure 4.2.

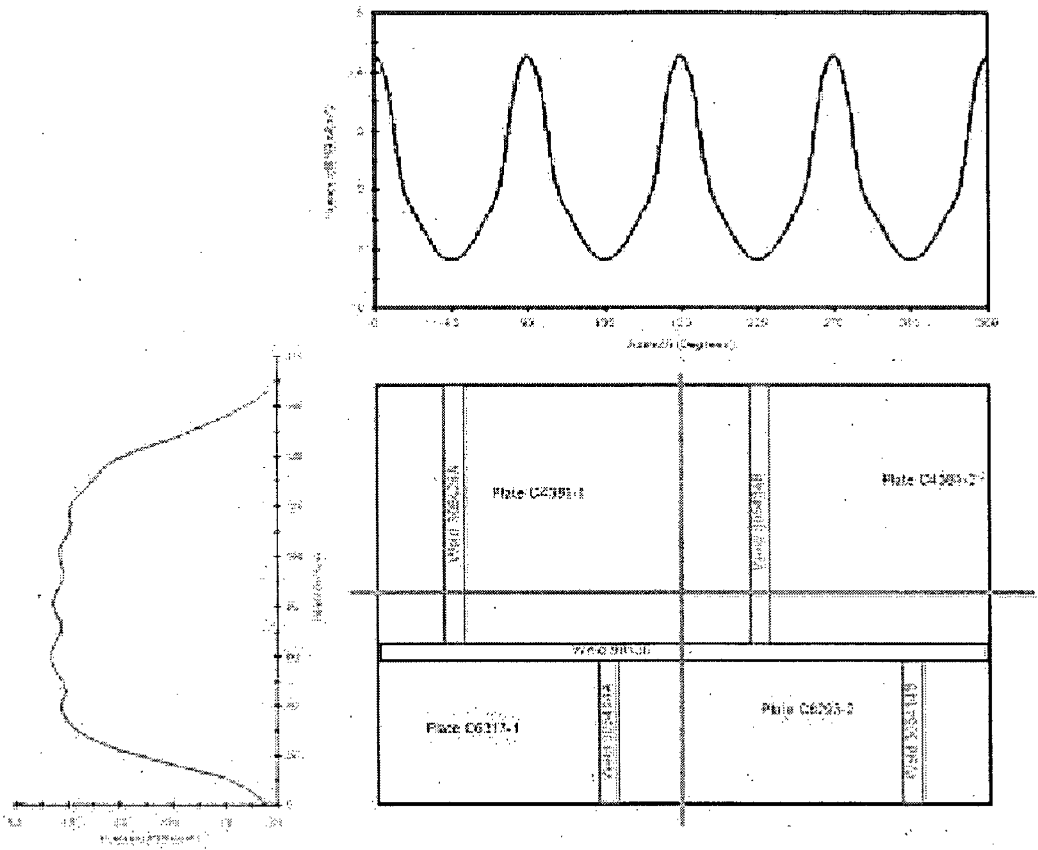

Figure B-1 Rollout Diagram of Beltline Materials and Representative Fluence Maps for BV1

6. Reactor vessel cladding details:

   a. Thickness: 0.156 inches

   b. Material properties are identified in Table B-1. This is consistent with the NRC PTS Risk Study [8, 9]:

Temperature (°F)	Thermal Conductivity (Btu/hr-ft-°F) "K"	Specific Heat (Btu/LBM-°F) "C"	Young's Modulus of Elasticity (KSI) "E"	Thermal Expansion Coefficient ($°F^{-1}$) "α"	Density (LBM/ft^3) "ρ"	Poisson's Ratio "ν"
0	-	-	-	-	489	.3
68	-	-	22045.7	-	489	.3
70	8.1	0.1158	-	-	489	.3
100	8.4	0.1185	-	8.55E-06	489	.3
150	8.6	0.1196	-	8.67E-06	489	.3
200	8.8	0.1208	-	8.79E-06	489	.3
250	9.1	0.1232	-	8.9E-06	489	.3
300	9.4	0.1256	-	9.0E-06	489	.3
302	-	-	20160.2	-	489	.3
350	9.6	0.1258	-	9.1E-06	489	.3
400	9.9	0.1281	-	9.19E-06	489	.3
450	10.1	0.1291	-	9.28E-06	489	.3
482	-	-	18419.8	-	489	.3
500	10.4	0.1305	-	9.37E-06	489	.3
550	10.6	0.1306	-	9.45E-06	489	.3
600	10.9	0.1327	-	9.53E-06	489	.3
650	11.1	0.1335	-	9.61E-06	489	.3
700	11.4	0.1348	-	9.69E-06	489	.3
750	11.6	0.1356	-	9.76E-06	489	.3
800	11.9	0.1367	-	9.82E-06	489	.3

Table B-1 Cladding Material Properties

c. Material including copper and nickel content: Material properties assigned to clad flaws are that of the underlying material be it base metal or weld. These properties are identified in Table B-3. This is consistent with the NRC PTS Risk Study [8, 9].

d. Material property uncertainties:

1) Bead width: 1 inch – bead widths vary for all plants. Based on the NRC PTS Risk Study [8, 9], a nominal dimension of 1 inch is selected for all analyses because this parameter is not expected to influence significantly the predicted reactor vessel failure probabilities.

2) Truncation limit: Cladding thickness rounded to the next 1/100th of the total reactor vessel thickness to be consistent with the NRC PTS Risk Study [8, 9].

3) Surface flaw depth: 0.161 inch

4) All cladding flaws are surface-breaking. Only flaws in cladding that would influence brittle fracture of the reactor vessel are brittle. This is consistent with the NRC PTS Risk Study [8, 9].

e. Additional cladding properties are identified in Table B-4.

7. Base metal:

a. Wall thickness: 7.875 inches

b. Material properties are identified in Table B-2 and B-3. This is consistent with the NRC PTS Risk Study [8, 9]:

Table B-2 Base Metal Material Properties

Temperature (°F)	Thermal Conductivity (Btu/hr-ft-°F) "K"	Specific Heat (Btu/LBM-°F) "C"	Young's Modulus of Elasticity (KSI) "E"	Thermal Expansion Coefficient ($°F^{-1}$) "α"	Density (LBM/ft^3) "ρ"	Poisson's Ratio "ν"
0	-	-	-	-	489	.3
70	24.8	0.1052	29200	-	489	.3
100	25	0.1072	-	7.06E-06	489	.3
150	25.1	0.1101	-	7.16E-06	489	.3
200	25.2	0.1135	28500	7.25E-06	489	.3
250	25.2	0.1166	-	7.34E-06	489	.3
300	25.1	0.1194	28000	7.43E-06	489	.3
350	25	0.1223	-	7.5E-06	489	.3
400	25.1	0.1267	27400	7.58E-06	489	.3
450	24.6	0.1277	-	7.63E-06	489	.3
500	24.3	0.1304	27000	7.7E-06	489	.3
550	24	0.1326	-	7.77E-06	489	.3
600	23.7	0.135	26400	7.83E-06	489	.3
650	23.4	0.1375	-	7.9E-06	489	.3
700	23	0.1404	25300	7.94E-06	489	.3
750	22.6	0.1435	-	8.0E-06	489	.3
800	22.2	0.1474	23900	8.05E-06	489	.3

Table B-3	BV1-Specific Material Values Drawn from the RVID (see Ref. 44, Table 4.1)							
Major Material Region Description				Cu [wt%]	Ni [wt%]	P [wt%]	Mn [wt%]	Un-Irradiated RT_{NDT} [°F]
#	Type	Heat	Location					
1	Axial Weld	305414A	Lower	0.337	0.609	0.012	1.440	- 56
2	Axial Weld	305414B	Lower	0.337	0.609	0.012	1.440	- 56
3	Axial Weld	305424A	Upper	0.273	0.629	0.013	1.440	- 56
4	Axial Weld	305424B	Upper	0.273	0.629	0.013	1.440	- 56
5	Circ Weld	90136	Intermediate	0.269	0.070	0.013	0.964	- 56
6	Plate	C6317-1	Lower	0.200	0.540	0.010	1.310	27
7	Plate	C6293-2	Lower	0.140	0.570	0.015	1.300	20
8	Plate	C4381-2	Upper	0.140	0.620	0.015	1.400	73
9	Plate	C4381-1	Upper	0.140	0.620	0.015	1.400	43

8. Weld metal details: Details of information used in addressing weld-specific information are taken directly from the NRC PTS Risk Study [44], Table 4.2. Summaries are reproduced as Table B-4. Values for SAW Weld Volume fraction and Repair Weld Volume fraction in Table B-4 were changed to 96.7% and 2.3% respectively per NUREG-1874 [9].

Table B-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution

Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Inner Radius (to cladding)	[in]	85.5	78.5	86	86	Vessel specific info
Base Metal Thickness	[in]	8.438	7.875	8.5	8.675	Vessel specific info
Total Wall Thickness	[in]	8.626	8.031	8.75	8.988	Vessel specific info

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SAW Weld	Volume fraction	[%]	97%				100% - SMAW% - REPAIR%
	Thru-Wall Bead Thickness	[in]	0.1875	0.1875	0.1875	0.1875	All plants report plant specific dimensions of 3/16-in.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table B-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SMAW Weld	Volume fraction	[%]	1%				Upper bound to all plant specific info provided by Steve Byrne (Westinghouse – Windsor).
	Thru-Wall Bead Thickness	[in]	0.21	0.20	0.22	0.25	Oconee is generic value based on average of all plants specific values (including Shoreham & PVRUF data). Other values are plant specific as reported by Steve Byrne.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table B-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Repair Weld	Volume fraction	[%]	2%				Judgment. A rounded integral percentage that exceeds the repaired volume observed for Shoreham and for PVRUF, which was 1.5%.
	Thru-Wall Bead Thickness	[in]	0.14				Generic value: As observed in PVRUF and Shoreham by PNNL.
	Truncation Limit	[in]	2				Judgment. Approx. 2X the largest repair flaw found in PVRUF & Shoreham. Also based on maximum expected width of repair cavity.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				The repair flaws had complex shapes and orientations that were not aligned with either the axial or circumferential welds; for consistency with the available treatments of flaws by the FAVOR code, a common treatment of orientations was adopted for flaws in SAW/SMAW and repair welds.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table B-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

	Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Cladding	Actual Thickness	[in]	0.188	0.156	0.25	0.313	Vessel specific info
	# of Layers	[#]	1	2	2	2	Vessel specific info
	Bead Width	[in]	1				Bead widths of 1 to 5-in. characteristic of machine deposited cladding. Bead widths down to ½-in. can occur over welds. Nominal dimension of 1-in. selected for all analyses because this parameter is not expected to influence significantly the predicted vessel failure probabilities. May need to refine this estimate later, particularly for Oconee who reported a 5-in bead width.
	Truncation Limit	[in]	Actual clad thickness rounded to the nearest 1/100th of the total vessel wall thickness				Judgment & computational convenience
	Surface flaw depth in FAVOR	[in]	0.259	0.161	0.263	0.360	
	Buried or Surface	--	All flaws are surface breaking				Judgment. Only flaws in cladding that would influence brittle fracture of the vessel are brittle. Material properties assigned to clad flaws are that of the underlying material, be it base or weld.
	Orientation	--	All circumferential.				Observation: All flaws observed in PVRUF & Shoreham were lack of inter-run fusion defects, and cladding is always deposited circumferentially
	Density basis	--	No surface flaws observed. Density is 1/1000 that of the observed buried flaws in cladding of vessels examined by PNNL. If there is more than one clad layer then there are no clad flaws.				Judgment
	Aspect ratio basis	--	Observations on buried flaws				Judgment
	Depth basis	--	Depth of all surface flaws is the actual clad thickness rounded up to the nearest 1/100th of the total vessel wall thickness.				Judgment.

Table B-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Plate	Truncation Limit	[in]	0.433				Judgment. Twice the depth of the largest flaw observed in all PNNL plate inspections.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Half of the simulated flaws are circumferential, half are axial.				Observation & Physics: No observed orientation preference, and no reason to suspect one (other than laminations which are benign.
	Density basis	--	1/10 of small weld flaw density, 1/40 of large weld flaw density of the PVRUF data				Judgment. Supported by limited data.
	Aspect ratio basis	--	Same as for PVRUF welds				Judgment
	Depth basis	--	Same as for PVRUF welds				Judgment. Supported by limited data.

9. $TWCF_{95\text{-}TOTAL}$ value calculated at 60 EFPY using correlations from NUREG-1874 (Reference 9): 1.76E-08 Events per year

APPENDIX C
BEAVER VALLEY UNIT 1 PROBSBFD OUTPUT

C-1: 10 Year ISI Only

```
                    STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
 WESTINGHOUSE           MONTE-CARLO SIMULATION PROGRAM PROBSBFD        VERSION
1.0

=============================================================================
==
      INPUT VARIABLES FOR CASE  3: BV1 HUCD 10 YR ISI ONLY

      NCYCLE =     80           NFAILS =  1001            NTRIAL =  1000
      NOVARS =     19           NUMSET =     2            NUMISI =     5
      NUMSSC =      4           NUMTRC =     4            NUMFMD =     4

    VARIABLE          DISTRIBUTION       MEDIAN       DEVIATION     SHIFT     USAGE
   NO.   NAME          TYPE      LOG       VALUE       OR FACTOR     MV/SD    NO. SUB

    1  FIFDepth     - CONSTANT -    2.0000D-02                               1  SET
    2  IFlawDen     - CONSTANT -    3.6589D-03                               2  SET
    3  ICy-ISI      - CONSTANT -    1.0000D+01                               1  ISI
    4  DCy-ISI      - CONSTANT -    8.0000D+01                               2  ISI
    5  MV-Depth     - CONSTANT -    1.5000D-02                               3  ISI
    6  SD-Depth     - CONSTANT -    1.8500D-01                               4  ISI
    7  CEff-ISI     - CONSTANT -    1.0000D+00                               5  ISI
    8  Aspect1      - CONSTANT -    2.0000D+00                               1  SSC
    9  Aspect2      - CONSTANT -    6.0000D+00                               2  SSC
   10  Aspect3      - CONSTANT -    1.0000D+01                               3  SSC
   11  Aspect4      - CONSTANT -    9.9000D+01                               4  SSC
   12  NoTr/Cy      - CONSTANT -    7.0000D+00                               1  TRC
   13  FCGThld      - CONSTANT -    1.5000D+00                               2  TRC
   14  FCGR-UC      NORMAL    NO    0.0000D+00   1.0000D+00       .00        3  TRC
   15  DKINFile     - CONSTANT -    1.0000D+00                               4  TRC
   16  Percent1     - CONSTANT -    5.6175D+01                               1  FMD
   17  Percent2     - CONSTANT -    3.0283D+01                               2  FMD
   18  Percent3     - CONSTANT -    3.9086D+00                               3  FMD
   19  Percent4     - CONSTANT -    9.6333D+00                               4  FMD

  INFORMATION GENERATED FROM FAVLOADS.DAT FILE
  AND SAVED IN DKINSAVE.DAT FILE:

  WALL THICKNESS =  8.0360 INCH

  FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

    TYPE 1 WITH AN ASPECT RATIO OF   2.

    8.03600D-02    2.41927D+00    1.03655D+01
    1.47862D-01    3.22858D+00    1.40170D+01
    4.01800D-01    1.29279D+01    1.75751D+01
    6.02700D-01    1.41327D+01    2.09080D+01
    8.03600D-01    1.49423D+01    2.33544D+01
    1.60720D+00    1.45812D+01    2.72710D+01
    2.41080D+00    1.02448D+01    2.63600D+01
    4.01800D+00    2.35823D+00    2.78623D+01
```

C-1: 10 Year ISI Only (cont.)

TYPE 2 WITH AN ASPECT RATIO OF 6.

8.03600D-02	3.63673D+00	1.56338D+01
1.47862D-01	4.95557D+00	2.15454D+01
4.01800D-01	1.90999D+01	2.63794D+01
6.02700D-01	2.31650D+01	3.16223D+01
8.03600D-01	2.48064D+01	3.60464D+01
1.60720D+00	2.65025D+01	4.51155D+01
2.41080D+00	2.31198D+01	4.76172D+01
4.01800D+00	1.54934D+01	5.27667D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.03600D-02	3.98451D+00	1.71374D+01
1.47862D-01	5.29827D+00	2.30393D+01
4.01800D-01	2.02922D+01	2.81955D+01
6.02700D-01	2.51750D+01	3.36684D+01
8.03600D-01	2.69393D+01	3.84779D+01
1.60720D+00	2.92755D+01	4.91684D+01
2.41080D+00	2.74642D+01	5.45509D+01
4.01800D+00	2.02195D+01	6.28814D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.03600D-02	6.51796D+00	1.75511D+01
1.60720D-01	1.01756D+01	2.28059D+01
2.41080D-01	1.54398D+01	2.23553D+01
4.01800D-01	2.18696D+01	2.94323D+01
6.02700D-01	2.69582D+01	3.66108D+01
8.03600D-01	2.88204D+01	4.17713D+01
1.60720D+00	3.37365D+01	5.67413D+01
2.41080D+00	3.35927D+01	6.64759D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

8	4.4254D-04	1.4320D-04	1.4728D-05	4.7035D-05
9	0.0000D+00	8.8686D-05	1.4703D-05	2.7347D-05
10	0.0000D+00	4.4175D-06	9.2631D-07	7.2598D-07
11	0.0000D+00	2.2821D-07	5.9150D-08	7.0131D-08
12	0.0000D+00	2.2665D-07	2.9099D-08	0.0000D+00
13	0.0000D+00	0.0000D+00	2.8861D-08	0.0000D+00

C-2: ISI Every 10 Years

```
                    STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
 WESTINGHOUSE            MONTE-CARLO SIMULATION PROGRAM PROBSBFD        VERSION
1.0

==============================================================================
==
      INPUT VARIABLES FOR CASE  2: BV1 HUCD 10 YR ISI INT

      NCYCLE =     80           NFAILS =  1001          NTRIAL =  1000
      NOVARS =     19           NUMSET =     2          NUMISI =     5
      NUMSSC =      4           NUMTRC =     4          NUMFMD =     4

    VARIABLE        DISTRIBUTION       MEDIAN       DEVIATION    SHIFT    USAGE
   NO.   NAME       TYPE      LOG       VALUE       OR FACTOR    MV/SD   NO. SUB

    1  FIFDepth    - CONSTANT -    2.0000D-02                              1  SET
    2  IFlawDen    - CONSTANT -    3.6589D-03                              2  SET
    3  ICy-ISI     - CONSTANT -    1.0000D+01                              1  ISI
    4  DCy-ISI     - CONSTANT -    1.0000D+01                              2  ISI
    5  MV-Depth    - CONSTANT -    1.5000D-02                              3  ISI
    6  SD-Depth    - CONSTANT -    1.8500D-01                              4  ISI
    7  CEff-ISI    - CONSTANT -    1.0000D+00                              5  ISI
    8  Aspect1     - CONSTANT -    2.0000D+00                              1  SSC
    9  Aspect2     - CONSTANT -    6.0000D+00                              2  SSC
   10  Aspect3     - CONSTANT -    1.0000D+01                              3  SSC
   11  Aspect4     - CONSTANT -    9.9000D+01                              4  SSC
   12  NoTr/Cy     - CONSTANT -    7.0000D+00                              1  TRC
   13  FCGThld     - CONSTANT -    1.5000D+00                              2  TRC
   14  FCGR-UC     NORMAL    NO    0.0000D+00   1.0000D+00      .00        3  TRC
   15  DKINFile    - CONSTANT -    1.0000D+00                              4  TRC
   16  Percent1    - CONSTANT -    5.6175D+01                              1  FMD
   17  Percent2    - CONSTANT -    3.0283D+01                              2  FMD
   18  Percent3    - CONSTANT -    3.9086D+00                              3  FMD
   19  Percent4    - CONSTANT -    9.6333D+00                              4  FMD

    INFORMATION GENERATED FROM FAVLOADS.DAT FILE
    AND SAVED IN DKINSAVE.DAT FILE:

    WALL THICKNESS =  8.0360 INCH

    FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

      TYPE 1 WITH AN ASPECT RATIO OF   2.

      8.03600D-02    2.41927D+00    1.03655D+01
      1.47862D-01    3.22858D+00    1.40170D+01
      4.01800D-01    1.29279D+01    1.75751D+01
      6.02700D-01    1.41327D+01    2.09080D+01
      8.03600D-01    1.49423D+01    2.33544D+01
      1.60720D+00    1.45812D+01    2.72710D+01
      2.41080D+00    1.02448D+01    2.63600D+01
      4.01800D+00    2.35823D+00    2.78623D+01
```

C-2: ISI Every 10 Years (cont.)

TYPE 2 WITH AN ASPECT RATIO OF 6.

8.03600D-02	3.63673D+00	1.56338D+01
1.47862D-01	4.95557D+00	2.15454D+01
4.01800D-01	1.90999D+01	2.63794D+01
6.02700D-01	2.31650D+01	3.16223D+01
8.03600D-01	2.48064D+01	3.60464D+01
1.60720D+00	2.65025D+01	4.51155D+01
2.41080D+00	2.31198D+01	4.76172D+01
4.01800D+00	1.54934D+01	5.27667D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.03600D-02	3.98451D+00	1.71374D+01
1.47862D-01	5.29827D+00	2.30393D+01
4.01800D-01	2.02922D+01	2.81955D+01
6.02700D-01	2.51750D+01	3.36684D+01
8.03600D-01	2.69393D+01	3.84779D+01
1.60720D+00	2.92755D+01	4.91684D+01
2.41080D+00	2.74642D+01	5.45509D+01
4.01800D+00	2.02195D+01	6.28814D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.03600D-02	6.51796D+00	1.75511D+01
1.60720D-01	1.01756D+01	2.28059D+01
2.41080D-01	1.54398D+01	2.23553D+01
4.01800D-01	2.18696D+01	2.94323D+01
6.02700D-01	2.69582D+01	3.66108D+01
8.03600D-01	2.88204D+01	4.17713D+01
1.60720D+00	3.37365D+01	5.67413D+01
2.41080D+00	3.35927D+01	6.64759D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

8	4.3486D-08	1.2355D-08	1.2447D-09	4.0471D-09
9	0.0000D+00	6.1902D-09	9.9626D-10	1.8380D-09
10	0.0000D+00	1.8825D-10	3.7663D-11	2.6218D-11
11	0.0000D+00	4.7355D-12	1.6752D-12	1.3302D-12
12	0.0000D+00	3.5199D-12	4.3837D-13	0.0000D+00
13	0.0000D+00	0.0000D+00	3.0423D-13	0.0000D+00

APPENDIX D
BEAVER VALLEY UNIT 1 PTS TRANSIENTS

Table D-1 PTS Transient Descriptions for BV1

Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Dominant*
1	002	3.59 cm [1.414 in] surge line break	None.	1.23E-04	No	No
2	003	5.08 cm [2 in] surge line break	None.	9.76E-05	No	No
3	007	2.54 cm [8 in] surge line break	None.	2.11E-05	No	Yes at 32, 60, 100, 200 EFPY
4	009	2.54 cm [16 in] hot leg break	None.	6.99E-06	No	Yes at 32, 60, 100, 200 EFPY
5	014	Reactor/turbine trip w/one stuck open pressurizer SRV	None.	2.22E-04	No	No
6	031	Reactor/turbine trip w/feed and bleed (Operator open all pressurizer PORVs and use all charging/HHSI pumps)	None.	3.10E-07	No	No
7	034	Reactor/turbine trip w/two stuck open pressurizer SRV's	None.	4.95E-07	No	No
8	056	10.16 cm [4.0 in] surge line break	None.	1.23E-04	Yes	Yes at 32, 60, 100, 200 EFPY
9	059	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 3,000 s.	None.	3.46E-04	No	No
10	060	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s.	None.	2.15E-05	No	Yes at 32, 60, 100 EFPY
11	061	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 3,000 s.	None.	1.79E-06	No	No
12	062	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 6,000 s.	None.	1.08E-07	No	No
13	064	Reactor/turbine trip w/two stuck open pressurizer SRV's	None.	8.67E-08	Yes	No
14	065	Reactor/turbine trip w/two stuck open pressurizer SRV's and HHSI failure	Operator opens all ASDVs 5 minutes after HHSI would have come on.	1.04E-09	No	No

Table D-1 PTS Transient Descriptions for BV1

Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Dominant*
15	066	Reactor/turbine trip w/two stuck open pressurizer SRV's. One valve recloses at 3000 seconds while the other valve remains open.	None.	1.18E-06	No	No
16	067	Reactor/turbine trip w/two stuck open pressurizer SRV's. One valve recloses at 6000 seconds while the other valve remains open.	None.	1.18E-06	No	No
17	068	Reactor/turbine trip w/two stuck open pressurizer SRV's that reclose at 6000 s with HHSI failure.	Operator opens all ASDVs 5 minutes after HHSI would have come on.	1.33E-08	No	No
18	069	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 3,000 s.	None.	2.09E-08	Yes	No
19	070	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 6,000 s.	None.	2.09E-08	Yes	No
20	071	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s.	None.	3.74E-06	Yes	Yes at 32 EFPY
21	072	Reactor/turbine trip w/one stuck open pressurizer SRV with HHSI failure.	Operator opens all ASDVs 5 minutes after HHSI would have come on.	5.14E-07	No	No
22	073	Reactor/turbine trip w/one stuck open pressurizer SRV with HHSI failure	Operator open all ASDVs 5 minutes after HHSI would have come on.	6.56E-08	Yes	No
23	074	Main steam line break with AFW continuing to feed affected generator	None.	1.46E-06	No	No
24	076	Reactor/turbine trip w/full MFW to all 3 SGs (MFW maintains SG level near top).	Operator trips reactor coolant pumps.	1.06E-04	Yes	No
25	078	Reactor/turbine trip with failure of MFW and AFW.	Operator opens all ASDVs to let condensate fill SGs.	3.25E-08	No	No
26	081	Main Steam Line Break with AFW continuing to feed affected generator and with HHSI failure initially.	Operator opens ADVs (on intact generators). HHSI is restored after CFTs discharge 50%.	2.65E-06	No	No
27	082	Reactor/turbine trip w/one stuck open pressurizer SRV (recloses at 6000 s) and with HHSI failure.	Operator opens all ASDVs 5 minutes after HHSI would have started.	1.51E-06	No	No

Table D-1 PTS Transient Descriptions for BV1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**Mean IE Frequency**	**HZP**	**Dominant***
28	083	2.54 cm [1.0 in] surge line break with HHSI failure and motor driven AFW failure. MFW is tripped. Level control failure causes all steam generators to be overfed with turbine AFW, with the level maintained at top of SGs.	Operator trips RCPs. Operator opens all ASDVs 5 minutes after HHSI would have come on.	3.51E-06	No	No
29	092	Reactor/turbine trip w/two stuck open pressurizer SRV's, one recloses at 3000 s.	None.	2.13E-07	Yes	No
30	093	Reactor/turbine trip w/two stuck open pressurizer SRV's. One valve recloses at 6000 seconds while the other valve remains open.	None.	2.13E-07	Yes	No
31	094	Reactor/turbine trip w/one stuck open pressurizer SRV.	None.	4.10E-05	Yes	No
32	097	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 3,000 s.	None.	3.74E-06	Yes	Yes at 32, 60 EFPY
33	102	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	1.02E-04	No	Yes at 100, 200 EFPY
34	103	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	1.07E-05	Yes	Yes at 60, 100, 200 EFPY
35	104	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 60 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	1.09E-04	No	Yes at 100, 200 EFPY

Table D-1 PTS Transient Descriptions for BV1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**Mean IE Frequency**	**HZP**	**Dominant***
36	105	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 60 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	1.07E-05	Yes	No
37	106	Main steam line break with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	2.21E-05	No	No
38	107	Main steam line break with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	4.31E-07	Yes	No
39	108	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed.	6.46E-04	Yes	No
40	109	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	6.81E-05	Yes	No
41	110	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes	Operator controls HHSI 60 minutes after allowed.	6.91E-04	No	Yes at 200 EFPY
42	111	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 60 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	6.82E-05	Yes	No

Table D-1 PTS Transient Descriptions for BV1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**Mean IE Frequency**	**HZP**	**Dominant***
43	112	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	1.41E-05	No	No
44	113	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	2.74E-06	Yes	No
45	114	7.18 cm [2.828 in] surge line break, summer conditions (HHSI, LHSI temp = 55°F, Accumulator Temp = 105°F), heat transfer coefficient increased 30% (modeled by increasing heat transfer surface area by 30% in passive heat structures).	None.	9.76E-05	No	No
46	115	7.18 cm [2.828 in] cold leg break	None.	9.76E-05	No	No
47	116	14.366 cm [5.657 in] cold leg break with break area increased 30%	None.	1.81E-05	No	No
48	117	14.366 cm [5.657 in] cold leg break, summer conditions (HHSI, LHSI temp = 55°F, Accumulator Temp = 105°F)	None.	2.11E-05	No	No
49	118	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator	None.	9.30E-06	No	No
50	119	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	6.84E-07	No	No
51	120	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	9.98E-07	No	No
52	121	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 3,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	1.33E-07	Yes	No

Table D-1 PTS Transient Descriptions for BV1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**Mean IE Frequency**	**HZP**	**Dominant***
53	122	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 6,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	1.33E-07	Yes	No
54	123	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 3,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	1.65E-07	Yes	Yes at 32 EFPY
55	124	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 6,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	1.65E-07	Yes	No
56	125	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	1.34E-04	No	No
57	126	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	1.87E-04	No	Yes at 32, 60, 100 EFPY
58	127	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	2.59E-05	Yes	No
59	128	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 3,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	2.59E-05	Yes	No
60	129	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	3.09E-05	Yes	Yes at 32, 60 EFPY
61	130	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 3,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	3.09E-05	Yes	Yes at 32, 60, 100 EFPY

Notes:
1. TH – Thermal hydraulics
2. LOCA – Loss-of-coolant accident
3. SBLOCA – Small-break loss-of-coolant accident
4. MBLOCA – Medium-break loss-of-coolant accident
5. LBLOCA – Large-break loss-of-coolant accident
6. HZP – Hot-zero power
7. SRV – Safety and relief valve
8. MSLB – Main steam line break
9. AFW – Auxiliary feedwater
10. HPI – High-pressure injection
11. RCPs – Reactor coolant pumps

* The arbitrary definition of a dominant transient is a transient that contributes 1% or more of the total Through-Wall Cracking Failure (TWCF).

APPENDIX E
BEAVER VALLEY UNIT 1 FAVPOST OUTPUT

E-1: 10 Year ISI Only

```
************************************************
*                                              *
*              WELCOME TO FAVOR                *
*                                              *
*   FRACTURE ANALYSIS OF VESSELS: OAK RIDGE    *
*               VERSION 06.1                   *
*                                              *
*     FAVPOST MODULE: POSTPROCESSOR MODULE     *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES  *
*        WITH RESULTS OF PFM ANALYSIS          *
*                                              *
*     PROBLEMS OR QUESTIONS REGARDING FAVOR    *
*            SHOULD BE DIRECTED TO             *
*                                              *
*               TERRY DICKSON                  *
*         OAK RIDGE NATIONAL LABORATORY        *
*                                              *
*          e-mail: dicksontl@ornl.gov          *
*                                              *
************************************************

***********************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
* Neither the United States, nor the United States        *
* Department of Energy, nor the United States Nuclear     *
* Regulatory Commission, nor any of their employees,      *
* nor any of their contractors, subcontractors, or their  *
* employees, makes any warranty, expressed or implied, or *
* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
***********************************************************


     DATE: 03-May-2007  TIME: 16:02:21


Begin echo of FAVPost input data deck          16:02:21   03-May-2007
End echo of FAVPost input data deck          16:02:21   03-May-2007


       FAVPOST INPUT  FILE NAME                         = postbv.in
       FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
       FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
       FAVPOST OUTPUT FILE NAME                         = 70000.out
```

E-1: 10 Year ISI Only (cont.)

```
**********************************
* NUMBER OF SIMULATIONS =   70000 *
**********************************
```

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
3	3.4447E-07	0.0000E+00	0.0000E+00	1.0487E-08	0.0000E+00	0.0000E+00	0.0304
7	2.4538E-03	5.7840E-03	2.9648E-02	8.9261E-06	1.7129E-04	1.6542E-04	0.0036
9	3.5917E-03	8.8320E-03	4.7025E-02	9.1001E-06	1.5080E-04	1.8094E-04	0.0025
14	2.8062E-09	0.0000E+00	0.0000E+00	3.9534E-11	0.0000E+00	0.0000E+00	0.0141
31	3.1040E-06	0.0000E+00	9.1224E-07	6.5429E-09	0.0000E+00	0.0000E+00	0.0021
34	3.6725E-06	0.0000E+00	8.8530E-06	1.3780E-08	0.0000E+00	0.0000E+00	0.0038
56	3.6233E-03	8.4587E-03	4.0163E-02	1.5372E-05	2.1788E-04	2.9507E-04	0.0042
59	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
60	1.8872E-05	2.1105E-04	1.4508E-04	1.8606E-05	2.1105E-04	1.4063E-04	0.9859
61	2.7682E-05	3.6093E-04	3.3554E-04	5.9860E-06	9.8116E-05	6.4286E-05	0.2162
62	8.8381E-06	0.0000E+00	3.9900E-05	5.3279E-06	0.0000E+00	5.4892E-06	0.6028
64	3.0356E-04	2.4273E-03	4.1264E-03	3.3431E-07	0.0000E+00	8.1349E-08	0.0011
65	1.2194E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
66	2.6208E-05	3.6093E-04	2.9981E-04	1.4196E-07	0.0000E+00	7.0609E-08	0.0054
67	4.2327E-06	0.0000E+00	4.7032E-05	2.0694E-07	0.0000E+00	1.1577E-06	0.0489
68	4.0785E-07	0.0000E+00	0.0000E+00	3.6585E-07	0.0000E+00	0.0000E+00	0.8970
69	6.2806E-04	1.6800E-03	9.3269E-03	4.6732E-04	1.6799E-03	6.5428E-03	0.7441
70	3.3801E-04	2.4275E-03	4.9362E-03	4.7246E-05	6.4369E-04	6.3369E-04	0.1398
71	1.7881E-05	3.7377E-04	1.5558E-04	1.6288E-05	3.7377E-04	1.0611E-04	0.9109
72	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
73	4.1554E-06	6.8843E-05	4.5561E-05	8.3562E-08	0.0000E+00	3.6136E-07	0.0201
74	1.1710E-07	0.0000E+00	0.0000E+00	4.4526E-09	0.0000E+00	0.0000E+00	0.0380
76	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
78	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
81	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
82	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000

E-1: 10 Year ISI Only (cont.)

83	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
92	2.3070E-04	1.1915E-03	3.0316E-03	1.0747E-06	6.0187E-05	8.1838E-06	0.0047
93	2.3070E-04	1.1915E-03	3.0316E-03	1.0747E-06	6.0187E-05	8.1838E-06	0.0047
94	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
97	7.7573E-05	6.7177E-04	1.2231E-03	7.4960E-05	6.7177E-04	1.1540E-03	0.9663
102	1.6387E-06	0.0000E+00	0.0000E+00	2.8950E-08	0.0000E+00	0.0000E+00	0.0177
103	2.5650E-05	3.9193E-04	2.2038E-04	2.0631E-06	0.0000E+00	2.3905E-06	0.0804
104	1.6387E-06	0.0000E+00	0.0000E+00	2.8950E-08	0.0000E+00	0.0000E+00	0.0177
105	1.8207E-07	0.0000E+00	0.0000E+00	9.2460E-09	0.0000E+00	0.0000E+00	0.0508
106	1.5553E-06	0.0000E+00	0.0000E+00	3.0059E-08	0.0000E+00	0.0000E+00	0.0193
107	2.5612E-05	3.5810E-04	1.8762E-04	2.6496E-06	0.0000E+00	3.9040E-06	0.1035
108	5.8945E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
109	5.1071E-08	0.0000E+00	0.0000E+00	2.9873E-09	0.0000E+00	0.0000E+00	0.0585
110	5.8945E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
111	5.1071E-08	0.0000E+00	0.0000E+00	2.9873E-09	0.0000E+00	0.0000E+00	0.0585
112	4.9435E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
113	3.8359E-07	0.0000E+00	0.0000E+00	1.1730E-09	0.0000E+00	0.0000E+00	0.0031
114	3.2501E-05	2.9625E-04	5.5873E-04	1.8948E-07	0.0000E+00	1.0536E-06	0.0058
115	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
116	3.2756E-05	1.4620E-04	7.1227E-04	2.0959E-07	0.0000E+00	1.8495E-06	0.0064
117	1.3498E-04	8.6111E-04	1.6687E-03	6.1235E-07	1.7625E-05	5.7374E-06	0.0045
118	1.0922E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
119	5.2979E-06	0.0000E+00	2.6217E-05	1.2116E-07	0.0000E+00	5.0497E-08	0.0229
120	2.2859E-05	6.2652E-04	1.8086E-04	1.8474E-05	6.2025E-04	9.6291E-05	0.8082
121	1.9161E-04	1.0949E-03	2.4748E-03	1.6094E-06	5.2227E-05	5.0140E-06	0.0084
122	1.9161E-04	1.0949E-03	2.4748E-03	5.0299E-07	0.0000E+00	1.0741E-07	0.0026
123	4.9718E-04	1.5265E-03	7.3133E-03	3.7011E-04	1.5265E-03	4.8876E-03	0.7444
124	2.1942E-04	1.0951E-03	2.8781E-03	3.2633E-05	1.0262E-03	2.0102E-04	0.1487
125	2.2644E-08	0.0000E+00	0.0000E+00	1.4703E-12	0.0000E+00	0.0000E+00	0.0001
126	3.2134E-06	0.0000E+00	1.0696E-05	3.0296E-06	0.0000E+00	9.4015E-06	0.9428
127	3.3065E-05	4.2287E-04	4.9126E-04	7.0810E-08	0.0000E+00	2.4099E-08	0.0021
128	3.3065E-05	4.2287E-04	4.9126E-04	7.0810E-08	0.0000E+00	2.4099E-08	0.0021
129	3.5962E-05	4.2289E-04	5.6359E-04	4.2114E-06	1.0047E-04	2.8955E-05	0.1171
130	6.4214E-05	4.8542E-04	9.0405E-04	3.7700E-05	4.8057E-04	4.3728E-04	0.5871

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

E-1: 10 Year ISI Only (cont.)

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**********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*         FOR THE FREQUENCY OF CRACK INITIATION           *
**********************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	16.2600	16.2600
1.3009E-06	78.7386	94.9986
3.9026E-06	2.5114	97.5100
6.5043E-06	1.0343	98.5443
9.1060E-06	0.5171	99.0614
1.1708E-05	0.2743	99.3357
1.4309E-05	0.1857	99.5214
1.6911E-05	0.1057	99.6271
1.9513E-05	0.0786	99.7057
2.2115E-05	0.0529	99.7586
2.4716E-05	0.0357	99.7943
2.7318E-05	0.0286	99.8229
2.9920E-05	0.0171	99.8400
3.2521E-05	0.0243	99.8643
3.5123E-05	0.0171	99.8814
3.7725E-05	0.0143	99.8957
4.0327E-05	0.0114	99.9071
4.2928E-05	0.0086	99.9157
4.5530E-05	0.0100	99.9257
4.8132E-05	0.0057	99.9314
5.0733E-05	0.0043	99.9357
5.3335E-05	0.0043	99.9400
5.5937E-05	0.0029	99.9429
5.8539E-05	0.0014	99.9443
6.1140E-05	0.0014	99.9457
6.3742E-05	0.0086	99.9543
6.6344E-05	0.0014	99.9557
6.8946E-05	0.0071	99.9629
7.1547E-05	0.0029	99.9657
7.4149E-05	0.0014	99.9671
7.6751E-05	0.0014	99.9686
7.9352E-05	0.0029	99.9714
8.7158E-05	0.0029	99.9743
8.9759E-05	0.0014	99.9757
9.2361E-05	0.0014	99.9771
9.4963E-05	0.0014	99.9786
9.7564E-05	0.0014	99.9800
1.0797E-04	0.0029	99.9829
1.1057E-04	0.0029	99.9857
1.1838E-04	0.0014	99.9871
1.2098E-04	0.0029	99.9900
1.2879E-04	0.0014	99.9914
1.3919E-04	0.0029	99.9943
1.5480E-04	0.0029	99.9971

E-1: 10 Year ISI Only (cont.)

1.8082E-04	0.0014	99.9986
2.3546E-04	0.0014	100.0000

```
==============================================
==      Summary Descriptive Statistics      ==
==============================================

Minimum                                = 0.0000E+00
Maximum                                = 2.3451E-04
Range                                  = 2.3451E-04

Number of Simulations                  =  70000

5th Percentile                         = 0.0000E+00
Median                                 = 1.0978E-08
95.0th Percentile                      = 1.3009E-06
99.0th Percentile                      = 8.7970E-06
99.9th Percentile                      = 3.8701E-05

Mean                                   = 5.9461E-07
Standard Deviation                     = 3.3139E-06
Standard Error                         = 1.2525E-08
Variance (unbiased)                    = 1.0982E-11
Variance (biased)                      = 1.0982E-11
Moment Coeff. of Skewness              = 2.4476E+01
Pearson's 2nd Coeff. of Skewness       = 5.3829E-01
Kurtosis                               = 1.0028E+03
```

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***********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)      *
***********************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	31.5414	31.5414
4.5123E-08	67.4686	99.0100
1.3537E-07	0.5000	99.5100
2.2561E-07	0.1786	99.6886
3.1586E-07	0.0843	99.7729
4.0611E-07	0.0543	99.8271
4.9635E-07	0.0400	99.8671
5.8660E-07	0.0257	99.8929
6.7684E-07	0.0186	99.9114
7.6709E-07	0.0157	99.9271
8.5733E-07	0.0100	99.9371
9.4758E-07	0.0100	99.9471

E-1: 10 Year ISI Only (cont.)

```
1.0378E-06          0.0100        99.9571
1.1281E-06          0.0029        99.9600
1.2183E-06          0.0029        99.9629
1.3086E-06          0.0014        99.9643
1.4891E-06          0.0014        99.9657
1.5793E-06          0.0043        99.9700
1.6695E-06          0.0014        99.9714
1.7598E-06          0.0043        99.9757
1.8500E-06          0.0014        99.9771
1.9403E-06          0.0014        99.9786
2.0305E-06          0.0043        99.9829
2.2110E-06          0.0014        99.9843
2.3013E-06          0.0029        99.9871
2.7525E-06          0.0014        99.9886
2.8427E-06          0.0043        99.9929
3.2940E-06          0.0014        99.9943
3.4745E-06          0.0014        99.9957
4.1964E-06          0.0014        99.9971
5.3696E-06          0.0014        99.9986
8.8892E-06          0.0014       100.0000


============================================
==       Summary Descriptive Statistics     ==
============================================

Minimum                                = 0.0000E+00
Maximum                                = 8.9343E-06
Range                                  = 8.9343E-06

Number of Simulations                  =  70000

5th Percentile                         = 0.0000E+00
Median                                 = 3.2629E-13
95.0th Percentile                      = 4.5123E-08
99.0th Percentile                      = 8.9609E-08
99.9th Percentile                      = 6.2131E-07

Mean                                   = 5.0396E-09
Standard Deviation                     = 6.7097E-08
Standard Error                         = 2.5360E-10
Variance (unbiased)                    = 4.5020E-15
Variance (biased)                      = 4.5019E-15
Moment Coeff. of Skewness              = 5.9654E+01
Pearson's 2nd Coeff. of Skewness       =-2.6549E-01
Kurtosis                               = 5.8120E+03


**********************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION    *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES      *
**********************************************************
```

E-1: 10 Year ISI Only (cont.)

	% of total frequency of crack initiation	% of total frequency of of TWC failure
2	0.00	0.00
3	0.01	0.03
7	2.62	1.25
9	1.18	0.37
14	0.00	0.00
31	0.00	0.00
34	0.00	0.00
56	93.99	43.32
59	0.00	0.00
60	0.08	8.80
61	0.01	0.23
62	0.00	0.02
64	0.00	0.00
65	0.00	0.00
66	0.01	0.00
67	0.00	0.01
68	0.00	0.00
69	0.00	0.23
70	0.00	0.03
71	0.01	1.29
72	0.00	0.00
73	0.00	0.00
74	0.00	0.00
76	0.00	0.00
78	0.00	0.00
81	0.00	0.00
82	0.00	0.00
83	0.00	0.00
92	0.01	0.00
93	0.01	0.01
94	0.00	0.00
97	0.05	6.07
102	0.03	0.08
103	0.05	0.44
104	0.03	0.04
105	0.00	0.00
106	0.00	0.00
107	0.00	0.02
108	0.00	0.00
109	0.00	0.00
110	0.00	0.00
111	0.00	0.00
112	0.00	0.00
113	0.00	0.00
114	0.72	0.46
115	0.00	0.00
116	0.03	0.03
117	0.14	0.08
118	0.00	0.00
119	0.00	0.00
120	0.00	0.42

E-1: 10 Year ISI Only (cont.)

121	0.00	0.01
122	0.00	0.00
123	0.02	1.34
124	0.01	0.14
125	0.00	0.00
126	0.10	10.65
127	0.18	0.04
128	0.15	0.04
129	0.21	2.56
130	0.32	22.00
TOTALS	100.00	100.00

DATE: 03-May-2007 TIME: 16:03:48

E-2: ISI Every 10 Years

```
**************************************************
*                                                *
*               WELCOME TO FAVOR                 *
*                                                *
*   FRACTURE ANALYSIS OF VESSELS: OAK RIDGE      *
*                VERSION 06.1                    *
*                                                *
*     FAVPOST MODULE: POSTPROCESSOR MODULE       *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES    *
*        WITH RESULTS OF PFM ANALYSIS            *
*                                                *
*     PROBLEMS OR QUESTIONS REGARDING FAVOR      *
*            SHOULD BE DIRECTED TO               *
*                                                *
*               TERRY DICKSON                    *
*        OAK RIDGE NATIONAL LABORATORY           *
*                                                *
*           e-mail: dicksontl@ornl.gov           *
*                                                *
**************************************************

***********************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
* Neither the United States, nor the United States        *
* Department of Energy, nor the United States Nuclear     *
* Regulatory Commission, nor any of their employees,      *
* nor any of their contractors, subcontractors, or their  *
* employees, makes any warranty, expressed or implied, or *
* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
***********************************************************


     DATE: 03-May-2007  TIME: 15:03:08


Begin echo of FAVPost input data deck          15:03:08   03-May-2007
End echo of FAVPost input data deck          15:03:08   03-May-2007


       FAVPOST INPUT  FILE NAME                      = postbv.in
       FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
       FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
       FAVPOST OUTPUT FILE NAME                      = 70000.out
```

E-2: ISI Every 10 Years (cont.)

```
**********************************
* NUMBER OF SIMULATIONS =  70000 *
**********************************
```

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
3	2.9800E-07	0.0000E+00	0.0000E+00	1.0009E-11	0.0000E+00	0.0000E+00	0.0000
7	2.3214E-03	5.3036E-03	3.1228E-02	9.5073E-06	1.6615E-04	1.5375E-04	0.0041
9	3.4412E-03	8.2533E-03	4.6458E-02	9.7557E-06	1.2533E-04	1.6701E-04	0.0028
14	9.1393E-09	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
31	2.9830E-06	0.0000E+00	5.6087E-07	6.0712E-09	0.0000E+00	0.0000E+00	0.0020
34	3.5119E-06	0.0000E+00	1.7572E-06	8.9484E-10	0.0000E+00	0.0000E+00	0.0003
56	3.4601E-03	7.9216E-03	3.9985E-02	1.6170E-05	2.0726E-04	2.7235E-04	0.0047
59	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
60	2.0168E-05	3.2408E-04	1.3589E-04	1.9908E-05	3.0787E-04	1.3395E-04	0.9871
61	2.7685E-05	8.4773E-04	3.2463E-04	6.6560E-06	1.7053E-04	6.1162E-05	0.2404
62	7.5949E-06	0.0000E+00	1.9721E-05	4.0857E-06	0.0000E+00	8.8105E-07	0.5380
64	2.2973E-04	2.4655E-03	3.8380E-03	1.4364E-07	0.0000E+00	1.7605E-09	0.0006
65	2.2001E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
66	2.6002E-05	8.4773E-04	2.8721E-04	2.3979E-07	0.0000E+00	1.4232E-08	0.0092
67	2.0237E-06	0.0000E+00	1.0314E-08	4.1104E-10	0.0000E+00	0.0000E+00	0.0002
68	3.1241E-07	0.0000E+00	0.0000E+00	2.5031E-07	0.0000E+00	0.0000E+00	0.8012
69	6.2793E-04	2.3821E-03	8.8419E-03	4.7965E-04	2.3081E-03	6.1921E-03	0.7639
70	2.6032E-04	2.4655E-03	4.5217E-03	3.5399E-05	4.5620E-04	3.0528E-04	0.1360
71	1.1361E-05	0.0000E+00	2.1871E-05	1.1359E-05	0.0000E+00	2.1795E-05	0.9998
72	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
73	2.6310E-06	0.0000E+00	6.8862E-06	1.1986E-08	0.0000E+00	0.0000E+00	0.0046
74	1.9735E-07	0.0000E+00	0.0000E+00	4.1257E-09	0.0000E+00	0.0000E+00	0.0209
76	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
78	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
81	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
82	2.3896E-12	0.0000E+00	0.0000E+00	1.4423E-12	0.0000E+00	0.0000E+00	0.6036

E-2: ISI Every 10 Years (cont.)

83	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
92	2.2151E-04	2.3171E-03	3.9602E-03	1.4445E-06	6.6444E-05	6.9896E-06	0.0065
93	2.2151E-04	2.3171E-03	3.9602E-03	1.4435E-06	6.6444E-05	6.9896E-06	0.0065
94	5.0580E-13	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
97	5.6862E-05	1.1055E-03	5.9676E-04	5.6592E-05	1.1055E-03	5.9102E-04	0.9953
102	2.3159E-06	0.0000E+00	0.0000E+00	1.7235E-08	0.0000E+00	0.0000E+00	0.0074
103	2.7193E-05	1.3062E-03	2.3097E-04	2.2622E-06	0.0000E+00	2.7032E-06	0.0832
104	2.3159E-06	0.0000E+00	0.0000E+00	1.7235E-08	0.0000E+00	0.0000E+00	0.0074
105	3.0512E-07	0.0000E+00	0.0000E+00	6.1501E-09	0.0000E+00	0.0000E+00	0.0202
106	2.2119E-06	0.0000E+00	0.0000E+00	2.0044E-08	0.0000E+00	0.0000E+00	0.0091
107	2.6872E-05	1.4079E-03	1.8487E-04	2.9860E-06	0.0000E+00	4.3421E-06	0.1111
108	6.2467E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
109	1.1837E-07	0.0000E+00	0.0000E+00	3.2204E-11	0.0000E+00	0.0000E+00	0.0003
110	6.2467E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
111	1.1837E-07	0.0000E+00	0.0000E+00	3.2204E-11	0.0000E+00	0.0000E+00	0.0003
112	6.2651E-10	0.0000E+00	0.0000E+00	2.9991E-18	0.0000E+00	0.0000E+00	0.0000
113	6.1401E-07	0.0000E+00	0.0000E+00	2.1350E-09	0.0000E+00	0.0000E+00	0.0035
114	2.4706E-05	7.7763E-04	3.1629E-04	5.7733E-08	0.0000E+00	0.0000E+00	0.0023
115	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
116	5.5954E-06	0.0000E+00	1.1859E-05	2.5625E-09	0.0000E+00	0.0000E+00	0.0005
117	1.2355E-04	1.5759E-03	2.1069E-03	8.5564E-07	3.3720E-05	3.4261E-06	0.0069
118	2.7278E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
119	4.5739E-06	0.0000E+00	3.4377E-06	1.5432E-08	0.0000E+00	0.0000E+00	0.0034
120	2.0556E-05	4.1262E-04	1.0487E-04	1.6115E-05	3.6562E-04	5.0320E-05	0.7839
121	1.8168E-04	2.1666E-03	3.1673E-03	1.5012E-06	7.2990E-05	2.5174E-06	0.0083
122	1.8168E-04	2.1666E-03	3.1673E-03	4.0757E-07	0.0000E+00	7.5126E-08	0.0022
123	5.0111E-04	2.1906E-03	6.5400E-03	3.8406E-04	2.0773E-03	4.9104E-03	0.7664
124	2.0576E-04	2.1666E-03	3.6134E-03	2.8451E-05	6.1944E-04	1.8793E-04	0.1383
125	5.2120E-08	0.0000E+00	0.0000E+00	2.5396E-13	0.0000E+00	0.0000E+00	0.0000
126	3.4102E-06	0.0000E+00	7.9349E-07	3.2667E-06	0.0000E+00	6.8717E-07	0.9579
127	2.6738E-05	2.5191E-04	3.1581E-04	2.6252E-12	0.0000E+00	0.0000E+00	0.0000
128	2.6738E-05	2.5191E-04	3.1581E-04	2.6252E-12	0.0000E+00	0.0000E+00	0.0000
129	2.9639E-05	2.5191E-04	3.7032E-04	2.9210E-06	0.0000E+00	7.9206E-07	0.0986
130	5.5626E-05	9.5051E-04	7.2434E-04	3.2049E-05	9.4101E-04	2.4706E-04	0.5762

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

E-2: ISI Every 10 Years (cont.)

```
*********************************************************
*     PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*        FOR THE FREQUENCY OF CRACK INITIATION           *
*********************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	16.2857	16.2857
1.2622E-06	78.7129	94.9986
3.7866E-06	2.5743	97.5729
6.3111E-06	0.9743	98.5471
8.8355E-06	0.4871	99.0343
1.1360E-05	0.2614	99.2957
1.3884E-05	0.1829	99.4786
1.6409E-05	0.1029	99.5814
1.8933E-05	0.0900	99.6714
2.1458E-05	0.0643	99.7357
2.3982E-05	0.0400	99.7757
2.6506E-05	0.0314	99.8071
2.9031E-05	0.0329	99.8400
3.1555E-05	0.0171	99.8571
3.4080E-05	0.0200	99.8771
3.6604E-05	0.0143	99.8914
3.9129E-05	0.0086	99.9000
4.1653E-05	0.0114	99.9114
4.4177E-05	0.0086	99.9200
4.6702E-05	0.0129	99.9329
4.9226E-05	0.0014	99.9343
5.1751E-05	0.0071	99.9414
5.4275E-05	0.0043	99.9457
5.6800E-05	0.0043	99.9500
5.9324E-05	0.0043	99.9543
6.1848E-05	0.0043	99.9586
6.6897E-05	0.0029	99.9614
6.9422E-05	0.0014	99.9629
7.1946E-05	0.0029	99.9657
7.6995E-05	0.0014	99.9671
8.4568E-05	0.0029	99.9700
8.7093E-05	0.0014	99.9714
8.9617E-05	0.0029	99.9743
9.2141E-05	0.0029	99.9771
9.4666E-05	0.0014	99.9786
9.7190E-05	0.0029	99.9814
1.0224E-04	0.0014	99.9829
1.0729E-04	0.0014	99.9843
1.0981E-04	0.0014	99.9857
1.1739E-04	0.0014	99.9871
1.2243E-04	0.0014	99.9886
1.3506E-04	0.0014	99.9900
1.5020E-04	0.0014	99.9914
1.5525E-04	0.0014	99.9929

E-2: ISI Every 10 Years (cont.)

1.5778E-04	0.0014	99.9943
1.7797E-04	0.0014	99.9957
2.2089E-04	0.0014	99.9971
2.2846E-04	0.0014	99.9986
2.3098E-04	0.0014	100.0000

```
=============================================
==      Summary Descriptive Statistics     ==
=============================================

Minimum                                = 0.0000E+00
Maximum                                = 2.3200E-04
Range                                  = 2.3200E-04

Number of Simulations                  =  70000

5th Percentile                         = 0.0000E+00
Median                                 = 9.7197E-09
95.0th Percentile                      = 1.2622E-06
99.0th Percentile                      = 8.6578E-06
99.9th Percentile                      = 3.9129E-05

Mean                                   = 5.8049E-07
Standard Deviation                     = 3.4364E-06
Standard Error                         = 1.2989E-08
Variance (unbiased)                    = 1.1809E-11
Variance (biased)                      = 1.1809E-11
Moment Coeff. of Skewness              = 2.8192E+01
Pearson's 2nd Coeff. of Skewness       = 5.0677E-01
Kurtosis                               = 1.3237E+03
```

```
***********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)      *
***********************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	32.3229	32.3229
3.7808E-08	66.6543	98.9771
1.1342E-07	0.4457	99.4229
1.8904E-07	0.2186	99.6414
2.6466E-07	0.0914	99.7329
3.4027E-07	0.0514	99.7843
4.1589E-07	0.0343	99.8186
4.9151E-07	0.0271	99.8457
5.6712E-07	0.0171	99.8629

E-2: ISI Every 10 Years (cont.)

6.4274E-07	0.0186	99.8814
7.1836E-07	0.0171	99.8986
7.9397E-07	0.0143	99.9129
8.6959E-07	0.0043	99.9171
9.4521E-07	0.0086	99.9257
1.0208E-06	0.0043	99.9300
1.0964E-06	0.0043	99.9343
1.1721E-06	0.0071	99.9414
1.2477E-06	0.0043	99.9457
1.3989E-06	0.0029	99.9486
1.4745E-06	0.0057	99.9543
1.5501E-06	0.0014	99.9557
1.6258E-06	0.0057	99.9614
1.7014E-06	0.0029	99.9643
1.7770E-06	0.0043	99.9686
2.0038E-06	0.0029	99.9714
2.1551E-06	0.0014	99.9729
2.3063E-06	0.0014	99.9743
2.6844E-06	0.0014	99.9757
2.9869E-06	0.0014	99.9771
3.0625E-06	0.0014	99.9786
3.1381E-06	0.0014	99.9800
3.2893E-06	0.0029	99.9829
3.5162E-06	0.0014	99.9843
3.6674E-06	0.0014	99.9857
3.8186E-06	0.0014	99.9871
3.9699E-06	0.0029	99.9900
4.0455E-06	0.0014	99.9914
4.2723E-06	0.0014	99.9929
5.2554E-06	0.0014	99.9943
5.5578E-06	0.0029	99.9971
6.0115E-06	0.0014	99.9986
7.4482E-06	0.0014	100.0000

```
=============================================
==       Summary Descriptive Statistics    ==
=============================================

Minimum                              = 0.0000E+00
Maximum                              = 7.4860E-06
Range                                = 7.4860E-06

Number of Simulations                =  70000

5th Percentile                       = 0.0000E+00
Median                               = 2.5574E-13
95.0th Percentile                    = 3.7808E-08
99.0th Percentile                    = 4.1686E-08
99.9th Percentile                    = 7.2592E-07

Mean                                 = 5.2336E-09
Standard Deviation                   = 8.2534E-08
```

E-2: ISI Every 10 Years (cont.)

```
Standard Error                                  = 3.1195E-10
    Variance (unbiased)                             = 6.8118E-15
    Variance (biased)                               = 6.8117E-15
    Moment Coeff. of Skewness                       = 4.7286E+01
    Pearson's 2nd Coeff. of Skewness               =-1.6306E-01
    Kurtosis                                        = 2.9147E+03
```

```
**********************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION    *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -   *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES     *
**********************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
2	0.00	0.00
3	0.00	0.00
7	2.58	1.01
9	1.13	0.39
14	0.00	0.00
31	0.00	0.00
34	0.00	0.00
56	94.32	43.78
59	0.00	0.00
60	0.10	11.13
61	0.01	0.28
62	0.00	0.01
64	0.00	0.00
65	0.00	0.00
66	0.01	0.01
67	0.00	0.00
68	0.00	0.00
69	0.00	0.22
70	0.00	0.02
71	0.01	0.78
72	0.00	0.00
73	0.00	0.00
74	0.00	0.00
76	0.00	0.00
78	0.00	0.00
81	0.00	0.00
82	0.00	0.00
83	0.00	0.00
92	0.01	0.01
93	0.01	0.01
94	0.00	0.00
97	0.05	5.07
102	0.07	0.12
103	0.06	0.51
104	0.05	0.04
105	0.00	0.00

E-2: ISI Every 10 Years (cont.)

106	0.00	0.00
107	0.00	0.03
108	0.00	0.00
109	0.00	0.00
110	0.00	0.00
111	0.00	0.00
112	0.00	0.00
113	0.01	0.00
114	0.47	0.13
115	0.00	0.00
116	0.01	0.00
117	0.13	0.09
118	0.00	0.00
119	0.00	0.00
120	0.00	0.34
121	0.00	0.00
122	0.00	0.00
123	0.02	1.33
124	0.01	0.10
125	0.00	0.00
126	0.09	10.04
127	0.13	0.00
128	0.15	0.00
129	0.20	1.27
130	0.36	23.28
TOTALS	100.00	100.00

```
*********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION   *
*     AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -     *
*                           BY                            *
*                RPV BELTLINE MAJOR REGION                *
*                   BY PARENT SUBREGION                   *
*                                                         *
*      WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT       *
*         TO FREQUENCY OF CRACK INITIATION AND            *
*        THROUGH-WALL CRACKING FREQUENCY (FAILURE)        *
*********************************************************
```

MAJOR REGION	RTndt (MAX)	% of total flaws	% of total frequency of crack initiation	% of total through-wall crack frequency cleavage	ductile	total
1	174.15	2.29	0.11	2.82	0.02	2.84
2	174.15	2.29	0.10	2.15	0.03	2.18
3	164.48	3.69	4.16	25.64	4.45	30.09
4	164.48	3.69	3.65	20.74	4.15	24.90
5	89.30	19.28	89.66	2.88	0.05	2.93

E-2: ISI Every 10 Years (cont.)

6	220.82	13.16	0.08	0.82	0.07	0.89
7	192.65	13.16	0.00	0.00	0.00	0.00
8	253.03	21.22	2.06	32.01	2.57	34.58
9	223.03	21.22	0.18	1.36	0.24	1.60
	TOTALS	100.00	100.00	88.43	11.57	100.00

```
***********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION   *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*                           BY                            *
*                 RPV BELTLINE MAJOR REGION               *
*                     BY CHILD SUBREGION                  *
*                                                         *
*       WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT      *
*           TO FREQUENCY OF CRACK INITIATION AND          *
*         THROUGH-WALL CRACKING FREQUENCY (FAILURE)       *
***********************************************************
```

MAJOR REGION	RTndt (MAX)	% of total flaws	% of total frequency of crack initiation	% of total through-wall crack frequency cleavage	ductile	total
1	174.15	2.29	0.00	0.00	0.00	0.00
2	174.15	2.29	0.00	0.00	0.00	0.00
3	164.48	3.69	0.00	0.00	0.00	0.00
4	164.48	3.69	0.00	0.00	0.00	0.00
5	89.30	19.28	0.00	0.00	0.00	0.00
6	220.82	13.16	2.89	5.80	0.12	5.92
7	192.65	13.16	0.00	0.00	0.00	0.00
8	253.03	21.22	86.26	81.11	11.21	92.32
9	223.03	21.22	10.85	1.51	0.25	1.76
	TOTALS	100.00	100.00	88.43	11.57	100.00

```
***********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION   *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*          MATERIAL, FLAW CATEGORY, AND FLAW DEPTH        *
*                                                         *
*       WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT      *
*           TO FREQUENCY OF CRACK INITIATION AND          *
*         THROUGH-WALL CRACKING FREQUENCY (FAILURE)       *
***********************************************************
```

E-2: ISI Every 10 Years (cont.)

```
*****************
* WELD MATERIAL *
*****************
```

FLAW DEPTH (in)	% of total frequency of crack initiation			% of total through-wall crack frequency		
	CAT I flaws	CAT 2 flaws	CAT 3 flaws	CAT 1 flaws	CAT 2 flaws	CAT 3 flaws
0.080	0.00	1.68	0.00	0.00	0.06	0.00
0.161	0.00	38.20	0.00	0.00	2.93	0.00
0.241	0.00	18.39	0.00	0.00	2.38	0.00
0.321	0.00	9.55	0.00	0.00	1.39	0.01
0.402	0.00	7.76	0.00	0.00	2.20	0.03
0.482	0.00	6.12	0.00	0.00	2.57	0.08
0.563	0.00	4.14	0.00	0.00	3.19	0.08
0.643	0.00	2.73	0.00	0.00	2.74	0.13
0.723	0.00	2.03	0.00	0.00	2.78	0.12
0.804	0.00	1.35	0.00	0.00	3.51	0.30
0.884	0.00	1.02	0.00	0.00	3.28	0.26
0.964	0.00	0.67	0.00	0.00	2.93	0.19
1.045	0.00	1.08	0.00	0.00	2.91	0.07
1.125	0.00	0.40	0.00	0.00	2.84	0.07
1.205	0.00	0.30	0.00	0.00	2.57	0.06
1.286	0.00	0.21	0.00	0.00	4.21	0.44
1.366	0.00	0.34	0.01	0.00	1.35	0.80
1.446	0.00	0.16	0.00	0.00	4.13	0.16
1.527	0.00	0.42	0.00	0.00	2.18	0.05
1.607	0.00	0.21	0.00	0.00	4.06	0.29
1.688	0.00	0.29	0.00	0.00	1.86	0.14
1.768	0.00	0.02	0.00	0.00	0.61	0.39
1.848	0.00	0.01	0.01	0.00	1.15	0.66
1.929	0.00	0.53	0.00	0.00	0.69	0.10
TOTALS	0.00	97.63	0.05	0.00	58.51	4.42

```
******************
* PLATE MATERIAL *
******************
```

FLAW DEPTH (in)	% of total frequency of crack initiation			% of total through-wall crack frequency		
	CAT I flaws	CAT 2 flaws	CAT 3 flaws	CAT 1 flaws	CAT 2 flaws	CAT 3 flaws
0.080	0.00	0.01	0.00	0.00	0.13	0.00
0.161	0.00	0.48	0.00	0.00	7.33	0.00
0.241	0.00	0.69	0.00	0.00	10.42	0.00
0.321	0.00	0.61	0.00	0.00	9.99	0.00

E-2: ISI Every 10 Years (cont.)

0.402	0.00	0.54	0.00	0.00	9.20	0.00
0.482	0.00	0.00	0.00	0.00	0.00	0.00
0.563	0.00	0.00	0.00	0.00	0.00	0.00
0.643	0.00	0.00	0.00	0.00	0.00	0.00
0.723	0.00	0.00	0.00	0.00	0.00	0.00
0.804	0.00	0.00	0.00	0.00	0.00	0.00
0.884	0.00	0.00	0.00	0.00	0.00	0.00
0.964	0.00	0.00	0.00	0.00	0.00	0.00
1.045	0.00	0.00	0.00	0.00	0.00	0.00
TOTALS	0.00	2.33	0.00	0.00	37.06	0.00

DATE: 03-May-2007 TIME: 15:04:07

APPENDIX F
INPUTS FOR THE PALISADES PILOT PLANT EVALUATION

A summary of the NDE inspection history based on Regulatory Guide 1.150 and pertinent input data for Palisades is as follows:

1. Number of ISIs performed (relative to initial pre-service and 10-year interval inspections) for full penetration Category B-A and B-D reactor vessel welds assuming all of the candidate welds were inspected: 2 (covering all welds of the specified categories).

2. The inspections performed covered: 100% for 13 Category B-A welds, >90% but <100% for 6 Category B-A welds, <90% for 8 Category B-A welds, and 100% of all Category B-D welds.

3. Number of indications found during most recent inservice inspection: 11
   This number includes consideration of the following additional information:

   a. Indications found that were reportable: 0
   b. Indications found that were within acceptable limits: 11
   c. Indications/anomalies currently being monitored: 0

4. Full penetration relief requests for the RV submitted and accepted by the NRC: 2 relief requests for limited converage for 12 welds

5. Fluence distribution at inside surface of RV beltline until end of life (EOL): see Figure F-1 taken from the NRC PTS Risk Study [9], Figure 4.3.

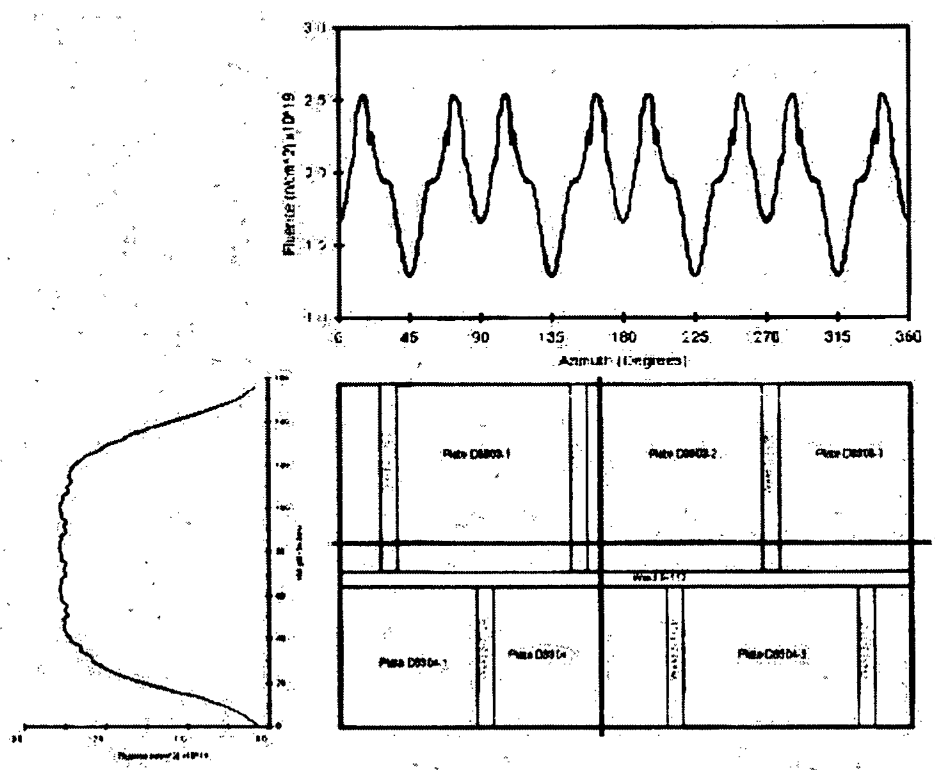

Figure F-1 Rollout Diagram of Beltline Materials and Representative Fluence Maps for Palisades

6. Reactor vessel cladding details:

a. Thickness: 0.25 inches

b. Material properties are identified in Table F-1:

Table F-1 Cladding Material Properties

Temperature (°F)	Thermal Conductivity (Btu/hr-ft-°F) "K"	Specific Heat (Btu/LBM-°F) "C"	Young's Modulus of Elasticity (KSI) "E"	Thermal Expansion Coefficient (°F^{-1}) "α"	Density (LBM/ft^3) "ρ"	Poisson's Ratio "ν"
0	-	-	-	-	489	.3
68	-	-	22045.7	-	489	.3
70	8.1	0.1158	-	-	489	.3
100	8.4	0.1185	-	8.55E-06	489	.3
150	8.6	0.1196	-	8.67E-06	489	.3
200	8.8	0.1208	-	8.79E-06	489	.3
250	9.1	0.1232	-	8.9E-06	489	.3
300	9.4	0.1256	-	9.0E-06	489	.3
302	-	-	20160.2	-	489	.3
350	9.6	0.1258	-	9.1E-06	489	.3
400	9.9	0.1281	-	9.19E-06	489	.3
450	10.1	0.1291	-	9.28E-06	489	.3
482	-	-	18419.8	-	489	.3
500	10.4	0.1305	-	9.37E-06	489	.3
550	10.6	0.1306	-	9.45E-06	489	.3
600	10.9	0.1327	-	9.53E-06	489	.3
650	11.1	0.1335	-	9.61E-06	489	.3
700	11.4	0.1348	-	9.69E-06	489	.3
750	11.6	0.1356	-	9.76E-06	489	.3
800	11.9	0.1367	-	9.82E-06	489	.3

c. Material including copper and nickel content: Material properties assigned to clad flaws are that of the underlying material be it base metal or weld. These properties are identified in Table F-3. This is consistent with the NRC PTS Risk Study [8, 9].

d. Material property uncertainties:

1) Bead width: 1 inch – bead widths vary for all plants. Based on the NRC PTS Risk Study [8, 9], a nominal dimension of 1 inch is selected for all analyses because this parameter is not expected to influence significantly the predicted vessel failure probabilities.

2) Truncation limit: Cladding thickness rounded to the next 1/100th of the total reactor vessel thickness to be consistent with the NRC PTS Risk Study [8, 9].

3) Surface flaw depth: 0.263 inch

4) All flaws are surface-breaking. Only flaws in cladding that would influence brittle fracture of the reactor vessel are brittle. This is consistent with the NRC PTS Risk Study [8, 9].

e. Additional cladding properties are identified in Table F-4. This is consistent with the NRC PTS Risk Study [8, 9].

7. Base metal:

a. Wall thickness: 8.5 inches

b. Material properties are identified in Tables F-2 and F-3:

Table F-2 Base Metal Material Properties

Temperature (°F)	Thermal Conductivity (Btu/hr-ft-°F) "K"	Specific Heat (Btu/LBM-°F) "C"	Young's Modulus of Elasticity (KSI) "E"	Thermal Expansion Coefficient (°F^{-1}) "α"	Density (LBM/ft^3) "ρ"	Poisson's Ratio "ν"
0	-	-	-	-	489	.3
70	24.8	0.1052	29200	-	489	.3
100	25	0.1072	-	7.06E-06	489	.3
150	25.1	0.1101	-	7.16E-06	489	.3
200	25.2	0.1135	28500	7.25E-06	489	.3
250	25.2	0.1166	-	7.34E-06	489	.3
300	25.1	0.1194	28000	7.43E-06	489	.3
350	25	0.1223	-	7.5E-06	489	.3
400	25.1	0.1267	27400	7.58E-06	489	.3
450	24.6	0.1277	-	7.63E-06	489	.3
500	24.3	0.1304	27000	7.7E-06	489	.3
550	24	0.1326	-	7.77E-06	489	.3
600	23.7	0.135	26400	7.83E-06	489	.3
650	23.4	0.1375	-	7.9E-06	489	.3
700	23	0.1404	25300	7.94E-06	489	.3
750	22.6	0.1435	-	8.0E-06	489	.3
800	22.2	0.1474	23900	8.05E-06	489	.3

Table F-3 Palisades-Specific Material Values Drawn from the RVID (see Ref. 44 Table 4.1)								
Major Material Region Description				Cu [wt%]	Ni [wt%]	P [wt%]	Mn [wt%]	Un-Irradiated RT_{NDT} [°F]
#	Type	Heat	Location					
1	Axial Weld	3-112A	lower	0.213	1.010	0.019	1.315	- 56
2	Axial Weld	3-112B	lower	0.213	1.010	0.019	1.315	- 56
3	Axial Weld	3-112C	lower	0.213	1.010	0.019	1.315	- 56
4	Axial Weld	2-112A	upper	0.213	1.010	0.019	1.315	- 56
5	Axial Weld	2-112B	upper	0.213	1.010	0.019	1.315	- 56
6	Axial Weld	2-112C	upper	0.213	1.010	0.019	1.315	- 56
7	Circ Weld	9-112	intermediate	0.203	1.018	0.013	1.147	- 56
8	Plate	D3804-1	lower	0.190	0.480	0.016	1.235	0
9	Plate	D3804-2	lower	0.190	0.500	0.015	1.235	-30
10	Plate	D3804-3	lower	0.120	0.550	0.010	1.270	-25
11	Plate	D3803-1	upper	0.240	0.510	0.009	1.293	-5
12	Plate	D3803-2	upper	0.240	0.520	0.010	1.350	-30
13	Plate	D3803-3	upper	0.240	0.500	0.011	1.293	-5

8. Weld metal details: Details of information used in addressing weld-specific information are taken directly from the NRC PTS Risk Study [44], Table 4.2. Summaries are reproduced as Table F-4. Values for SAW Weld Volume fraction and Repair Weld Volume fraction in Table F-4 were changed to 96.7% and 2.3% respectively per NUREG-1874 [9].

Table F-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution

Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Inner Radius (to cladding)	[in]	85.5	78.5	86	86	Vessel specific info
Base Metal Thickness	[in]	8.438	7.875	8.5	8.675	Vessel specific info
Total Wall Thickness	[in]	8.626	8.031	8.75	8.988	Vessel specific info

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SAW Weld	Volume fraction	[%]	97%				100% - SMAW% - REPAIR%
	Thru-Wall Bead Thickness	[in]	0.1875	0.1875	0.1875	0.1875	All plants report plant specific dimensions of 3/16-in.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates; when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table F-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SMAW Weld	Volume fraction	[%]	1%				Upper bound to all plant specific info provided by Steve Byrne (Westinghouse – Windsor).
	Thru-Wall Bead Thickness	[in]	0.21	0.20	0.22	0.25	Oconee is generic value based on average of all plants specific values (including Shoreham & PVRUF data). Other values are plant specific as reported by Steve Byrne.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table F-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Repair Weld	Volume fraction	[%]	2%				Judgment. A rounded integral percentage that exceeds the repaired volume observed for Shoreham and for PVRUF, which was 1.5%.
	Thru-Wall Bead Thickness	[in]	0.14				Generic value: As observed in PVRUF and Shoreham by PNNL.
	Truncation Limit	[in]	2				Judgment. Approx. 2X the largest repair flaw found in PVRUF & Shoreham. Also based on maximum expected width of repair cavity.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				The repair flaws had complex shapes and orientations that were not aligned with either the axial or circumferential welds; for consistency with the available treatments of flaws by the FAVOR code, a common treatment of orientations was adopted for flaws in SAW/SMAW and repair welds.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table F-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Cladding	Actual Thickness	[in]	0.188	0.156	0.25	0.313	Vessel specific info
	# of Layers	[#]	1	2	2	2	Vessel specific info
	Bead Width	[in]	1				Bead widths of 1 to 5-in. characteristic of machine deposited cladding. Bead widths down to ½-in. can occur over welds. Nominal dimension of 1-in. selected for all analyses because this parameter is not expected to influence significantly the predicted vessel failure probabilities. May need to refine this estimate later, particularly for Oconee who reported a 5-in bead width.
	Truncation Limit	[in]	Actual clad thickness rounded to the nearest 1/100th of the total vessel wall thickness				Judgment & computational convenience
	Surface flaw depth in FAVOR	[in]	0.259	0.161	0.263	0.360	
	Buried or Surface	--	All flaws are surface breaking				Judgment. Only flaws in cladding that would influence brittle fracture of the vessel are brittle. Material properties assigned to clad flaws are that of the underlying material, be it base or weld.
	Orientation	--	All circumferential.				Observation: All flaws observed in PVRUF & Shoreham were lack of inter-run fusion defects, and cladding is always deposited circumferentially
	Density basis	--	No surface flaws observed. Density is 1/1000 that of the observed buried flaws in cladding of vessels examined by PNNL. If there is more than one clad layer then there are no clad flaws.				Judgment
	Aspect ratio basis	--	Observations on buried flaws				Judgment
	Depth basis	--	Depth of all surface flaws is the actual clad thickness rounded up to the nearest 1/100th of the total vessel wall thickness.				Judgment.

Table F-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Plate	Truncation Limit	[in]	0.433				Judgment. Twice the depth of the largest flaw observed in all PNNL plate inspections.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Half of the simulated flaws are circumferential, half are axial.				Observation & Physics: No observed orientation preference, and no reason to suspect one (other than laminations which are benign.
	Density basis	--	1/10 of small weld flaw density. 1/40 of large weld flaw density of the PVRUF data				Judgment. Supported by limited data.
	Aspect ratio basis	--	Same as for PVRUF welds				Judgment
	Depth basis	--	Same as for PVRUF welds				Judgment. Supported by limited data.

9. $TWCF_{95\text{-}TOTAL}$ value calculated at 60 EFPY using correlations from NUREG-1874 (Reference 9): 3.16E-7 Events per year

APPENDIX G
PALISADES PROBSBFD OUTPUT

G-1: 10 Year ISI Only

```
                STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE         MONTE-CARLO SIMULATION PROGRAM PROBSBFD    VERSION 1.0

============================================================================
     INPUT VARIABLES FOR CASE  2: PAL 10 YEAR ISI ONLY

     NCYCLE =     80           NFAILS =   1001          NTRIAL =  1000
     NOVARS =     19           NUMSET =      2          NUMISI =     5
     NUMSSC =      4           NUMTRC =      4          NUMFMD =     4

   VARIABLE         DISTRIBUTION       MEDIAN       DEVIATION     SHIFT    USAGE
  NO.   NAME        TYPE      LOG       VALUE       OR FACTOR     MV/SD   NO. SUB

   1  FIFDepth     - CONSTANT -    3.0000D-02                              1  SET
   2  IFlawDen     - CONSTANT -    3.6589D-03                              2  SET
   3  ICy-ISI      - CONSTANT -    1.0000D+01                              1  ISI
   4  DCy-ISI      - CONSTANT -    8.0000D+01                              2  ISI
   5  MV-Depth     - CONSTANT -    1.5000D-02                              3  ISI
   6  SD-Depth     - CONSTANT -    1.8500D-01                              4  ISI
   7  CEff-ISI     - CONSTANT -    1.0000D+00                              5  ISI
   8  Aspect1      - CONSTANT -    2.0000D+00                              1  SSC
   9  Aspect2      - CONSTANT -    6.0000D+00                              2  SSC
  10  Aspect3      - CONSTANT -    1.0000D+01                              3  SSC
  11  Aspect4      - CONSTANT -    9.9000D+01                              4  SSC
  12  NoTr/Cy      - CONSTANT -    1.3000D+01                              1  TRC
  13  FCGThld      - CONSTANT -    1.5000D+00                              2  TRC
  14  FCGR-UC      NORMAL    NO    0.0000D+00    1.0000D+00      .00       3  TRC
  15  DKINFile     - CONSTANT -    1.0000D+00                              4  TRC
  16  Percent1     - CONSTANT -    7.8870D+01                              1  FMD
  17  Percent2     - CONSTANT -    1.0720D+01                              2  FMD
  18  Percent3     - CONSTANT -    4.3807D+00                              3  FMD
  19  Percent4     - CONSTANT -    6.0298D+00                              4  FMD

 INFORMATION GENERATED FROM FAVLOADS.DAT FILE
 AND SAVED IN DKINSAVE.DAT FILE:

 WALL THICKNESS =  8.7500 INCH

 FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

   TYPE 1 WITH AN ASPECT RATIO OF   2.

   8.75000D-02    2.69285D+00    1.08492D+01
   1.61000D-01    3.60064D+00    1.46562D+01
   4.37500D-01    1.26609D+01    2.00367D+01
   6.56250D-01    1.49279D+01    2.39231D+01
   8.75000D-01    1.53491D+01    2.67406D+01
   1.75000D+00    1.37876D+01    3.14212D+01
   2.62500D+00    8.13906D+00    3.01520D+01
   4.37500D+00   -2.32655D+00    2.91175D+01

   TYPE 2 WITH AN ASPECT RATIO OF   6.

   8.75000D-02    4.04516D+00    1.64003D+01
```

G-1: 10 Year ISI Only (cont.)

1.61000D-01	5.52109D+00	2.25832D+01
4.37500D-01	1.80126D+01	3.03772D+01
6.56250D-01	2.31235D+01	3.61026D+01
8.75000D-01	2.65795D+01	4.11957D+01
1.75000D+00	2.62424D+01	5.18633D+01
2.62500D+00	2.10650D+01	5.45640D+01
4.37500D+00	9.61580D+00	5.85179D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.75000D-02	4.43154D+00	1.79837D+01
1.61000D-01	5.90218D+00	2.41564D+01
4.37500D-01	1.90406D+01	3.24750D+01
6.56250D-01	2.45354D+01	3.85918D+01
8.75000D-01	2.87821D+01	4.40958D+01
1.75000D+00	2.91774D+01	5.64674D+01
2.62500D+00	2.54877D+01	6.25646D+01
4.37500D+00	1.38132D+01	7.03917D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.75000D-02	7.10780D+00	1.85180D+01
1.75000D-01	1.00487D+01	2.59141D+01
2.62500D-01	1.38195D+01	2.86661D+01
4.37500D-01	2.16458D+01	3.45538D+01
6.56250D-01	2.85157D+01	4.23747D+01
8.75000D-01	3.03911D+01	4.83133D+01
1.75000D+00	3.36289D+01	6.57043D+01
2.62500D+00	3.16032D+01	7.68320D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

12	2.5402D-04	4.5317D-06	1.3489D-06	2.1739D-06
13	5.8986D-06	1.9521D-05	7.3792D-06	9.7312D-06
14	0.0000D+00	7.0234D-06	3.2977D-06	4.3086D-06
15	0.0000D+00	1.9775D-06	1.1450D-06	1.7029D-06
16	0.0000D+00	5.8037D-07	4.0809D-07	6.4975D-07
17	0.0000D+00	3.4736D-07	1.5441D-07	2.2919D-07
18	0.0000D+00	1.5414D-07	8.8627D-08	1.7208D-07
19	0.0000D+00	9.1024D-08	6.1738D-08	5.0696D-08
20	0.0000D+00	0.0000D+00	3.6375D-08	8.2449D-08
21	0.0000D+00	0.0000D+00	0.0000D+00	3.2256D-08
23	0.0000D+00	2.7971D-08	0.0000D+00	0.0000D+00
25	0.0000D+00	0.0000D+00	1.1041D-08	0.0000D+00
26	0.0000D+00	2.6821D-08	0.0000D+00	0.0000D+00
27	0.0000D+00	0.0000D+00	0.0000D+00	1.4338D-08
29	0.0000D+00	0.0000D+00	1.0518D-08	0.0000D+00
31	0.0000D+00	0.0000D+00	0.0000D+00	1.3440D-08

G-2: ISI Every 10 Years

```
                    STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
 WESTINGHOUSE           MONTE-CARLO SIMULATION PROGRAM PROBSBFD    VERSION 1.0

==========================================================================
      INPUT VARIABLES FOR CASE  2: PAL 10 YEAR INT

      NCYCLE =     80            NFAILS =  1001            NTRIAL =  1000
      NOVARS =     19            NUMSET =     2            NUMISI =     5
      NUMSSC =      4            NUMTRC =     4            NUMFMD =     4

    VARIABLE         DISTRIBUTION        MEDIAN      DEVIATION    SHIFT    USAGE
   NO.    NAME       TYPE      LOG        VALUE      OR FACTOR    MV/SD   NO. SUB

    1  FIFDepth     - CONSTANT -     3.0000D-02                            1  SET
    2  IFlawDen     - CONSTANT -     3.6589D-03                            2  SET
    3  ICy-ISI      - CONSTANT -     1.0000D+01                            1  ISI
    4  DCy-ISI      - CONSTANT -     1.0000D+01                            2  ISI
    5  MV-Depth     - CONSTANT -     1.5000D-02                            3  ISI
    6  SD-Depth     - CONSTANT -     1.8500D-01                            4  ISI
    7  CEff-ISI     - CONSTANT -     1.0000D+00                            5  ISI
    8  Aspect1      - CONSTANT -     2.0000D+00                            1  SSC
    9  Aspect2      - CONSTANT -     6.0000D+00                            2  SSC
   10  Aspect3      - CONSTANT -     1.0000D+01                            3  SSC
   11  Aspect4      - CONSTANT -     9.9000D+01                            4  SSC
   12  NoTr/Cy      - CONSTANT -     1.3000D+01                            1  TRC
   13  FCGThld      - CONSTANT -     1.5000D+00                            2  TRC
   14  FCGR-UC      NORMAL    NO     0.0000D+00   1.0000D+00      .00      3  TRC
   15  DKINFile     - CONSTANT -     1.0000D+00                            4  TRC
   16  Percent1     - CONSTANT -     7.8870D+01                            1  FMD
   17  Percent2     - CONSTANT -     1.0720D+01                            2  FMD
   18  Percent3     - CONSTANT -     4.3807D+00                            3  FMD
   19  Percent4     - CONSTANT -     6.0298D+00                            4  FMD

  INFORMATION GENERATED FROM FAVLOADS.DAT FILE
  AND SAVED IN DKINSAVE.DAT FILE:

  WALL THICKNESS =  8.7500 INCH

  FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

    TYPE 1 WITH AN ASPECT RATIO OF   2.

    8.75000D-02     2.69285D+00     1.08492D+01
    1.61000D-01     3.60064D+00     1.46562D+01
    4.37500D-01     1.26609D+01     2.00367D+01
    6.56250D-01     1.49279D+01     2.39231D+01
    8.75000D-01     1.53491D+01     2.67406D+01
    1.75000D+00     1.37876D+01     3.14212D+01
    2.62500D+00     8.13906D+00     3.01520D+01
    4.37500D+00    -2.32655D+00     2.91175D+01

    TYPE 2 WITH AN ASPECT RATIO OF   6.

    8.75000D-02     4.04516D+00     1.64003D+01
```

G-2: ISI Every 10 Years (cont.)

1.61000D-01	5.52109D+00	2.25832D+01
4.37500D-01	1.80126D+01	3.03772D+01
6.56250D-01	2.31235D+01	3.61026D+01
8.75000D-01	2.65795D+01	4.11957D+01
1.75000D+00	2.62424D+01	5.18633D+01
2.62500D+00	2.10650D+01	5.45640D+01
4.37500D+00	9.61580D+00	5.85179D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.75000D-02	4.43154D+00	1.79837D+01
1.61000D-01	5.90218D+00	2.41564D+01
4.37500D-01	1.90406D+01	3.24750D+01
6.56250D-01	2.45354D+01	3.85918D+01
8.75000D-01	2.87821D+01	4.40958D+01
1.75000D+00	2.91774D+01	5.64674D+01
2.62500D+00	2.54877D+01	6.25646D+01
4.37500D+00	1.38132D+01	7.03917D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.75000D-02	7.10780D+00	1.85180D+01
1.75000D-01	1.00487D+01	2.59141D+01
2.62500D-01	1.38195D+01	2.86661D+01
4.37500D-01	2.16458D+01	3.45538D+01
6.56250D-01	2.85157D+01	4.23747D+01
8.75000D-01	3.03911D+01	4.83133D+01
1.75000D+00	3.36289D+01	6.57043D+01
2.62500D+00	3.16032D+01	7.68320D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

12	1.2465D-10	1.8940D-12	5.5678D-13	9.1111D-13
13	1.9983D-12	5.5048D-12	2.0459D-12	2.7226D-12
14	0.0000D+00	9.6570D-13	4.5289D-13	5.8811D-13
15	0.0000D+00	1.2835D-13	7.5032D-14	1.0930D-13
16	0.0000D+00	1.8170D-14	1.2594D-14	1.8759D-14
17	0.0000D+00	5.2179D-15	2.1701D-15	2.9926D-15
18	0.0000D+00	9.4118D-16	6.6938D-16	9.6145D-16
19	0.0000D+00	2.9809D-16	1.7580D-16	1.4879D-16
20	0.0000D+00	0.0000D+00	4.8987D-17	9.2976D-17
21	0.0000D+00	0.0000D+00	0.0000D+00	1.4658D-17
23	0.0000D+00	2.2110D-18	0.0000D+00	0.0000D+00
25	0.0000D+00	0.0000D+00	1.5152D-19	0.0000D+00
26	0.0000D+00	2.1470D-19	0.0000D+00	0.0000D+00
27	0.0000D+00	0.0000D+00	0.0000D+00	2.4461D-20
29	0.0000D+00	0.0000D+00	7.9308D-21	0.0000D+00
31	0.0000D+00	0.0000D+00	0.0000D+00	5.2922D-22

APPENDIX H
PALISADES PTS TRANSIENTS

Table H-1 PTS Transient Descriptions for Palisades

Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	HiK	Dominant*
1	2	3.59 cm (1.414 in) surge line break. Containment sump recirculation included in the analysis.	None	2.66E-04	No	Yes	No
2	16	Turbine/reactor trip with 2 stuck-open ADVs on SG-A combined with controller failure resulting in the flow from two AFW pumps into affected steam generator.	Operator starts second AFW pump. Operator isolates AFW to affected SG at 30 minutes after initiation. Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level between 40 and 60 %.	1.23E-04	No	No	No
3	18	Turbine/reactor trip with 1 stuck-open ADV on SG-A. Failure of both MSIVs (SG-A and SG-B) to close.	Operator does not isolate AFW on affected SG. Normal AFW flow assumed (200 gpm). Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level between 40 and 60 %.	4.71E-03	No	No	No
4	19	Reactor trip with 1 stuck-open ADV on SG-A.	None. Operator does not throttle HPI.	2.29E-03	Yes	No	Yes at 60, 200, 500 EFPY
5	22	Turbine/reactor trip with loss of MFW and AFW.	Operator depressurizes through ADVs and feeds SG's using condensate booster pumps. Operators maintain a cooldown rate within technical specification limits and throttle condensate flow at 84 % level in the steam generator.	6.67E-05	No	No	No
6	24	Main steam line break with the break assumed to be inside containment causing containment spray actuation.	None	2.43E-06	No	No	No

Table H-1 PTS Transient Descriptions for Palisades							
Count	**TH Case #**	**System Failure**	**Operator Action**	**Mean IE Frequency**	**HZP**	**HiK**	**Dominant***
7	26	Main steam line break with the break assumed to be inside containment causing containment spray actuation.	Operator isolates AFW to affected SG at 30 minutes after initiation.	5.69E-04	No	No	No
8	27	Main steam line break with controller failure resulting in the flow from two AFW pumps into affected steam generator. Break assumed to be inside containment causing containment spray actuation.	Operator starts second AFW pump.	3.65E-05	No	No	No
9	29	Main steam line break with break assumed to be inside containment causing containment spray actuation.	None. Operator does not throttle HPI.	4.20E-08	Yes	No	No
10	31	Turbine/reactor trip with failure of MFW and AFW. Containment spray actuation assumed due to PORV discharge.	Operator maintains core cooling by "feed and bleed" using HPI to feed and two PORVs to bleed.	1.29E-05	No	No	No
11	32	Turbine/reactor trip with failure of MFW and AFW. Containment spray actuation assumed due to PORV discharge.	Operator maintains core cooling by "feed and bleed" using HPI to feed and two PORV to bleed. AFW is recovered 15 minutes after initiation of "feed and bleed" cooling. Operator closes PORVs when SG level reaches 60 percent.	1.08E-06	No	No	No
12	34	Main steam line break concurrent with a single tube failure in SG-A due to MSLB vibration.	Operator isolates AFW to affected SG at 15 minutes after initiation. Operator trips RCPs assuming that they do not trip as a result of the event. Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level between 40 and 60 %.	1.48E-05	No	No	No
13	40	40.64 cm (16 in) hot leg break. Containment sump recirculation included in the analysis.	None. Operator does not throttle HPI.	3.22E-05	No	Yes	Yes at 32, 60, 200, 500 EFPY

Table H-1	PTS Transient Descriptions for Palisades						
Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	HiK	Dominant*
14	42	Turbine/reactor trip with two stuck open pressurizer SRVs. Containment spray is assumed not to actuate.	Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level between 40 and 60 %.	7.67E-07	No	No	No
15	48	Two stuck-open pressurizer SRVs that reclose at 6000 sec after initiation. Containment spray is assumed not to actuate.	None. Operator does not throttle HPI.	7.67E-07	Yes	No	Yes at 32 EFPY
16	49	Main steam line break with the break assumed to be inside containment causing containment spray actuation.	Operator isolates AFW to affected SG at 30 minutes after initiation. Operator does not throttle HPI.	1.00e-05	Yes	No	No
17	50	Main steam line break with controller failure resulting in the flow from two AFW pumps into affected steam generator. Break assumed to be inside containment causing containment spray actuation.	Operator starts second AFW pump. Operator does not throttle HPI.	5.81E-07	Yes	No	No
18	51	Main steam line break with failure of both MSIVs to close. Break assumed to be inside containment causing containment spray actuation.	Operator does not isolate AFW on affected SG. Operator does not throttle HPI.	7.51E-08	Yes	No	No
19	52	Reactor trip with 1 stuck-open ADV on SG-A. Failure of both MSIVs (SG-A and SG-B) to close.	Operator does not isolate AFW on affected SG. Normal AFW flow assumed (200 gpm). Operator does not throttle HPI.	6.37E-04	Yes	No	Yes at 500 EFPY
20	53	Turbine/reactor trip with two stuck-open pressurizer SRVs that reclose at 6000 sec after initiation. Containment spray is assumed not to actuate.	None. Operator does not throttle HPI.	1.09E-03	No	No	Yes at 500 EFPY
21	54	Main steam line break with failure of both MSIVs to close. Break assumed to be inside containment causing containment spray actuation.	Operator does not isolate AFW on affected SG. Operator does not throttle HPI.	4.26E-06	No	No	Yes at 32, 60, 200, 500 EFPY

Table H-1 PTS Transient Descriptions for Palisades							
Count	**TH Case #**	**System Failure**	**Operator Action**	**Mean IE Frequency**	**HZP**	**HiK**	**Dominant***
22	55	Turbine/reactor trip with 2 stuck-open ADVs on SG-A combined with controller failure resulting in the flow from two AFW pumps into affected steam generator.	Operator starts second AFW pump.	2.74E-03	No	No	Yes at 32, 60, 200, 500 EFPY
23	58	10.16 cm (4 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	2.66E-04	No	Yes	Yes at 32, 60, 200, 500 EFPY
24	59	10.16 cm (4 in) cold leg break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	None. Operator does not throttle HPI.	2.09E-04	No	Yes	Yes at 500 EFPY
25	60	5.08 cm (2 in) surge line break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	2.09E-04	No	Yes	Yes at 60, 200, 500 EFPY
26	61	7.18 cm (2.8 in) cold leg break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	None. Operator does not throttle HPI.	2.09E-04	No	Yes	No
27	62	20.32 cm (8 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	7.07E-06	No	Yes	Yes at 32, 60, 200, 500 EFPY
28	63	14.37 cm (5.656 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	6.06E-06	No	Yes	Yes at 60, 200, 500 EFPY
29	64	10.16 cm (4 in) surge line break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	None. Operator does not throttle HPI.	7.07E-06	No	Yes	Yes at 32, 60, 200, 500 EFPY
30	65	One stuck-open pressurizer SRV that recloses at 6000 sec after initiation. Containment spray is assumed not to actuate.	None. Operator does not throttle HPI.	1.24E-04	Yes	No	Yes at 32, 60, 200, 500 EFPY

Notes:
1. TH ### – Thermal hydraulics run number ###
2. LOCA – Loss-of-coolant accident
3. SBLOCA – Small-break loss-of-coolant accident
4. MBLOCA – Medium-break loss-of-coolant accident
5. LBLOCA – Large-break loss-of-coolant accident
6. HZP – Hot-zero power
7. ADV – Atmospheric dump valve
8. SRV – Safety and relief valve
9. MSLB – Main steam line break
10. AFW – Auxiliary feedwater
11. HPI – High-pressure injection
12. RCP – Reactor coolant pump
13. SG – Steam generator

* The arbitrary definition of a dominant transient is a transient that contributes 1% or more of the total Through-Wall Cracking Failure (TWCF).

APPENDIX I
PALISADES FAVPOST OUTPUT

I-1: 10 Year ISI only

```
**************************************************
*                                                *
*               WELCOME TO FAVOR                 *
*                                                *
*   FRACTURE ANALYSIS OF VESSELS: OAK RIDGE      *
*              VERSION 06.1                      *
*                                                *
*     FAVPOST MODULE: POSTPROCESSOR MODULE       *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES    *
*        WITH RESULTS OF PFM ANALYSIS            *
*                                                *
*     PROBLEMS OR QUESTIONS REGARDING FAVOR      *
*            SHOULD BE DIRECTED TO               *
*                                                *
*               TERRY DICKSON                    *
*        OAK RIDGE NATIONAL LABORATORY           *
*                                                *
*        e-mail: dicksontl@ornl.gov              *
*                                                *
**************************************************

***********************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
* Neither the United States, nor the United States        *
* Department of Energy, nor the United States Nuclear     *
* Regulatory Commission, nor any of their employees,      *
* nor any of their contractors, subcontractors, or their  *
* employees, makes any warranty, expressed or implied, or *
* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
***********************************************************

     DATE: 27-Jun-2007  TIME: 10:25:13


 Begin echo of FAVPost input data deck         10:25:13   27-Jun-2007
 End echo of FAVPost input data deck         10:25:13   27-Jun-2007

     FAVPOST INPUT  FILE NAME                        = postpl.in
     FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
     FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
     FAVPOST OUTPUT FILE NAME                        = plpost10yronly.out
```

I-1: 10 Year ISI only (cont.)

```
*********************************
* NUMBER OF SIMULATIONS =  70000 *
*********************************
```

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
16	2.0991E-09	0.0000E+00	0.0000E+00	2.6647E-10	0.0000E+00	0.0000E+00	0.1269
18	4.1893E-11	0.0000E+00	0.0000E+00	2.0525E-11	0.0000E+00	0.0000E+00	0.4899
19	5.1457E-07	0.0000E+00	0.0000E+00	3.4107E-07	0.0000E+00	0.0000E+00	0.6628
22	6.0538E-09	0.0000E+00	0.0000E+00	1.7933E-09	0.0000E+00	0.0000E+00	0.2962
24	7.2366E-07	0.0000E+00	0.0000E+00	8.5491E-08	0.0000E+00	0.0000E+00	0.1181
26	7.2366E-07	0.0000E+00	0.0000E+00	9.3663E-08	0.0000E+00	0.0000E+00	0.1294
27	2.0421E-05	4.3132E-04	2.0173E-04	5.9362E-06	1.6195E-04	6.0880E-05	0.2907
29	6.8282E-07	0.0000E+00	0.0000E+00	4.6244E-07	0.0000E+00	0.0000E+00	0.6772
31	2.3780E-05	2.9665E-04	2.6221E-04	5.4087E-06	9.2535E-05	5.7966E-05	0.2274
32	4.3551E-07	0.0000E+00	0.0000E+00	3.0097E-07	0.0000E+00	0.0000E+00	0.6911
34	3.4459E-06	0.0000E+00	1.1896E-06	6.4355E-07	0.0000E+00	1.6486E-07	0.1868
40	6.0211E-03	1.2838E-02	6.9324E-02	3.8794E-04	8.0216E-04	5.1013E-03	0.0644
42	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
48	5.4079E-04	2.4464E-03	7.3391E-03	5.3190E-04	2.4464E-03	7.2226E-03	0.9835
49	2.8452E-07	0.0000E+00	0.0000E+00	4.9008E-08	0.0000E+00	0.0000E+00	0.1722
50	4.4689E-05	6.9298E-04	6.6733E-04	1.4807E-05	2.7591E-04	2.2267E-04	0.3313
51	2.3653E-04	1.5112E-03	3.3702E-03	1.1828E-04	8.5384E-04	1.6646E-03	0.5001
52	6.0157E-07	0.0000E+00	0.0000E+00	3.9569E-07	0.0000E+00	0.0000E+00	0.6578
53	2.6708E-07	0.0000E+00	0.0000E+00	1.8524E-07	0.0000E+00	0.0000E+00	0.6936
54	4.7146E-04	2.0579E-03	6.9604E-03	2.1455E-04	1.1565E-03	3.1205E-03	0.4551
55	1.3039E-06	0.0000E+00	0.0000E+00	9.5266E-07	0.0000E+00	0.0000E+00	0.7306
58	3.9842E-04	1.9057E-03	5.2952E-03	7.5527E-05	5.0201E-04	8.2070E-04	0.1896
59	1.6591E-05	2.2922E-04	1.5697E-04	1.1995E-06	3.1036E-05	9.8600E-06	0.0723
60	4.2014E-05	7.4348E-04	5.9014E-04	5.3889E-06	8.0611E-05	7.5170E-05	0.1283
61	9.0696E-07	0.0000E+00	2.2070E-10	3.9814E-08	0.0000E+00	1.5568E-12	0.0439
62	4.8426E-03	1.0080E-02	5.6235E-02	5.1473E-04	1.0533E-03	6.6194E-03	0.1063
63	1.7066E-03	3.3225E-03	2.3408E-02	2.4011E-04	8.3884E-04	3.0140E-03	0.1407
64	1.9466E-03	4.0456E-03	2.5502E-02	2.8279E-04	5.7833E-04	3.9028E-03	0.1453

I-1: 10 Year ISI only (cont.)

65	1.7674E-04	1.5370E-03	2.8655E-03	1.7065E-04	1.5297E-03	2.7897E-03	0.9655

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

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**********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*         FOR THE FREQUENCY OF CRACK INITIATION           *
**********************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	1.7957	1.7957
1.8255E-06	95.9914	97.7871
5.4764E-06	1.1900	98.9771
9.1273E-06	0.3786	99.3557
1.2778E-05	0.1929	99.5486
1.6429E-05	0.1129	99.6614
2.0080E-05	0.0829	99.7443
2.3731E-05	0.0514	99.7957
2.7382E-05	0.0357	99.8314
3.1033E-05	0.0343	99.8657
3.4684E-05	0.0143	99.8800
3.8335E-05	0.0171	99.8971
4.1986E-05	0.0171	99.9143
4.5636E-05	0.0129	99.9271
4.9287E-05	0.0071	99.9343
5.2938E-05	0.0086	99.9429
5.6589E-05	0.0086	99.9514
6.0240E-05	0.0043	99.9557
6.3891E-05	0.0014	99.9571
6.7542E-05	0.0043	99.9614

I-1: 10 Year ISI only (cont.)

7.1193E-05	0.0029	99.9643
7.4844E-05	0.0014	99.9657
7.8495E-05	0.0043	99.9700
8.2146E-05	0.0029	99.9729
8.9447E-05	0.0029	99.9757
9.3098E-05	0.0014	99.9771
9.6749E-05	0.0014	99.9786
1.0040E-04	0.0014	99.9800
1.0405E-04	0.0029	99.9829
1.0770E-04	0.0014	99.9843
1.1500E-04	0.0014	99.9857
1.1865E-04	0.0029	99.9886
1.3326E-04	0.0014	99.9900
1.3691E-04	0.0014	99.9914
1.4056E-04	0.0014	99.9929
1.4786E-04	0.0014	99.9943
1.5516E-04	0.0014	99.9957
2.0263E-04	0.0014	99.9971
3.0120E-04	0.0014	99.9986
3.5962E-04	0.0014	100.0000

```
=============================================
==      Summary Descriptive Statistics     ==
=============================================

Minimum                              = 0.0000E+00
Maximum                              = 3.6144E-04
Range                                = 3.6144E-04

Number of Simulations                =  70000

5th Percentile                       = 1.0576E-11
Median                               = 3.1534E-08
95.0th Percentile                    = 1.8255E-06
99.0th Percentile                    = 5.6968E-06
99.9th Percentile                    = 3.8943E-05

Mean                                 = 4.7962E-07
Standard Deviation                   = 3.4846E-06
Standard Error                       = 1.3171E-08
Variance (unbiased)                  = 1.2142E-11
Variance (biased)                    = 1.2142E-11
Moment Coeff. of Skewness            = 4.1589E+01
Pearson's 2nd Coeff. of Skewness     = 4.1292E-01
Kurtosis                             = 3.0077E+03


***********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)      *
```

****

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	2.7414	2.7414
9.9132E-07	96.6714	99.4129
2.9740E-06	0.3271	99.7400
4.9566E-06	0.0971	99.8371
6.9392E-06	0.0557	99.8929
8.9219E-06	0.0286	99.9214
1.0904E-05	0.0186	99.9400
1.2887E-05	0.0100	99.9500
1.4870E-05	0.0100	99.9600
1.6852E-05	0.0071	99.9671
1.8835E-05	0.0071	99.9743
2.0818E-05	0.0029	99.9771
2.2800E-05	0.0057	99.9829
2.4783E-05	0.0029	99.9857
2.8748E-05	0.0014	99.9871
3.2713E-05	0.0014	99.9886
3.4696E-05	0.0014	99.9900
3.6679E-05	0.0014	99.9914
4.2627E-05	0.0014	99.9929
4.4609E-05	0.0014	99.9943
4.6592E-05	0.0029	99.9971
1.0409E-04	0.0014	99.9986
1.9727E-04	0.0014	100.0000

```
============================================
==     Summary Descriptive Statistics     ==
============================================

Minimum                              = 0.0000E+00
Maximum                              = 1.9628E-04
Range                                = 1.9628E-04

Number of Simulations                =  70000

5th Percentile                       = 1.2334E-13
Median                               = 2.0809E-09
95.0th Percentile                    = 9.9132E-07
99.0th Percentile                    = 1.1299E-06
99.9th Percentile                    = 7.4349E-06

Mean                                 = 7.6237E-08
Standard Deviation                   = 1.0800E-06
Standard Error                       = 4.0821E-09
Variance (unbiased)                  = 1.1664E-12
Variance (biased)                    = 1.1664E-12
Moment Coeff. of Skewness            = 1.0725E+02
Pearson's 2nd Coeff. of Skewness     = 1.2417E-01
Kurtosis                             = 1.7088E+04
```

I-1: 10 Year ISI only (cont.)

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*************************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION      *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -     *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES       *
*************************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
2	0.00	0.00
16	0.00	0.00
18	0.00	0.00
19	0.28	1.10
22	0.00	0.00
24	0.00	0.00
26	0.10	0.08
27	0.16	0.30
29	0.00	0.00
31	0.07	0.10
32	0.00	0.00
34	0.01	0.01
40	50.98	20.66
42	0.00	0.00
48	0.09	0.58
49	0.00	0.00
50	0.01	0.01
51	0.01	0.02
52	0.05	0.21
53	0.06	0.26
54	0.57	1.62
55	1.02	4.91
58	24.02	27.50
59	0.79	0.36
60	1.91	1.55
61	0.05	0.01
62	8.97	5.93
63	2.59	2.33
64	3.42	3.01
65	4.84	29.44
TOTALS	100.00	100.00

I-1: 10 Year ISI only (cont.)

```
**********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION  *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -   *
*                           BY                           *
*                RPV BELTLINE MAJOR REGION               *
*                   BY PARENT SUBREGION                  *
*                                                        *
*      WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT      *
*         TO FREQUENCY OF CRACK INITIATION AND           *
*       THROUGH-WALL CRACKING FREQUENCY (FAILURE)        *
**********************************************************
```

MAJOR REGION	RTndt (MAX)	% of total flaws	% of total frequency of crack initiation	% of total through-wall crack frequency cleavage	ductile	total
1	237.44	2.04	7.75	3.95	2.55	6.51
2	246.96	2.04	10.50	6.85	4.40	11.24
3	246.96	2.04	11.34	7.15	4.89	12.04
4	247.43	3.16	23.77	19.46	10.46	29.92
5	237.77	3.16	16.33	10.62	5.75	16.36
6	247.43	3.16	22.11	14.52	9.40	23.92
7	231.49	19.12	7.68	0.01	0.00	0.01
8	195.14	8.55	0.04	0.00	0.00	0.00
9	166.15	8.55	0.00	0.00	0.00	0.00
10	130.00	8.55	0.00	0.00	0.00	0.00
11	206.88	13.21	0.18	0.00	0.00	0.00
12	186.36	13.21	0.03	0.00	0.00	0.00
13	208.62	13.21	0.26	0.00	0.00	0.00
	TOTALS	100.00	100.00	62.56	37.44	100.00

```
**********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION  *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -   *
*                           BY                           *
*                RPV BELTLINE MAJOR REGION               *
*                   BY CHILD SUBREGION                   *
*                                                        *
*      WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT      *
*         TO FREQUENCY OF CRACK INITIATION AND           *
*       THROUGH-WALL CRACKING FREQUENCY (FAILURE)        *
**********************************************************
```

I-1: 10 Year ISI only (cont.)

MAJOR REGION	RTndt (MAX)	% of total flaws	% of total frequency of crack initiation	% of total through-wall crack frequency cleavage	ductile	total
1	237.44	2.04	7.75	3.95	2.55	6.51
2	246.96	2.04	10.50	6.84	4.40	11.24
3	246.96	2.04	11.34	7.15	4.88	12.04
4	247.43	3.16	23.77	19.46	10.46	29.92
5	237.77	3.16	16.33	10.62	5.75	16.36
6	247.43	3.16	22.11	14.52	9.40	23.92
7	231.49	19.12	7.32	0.01	0.00	0.01
8	195.14	8.55	0.04	0.00	0.00	0.00
9	166.15	8.55	0.00	0.00	0.00	0.00
10	130.00	8.55	0.00	0.00	0.00	0.00
11	206.88	13.21	0.18	0.00	0.00	0.00
12	186.36	13.21	0.03	0.00	0.00	0.00
13	208.62	13.21	0.61	0.00	0.00	0.00
	TOTALS	100.00	100.00	62.56	37.44	100.00

```
**********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION   *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*         MATERIAL, FLAW CATEGORY, AND FLAW DEPTH         *
*                                                         *
*       WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT      *
*           TO FREQUENCY OF CRACK INITIATION AND          *
*          THROUGH-WALL CRACKING FREQUENCY (FAILURE)      *
**********************************************************
```

```
*****************
* WELD MATERIAL *
*****************
```

FLAW DEPTH (in)	% of total frequency of crack initiation CAT I flaws	CAT 2 flaws	CAT 3 flaws	% of total through-wall crack frequency CAT 1 flaws	CAT 2 flaws	CAT 3 flaws
0.088	0.00	3.48	0.00	0.00	1.59	0.00
0.175	0.00	36.00	0.00	0.00	19.98	0.00
0.263	0.00	13.27	0.00	0.00	10.08	0.00
0.350	0.00	8.79	0.00	0.00	9.08	0.01
0.438	0.00	7.99	0.01	0.00	9.15	0.03
0.525	0.00	6.37	0.01	0.00	9.36	0.06

I-1: 10 Year ISI only (cont.)

0.613	0.00	4.47	0.01	0.00	5.81	0.05
0.700	0.00	3.48	0.01	0.00	5.76	0.08
0.787	0.00	2.88	0.02	0.00	4.99	0.10
0.875	0.00	2.33	0.01	0.00	3.81	0.05
0.963	0.00	1.55	0.03	0.00	2.79	0.18
1.050	0.06	2.27	0.04	0.00	6.36	0.20
1.137	1.00	1.78	0.02	0.03	3.20	0.13
1.225	0.02	1.27	0.01	0.00	2.32	0.09
1.313	0.09	0.76	0.00	0.00	1.53	0.02
1.400	0.00	0.35	0.01	0.00	0.63	0.03
1.488	0.00	0.19	0.00	0.00	0.37	0.02
1.575	0.00	0.29	0.00	0.00	0.49	0.00
1.663	0.00	0.03	0.00	0.00	0.10	0.01
1.750	0.00	0.26	0.01	0.00	0.72	0.05
1.837	0.03	0.00	0.00	0.00	0.03	0.00
1.925	0.00	0.28	0.00	0.00	0.68	0.02
2.013	0.00	0.00	0.00	0.00	0.00	0.00
2.100	0.00	0.00	0.00	0.00	0.00	0.00
2.188	0.00	0.00	0.00	0.00	0.00	0.00
2.275	0.00	0.00	0.00	0.00	0.00	0.00
2.363	0.00	0.00	0.00	0.00	0.00	0.00
2.450	0.00	0.00	0.00	0.00	0.00	0.00
2.537	0.00	0.00	0.00	0.00	0.00	0.00
2.625	0.00	0.00	0.00	0.00	0.00	0.00
2.712	0.00	0.00	0.00	0.00	0.00	0.00
TOTALS	1.20	98.08	0.19	0.03	98.83	1.13

****
* PLATE MATERIAL *
****

FLAW DEPTH (in)	% of total frequency of crack initiation CAT I flaws	CAT 2 flaws	CAT 3 flaws	% of total through-wall crack frequency CAT 1 flaws	CAT 2 flaws	CAT 3 flaws
0.088	0.00	0.00	0.00	0.00	0.00	0.00
0.175	0.00	0.00	0.00	0.00	0.00	0.00
0.263	0.00	0.00	0.00	0.00	0.00	0.00
0.350	0.00	0.00	0.00	0.00	0.00	0.00
0.438	0.00	0.00	0.00	0.00	0.00	0.00
0.525	0.00	0.00	0.00	0.00	0.00	0.00
0.613	0.00	0.00	0.00	0.00	0.00	0.00
0.700	0.00	0.00	0.00	0.00	0.00	0.00
0.787	0.00	0.00	0.00	0.00	0.00	0.00
0.875	0.00	0.00	0.00	0.00	0.00	0.00
0.963	0.00	0.00	0.00	0.00	0.00	0.00
1.050	0.07	0.00	0.00	0.00	0.00	0.00
1.137	0.28	0.00	0.00	0.00	0.00	0.00

I-1: 10 Year ISI only (cont.)

1.225	0.10	0.00	0.00	0.00	0.00	0.00
1.313	0.02	0.00	0.00	0.00	0.00	0.00
1.400	0.04	0.00	0.00	0.00	0.00	0.00
1.488	0.00	0.00	0.00	0.00	0.00	0.00
1.575	0.00	0.00	0.00	0.00	0.00	0.00
1.663	0.00	0.00	0.00	0.00	0.00	0.00
1.750	0.01	0.00	0.00	0.00	0.00	0.00
1.837	0.00	0.00	0.00	0.00	0.00	0.00
1.925	0.00	0.00	0.00	0.00	0.00	0.00
2.013	0.00	0.00	0.00	0.00	0.00	0.00
2.100	0.00	0.00	0.00	0.00	0.00	0.00
2.188	0.00	0.00	0.00	0.00	0.00	0.00
2.275	0.00	0.00	0.00	0.00	0.00	0.00
2.363	0.00	0.00	0.00	0.00	0.00	0.00
2.450	0.00	0.00	0.00	0.00	0.00	0.00
2.537	0.00	0.00	0.00	0.00	0.00	0.00
2.625	0.00	0.00	0.00	0.00	0.00	0.00
2.712	0.00	0.00	0.00	0.00	0.00	0.00
TOTALS	0.52	0.00	0.00	0.00	0.00	0.00

DATE: 27-Jun-2007 TIME: 10:25:45

I-2: ISI Every 10 Years

```
************************************************
*                                              *
*              WELCOME TO FAVOR                *
*                                              *
*   FRACTURE ANALYSIS OF VESSELS: OAK RIDGE    *
*                VERSION 06.1                  *
*                                              *
*     FAVPOST MODULE: POSTPROCESSOR MODULE     *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES  *
*        WITH RESULTS OF PFM ANALYSIS          *
*                                              *
*     PROBLEMS OR QUESTIONS REGARDING FAVOR    *
*            SHOULD BE DIRECTED TO             *
*                                              *
*               TERRY DICKSON                  *
*        OAK RIDGE NATIONAL LABORATORY         *
*                                              *
*          e-mail: dicksontl@ornl.gov          *
*                                              *
************************************************

**********************************************************
* This computer program was prepared as an account of    *
* work sponsored by the United States Government         *
* Neither the United States, nor the United States       *
* Department of Energy, nor the United States Nuclear    *
* Regulatory Commission, nor any of their employees,     *
* nor any of their contractors, subcontractors, or their *
* employees, makes any warranty, expressed or implied, or *
* assumes any legal liability or responsibility for the  *
* accuracy, completeness, or usefulness of any           *
* information, apparatus, product, or process disclosed, *
* or represents that its use would not infringe          *
* privately-owned rights.                                *
**********************************************************


     DATE: 27-Jun-2007  TIME: 10:23:32


Begin echo of FAVPost input data deck          10:23:32   27-Jun-2007
End echo of FAVPost input data deck          10:23:32   27-Jun-2007


     FAVPOST INPUT  FILE NAME                        = postpl.in
     FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
     FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
     FAVPOST OUTPUT FILE NAME                        = plpost10yrint.out
```

I-2: ISI Every 10 Years (cont.)

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*********************************
* NUMBER OF SIMULATIONS =  70000 *
*********************************
```

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
16	1.2905E-10	0.0000E+00	0.0000E+00	1.9672E-11	0.0000E+00	0.0000E+00	0.1524
18	1.9999E-08	0.0000E+00	0.0000E+00	1.3349E-08	0.0000E+00	0.0000E+00	0.6675
19	4.3542E-07	0.0000E+00	0.0000E+00	2.9306E-07	0.0000E+00	0.0000E+00	0.6730
22	4.7785E-10	0.0000E+00	0.0000E+00	1.6163E-10	0.0000E+00	0.0000E+00	0.3382
24	4.2145E-07	0.0000E+00	0.0000E+00	5.5149E-08	0.0000E+00	0.0000E+00	0.1309
26	4.2145E-07	0.0000E+00	0.0000E+00	5.6127E-08	0.0000E+00	0.0000E+00	0.1332
27	1.6689E-05	4.0851E-04	1.8246E-04	5.0727E-06	1.3999E-04	5.5346E-05	0.3040
29	3.2222E-07	0.0000E+00	0.0000E+00	2.2848E-07	0.0000E+00	0.0000E+00	0.7091
31	1.9968E-05	3.1503E-04	2.2641E-04	5.2229E-06	8.4354E-05	5.6231E-05	0.2616
32	4.1319E-07	0.0000E+00	0.0000E+00	2.7634E-07	0.0000E+00	0.0000E+00	0.6688
34	2.0638E-06	0.0000E+00	9.6818E-07	3.9138E-07	0.0000E+00	1.4814E-07	0.1896
40	5.8654E-03	1.3025E-02	6.3577E-02	3.8441E-04	8.2240E-04	4.9153E-03	0.0655
42	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
48	5.1460E-04	2.9847E-03	7.2004E-03	5.0771E-04	2.9613E-03	7.1338E-03	0.9866
49	1.2025E-07	0.0000E+00	0.0000E+00	2.3074E-08	0.0000E+00	0.0000E+00	0.1919
50	3.8097E-05	6.1000E-04	6.0949E-04	1.3058E-05	2.3488E-04	2.0982E-04	0.3428
51	2.1960E-04	1.5122E-03	3.0705E-03	1.1283E-04	6.4702E-04	1.6553E-03	0.5138
52	5.4043E-07	0.0000E+00	0.0000E+00	3.6486E-07	0.0000E+00	0.0000E+00	0.6751
53	2.6394E-07	0.0000E+00	0.0000E+00	1.9008E-07	0.0000E+00	0.0000E+00	0.7202
54	4.4223E-04	2.0292E-03	6.0638E-03	2.0682E-04	9.2843E-04	2.9785E-03	0.4677
55	1.1909E-06	0.0000E+00	0.0000E+00	8.8283E-07	0.0000E+00	0.0000E+00	0.7413
58	3.4658E-04	1.0963E-03	5.2758E-03	7.1215E-05	3.8156E-04	9.2302E-04	0.2055
59	1.3685E-05	2.8125E-04	1.3312E-04	1.1435E-06	2.6929E-05	8.8218E-06	0.0836

I-2: ISI Every 10 Years (cont.)

60	3.4196E-05	5.1697E-04	4.9575E-04	5.1734E-06	1.0188E-04	6.8749E-05	0.1513
61	6.3437E-07	0.0000E+00	2.4648E-11	3.7831E-08	0.0000E+00	2.4484E-14	0.0596
62	4.6580E-03	1.0090E-02	5.1261E-02	5.0659E-04	1.0724E-03	6.4912E-03	0.1088
63	1.5523E-03	3.1172E-03	2.1441E-02	2.2840E-04	6.1778E-04	3.0185E-03	0.1471
64	1.8747E-03	4.0838E-03	2.2910E-02	2.8006E-04	6.6854E-04	3.7212E-03	0.1494
65	1.6533E-04	1.9541E-03	2.8139E-03	1.6078E-04	1.9452E-03	2.7094E-03	0.9725

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

I-2: ISI Every 10 Years (cont.)

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**********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*         FOR THE FREQUENCY OF CRACK INITIATION           *
**********************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	1.8657	1.8657
1.3669E-06	95.1129	96.9786
4.1008E-06	1.6643	98.6429
6.8346E-06	0.5400	99.1829
9.5685E-06	0.2614	99.4443
1.2302E-05	0.1471	99.5914
1.5036E-05	0.0986	99.6900
1.7770E-05	0.0771	99.7671
2.0504E-05	0.0329	99.8000
2.3238E-05	0.0300	99.8300
2.5972E-05	0.0186	99.8486
2.8705E-05	0.0229	99.8714
3.1439E-05	0.0157	99.8871
3.4173E-05	0.0186	99.9057
3.6907E-05	0.0071	99.9129
3.9641E-05	0.0043	99.9171
4.2375E-05	0.0029	99.9200
4.5109E-05	0.0057	99.9257
4.7842E-05	0.0157	99.9414
5.0576E-05	0.0086	99.9500
5.3310E-05	0.0029	99.9529
5.6044E-05	0.0029	99.9557
6.1512E-05	0.0029	99.9586
6.4246E-05	0.0071	99.9657
6.6979E-05	0.0043	99.9700
6.9713E-05	0.0043	99.9743
7.2447E-05	0.0029	99.9771
7.5181E-05	0.0014	99.9786
8.0649E-05	0.0014	99.9800
8.3383E-05	0.0014	99.9814
8.8850E-05	0.0043	99.9857
9.4318E-05	0.0029	99.9886
9.7052E-05	0.0029	99.9914
9.9786E-05	0.0014	99.9929
1.1072E-04	0.0014	99.9943
1.2166E-04	0.0014	99.9957
1.7907E-04	0.0014	99.9971
2.4195E-04	0.0014	99.9986
2.6928E-04	0.0014	100.0000

I-2: ISI Every 10 Years (cont.)

```
============================================
==      Summary Descriptive Statistics    ==
============================================

Minimum                              = 0.0000E+00
Maximum                              = 2.7065E-04
Range                                = 2.7065E-04

Number of Simulations                =  70000

5th Percentile                       = 9.2117E-12
Median                               = 3.0552E-08
95.0th Percentile                    = 1.3669E-06
99.0th Percentile                    = 5.9089E-06
99.9th Percentile                    = 3.3332E-05

Mean                                 = 4.4736E-07
Standard Deviation                   = 2.9010E-06
Standard Error                       = 1.0965E-08
Variance (unbiased)                  = 8.4156E-12
Variance (biased)                    = 8.4155E-12
Moment Coeff. of Skewness            = 3.6027E+01
Pearson's 2nd Coeff. of Skewness     = 4.6263E-01
Kurtosis                             = 2.3099E+03
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**********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)     *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)     *
**********************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	2.7414	2.7414
7.0253E-07	96.4900	99.2314
2.1076E-06	0.3986	99.6300
3.5126E-06	0.1529	99.7829
4.9177E-06	0.0600	99.8429
6.3228E-06	0.0414	99.8843
7.7278E-06	0.0286	99.9129
9.1329E-06	0.0257	99.9386
1.0538E-05	0.0043	99.9429
1.1943E-05	0.0086	99.9514
1.3348E-05	0.0114	99.9629
1.4753E-05	0.0014	99.9643
1.6158E-05	0.0043	99.9686
1.7563E-05	0.0071	99.9757
2.4589E-05	0.0029	99.9786

I-2: ISI Every 10 Years (cont.)

2.7399E-05	0.0029	99.9814
3.0209E-05	0.0014	99.9829
3.1614E-05	0.0029	99.9857
3.4424E-05	0.0014	99.9871
3.5829E-05	0.0029	99.9900
3.7234E-05	0.0014	99.9914
3.8639E-05	0.0014	99.9929
4.0044E-05	0.0014	99.9943
4.4259E-05	0.0014	99.9957
6.9550E-05	0.0014	99.9971
1.3278E-04	0.0014	99.9986
1.3980E-04	0.0014	100.0000

```
=============================================
==      Summary Descriptive Statistics     ==
=============================================

Minimum                               = 0.0000E+00
Maximum                               = 1.3910E-04
Range                                 = 1.3910E-04

Number of Simulations                 =  70000

5th Percentile                        = 1.2899E-13
Median                                = 2.0727E-09
95.0th Percentile                     = 7.0253E-07
99.0th Percentile                     = 1.0895E-06
99.9th Percentile                     = 7.0955E-06

Mean                                  = 7.3880E-08
Standard Deviation                    = 1.0054E-06
Standard Error                        = 3.8001E-09
Variance (unbiased)                   = 1.0108E-12
Variance (biased)                     = 1.0108E-12
Moment Coeff. of Skewness             = 8.5112E+01
Pearson's 2nd Coeff. of Skewness      = 1.2928E-01
Kurtosis                              = 1.0251E+04


***********************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION     *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES      *
***********************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
2	0.00	0.00
16	0.00	0.00
18	0.00	0.01
19	0.16	0.66

I-2: ISI Every 10 Years (cont.)

22	0.00	0.00
24	0.00	0.00
26	0.05	0.04
27	0.18	0.34
29	0.00	0.00
31	0.06	0.10
32	0.00	0.00
34	0.01	0.01
40	52.00	21.41
42	0.00	0.00
48	0.10	0.57
49	0.00	0.00
50	0.01	0.01
51	0.01	0.02
52	0.09	0.36
53	0.08	0.37
54	0.57	1.63
55	0.66	3.11
58	22.68	28.39
59	0.69	0.33
60	1.80	1.68
61	0.03	0.01
62	9.31	6.13
63	2.87	2.66
64	3.76	3.33
65	4.90	28.85
TOTALS	100.00	100.00

```
***********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION   *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*                           BY                            *
*                 RPV BELTLINE MAJOR REGION               *
*                    BY PARENT SUBREGION                  *
*                                                         *
*       WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT      *
*          TO FREQUENCY OF CRACK INITIATION AND           *
*        THROUGH-WALL CRACKING FREQUENCY (FAILURE)        *
***********************************************************
```

MAJOR REGION	RTndt (MAX)	% of total flaws	% of total frequency of crack initiation	% of total through-wall crack frequency cleavage	ductile	total
1	237.44	2.04	8.49	6.27	3.00	9.27
2	246.96	2.04	11.97	8.50	5.03	13.53
3	246.96	2.04	11.45	7.14	4.59	11.73
4	247.43	3.16	21.86	12.94	10.00	22.94

I-2: ISI Every 10 Years (cont.)

5	237.77	3.16	15.63	8.90	5.70	14.60
6	247.43	3.16	23.36	17.70	10.23	27.93
7	231.49	19.12	7.26	0.00	0.00	0.00
8	195.14	8.55	0.00	0.00	0.00	0.00
9	166.15	8.55	0.00	0.00	0.00	0.00
10	130.00	8.55	0.00	0.00	0.00	0.00
11	206.88	13.21	0.00	0.00	0.00	0.00
12	186.36	13.21	0.00	0.00	0.00	0.00
13	208.62	13.21	0.00	0.00	0.00	0.00
	TOTALS	100.00	100.00	61.45	38.55	100.00

```
***********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION   *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*                           BY                            *
*                 RPV BELTLINE MAJOR REGION               *
*                     BY CHILD SUBREGION                  *
*                                                         *
*       WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT      *
*           TO FREQUENCY OF CRACK INITIATION AND          *
*        THROUGH-WALL CRACKING FREQUENCY (FAILURE)        *
***********************************************************
```

MAJOR REGION	RTndt (MAX)	% of total flaws	% of total frequency of crack initiation	% of total through-wall crack frequency cleavage	ductile	total
1	237.44	2.04	8.49	6.27	3.00	9.27
2	246.96	2.04	11.97	8.50	5.03	13.53
3	246.96	2.04	11.45	7.14	4.59	11.73
4	247.43	3.16	21.86	12.94	10.00	22.94
5	237.77	3.16	15.63	8.90	5.70	14.60
6	247.43	3.16	23.36	17.70	10.23	27.93
7	231.49	19.12	6.96	0.00	0.00	0.00
8	195.14	8.55	0.00	0.00	0.00	0.00
9	166.15	8.55	0.00	0.00	0.00	0.00
10	130.00	8.55	0.00	0.00	0.00	0.00
11	206.88	13.21	0.00	0.00	0.00	0.00
12	186.36	13.21	0.00	0.00	0.00	0.00
13	208.62	13.21	0.29	0.00	0.00	0.00
	TOTALS	100.00	100.00	61.45	38.55	100.00

I-2: ISI Every 10 Years (cont.)

```
**********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION  *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -   *
*         MATERIAL, FLAW CATEGORY, AND FLAW DEPTH        *
*                                                        *
*       WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT     *
*           TO FREQUENCY OF CRACK INITIATION AND         *
*         THROUGH-WALL CRACKING FREQUENCY (FAILURE)      *
**********************************************************
```

```
*****************
* WELD MATERIAL *
*****************
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FLAW DEPTH (in)	% of total frequency of crack initiation CAT I flaws	CAT 2 flaws	CAT 3 flaws	% of total through-wall crack frequency CAT 1 flaws	CAT 2 flaws	CAT 3 flaws
0.088	0.00	3.70	0.00	0.00	1.68	0.00
0.175	0.00	37.45	0.00	0.00	20.91	0.00
0.263	0.00	13.39	0.00	0.00	10.25	0.00
0.350	0.00	10.03	0.00	0.00	9.61	0.01
0.438	0.00	7.97	0.01	0.00	9.27	0.05
0.525	0.00	6.65	0.01	0.00	10.34	0.06
0.613	0.00	5.13	0.01	0.00	7.94	0.06
0.700	0.00	3.58	0.03	0.00	5.22	0.14
0.787	0.00	2.47	0.02	0.00	3.86	0.13
0.875	0.00	1.59	0.00	0.00	1.90	0.02
0.963	0.00	2.29	0.01	0.00	4.94	0.05
1.050	0.00	0.96	0.02	0.00	1.60	0.10
1.137	0.00	1.31	0.01	0.00	2.78	0.03
1.225	0.00	0.78	0.00	0.00	1.67	0.01
1.313	0.00	0.34	0.01	0.00	0.43	0.04
1.400	0.00	0.54	0.00	0.00	1.31	0.03
1.488	0.00	0.40	0.00	0.00	1.14	0.01
1.575	0.00	0.05	0.00	0.00	0.13	0.01
1.663	0.00	0.84	0.01	0.00	3.14	0.04
1.750	0.00	0.13	0.00	0.00	0.42	0.00
1.837	0.00	0.17	0.00	0.00	0.60	0.00
1.925	0.00	0.06	0.00	0.00	0.04	0.02
TOTALS	0.00	99.86	0.14	0.00	99.17	0.83

I-2: ISI Every 10 Years (cont.)

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******************
* PLATE MATERIAL *
******************
```

FLAW DEPTH (in)	% of total frequency of crack initiation CAT I flaws	CAT 2 flaws	CAT 3 flaws	% of total through-wall crack frequency CAT 1 flaws	CAT 2 flaws	CAT 3 flaws
0.088	0.00	0.00	0.00	0.00	0.00	0.00
0.175	0.00	0.00	0.00	0.00	0.00	0.00
0.263	0.00	0.00	0.00	0.00	0.00	0.00
0.350	0.00	0.00	0.00	0.00	0.00	0.00
0.438	0.00	0.00	0.00	0.00	0.00	0.00
0.525	0.00	0.00	0.00	0.00	0.00	0.00
0.613	0.00	0.00	0.00	0.00	0.00	0.00
0.700	0.00	0.00	0.00	0.00	0.00	0.00
0.787	0.00	0.00	0.00	0.00	0.00	0.00
0.875	0.00	0.00	0.00	0.00	0.00	0.00
0.963	0.00	0.00	0.00	0.00	0.00	0.00
1.050	0.00	0.00	0.00	0.00	0.00	0.00
1.137	0.00	0.00	0.00	0.00	0.00	0.00
TOTALS	0.00	0.00	0.00	0.00	0.00	0.00

DATE: 27-Jun-2007 TIME: 10:24:06

APPENDIX J
INPUTS FOR THE OCONEE UNIT 1 PILOT PLANT EVALUATION

A summary of the NDE inspection history based on Regulatory Guide 1.150 and pertinent input data for OC1 is as follows:

1. Number of inservice inspections performed (relative to initial pre-service and 10 year interval inspections) for full penetration category B-A and B-D vessel welds assuming all of the candidate welds were inspected: 3 (covering all welds of the specified categories).

2. The inspections performed covered: 62 total examinations. 23 items with 100% coverage, 22 items with < 90% coverage and 17 items with coverage >90% but less than 100%.

3. Number of indications found during most recent inservice inspection: 44
   This number includes consideration of the following additional information.

   a. Indications found that were reportable: 0
   b. Indications found that were within acceptable limits: 44
   c. Indications/anomalies currently being monitored: 0

4. Full Penetration Relief requests for the reactor vessel submitted and accepted by the NRC: 2 relief requests for limited coverage for 22 items, as noted in item 2

5. Fluence distribution at inside surface of RV Beltline until end of life is shown in: see Figure J-1 taken from the NRC PTS Risk Study [44], Figure 4.1.

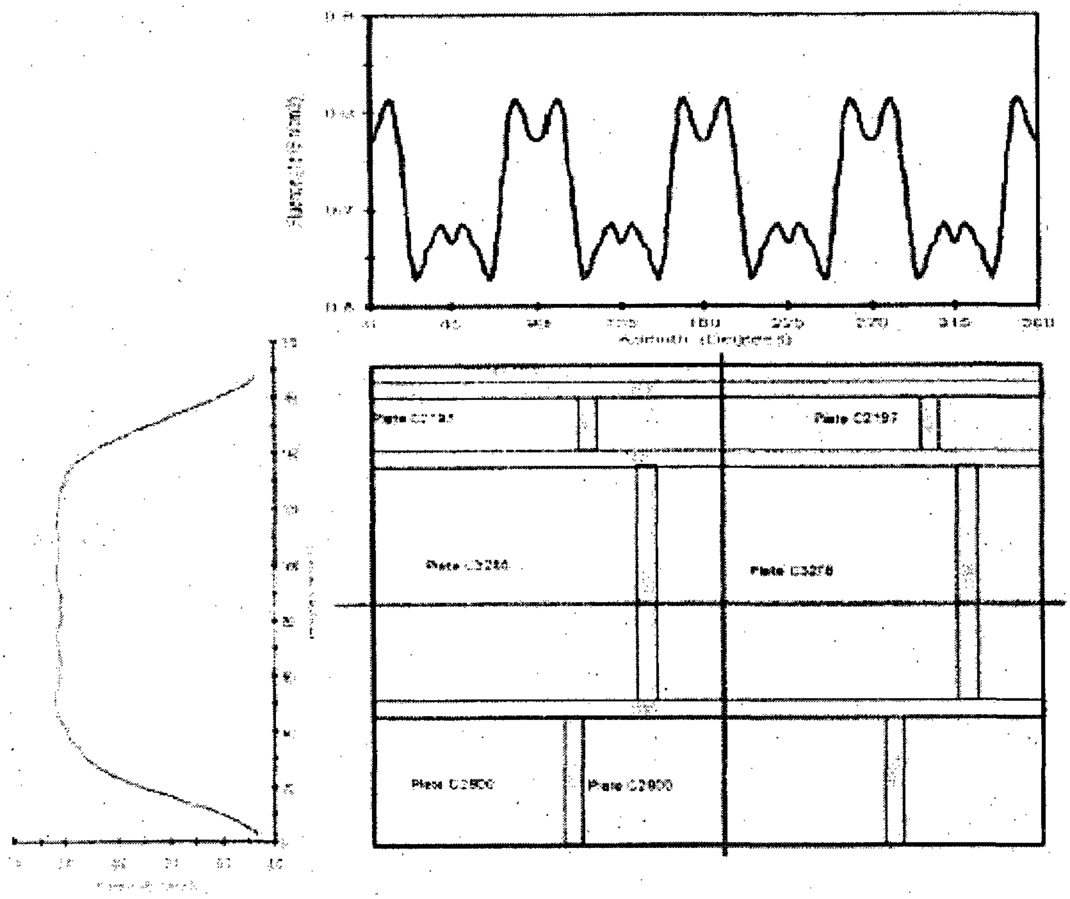

Figure J-1 Rollout Diagram of Beltline Materials and Representative Fluence Maps for OC1

6. Reactor vessel cladding details:
    a. Number of layers: 1
    b. Thickness: 0.188
    c. Material properties are identified in Table J-1:

Table J-1 Cladding Material Properties						
Temperature (°F)	Thermal Conductivity (Btu/hr-ft-°F) "K"	Specific Heat (Btu/LBM-°F) "C"	Young's Modulus of Elasticity (KSI) "E"	Thermal Expansion Coefficient ($°F^{-1}$) "α"	Density (LBM/ft^3) "ρ"	Poisson's Ratio "ν"
0	-	-	-	-	489	.3
68	-	-	22045.7	-	489	.3
70	8.1	0.1158	-	-	489	.3
100	8.4	0.1185	-	8.55E-06	489	.3
150	8.6	0.1196	-	8.67E-06	489	.3
200	8.8	0.1208	-	8.79E-06	489	.3
250	9.1	0.1232	-	8.9E-06	489	.3
300	9.4	0.1256	-	9.0E-06	489	.3
302	-	-	20160.2	-	489	.3
350	9.6	0.1258	-	9.1E-06	489	.3
400	9.9	0.1281	-	9.19E-06	489	.3
450	10.1	0.1291	-	9.28E-06	489	.3
482	-	-	18419.8	-	489	.3
500	10.4	0.1305	-	9.37E-06	489	.3
550	10.6	0.1306	-	9.45E-06	489	.3
600	10.9	0.1327	-	9.53E-06	489	.3
650	11.1	0.1335	-	9.61E-06	489	.3
700	11.4	0.1348	-	9.69E-06	489	.3
750	11.6	0.1356	-	9.76E-06	489	.3
800	11.9	0.1367	-	9.82E-06	489	.3

d. Material including copper and nickel content: Material properties assigned to clad flaws are that of the underlying material be it base metal or weld. These properties are identified in Table J-3. This is consistent with the PTS evaluation [8, 9].

e. Material property uncertainties:

1) Bead width: 1 inch – bead widths vary for all plants. Based on the NRC PTS Risk Study [8, 9], a nominal dimension of 1 inch is selected for all analyses because this parameter is not expected to significantly influence the predicted vessel failure probabilities.
2) Truncation Limit: Cladding thickness rounded up to the next 1/100th of the total vessel thickness to be consistent with the NRC PTS Risk Study [8, 9].
3) Surface flaw depth: 0.03 x 8.626 = 0.259 in
4) All flaws are surface-breaking. Only flaws in cladding that would influence brittle fracture of the reactor vessel are brittle. This is consistent with the NRC PTS Risk Study [8, 9].

f. Additional cladding properties are identified in Table J-4

7. Base metal:

a. Wall thickness: 8.438 inches
b. Material properties are identified in Tables J-2 and J-3:

Table J-2 Base Metal Material Properties

Temperature (°F)	Thermal Conductivity (Btu/hr-ft-°F) "K"	Specific Heat (Btu/LBM-°F) "C"	Young's Modulus of Elasticity (KSI) "E"	Thermal Expansion Coefficient (°F^{-1}) "α"	Density (LBM/ft^3) "ρ"	Poisson's Ratio "ν"
0	-	-	-	-	489	.3
70	24.8	0.1052	29200	-	489	.3
100	25	0.1072	-	7.06E-06	489	.3
150	25.1	0.1101	-	7.16E-06	489	.3
200	25.2	0.1135	28500	7.25E-06	489	.3
250	25.2	0.1166	-	7.34E-06	489	.3
300	25.1	0.1194	28000	7.43E-06	489	.3
350	25	0.1223	-	7.5E-06	489	.3
400	25.1	0.1267	27400	7.58E-06	489	.3
450	24.6	0.1277	-	7.63E-06	489	.3
500	24.3	0.1304	27000	7.7E-06	489	.3
550	24	0.1326	-	7.77E-06	489	.3
600	23.7	0.135	26400	7.83E-06	489	.3
650	23.4	0.1375	-	7.9E-06	489	.3
700	23	0.1404	25300	7.94E-06	489	.3
750	22.6	0.1435	-	8.0E-06	489	.3
800	22.2	0.1474	23900	8.05E-06	489	.3

Table J-3	OC1-Specific Material Values Drawn from the RVID (see Ref. 44 Table 4.1)							
Major Material Region Description								
#	Type	Heat	Location	Cu [wt%]	Ni [wt%]	P [wt%]	Mn [wt%]	Un-Irradiated RT_{NDT} [°F]
1	Axial Weld	SA-1430	Lower	0.190	0.570	0.017	1.480	-5
2	Axial Weld	SA-1493	Intermediate	0.190	0.570	0.017	1.480	-5
3	Axial Weld	SA-1073	Upper	0.210	0.640	0.025	1.380	-5
4	Circ Weld	SA-1585	Lower	0.220	0.540	0.016	1.436	-5
5	Circ Weld	SA-1229	Intermediate	0.230	0.590	0.021	1.488	10
6	Circ Weld	SA-1135	Upper	0.230	0.520	0.011	1.404	- 5
7	Plate	C-2800	Lower	0.110	0.630	0.012	1.400	1
8	Plate	C3265-1	Intermediate	0.100	0.500	0.015	1.420	1
9	Plate	C3278-1	Intermediate	0.120	0.600	0.010	1.260	1
10	Plate	C2197-2	Upper	0.150	0.500	0.008	1.280	1
11	Forging	ZV2861	Upper	0.160	0.650	0.006	0.800	3

8. Weld metal details: Details of information used in addressing weld-specific information are taken directly from the NRC PTS Risk Study [44], Table 4.2. Summaries are reproduced as Table J-4. Values for SAW Weld Volume fraction and Repair Weld Volume fraction in Table J-4 were changed to 96.7% and 2.3% respectively per NUREG-1874 [9].

Table J-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution

Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Inner Radius (to cladding)	[in]	85.5	78.5	86	86	Vessel specific info
Base Metal Thickness	[in]	8.438	7.875	8.5	8.675	Vessel specific info
Total Wall Thickness	[in]	8.626	8.031	8.75	8.988	Vessel specific info

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SAW Weld	Volume fraction	[%]	97%				100% - SMAW% - REPAIR%
	Thru-Wall Bead Thickness	[in]	0.1875	0.1875	0.1875	0.1875	All plants report plant specific dimensions of 3/16-in.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table J-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SMAW Weld	Volume fraction	[%]	1%				Upper bound to all plant specific info provided by Steve Byrne (Westinghouse – Windsor).
	Thru-Wall Bead Thickness	[in]	0.21	0.20	0.22	0.25	Oconee is generic value based on average of all plants specific values (including Shoreham & PVRUF data). Other values are plant specific as reported by Steve Byrne.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table J-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Repair Weld	Volume fraction	[%]	2%				Judgment. A rounded integral percentage that exceeds the repaired volume observed for Shoreham and for PVRUF, which was 1.5%.
	Thru-Wall Bead Thickness	[in]	0.14				Generic value: As observed in PVRUF and Shoreham by PNNL.
	Truncation Limit	[in]	2				Judgment. Approx. 2X the largest repair flaw found in PVRUF & Shoreham. Also based on maximum expected width of repair cavity.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				The repair flaws had complex shapes and orientations that were not aligned with either the axial or circumferential welds; for consistency with the available treatments of flaws by the FAVOR code, a common treatment of orientations was adopted for flaws in SAW/SMAW and repair welds.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table J-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

	Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Cladding	Actual Thickness	[in]	0.188	0.156	0.25	0.313	Vessel specific info
	# of Layers	[#]	1	2	2	2	Vessel specific info
	Bead Width	[in]	1				Bead widths of 1 to 5-in. characteristic of machine deposited cladding. Bead widths down to ½-in. can occur over welds. Nominal dimension of 1-in. selected for all analyses because this parameter is not expected to influence significantly the predicted vessel failure probabilities. May need to refine this estimate later, particularly for Oconee who reported a 5-in bead width.
	Truncation Limit	[in]	Actual clad thickness rounded to the nearest 1/100th of the total vessel wall thickness				Judgment & computational convenience
	Surface flaw depth in FAVOR	[in]	0.259	0.161	0.263	0.360	
	Buried or Surface	--	All flaws are surface breaking				Judgment. Only flaws in cladding that would influence brittle fracture of the vessel are brittle. Material properties assigned to clad flaws are that of the underlying material, be it base or weld.
	Orientation	--	All circumferential.				Observation: All flaws observed in PVRUF & Shoreham were lack of inter-run fusion defects, and cladding is always deposited circumferentially
	Density basis	--	No surface flaws observed. Density is 1/1000 that of the observed buried flaws in cladding of vessels examined by PNNL. If there is more than one clad layer then there are no clad flaws.				Judgment
	Aspect ratio basis	--	Observations on buried flaws				Judgment
	Depth basis	--	Depth of all surface flaws is the actual clad thickness rounded up to the nearest 1/100th of the total vessel wall thickness.				Judgment.

Table J-4 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Plate	Truncation Limit	[in]	0.433				Judgment. Twice the depth of the largest flaw observed in all PNNL plate inspections.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Half of the simulated flaws are circumferential, half are axial.				Observation & Physics: No observed orientation preference, and no reason to suspect one (other than laminations which are benign.
	Density basis	--	1/10 of small weld flaw density, 1/40 of large weld flaw density of the PVRUF data				Judgment. Supported by limited data.
	Aspect ratio basis	--	Same as for PVRUF welds				Judgment
	Depth basis	--	Same as for PVRUF welds				Judgment. Supported by limited data.

9. $TWCF_{95\text{-}TOTAL}$ value calculated at 500 EFPY using correlations from NUREG-1874 (Reference 9): 3.16E-07 Events per year

APPENDIX K
OCONEE UNIT 1 PROBSBFD OUTPUT

K-1: 10 Year ISI only

```
                  STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
 WESTINGHOUSE         MONTE-CARLO SIMULATION PROGRAM PROBSBFD    VERSION 1.0

=============================================================================
      INPUT VARIABLES FOR CASE  3: OC1 10 YEAR ISI ONLY

      NCYCLE =    80            NFAILS =  1001           NTRIAL =  1000
      NOVARS =    19            NUMSET =     2           NUMISI =     5
      NUMSSC =     4            NUMTRC =     4           NUMFMD =     4

    VARIABLE        DISTRIBUTION       MEDIAN      DEVIATION    SHIFT    USAGE
   NO.   NAME       TYPE      LOG       VALUE      OR FACTOR    MV/SD   NO. SUB

    1  FIFDepth     - CONSTANT -     3.0000D-02                          1  SET
    2  IFlawDen     - CONSTANT -     3.6589D-03                          2  SET
    3  ICy-ISI      - CONSTANT -     1.0000D+01                          1  ISI
    4  DCy-ISI      - CONSTANT -     8.0000D+01                          2  ISI
    5  MV-Depth     - CONSTANT -     1.5000D-02                          3  ISI
    6  SD-Depth     - CONSTANT -     1.8500D-01                          4  ISI
    7  CEff-ISI     - CONSTANT -     1.0000D+00                          5  ISI
    8  Aspect1      - CONSTANT -     2.0000D+00                          1  SSC
    9  Aspect2      - CONSTANT -     6.0000D+00                          2  SSC
   10  Aspect3      - CONSTANT -     1.0000D+01                          3  SSC
   11  Aspect4      - CONSTANT -     9.9000D+01                          4  SSC
   12  NoTr/Cy      - CONSTANT -     1.2000D+01                          1  TRC
   13  FCGThld      - CONSTANT -     1.5000D+00                          2  TRC
   14  FCGR-UC      NORMAL    NO     0.0000D+00    1.0000D+00     .00    3  TRC
   15  DKINFile     - CONSTANT -     1.0000D+00                          4  TRC
   16  Percent1     - CONSTANT -     6.7450D+01                          1  FMD
   17  Percent2     - CONSTANT -     2.0769D+01                          2  FMD
   18  Percent3     - CONSTANT -     3.9642D+00                          3  FMD
   19  Percent4     - CONSTANT -     7.8166D+00                          4  FMD

  INFORMATION GENERATED FROM FAVLOADS.DAT FILE
  AND SAVED IN DKINSAVE.DAT FILE:

  WALL THICKNESS =  8.6260 INCH

  FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

    TYPE 1 WITH AN ASPECT RATIO OF   2.

    8.62600D-02    2.26895D+00    1.06757D+01
    1.58718D-01    3.02106D+00    1.44232D+01
    4.31300D-01    1.30893D+01    2.08943D+01
    6.46950D-01    1.39096D+01    2.49826D+01
    8.62600D-01    1.44263D+01    2.80058D+01
    1.72520D+00    1.30110D+01    3.31903D+01
    2.58780D+00    7.51977D+00    3.23837D+01
    4.31300D+00   -2.67288D+00    3.20852D+01

    TYPE 2 WITH AN ASPECT RATIO OF   6.
```

K-1: 10 Year ISI only (cont.)

8.62600D-02	3.40901D+00	1.61172D+01
1.58718D-01	4.63620D+00	2.21942D+01
4.31300D-01	1.99455D+01	3.13897D+01
6.46950D-01	2.33230D+01	3.76625D+01
8.62600D-01	2.45197D+01	4.30412D+01
1.72520D+00	2.46021D+01	5.46183D+01
2.58780D+00	1.95704D+01	5.81373D+01
4.31300D+00	8.31986D+00	6.38027D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.62600D-02	3.73472D+00	1.76698D+01
1.58718D-01	4.95671D+00	2.37364D+01
4.31300D-01	2.11257D+01	3.35265D+01
6.46950D-01	2.53490D+01	4.01563D+01
8.62600D-01	2.66367D+01	4.59818D+01
1.72520D+00	2.73025D+01	5.94651D+01
2.58780D+00	2.36720D+01	6.65485D+01
4.31300D+00	1.21426D+01	7.64376D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.62600D-02	6.74437D+00	1.82354D+01
1.72520D-01	9.55233D+00	2.55450D+01
2.58780D-01	1.62039D+01	2.74271D+01
4.31300D-01	2.37153D+01	3.58624D+01
6.46950D-01	2.70360D+01	4.44287D+01
8.62600D-01	2.84566D+01	5.07281D+01
1.72520D+00	3.19293D+01	6.96665D+01
2.58780D+00	2.97815D+01	8.22041D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

12	2.2380D-04	1.0377D-05	1.4547D-06	1.1205D-05
13	6.5980D-06	4.0083D-05	7.1947D-06	1.1813D-05
14	0.0000D+00	1.2906D-05	2.8652D-06	2.3081D-06
15	0.0000D+00	3.4523D-06	1.0131D-06	4.5211D-07
16	0.0000D+00	1.1683D-06	2.9704D-07	2.7150D-07
17	0.0000D+00	5.0981D-07	1.5720D-07	1.2084D-07
18	0.0000D+00	3.1177D-07	3.5675D-08	7.1479D-08
19	0.0000D+00	1.2295D-07	5.8386D-08	0.0000D+00
20	0.0000D+00	0.0000D+00	2.2976D-08	0.0000D+00
22	0.0000D+00	5.7099D-08	0.0000D+00	0.0000D+00
24	0.0000D+00	0.0000D+00	0.0000D+00	2.2058D-08
25	0.0000D+00	5.4884D-08	1.0551D-08	0.0000D+00
28	0.0000D+00	0.0000D+00	1.0078D-08	2.1150D-08

K-2: ISI Every 10 Years

```
                 STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE         MONTE-CARLO SIMULATION PROGRAM PROBSBFD    VERSION 1.0

=============================================================================
     INPUT VARIABLES FOR CASE  2: OC1 10 YEAR INTERVAL

     NCYCLE =     80           NFAILS =  1001          NTRIAL =  1000
     NOVARS =     19           NUMSET =     2          NUMISI =     5
     NUMSSC =      4           NUMTRC =     4          NUMFMD =     4

   VARIABLE        DISTRIBUTION       MEDIAN      DEVIATION    SHIFT   USAGE
  NO.   NAME       TYPE      LOG       VALUE      OR FACTOR    MV/SD  NO. SUB

   1  FIFDepth    - CONSTANT -    3.0000D-02                           1  SET
   2  IFlawDen    - CONSTANT -    3.6589D-03                           2  SET
   3  ICy-ISI     - CONSTANT -    1.0000D+01                           1  ISI
   4  DCy-ISI     - CONSTANT -    1.0000D+01                           2  ISI
   5  MV-Depth    - CONSTANT -    1.5000D-02                           3  ISI
   6  SD-Depth    - CONSTANT -    1.8500D-01                           4  ISI
   7  CEff-ISI    - CONSTANT -    1.0000D+00                           5  ISI
   8  Aspect1     - CONSTANT -    2.0000D+00                           1  SSC
   9  Aspect2     - CONSTANT -    6.0000D+00                           2  SSC
  10  Aspect3     - CONSTANT -    1.0000D+01                           3  SSC
  11  Aspect4     - CONSTANT -    9.9000D+01                           4  SSC
  12  NoTr/Cy     - CONSTANT -    1.2000D+01                           1  TRC
  13  FCGThld     - CONSTANT -    1.5000D+00                           2  TRC
  14  FCGR-UC     NORMAL    NO    0.0000D+00   1.0000D+00      .00     3  TRC
  15  DKINFile    - CONSTANT -    1.0000D+00                           4  TRC
  16  Percent1    - CONSTANT -    6.7450D+01                           1  FMD
  17  Percent2    - CONSTANT -    2.0769D+01                           2  FMD
  18  Percent3    - CONSTANT -    3.9642D+00                           3  FMD
  19  Percent4    - CONSTANT -    7.8166D+00                           4  FMD

 INFORMATION GENERATED FROM FAVLOADS.DAT FILE
 AND SAVED IN DKINSAVE.DAT FILE:

 WALL THICKNESS =  8.6260 INCH

 FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

   TYPE 1 WITH AN ASPECT RATIO OF   2.

   8.62600D-02    2.26895D+00    1.06757D+01
   1.58718D-01    3.02106D+00    1.44232D+01
   4.31300D-01    1.30893D+01    2.08943D+01
   6.46950D-01    1.39096D+01    2.49826D+01
   8.62600D-01    1.44263D+01    2.80058D+01
   1.72520D+00    1.30110D+01    3.31903D+01
   2.58780D+00    7.51977D+00    3.23837D+01
   4.31300D+00   -2.67288D+00    3.20852D+01

   TYPE 2 WITH AN ASPECT RATIO OF   6.
```

K-2: ISI Every 10 Years (cont.)

8.62600D-02	3.40901D+00	1.61172D+01
1.58718D-01	4.63620D+00	2.21942D+01
4.31300D-01	1.99455D+01	3.13897D+01
6.46950D-01	2.33230D+01	3.76625D+01
8.62600D-01	2.45197D+01	4.30412D+01
1.72520D+00	2.46021D+01	5.46183D+01
2.58780D+00	1.95704D+01	5.81373D+01
4.31300D+00	8.31986D+00	6.38027D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.62600D-02	3.73472D+00	1.76698D+01
1.58718D-01	4.95671D+00	2.37364D+01
4.31300D-01	2.11257D+01	3.35265D+01
6.46950D-01	2.53490D+01	4.01563D+01
8.62600D-01	2.66367D+01	4.59818D+01
1.72520D+00	2.73025D+01	5.94651D+01
2.58780D+00	2.36720D+01	6.65485D+01
4.31300D+00	1.21426D+01	7.64376D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.62600D-02	6.74437D+00	1.82354D+01
1.72520D-01	9.55233D+00	2.55450D+01
2.58780D-01	1.62039D+01	2.74271D+01
4.31300D-01	2.37153D+01	3.58624D+01
6.46950D-01	2.70360D+01	4.44287D+01
8.62600D-01	2.84566D+01	5.07281D+01
1.72520D+00	3.19293D+01	6.96665D+01
2.58780D+00	2.97815D+01	8.22041D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

12	1.3580D-10	5.4482D-12	7.5613D-13	6.1767D-12
13	2.8117D-12	1.4377D-11	2.5387D-12	4.4630D-12
14	0.0000D+00	2.2869D-12	5.0820D-13	4.3208D-13
15	0.0000D+00	2.9908D-13	8.6948D-14	4.2493D-14
16	0.0000D+00	4.7816D-14	1.1866D-14	1.3716D-14
17	0.0000D+00	1.0793D-14	2.7598D-15	2.7273D-15
18	0.0000D+00	2.8658D-15	3.3064D-16	8.9749D-16
19	0.0000D+00	6.3484D-16	2.5927D-16	0.0000D+00
20	0.0000D+00	0.0000D+00	5.0956D-17	0.0000D+00
22	0.0000D+00	1.1431D-17	0.0000D+00	0.0000D+00
24	0.0000D+00	0.0000D+00	0.0000D+00	5.0464D-18
25	0.0000D+00	1.4911D-18	3.6983D-19	0.0000D+00
28	0.0000D+00	0.0000D+00	2.2911D-20	2.7483D-19

APPENDIX L
OCONEE UNIT 1 PTS TRANSIENTS

Table L-1 PTS Transient Descriptions for OC1

Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Hi K	Dominant*
1	8	2.54 cm [1 in] surge line break with 1 stuck open safety valve in SG-A.	None	9.68E-08	No	No	No
2	12	2.54 cm [1 in] surge line break with 1 stuck open safety valve in SG-A.	HPI throttled to maintain 27.8 K [50° F] subcooling margin	9.24E-07	No	No	No
3	15	2.54 cm [1 in] surge line break with HPI Failure	At 15 minutes after transient initiation, operator opens all TBVs to lower primary system pressure and allow CFT and LPI injection.	3.39E-08	No	No	No
4	27	MSLB without trip of turbine driven emergency feedwater.	Operator throttles HPI to maintain 27.8 K [50° F] subcooling margin.	2.13E-06	No	No	No
5	28	Reactor/turbine trip with 1 stuck open safety valve in SG-A	None	7.53E-08	No	No	No
6	29	Reactor/turbine trip with 1 stuck open safety valve in SG-A and a second stuck open safety valve in SG-B	None	3.09E-07	No	No	No
7	30	Reactor/turtine trip with 1 stuck open safety valve in SG-A	None	1.46E-07	Yes	No	No
8	31	Reactor/turbine trip with 1 stuck open safety valve in SG-A and a second stuck open safety valve in SG-B	None	8.36E-09	Yes	No	No
9	36	Reactor/turbine trip with 1 stuck open safety valve in SG-A and a second stuck open safety valve in SG-B	Operator throttles HPI to maintain 27.8 K [50° F] subcooling and 304.8 cm [120 in] pressurizer level.	1.40E-05	No	No	No
10	37	Reactor/turbine trip with 1 stuck open safety valve in SG-A	Operator throttles HPI to maintain 27.8 K [50° F] subcooling and 304.8 cm [120 in] pressurizer level.	1.41E-06	Yes	No	No
11	38	Reactor/turbine trip with 1 stuck open safety valve in SG-A and a second stuck open safety valve in SG-B	Operator throttles HPI to maintain 27.8 K [50° F] subcooling and 304.8 cm [120 in] pressurizer level.	2.65E-06	Yes	No	No

Table L-1 PTS Transient Descriptions for OC1							
Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Hi K	Dominant*
12	44	2.54 cm [1 in] surge line break with HPI Failure	At 15 minutes after initiation, operators open all TBVs to depressurize the system to the CFT setpoint. When the CFTs are 50 percent discharged, HPI is assumed to be recovered. The TBVs are assumed remain open for the duration of the transient.	2.69E-07	No	No	No
13	89	Reactor/turbine trip with Loss of MFW and EFW.	Operator opens all TBVs to depressurize the secondary side to below the condensate booster pump shutoff head so that these pumps feed the steam generators. Booster pumps are assumed to be initially uncontrolled so that the steam generators are overfilled (609 cm [240 in] startup level). Operator controls booster pump flow to maintain SG level at 76 cm [30 in] due to continued RCP operation. Operator also throttles HPI to maintain 55 K [100EF] subcooling and a pressurizer level of 254 cm [100 in]. The TBVs are kept fully opened due to operator error.	5.38E-07	No	No	No
14	90	Reactor/turbine trip with 2 stuck open safety valves in SG-A	Operator throttles HPI 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached [throttling criteria is 27.8 K [50°F] subcooling].	6.29E-07	No	No	No

Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Hi K	Dominant*
15	98	Reactor/turbine trip with loss of MFW and EFW	Operator opens all TBVs to depressurize the secondary side to below the condensate booster pump shutoff head so that these pumps feed the steam generators. Booster pumps are assumed to be initially uncontrolled so that the steam generators are overfilled (610 cm [240 in] startup level). Operator controls booster pump flow to maintain SG level at 76 cm [30 in] due to continued RCP operation. Operator also throttles HPI to maintain 55 K [100EF] subcooling and a pressurizer level of 254 cm [100 in]. The TBVs are kept fully opened due to operator error.	9.96E-08	Yes	No	No
16	99	MSLB with trip of turbine driven EFW by MSLB Circuitry	HPI is throttled 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	2.44E-07	No	No	No
17	100	MSLB with trip of turbine driven EFW by MSLB Circuitry	Operator throttles HPI 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	5.11E-08	Yes	No	No
18	101	MSLB without trip of turbine driven EFW by MSLB Circuitry	Operator throttles HPI to maintain 27.8 K [50° F] subcooling margin (throttling criteria is 27.8 K [50°F] subcooling).	3.86E-07	Yes	No	No
19	102	Reactor/turbine trip with 2 stuck open safety valves in SG-A	Operator throttles HPI 20 minutes after 2.77 K [5°F] subcooling and 254 cm [100 in] pressurizer level is reached (throttling criteria is 27 K [50°F] subcooling).	2.03E-07	Yes	No	No

Table L-1 PTS Transient Descriptions for OC1

Table L-1 PTS Transient Descriptions for OC1

Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Hi K	Dominant*
20	109	Stuck open pressurizer safety valve. Valve recloses at 6000 secs [RCS low pressure point].	None	9.58E-06	No	Yes	No
21	110	5.08 cm [2 inch] surge line break with HPI failure	At 15 minutes after transient initiation, operator opens both TBV to lower primary system pressure and allow CFT and LPI injection.	3.42E-06	No	Yes	Yes at 1000 EFPY
22	111	2.54 cm [1 in] surge line break with HPI failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. At 3000 seconds after initiation, operator starts throttling HPI to 55 K [100°F] subcooling and 254 cm [100"] pressurizer level.	4.16E-07	No	Yes	No
23	112	Stuck open pressurizer safety valve. Valve recloses at 6000 secs.	After valve recloses, operator throttles HPI 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27 K [50°F] subcooling)	1.25E-04	No	Yes	No
24	113	Stuck open pressurizer safety valve. Valve recloses at 6000 secs.	After valve recloses, operator throttles HPI 10 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling)	5.07E-05	No	Yes	No
25	114	Stuck open pressurizer safety valve. Valve recloses at 3000 secs.	After valve recloses, operator throttles HPI 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 50°F subcooling)	1.25E-04	No	Yes	No
26	115	Stuck open pressurizer Safety Valve. Valve recloses at 3000 secs.	After valve recloses, operator throttles HPI 10 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 50°F subcooling)	5.07E-05	No	Yes	No

Table L-1 PTS Transient Descriptions for OC1

Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Hi K	Dominant*
27	116	Stuck open pressurizer safety valve and HPI failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. The HPI is throttled 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 50°F subcooling).	2.60E-07	No	Yes	No
28	117	Stuck open pressurizer safety valve and HPI failure	At 15 minutes after initiation, operator opens all TBV to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. The SRV is closed 5 minutes after HPI recovered. HPI is throttled at 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	5.38E-07	No	Yes	No
29	119	2.54 cm [1 in] surge line break with HPI Failure	At 15 minutes after transient initiation, the operator opens all turbine bypass valves to lower primary system pressure and allow core flood tank and LPI injection.	4.41E-07	Yes	Yes	No
30	120	2.54 cm [1 in] surge line break with HPI Failure	At 15 minutes after sequence initiation, operators open all TBVs to depressurize the system to the CFT setpoint. When the CFTs are 50 percent discharged, HPI is assumed to be recovered. The TBVs are assumed remain opened for the duration of the transient.	4.22E-08	Yes	Yes	No
31	121	Stuck open pressurizer safety valve. Valve recloses at 6000 secs .	Operator throttles HPI at 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached [throttling criteria is 27.8 K [50°F] subcooling].	2.28E-05	Yes	Yes	No

Table L-1 PTS Transient Descriptions for OC1							
Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Hi K	Dominant*
32	122	Stuck open pressurizer safety valve. Valve recloses at 6000 secs.	Operator throttles HPI at 10 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	7.57E-06	Yes	Yes	Yes at 32, 60, 500, 1000 EFPY
33	123	Stuck open pressurizer safety valve. Valve recloses at 3000 secs.	Operator throttles HPI at 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	2.28E-05	Yes	Yes	No
34	124	Stuck open pressurizer safety valve. Valve recloses at 3000 secs.	Operator throttles HPI at 10 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	7.57E-06	Yes	Yes	Yes at 60, 500, 1000 EFPY
35	125	Stuck open pressurizer safety valve and HPI Failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. HPI is throttled 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	4.61E-08	Yes	Yes	No
36	126	Stuck open pressurizer safety valve and HPI Failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. SRV is closed at 5 minutes after HPI is recovered. HPI is throttled at 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	8.41E-08	Yes	Yes	No

Table L-1 PTS Transient Descriptions for OC1

Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Hi K	Dominant*
37	127	SGTR with a stuck open SRV in SG-B. A reactor trip is assumed to occur at the time of the tube rupture. Stuck safety relief valve is assumed to reclose 10 minutes after initiation.	Operator trips RCP's 1 minute after initiation. Operator also throttles HPI 10 minutes after 2.77 K [5° F] subcooling and 254 cm [100 in] pressurizer level is reached (assumed throttling criteria is 27 K [50°F] subcooling).	1.25E-07	Yes	Yes	No
38	141	8.19 cm [3.22 in] surge line break [Break flow area increased by 30% from 7.18 cm [2.828 in] break].	None	1.06E-04	No	Yes	Yes at 500, 1000 EFPY
39	142	6.01 cm [2.37 in] surge line break [Break flow area decreased by 30% from 7.18 cm [2.828 in] break].	None	1.06E-04	No	Yes	No
40	145	4.34 cm [1.71 in] surge line break [Break flow area increased by 30% from 3.81 cm [1.5 in] break]. Winter conditions assumed [HPI, LPI temp = 277 K [40° F] and CFT temp = 294 K [70° F]].	None	1.34E-04	No	Yes	No
41	146	TT/RT with stuck open pzr SRV [valve flow area reduced by 30 percent]. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]]. Vent valves do not function.	None	4.23E-05	No	Yes	No
42	147	TT/RT with stuck open pzr SRV. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]].	None	3.63E-05	No	Yes	No
43	148	TT/RT with partially stuck open pzr SRV [flow area equivalent to 1.5 in diameter opening]. HTC coefficients increased by 1.3.	None	4.23E-05	No	Yes	No
44	149	TT/RT with stuck open pzr SRV. SRV assumed to reclose at 3000 secs. Operator does not throttle HPI.	None	9.58E-06	No	Yes	No

Table L-1 PTS Transient Descriptions for OC1							
Count	**TH Case #**	**System Failure**	**Operator Action**	**Mean IE Frequency**	**HZP**	**Hi K**	**Dominant***
45	154	8.53 cm [3.36 in] surge line break [Break flow area reduced by 30% from 10.16 cm [4 in] break]. Vent valves do not function. ECC suction switch to the containment sump included in the analysis.	None	1.34E-04	No	Yes	No
46	156	40.64 cm [16 in] hot leg break. ECC suction switch to the containment sump included in the analysis.	None	7.03E-06	No	Yes	Yes at 500, 1000 EFPY
47	160	14.37 cm [5.656 in] surge line break. ECC suction switch to the containment sump included in the analysis.	None	1.82E-05	No	Yes	Yes at 500, 1000 EFPY
48	164	20.32 cm [8 inch] surge line break. ECC suction switch to the containment sump included in the analysis.	None	2.12E-05	No	Yes	Yes at 60, 500, 1000 EFPY
49	165	Stuck open pressurizer safety valve. Valve recloses at 6000 secs [RCS low pressure point].	None	1.76E-06	Yes	Yes	Yes at 32, 60, 500, 1000 EFPY
50	168	TT/RT with stuck open pzr SRV. SRV assumed to reclose at 3000 secs. Operator does not throttle HPI.	None	1.76E-06	Yes	Yes	Yes at 500, 1000 EFPY
51	169	TT/RT with stuck open pzr SRV [valve flow area reduced by 30 percent]. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]]. Vent valves do not function.	None	7.33E-06	Yes	Yes	No
52	170	TT/RT with stuck open pzr SRV. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]].	None	6.28E-06	Yes	Yes	No
53	171	TT/RT with partially stuck open pzr SRV [flow area equivalent to 1.5 in diameter opening]. HTC coefficients increased by 1.3.	None	7.33E-06	Yes	Yes	No

Table L-1 PTS Transient Descriptions for OC1

Count	TH Case #	System Failure	Operator Action	Mean IE Frequency	HZP	Hi K	Dominant*
54	172	10.16 cm [4 in] cold leg break. ECC suction switch to the containment sump included in the analysis.	None	1.06E-04	No	Yes	Yes at 1000 EFPY
55	178	8.53 cm [3.36 in] surge line break [Break flow area reduced by 30% from 10.16 cm [4 in] break]. Vent valves do not function. ECC suction switch to the containment sump included in the analysis.	None	2.12E-05	No	Yes	No

Notes:
1. TH – Thermal hydraulics
2. LOCA – Loss-of-coolant accident
3. SBLOCA – Small-break loss-of-coolant accident
4. MBLOCA – Medium-break loss-of-coolant accident
5. LBLOCA – Large-break loss-of-coolant accident
6. HZP – Hot-zero power
7. SRV – Safety and relief valve
8. MSLB – Main steam line break
9. AFW – Auxiliary feedwater
10. HPI – High-pressure injection
11. RCPs – Reactor coolant pumps

* The arbitrary definition of a dominant transient is a transient that contributes 1% or more of the total Through-Wall Cracking Failure (TWCF).

APPENDIX M
OCONEE UNIT 1 FAVPOST OUTPUT

M-1: 10 Year ISI only

```
************************************************
*                                              *
*               WELCOME TO FAVOR               *
*                                              *
*   FRACTURE ANALYSIS OF VESSELS: OAK RIDGE    *
*                 VERSION 06.1                 *
*                                              *
*     FAVPOST MODULE: POSTPROCESSOR MODULE     *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES  *
*        WITH RESULTS OF PFM ANALYSIS          *
*                                              *
*     PROBLEMS OR QUESTIONS REGARDING FAVOR    *
*            SHOULD BE DIRECTED TO             *
*                                              *
*               TERRY DICKSON                  *
*        OAK RIDGE NATIONAL LABORATORY         *
*                                              *
*          e-mail: dicksontl@ornl.gov          *
*                                              *
************************************************

***********************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
* Neither the United States, nor the United States        *
* Department of Energy, nor the United States Nuclear     *
* Regulatory Commission, nor any of their employees,      *
* nor any of their contractors, subcontractors, or their  *
* employees, makes any warranty, expressed or implied, or *
* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
***********************************************************

     DATE: 27-Jun-2007  TIME: 10:37:21


Begin echo of FAVPost input data deck          10:37:21   27-Jun-2007
End echo of FAVPost input data deck          10:37:21   27-Jun-2007


        FAVPOST INPUT  FILE NAME                        = postoc55.in
        FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
        FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
        FAVPOST OUTPUT FILE NAME                        = ocpost10yronly.out
```

M-1: 10 Year ISI only (cont.)

```
**********************************
* NUMBER OF SIMULATIONS =   42000 *
**********************************
```

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
8	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
12	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
15	1.3573E-07	0.0000E+00	0.0000E+00	1.8219E-12	0.0000E+00	0.0000E+00	0.0000
27	3.5321E-06	0.0000E+00	9.7209E-06	1.0415E-08	0.0000E+00	0.0000E+00	0.0029
28	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
29	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
30	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
31	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
36	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
37	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
38	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
44	4.1737E-06	0.0000E+00	4.4260E-06	2.8306E-06	0.0000E+00	1.7178E-06	0.6782
89	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
90	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
98	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
99	5.9408E-07	0.0000E+00	0.0000E+00	1.4020E-08	0.0000E+00	0.0000E+00	0.0236
100	1.7278E-06	0.0000E+00	3.6037E-07	4.0017E-07	0.0000E+00	0.0000E+00	0.2316
101	1.4057E-05	0.0000E+00	5.7634E-05	1.4131E-08	0.0000E+00	0.0000E+00	0.0010
102	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
109	1.2464E-07	0.0000E+00	0.0000E+00	4.0517E-08	0.0000E+00	0.0000E+00	0.3251
110	2.5705E-03	5.9843E-03	3.2961E-02	1.7099E-05	1.1551E-04	2.3858E-04	0.0067
111	4.5456E-08	0.0000E+00	0.0000E+00	1.4016E-11	0.0000E+00	0.0000E+00	0.0003
112	8.3691E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
113	3.1674E-08	0.0000E+00	0.0000E+00	2.9159E-08	0.0000E+00	0.0000E+00	0.9206
114	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000

M-1: 10 Year ISI only (cont.)

115	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
116	2.6284E-06	0.0000E+00	3.2436E-07	1.1695E-09	0.0000E+00	0.0000E+00	0.0004
117	5.3511E-05	5.7899E-04	1.0032E-03	6.4161E-08	0.0000E+00	1.1859E-13	0.0012
119	9.2288E-05	5.7141E-04	1.4015E-03	6.2724E-07	1.2359E-05	3.9221E-06	0.0068
120	3.3053E-05	4.2682E-04	4.6996E-04	2.4353E-05	4.0226E-04	3.3238E-04	0.7368
121	7.5802E-08	0.0000E+00	0.0000E+00	2.9336E-12	0.0000E+00	0.0000E+00	0.0000
122	4.2840E-04	2.6185E-03	4.5275E-03	4.2818E-04	2.6185E-03	4.5275E-03	0.9995
123	7.5802E-08	0.0000E+00	0.0000E+00	2.9336E-12	0.0000E+00	0.0000E+00	0.0000
124	1.6752E-04	1.1252E-03	2.2503E-03	1.6517E-04	1.1140E-03	2.2106E-03	0.9860
125	7.0273E-05	5.4000E-04	9.7650E-04	2.7388E-07	0.0000E+00	7.8181E-07	0.0039
126	1.1462E-06	0.0000E+00	6.9098E-08	8.8915E-10	0.0000E+00	0.0000E+00	0.0008
127	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
141	1.3088E-04	6.5253E-04	2.1093E-03	2.4722E-06	5.4576E-05	2.1956E-05	0.0189
142	4.5999E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
145	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
146	2.7129E-06	0.0000E+00	6.0255E-07	9.5223E-08	0.0000E+00	0.0000E+00	0.0351
147	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
148	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
149	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
154	1.4610E-04	9.2152E-04	2.2196E-03	5.0292E-07	0.0000E+00	0.0000E+00	0.0034
156	2.2010E-02	5.2143E-02	1.7530E-01	8.9468E-05	3.2180E-04	1.4740E-03	0.0041
160	1.3270E-02	2.8340E-02	1.0630E-01	1.7457E-04	4.0235E-04	2.9084E-03	0.0132
164	1.3995E-02	3.2240E-02	1.2584E-01	8.8292E-05	2.5946E-04	1.5275E-03	0.0063
165	3.9220E-04	2.6738E-03	3.0921E-03	3.9180E-04	2.6738E-03	3.0921E-03	0.9990
168	2.0391E-04	1.2392E-03	2.8916E-03	2.0139E-04	1.2392E-03	2.8344E-03	0.9876
169	2.1163E-04	1.0072E-03	3.3572E-03	6.7665E-06	1.0975E-04	8.2906E-05	0.0320
170	4.6761E-08	0.0000E+00	0.0000E+00	4.6483E-13	0.0000E+00	0.0000E+00	0.0000
171	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
172	8.0788E-05	5.5702E-04	1.1954E-03	3.2570E-07	0.0000E+00	1.2316E-06	0.0040
178	1.4610E-04	9.2152E-04	2.2196E-03	5.0292E-07	0.0000E+00	0.0000E+00	0.0034

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

M-1: 10 Year ISI only (cont.)

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*********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)    *
*         FOR THE FREQUENCY OF CRACK INITIATION         *
*********************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	2.9119	2.9119
2.1448E-06	92.0857	94.9976
6.4345E-06	2.9429	97.9405
1.0724E-05	0.8714	98.8119
1.5014E-05	0.4571	99.2690
1.9304E-05	0.2405	99.5095
2.3593E-05	0.1524	99.6619
2.7883E-05	0.0905	99.7524
3.2173E-05	0.0619	99.8143
3.6462E-05	0.0357	99.8500
4.0752E-05	0.0214	99.8714
4.5042E-05	0.0214	99.8929
4.9331E-05	0.0214	99.9143
5.3621E-05	0.0167	99.9310
5.7911E-05	0.0095	99.9405
6.2200E-05	0.0048	99.9452
6.6490E-05	0.0071	99.9524
7.0780E-05	0.0095	99.9619
7.5069E-05	0.0071	99.9690
7.9359E-05	0.0048	99.9738
8.7938E-05	0.0024	99.9762
9.6518E-05	0.0048	99.9810
1.0081E-04	0.0024	99.9833
1.0510E-04	0.0048	99.9881
1.1368E-04	0.0024	99.9905
1.3512E-04	0.0024	99.9929
1.3941E-04	0.0024	99.9952
1.6944E-04	0.0024	99.9976
2.0376E-04	0.0024	100.0000

```
=============================================
==      Summary Descriptive Statistics     ==
=============================================
```

Minimum	= 0.0000E+00
Maximum	= 2.0449E-04
Range	= 2.0449E-04
Number of Simulations	= 42000
5th Percentile	= 3.8158E-12
Median	= 1.1260E-07
95.0th Percentile	= 2.1448E-06

M-1: 10 Year ISI only (cont.)

```
99.0th Percentile                  = 1.2489E-05
99.9th Percentile                  = 4.6472E-05

Mean                               = 9.9210E-07
Standard Deviation                 = 3.7931E-06
Standard Error                     = 1.8508E-08
Variance (unbiased)                = 1.4387E-11
Variance (biased)                  = 1.4387E-11
Moment Coeff. of Skewness          = 1.6997E+01
Pearson's 2nd Coeff. of Skewness   = 7.8467E-01
Kurtosis                           = 5.3092E+02
```

```
***********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)      *
***********************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	23.0000	23.0000
3.5468E-07	76.3452	99.3452
1.0641E-06	0.3333	99.6786
1.7734E-06	0.1357	99.8143
2.4828E-06	0.0429	99.8571
3.1922E-06	0.0405	99.8976
3.9015E-06	0.0119	99.9095
4.6109E-06	0.0143	99.9238
5.3203E-06	0.0119	99.9357
6.0296E-06	0.0095	99.9452
6.7390E-06	0.0143	99.9595
7.4484E-06	0.0071	99.9667
8.1577E-06	0.0048	99.9714
8.8671E-06	0.0048	99.9762
1.0286E-05	0.0048	99.9810
1.0995E-05	0.0024	99.9833
1.1705E-05	0.0048	99.9881
1.3833E-05	0.0024	99.9905
1.4542E-05	0.0024	99.9929
2.6601E-05	0.0024	99.9952
5.4267E-05	0.0024	99.9976
7.0582E-05	0.0024	100.0000

```
============================================
==     Summary Descriptive Statistics     ==
============================================

Minimum                            = 0.0000E+00
```

M-1: 10 Year ISI only (cont.)

```
Maximum                              = 7.0227E-05
Range                                = 7.0227E-05

Number of Simulations                =  42000

5th Percentile                       = 0.0000E+00
Median                               = 9.3729E-11
95.0th Percentile                    = 3.5468E-07
99.0th Percentile                    = 4.7205E-07
99.9th Percentile                    = 3.3340E-06

Mean                                 = 3.1118E-08
Standard Deviation                   = 5.2306E-07
Standard Error                       = 2.5523E-09
Variance (unbiased)                  = 2.7359E-13
Variance (biased)                    = 2.7358E-13
Moment Coeff. of Skewness            = 9.0777E+01
Pearson's 2nd Coeff. of Skewness     =-4.6734E-01
Kurtosis                             = 1.0697E+04
```

```
***********************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION     *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES      *
***********************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
8	0.00	0.00
12	0.00	0.00
15	0.00	0.00
27	0.00	0.00
28	0.00	0.00
29	0.00	0.00
30	0.00	0.00
31	0.00	0.00
36	0.00	0.00
37	0.00	0.00
38	0.00	0.00
44	0.00	0.00
89	0.00	0.00
90	0.00	0.00
98	0.00	0.00
99	0.00	0.00
100	0.00	0.00
101	0.00	0.00
102	0.00	0.00
109	0.00	0.00
110	1.11	0.22
111	0.00	0.00
112	0.00	0.00

M-1: 10 Year ISI only (cont.)

113	0.00	0.02
114	0.00	0.00
115	0.00	0.00
116	0.00	0.00
117	0.00	0.00
119	0.00	0.00
120	0.00	0.00
121	0.00	0.00
122	1.38	44.11
123	0.00	0.00
124	0.52	16.27
125	0.00	0.00
126	0.00	0.00
127	0.00	0.00
141	1.58	0.91
142	0.00	0.00
145	0.00	0.00
146	0.04	0.04
147	0.00	0.00
148	0.00	0.00
149	0.00	0.00
154	2.36	0.34
156	21.78	2.62
160	31.29	12.37
164	37.57	7.32
165	0.30	9.70
168	0.17	5.29
169	0.58	0.60
170	0.00	0.00
171	0.00	0.00
172	0.93	0.12
178	0.37	0.05
TOTALS	100.00	100.00

```
**********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION  *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -   *
*                           BY                           *
*               RPV BELTLINE MAJOR REGION                *
*                  BY PARENT SUBREGION                   *
*                                                        *
*       WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT     *
*          TO FREQUENCY OF CRACK INITIATION AND          *
*        THROUGH-WALL CRACKING FREQUENCY (FAILURE)       *
**********************************************************
```

M-1: 10 Year ISI only (cont.)

MAJOR REGION	RTndt (MAX)	% of total flaws	% of total frequency of crack initiation	% of total through-wall crack frequency cleavage	ductile	total
1	223.80	3.03	2.10	18.32	2.77	21.09
2	220.74	3.54	4.01	36.95	5.25	42.21
3	253.04	1.43	3.54	26.25	10.23	36.48
4	236.56	13.81	14.70	0.06	0.00	0.06
5	277.07	13.81	75.50	0.14	0.00	0.14
6	168.76	13.81	0.09	0.00	0.00	0.00
7	158.07	18.93	0.02	0.01	0.00	0.01
8	152.82	11.05	0.01	0.00	0.00	0.00
9	154.58	11.05	0.02	0.01	0.00	0.02
10	155.89	8.92	0.01	0.00	0.00	0.00
11	101.31	0.62	0.00	0.00	0.00	0.00
	TOTALS	100.00	100.00	81.74	18.26	100.00

```
***********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION   *
*     AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -     *
*                           BY                            *
*                RPV BELTLINE MAJOR REGION                *
*                    BY CHILD SUBREGION                   *
*                                                         *
*      WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT       *
*          TO FREQUENCY OF CRACK INITIATION AND           *
*        THROUGH-WALL CRACKING FREQUENCY (FAILURE)        *
***********************************************************
```

MAJOR REGION	RTndt (MAX)	% of total flaws	% of total frequency of crack initiation	% of total through-wall crack frequency cleavage	ductile	total
1	223.80	3.03	2.10	18.32	2.77	21.09
2	220.74	3.54	4.01	36.95	5.25	42.21
3	253.04	1.43	3.54	26.25	10.23	36.48
4	236.56	13.81	14.70	0.06	0.00	0.06
5	277.07	13.81	75.50	0.14	0.00	0.14
6	168.76	13.81	0.09	0.00	0.00	0.00
7	158.07	18.93	0.02	0.01	0.00	0.01
8	152.82	11.05	0.01	0.00	0.00	0.00
9	154.58	11.05	0.02	0.01	0.00	0.02
10	155.89	8.92	0.01	0.00	0.00	0.00
11	101.31	0.62	0.00	0.00	0.00	0.00
	TOTALS	100.00	100.00	81.74	18.26	100.00

M-1: 10 Year ISI only (cont.)

```
***********************************************************
*    FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION    *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -      *
*         MATERIAL, FLAW CATEGORY, AND FLAW DEPTH           *
*                                                           *
*       WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT        *
*          TO FREQUENCY OF CRACK INITIATION AND             *
*        THROUGH-WALL CRACKING FREQUENCY (FAILURE)          *
***********************************************************
```

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*****************
* WELD MATERIAL *
*****************
```

FLAW DEPTH (in)	% of total frequency of crack initiation CAT I flaws	CAT 2 flaws	CAT 3 flaws	% of total through-wall crack frequency CAT 1 flaws	CAT 2 flaws	CAT 3 flaws
0.086	0.00	3.46	0.00	0.00	0.89	0.00
0.173	0.00	43.13	0.00	0.00	12.86	0.00
0.259	0.00	15.98	0.00	0.00	6.44	0.00
0.345	0.00	9.47	0.00	0.00	5.39	0.05
0.431	0.00	7.02	0.01	0.00	6.88	0.17
0.518	0.00	4.91	0.01	0.00	7.10	0.31
0.604	0.00	3.69	0.02	0.00	4.99	0.50
0.690	0.00	2.24	0.02	0.00	5.04	0.48
0.776	0.00	1.85	0.01	0.00	4.00	0.45
0.863	0.00	1.97	0.02	0.00	5.08	0.52
0.949	0.00	0.88	0.02	0.00	4.28	0.65
1.035	0.03	1.03	0.01	0.00	4.67	0.20
1.121	0.04	0.70	0.00	0.00	2.85	0.10
1.208	0.01	0.61	0.01	0.00	2.08	0.21
1.294	0.00	0.66	0.01	0.00	3.93	0.23
1.380	0.00	0.86	0.01	0.00	3.34	0.30
1.466	0.00	0.20	0.01	0.00	3.29	0.19
1.553	0.00	0.25	0.00	0.00	2.25	0.13
1.639	0.00	0.24	0.01	0.00	0.81	0.20
1.725	0.00	0.12	0.00	0.00	1.45	0.03
1.811	0.00	0.38	0.00	0.00	5.61	0.06
1.898	0.00	0.05	0.00	0.00	1.62	0.02
1.984	0.00	0.04	0.00	0.00	0.25	0.02
2.070	0.00	0.00	0.00	0.00	0.00	0.00
2.157	0.00	0.00	0.00	0.00	0.00	0.00
2.243	0.00	0.00	0.00	0.00	0.00	0.00
2.329	0.00	0.00	0.00	0.00	0.00	0.00
2.415	0.00	0.00	0.00	0.00	0.00	0.00
TOTALS	0.08	99.72	0.15	0.00	95.13	4.85

M-1: 10 Year ISI only (cont.)

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******************
* PLATE MATERIAL *
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FLAW DEPTH (in)	% of total frequency of crack initiation CAT I flaws	CAT 2 flaws	CAT 3 flaws	% of total through-wall crack frequency CAT 1 flaws	CAT 2 flaws	CAT 3 flaws
0.086	0.00	0.00	0.00	0.00	0.00	0.00
0.173	0.00	0.00	0.00	0.00	0.01	0.00
0.259	0.00	0.00	0.00	0.00	0.01	0.00
0.345	0.00	0.00	0.00	0.00	0.00	0.00
0.431	0.00	0.00	0.00	0.00	0.01	0.00
0.518	0.00	0.00	0.00	0.00	0.00	0.00
0.604	0.00	0.00	0.00	0.00	0.00	0.00
0.690	0.00	0.00	0.00	0.00	0.00	0.00
0.776	0.00	0.00	0.00	0.00	0.00	0.00
0.863	0.00	0.00	0.00	0.00	0.00	0.00
0.949	0.00	0.00	0.00	0.00	0.00	0.00
1.035	0.02	0.00	0.00	0.00	0.00	0.00
1.121	0.02	0.00	0.00	0.00	0.00	0.00
1.208	0.00	0.00	0.00	0.00	0.00	0.00
1.294	0.00	0.00	0.00	0.00	0.00	0.00
1.380	0.00	0.00	0.00	0.00	0.00	0.00
1.466	0.00	0.00	0.00	0.00	0.00	0.00
1.553	0.00	0.00	0.00	0.00	0.00	0.00
1.639	0.00	0.00	0.00	0.00	0.00	0.00
1.725	0.00	0.00	0.00	0.00	0.00	0.00
1.811	0.00	0.00	0.00	0.00	0.00	0.00
1.898	0.00	0.00	0.00	0.00	0.00	0.00
1.984	0.00	0.00	0.00	0.00	0.00	0.00
2.070	0.00	0.00	0.00	0.00	0.00	0.00
2.157	0.00	0.00	0.00	0.00	0.00	0.00
2.243	0.00	0.00	0.00	0.00	0.00	0.00
2.329	0.00	0.00	0.00	0.00	0.00	0.00
2.415	0.00	0.00	0.00	0.00	0.00	0.00
TOTALS	0.05	0.00	0.00	0.00	0.03	0.00

DATE: 27-Jun-2007 TIME: 10:38:00

M-2: ISI Every 10 Years

```
***********************************************
*                                             *
*              WELCOME TO FAVOR               *
*                                             *
*   FRACTURE ANALYSIS OF VESSELS: OAK RIDGE   *
*               VERSION 06.1                  *
*                                             *
*     FAVPOST MODULE: POSTPROCESSOR MODULE    *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES *
*        WITH RESULTS OF PFM ANALYSIS         *
*                                             *
*     PROBLEMS OR QUESTIONS REGARDING FAVOR   *
*             SHOULD BE DIRECTED TO           *
*                                             *
*                TERRY DICKSON                *
*        OAK RIDGE NATIONAL LABORATORY        *
*                                             *
*          e-mail: dicksontl@ornl.gov         *
*                                             *
***********************************************

**********************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
* Neither the United States, nor the United States        *
* Department of Energy, nor the United States Nuclear     *
* Regulatory Commission, nor any of their employees,      *
* nor any of their contractors, subcontractors, or their  *
* employees, makes any warranty, expressed or implied, or *
* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
**********************************************************

     DATE: 27-Jun-2007  TIME: 10:36:09


 Begin echo of FAVPost input data deck        10:36:09   27-Jun-2007
 End echo of FAVPost input data deck        10:36:09   27-Jun-2007


        FAVPOST INPUT  FILE NAME                    = postoc55.in
        FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
        FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
        FAVPOST OUTPUT FILE NAME                    = ocpost10yrint.out
```

M-2: ISI Every 10 Years (cont.)

```
**********************************
* NUMBER OF SIMULATIONS =   42000 *
**********************************
```

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
8	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
12	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
15	6.9014E-08	0.0000E+00	0.0000E+00	3.6598E-12	0.0000E+00	0.0000E+00	0.0001
27	3.0898E-06	0.0000E+00	9.3807E-06	2.6045E-08	0.0000E+00	0.0000E+00	0.0084
28	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
29	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
30	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
31	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
36	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
37	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
38	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
44	4.4385E-06	0.0000E+00	3.1287E-06	3.3673E-06	0.0000E+00	1.6329E-06	0.7586
89	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
90	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
98	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
99	4.3209E-07	0.0000E+00	0.0000E+00	4.2164E-08	0.0000E+00	0.0000E+00	0.0976
100	1.6582E-06	0.0000E+00	4.6005E-07	6.5525E-07	0.0000E+00	0.0000E+00	0.3952

M-2: ISI Every 10 Years (cont.)

101	9.7863E-06	0.0000E+00	5.0383E-05	1.9741E-08	0.0000E+00	0.0000E+00	0.0020
102	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
109	1.1900E-07	0.0000E+00	0.0000E+00	7.7323E-08	0.0000E+00	0.0000E+00	0.6498
110	2.6087E-03	6.2367E-03	3.2743E-02	1.7303E-05	2.1472E-04	3.0001E-04	0.0066
111	1.4966E-08	0.0000E+00	0.0000E+00	5.2442E-11	0.0000E+00	0.0000E+00	0.0035
112	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
113	6.2055E-08	0.0000E+00	0.0000E+00	5.9294E-08	0.0000E+00	0.0000E+00	0.9555
114	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
115	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
116	2.2291E-06	0.0000E+00	7.0608E-07	5.1321E-09	0.0000E+00	0.0000E+00	0.0023
117	5.5617E-05	3.4579E-04	7.7732E-04	9.5098E-08	0.0000E+00	0.0000E+00	0.0017
119	9.1956E-05	3.0210E-04	1.5073E-03	7.8504E-07	2.9569E-05	3.7212E-06	0.0085
120	3.1952E-05	2.8404E-04	4.5357E-04	2.5547E-05	2.6061E-04	3.1323E-04	0.7995
121	2.7485E-08	0.0000E+00	0.0000E+00	7.2283E-10	0.0000E+00	0.0000E+00	0.0263
122	3.8697E-04	2.6683E-03	3.8068E-03	3.8690E-04	2.6683E-03	3.8068E-03	0.9998
123	2.7485E-08	0.0000E+00	0.0000E+00	7.2283E-10	0.0000E+00	0.0000E+00	0.0263
124	1.7117E-04	8.5298E-04	2.3535E-03	1.6870E-04	8.5298E-04	2.3226E-03	0.9856
125	7.0998E-05	3.0103E-04	1.1539E-03	3.7122E-07	0.0000E+00	7.3397E-07	0.0052
126	8.9419E-07	0.0000E+00	1.1998E-07	3.0547E-09	0.0000E+00	0.0000E+00	0.0034
127	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
141	1.2407E-04	3.1186E-04	2.2591E-03	2.6331E-06	4.3366E-05	2.3103E-05	0.0212
142	4.5517E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
145	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
146	2.2255E-06	0.0000E+00	7.2719E-07	1.8547E-07	0.0000E+00	0.0000E+00	0.0833
147	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
148	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
149	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
154	1.4390E-04	4.9507E-04	2.3339E-03	3.8809E-07	0.0000E+00	0.0000E+00	0.0027
156	2.2374E-02	5.3757E-02	1.8079E-01	8.5646E-05	4.1552E-04	1.2331E-03	0.0038
160	1.3434E-02	2.9083E-02	1.0573E-01	1.7218E-04	5.9581E-04	2.6851E-03	0.0128
164	1.4231E-02	3.3616E-02	1.2566E-01	8.7527E-05	5.4722E-04	1.2941E-03	0.0062
165	3.5089E-04	2.6865E-03	2.7402E-03	3.5077E-04	2.6865E-03	2.7402E-03	0.9997
168	2.0794E-04	9.4460E-04	2.8225E-03	2.0523E-04	9.4460E-04	2.8108E-03	0.9870
169	2.0891E-04	5.2788E-04	3.5882E-03	7.6133E-06	1.4604E-04	8.9295E-05	0.0364
170	1.7072E-08	0.0000E+00	0.0000E+00	3.3194E-10	0.0000E+00	0.0000E+00	0.0194
171	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
172	8.2142E-05	3.0025E-04	1.3571E-03	4.2000E-07	0.0000E+00	9.8847E-07	0.0051

M-2: ISI Every 10 Years (cont.)

178	1.4390E-04	4.9507E-04	2.3339E-03	3.8804E-07	0.0000E+00	0.0000E+00	0.0027

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

M-2: ISI Every 10 Years (cont.)

```
*********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*          FOR THE FREQUENCY OF CRACK INITIATION          *
*********************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	2.8976	2.8976
2.1486E-06	92.1000	94.9976
6.4459E-06	2.9714	97.9690
1.0743E-05	0.8738	98.8429
1.5040E-05	0.3714	99.2143
1.9338E-05	0.2548	99.4690
2.3635E-05	0.1476	99.6167
2.7932E-05	0.1000	99.7167
3.2230E-05	0.0643	99.7810
3.6527E-05	0.0381	99.8190
4.0824E-05	0.0262	99.8452
4.5121E-05	0.0262	99.8714
4.9419E-05	0.0143	99.8857
5.3716E-05	0.0262	99.9119
5.8013E-05	0.0190	99.9310
6.2311E-05	0.0119	99.9429
6.6608E-05	0.0095	99.9524
7.0905E-05	0.0071	99.9595
7.5202E-05	0.0048	99.9643
7.9500E-05	0.0024	99.9667
8.3797E-05	0.0048	99.9714
9.2392E-05	0.0048	99.9762
1.0099E-04	0.0024	99.9786
1.0528E-04	0.0048	99.9833
1.0958E-04	0.0071	99.9905
1.1818E-04	0.0024	99.9929
1.4826E-04	0.0048	99.9976
1.6115E-04	0.0024	100.0000

```
=============================================
==      Summary Descriptive Statistics     ==
=============================================

Minimum                              = 0.0000E+00
Maximum                              = 1.6165E-04
Range                                = 1.6165E-04

Number of Simulations                =  42000

5th Percentile                       = 3.8070E-12
Median                               = 1.0893E-07
95.0th Percentile                    = 2.1486E-06
```

M-2: ISI Every 10 Years (cont.)

```
99.0th Percentile                     = 1.2561E-05
99.9th Percentile                     = 5.1763E-05

Mean                                  = 1.0085E-06
Standard Deviation                    = 3.8747E-06
Standard Error                        = 1.8907E-08
Variance (unbiased)                   = 1.5013E-11
Variance (biased)                     = 1.5013E-11
Moment Coeff. of Skewness             = 1.4853E+01
Pearson's 2nd Coeff. of Skewness      = 7.8082E-01
Kurtosis                              = 3.6493E+02
```

```
***********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)      *
***********************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	22.6333	22.6333
9.0774E-08	74.9690	97.6024
2.7232E-07	1.1857	98.7881
4.5387E-07	0.4024	99.1905
6.3542E-07	0.1905	99.3810
8.1696E-07	0.1286	99.5095
9.9851E-07	0.0810	99.5905
1.1801E-06	0.0952	99.6857
1.3616E-06	0.0405	99.7262
1.5432E-06	0.0143	99.7405
1.7247E-06	0.0167	99.7571
1.9062E-06	0.0167	99.7738
2.0878E-06	0.0238	99.7976
2.2693E-06	0.0167	99.8143
2.4509E-06	0.0143	99.8286
2.6324E-06	0.0310	99.8595
2.8140E-06	0.0071	99.8667
2.9955E-06	0.0048	99.8714
3.1771E-06	0.0071	99.8786
3.3586E-06	0.0214	99.9000
3.5402E-06	0.0167	99.9167
3.7217E-06	0.0095	99.9262
4.0848E-06	0.0119	99.9381
4.2664E-06	0.0048	99.9429
4.4479E-06	0.0024	99.9452
4.6295E-06	0.0095	99.9548
4.8110E-06	0.0095	99.9643
6.2634E-06	0.0024	99.9667
6.4449E-06	0.0024	99.9690

M-2: ISI Every 10 Years (cont.)

6.8080E-06	0.0048	99.9738
6.9896E-06	0.0024	99.9762
7.1711E-06	0.0024	99.9786
7.3527E-06	0.0024	99.9810
7.8973E-06	0.0024	99.9833
8.4419E-06	0.0024	99.9857
1.0076E-05	0.0024	99.9881
1.0802E-05	0.0048	99.9929
1.7156E-05	0.0024	99.9952
1.7882E-05	0.0024	99.9976
1.8064E-05	0.0024	100.0000

```
==============================================
==      Summary Descriptive Statistics      ==
==============================================

Minimum                                  = 0.0000E+00
Maximum                                  = 1.7973E-05
Range                                    = 1.7973E-05

Number of Simulations                    =  42000

5th Percentile                           = 0.0000E+00
Median                                   = 8.9093E-11
95.0th Percentile                        = 9.0774E-08
99.0th Percentile                        = 3.6793E-07
99.9th Percentile                        = 3.3586E-06

Mean                                     = 2.6200E-08
Standard Deviation                       = 2.6170E-07
Standard Error                           = 1.2770E-09
Variance (unbiased)                      = 6.8486E-14
Variance (biased)                        = 6.8485E-14
Moment Coeff. of Skewness                = 3.5238E+01
Pearson's 2nd Coeff. of Skewness         =-9.4834E-01
Kurtosis                                 = 1.8441E+03


************************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION      *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -     *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES       *
************************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
8	0.00	0.00
12	0.00	0.00
15	0.00	0.00
27	0.00	0.00
28	0.00	0.00

M-2: ISI Every 10 Years (cont.)

29	0.00	0.00
30	0.00	0.00
31	0.00	0.00
36	0.00	0.00
37	0.00	0.00
38	0.00	0.00
44	0.00	0.01
89	0.00	0.00
90	0.00	0.00
98	0.00	0.00
99	0.00	0.00
100	0.00	0.00
101	0.00	0.00
102	0.00	0.00
109	0.00	0.02
110	1.06	0.30
111	0.00	0.00
112	0.00	0.00
113	0.00	0.01
114	0.00	0.00
115	0.00	0.00
116	0.00	0.00
117	0.01	0.00
119	0.01	0.00
120	0.00	0.01
121	0.00	0.00
122	0.96	36.96
123	0.00	0.00
124	0.44	16.78
125	0.00	0.00
126	0.00	0.00
127	0.00	0.00
141	1.43	1.04
142	0.00	0.00
145	0.00	0.00
146	0.03	0.12
147	0.00	0.00
148	0.00	0.00
149	0.00	0.00
154	2.13	0.22
156	21.66	3.30
160	30.94	14.65
164	38.99	9.54
165	0.24	9.06
168	0.17	6.32
169	0.60	1.45
170	0.00	0.00
171	0.00	0.00
172	0.99	0.20
178	0.36	0.02
TOTALS	100.00	100.00

DATE: 27-Jun-2007 TIME: 10:36:45

APPENDIX N
RESPONSES TO THE NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE REVIEW OF WCAP-16168-NP, REVISION 1

PWROG
PWR Owners Group

Program Management Office
20 International Drive
Windsor, Connecticut 06095

October 16, 2007

OG-07-455

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

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington DC 20555-0001

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

References:

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

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

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

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington DC 20555-0001

October 16, 2007
Page 2 of 2
OG-07-455

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

Regards,

Frederick P. "Ted" Schiffley, II, Chairman
PWR Owners Group

FPS:JPM:las

Enclosures: (2)

cc:
M. Mitchell, USNRC
S. Peters, USNRC
T. Mensah, USNRC
S. Rosenberg, USNRC
C. Brinkman, W
C. Boggess, W
N. Palm, W
B. Bishop, W
J. Andrachek, W
J. Carlson, W
PWROG MSC Participants in the RV ISI Program
PWROG Management Participants in the RV ISI Program
PWROG PMO

REQUEST FOR ADDITIONAL INFORMATION

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

PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT (TR)

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

REACTOR VESSEL IN-SERVICE INSPECTION INTERVAL"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

Materials

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

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

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

-1-

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

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

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

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

Table 1a: Westinghouse NSSS Design Transients

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

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

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

Existing analyses of these transients had been performed using a 10% through-wall initial flaw. Therefore, sensitivity studies were performed on the four contributing transients using the PROBSBFD Code with an initial flaw depth equivalent to the thickness of the cladding (then rounded up to the nearest whole percent of the wall thickness). The analysis showed that the only design basis transient that resulted in significant crack growth was the cooldown transient. The sensitivity study using the PROFSBFD indicated that the flaw growth contribution of the Feedwater Cycling and Inadvertant Depressurization transients was at least an order of magnitude less than the contribution from the heatup/cooldown transient. Pressure test transients were enveloped by the heatup/cooldown transient. To envelope the contribution of the Feedwater Cycling and Inadvertent Depressurization transients and any partial cooldowns, 2 additional

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cooldown transients per year were conservatively added to the design basis of 5 cooldown cycles per year. Therefore, 7 cooldown cycles per year were evaluated with PROBSBFD to determine the surface breaking flaw density for the Westinghouse NSSS design pilot plant.

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

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

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

Table 2: CE NSSS Design Transients

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

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Table 2: CE NSSS Design Transients	
Transient	Description
	safety valve actuation causes the most severe transient and is commonly used as an umbrella case to conservatively represent the impact on the system from any of the other transients.

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

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

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

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

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

Input variable 2 (IFlawDen) gives the initial flaw density as 0.0036589 flaws per square foot in Figure 9.17. PROBSBFD input variables 16 to 19 (Percent1-Percent4) gives the percentages for the 4 values of aspect ratio specified in input variables 8 to 11 (Aspect1-Aspect4) of 2, 6, 10 and 99 (infinite) as 67.450, 20.769, 3.9642 and 7.8166, respectively, which agrees with the values in

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Figure 9.17 of the NUREG/CR Report. The same input to VFLAW03 was used, except for plant-specific values of vessel wall thickness and cladding thickness, which were set equal to the bead size for a single-pass cladding, to generate the initial surface flaw distribution input to PROBSBFD for Beaver Valley Unit 1 in Sections C-1 and C-2 of Appendix C and for Palisades in Sections G-1 and G-2 of Appendix G in the WCAP Report. As indicated in the WCAP Report, the information for calculating cladding (surface breaking) flaws in Tables B-2 (page B-8), F-2 (page F-8) and J-2 (page J-8) is taken directly from Table 4-2 of the December 2002 Draft NUREG Report on the Technical Basis for Revision of the PTS Rule (ADAMS: ML030090626). Note that the citation for this reference within Appendices B, F and J in Revision 1 of WCAP-16168-NP will be changed from [7] to the correct reference number of [8].

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

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

$R \leq 0.25$

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

$$Q = \exp(-0.408 + 0.542 C_r)$$

$0.25 < R < 0.65$

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

$$f_1 = 1.02 \times 10^{-12} (26.9R - 5.725)$$
$$f_2 = 1.01 \times 10^{-07} (3.75R + 0.06)$$

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$R \geq 0.65$

$$f_3 = (f_2 / f_1)^{1/4}$$
$$Q = \exp[0.1025R - 0.433625 + (0.6875R + 0.370125)C_F]$$

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

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

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

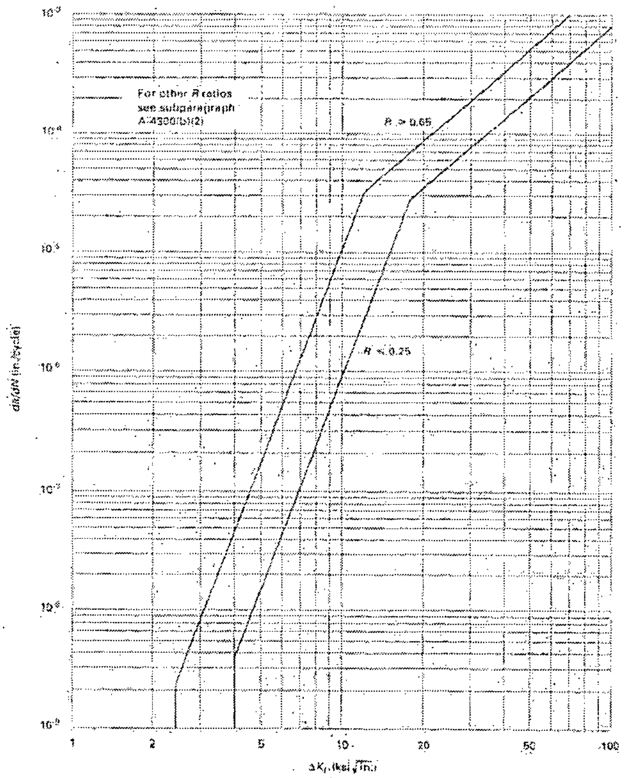

Figure A-4300-2, Reference Fatigue Crack Growth Curves for Carbon and Low Alloy Ferritic Steels
Exposed to Water Environments
(From Appendix A of Section XI of the ASME BPV Code)

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

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

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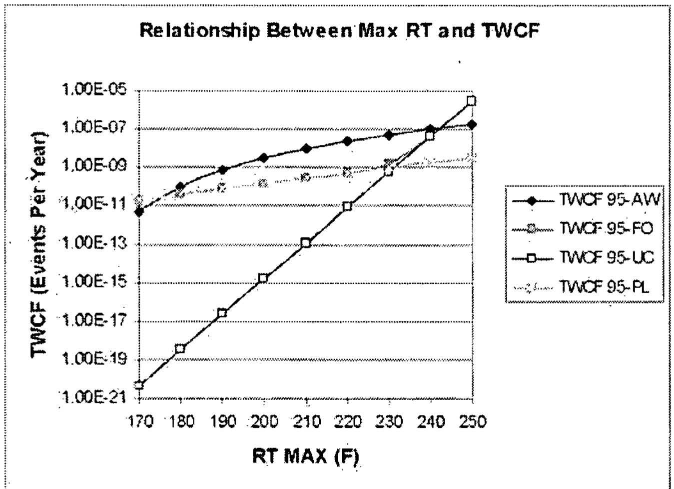
under-clad cracking (UC) on the same graph as shown below.



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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

6. The Probabilistic Fracture Mechanics Computer Tool and Methodology portion of Section 3.2 of WCAP-16168-NP, Revision 1, indicates that the failure frequency and distribution for all flaws in the

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reactor vessels were calculated using the latest version (05.1) of the FAVOR code. This code has been significantly revised by Oak Ridge National Laboratory. Provide an analysis that demonstrates the impact of using the latest version of FAVOR on the failure frequency and distributions documented in WCAP-16168-NP, Revision 1. Describe how the results from the latest version of FAVOR code would impact the conclusions in the WCAP.

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

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

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

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

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

Risk

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

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expect the FAVOR code to undergo additional changes before the technical basis for the PTS rulemaking is completed, but concludes that the version of the FAVOR code ultimately found acceptable in the PTS technical basis will be the version that will be acceptable for reference by the WOG in this Topical).

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

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

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

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

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

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

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

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

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

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04.1, 2007; 3) Sections 3 and 4 (Equations 1 to 132) of NUREG/CR-6854, *Fracture Analysis of Vessels – Oak Ridge FAVOR, v04.1, Computer Code: Theory and Implementation of Algorithms, Methods and Correlations*, 2006 and 4) Section 2 and Appendix A of NUREG-1874, *Recommended Screening Limits for Pressurized Thermal Shock (PTS)*, 2007. Appendix A of NUREG-1874 describes the requested changes in going from FAVOR version 05.1, which was used in Revision 1 of WCAP-16168-NP, to FAVOR version 06.1, which was used to calculate the CPF and TWCF results in NUREG-1874.

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

The WCAP will be revised to show that:

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

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

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

Response: Because vessel failure during a postulated PTS event is more likely to occur with a higher degree of embrittlement, which increases with operating time due to the accumulated neutron fluence, an operating time that is realistic but not overly conservative was desired. Another consideration was trying to bound most of the plants of each nuclear steam supply system (NSSS) vendor's design using one of the operating conditions in the PTS Risk Study performed by the NRC (i.e. from Table 8.5 of NUREG-1806) through the end of the first license renewal period (60 years), as requested by the PWR Owners Group. Using the information in Table 9.5, Plant List for Generalization Study, in NUREG-1806, Beaver Valley 1 at 60 EFPY was judged to be bounding for embrittlement at all the plants with a Westinghouse NSSS design, including more embrittled Salem 1, at 60 calendar years. Likewise, Palisades at 60 EFPY was judged to be bounding for embrittlement at all the plants with a Combustion-Engineering NSSS design, including more embrittled Fort Calhoun, at 60 calendar years. However, Oconee 1 at 60 EFPY would not be bounding for embrittlement at all the plants with a Babcock & Wilcox NSSS design, specifically TMI-1, at 60 calendar years so the next higher extended condition A was used

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for this pilot plant. These EFPY conditions set the vessel accumulated fluence and material embrittlement levels. A maximum operating time of 80 calendar years was used instead of 60 years. This longer operating time is not only considered to be bounding but is also conservative for two reasons. First, the transients in the plant design duty cycle that could produce fatigue crack growth are specified using a given rate per calendar year as described in the response to Part a) of RAI 1. Therefore the fatigue crack growth would be about 33% higher due to the larger total number of fatigue transients. Second, the effects of in-service inspections at 60, 70 and 80 calendar years are added in the cases for ISI every 10 years (a 60% increase relative to the last inspection after 50 years). Both of these conservatisms would tend to maximize the differences in TWCF for ISI every 10 years relative to the cases for 10-year ISI only.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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- Defenses against human errors are preserved:

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

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

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

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

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

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

a) Please explain why the mean failure estimates are sometimes opposite of what is expected. The explanation should include a justification that this analysis is precise enough to support the change in risk estimates instead of further investigating the apparent discrepancy and developing results that no longer

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appear contradictory. If the answer to b) below removes this apparent discrepancy the answer to this question can be referred to b).

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

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

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

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

Response: The information in Table 4-1 is a summary of the vessel failure frequency results for each of the three pilot plants that are given in Tables 3-2, 3-3 and 3-4, respectively, in the WCAP. For the first plant (Beaver Valley Unit 1), the mean value and standard error for 10-Year ISI Only of 5.04E-09 and 4.83E-10, respectively, in Table 3-2 on page 3-18 were taken from the FAVPOST Output values of 5.0405E-09 and 4.8272E-10, respectively, on page E-5 in Section E-

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

b) Given the response to a), please propose how the probabilistic risk assessment analyses that will be relied upon to support the relief requests will be demonstrated to be of sufficient technical adequacy so

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that there is confidence that the increases in core damage frequency or risk caused by the extension of the reactor vessel weld inspection interval from 10 to 20 years is small. One acceptable approach to assess technical adequacy is to assess the analysis against endorsed standard as described in RG 1.200.

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

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

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

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

Response: The PWROG proposes that for plants implementing the extended interval, TWCF be re-evaluated any time fluence is projected to increase by more than 10 percent, which is less than one standard deviation on the global fluence that is input to FAVOR. Fluence may be projected to increase as a result of core reloading, core loading pattern, power uprating, or when a surveillance capsule is removed from the reactor vessel and evaluated. This is consistent with the

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current approach for re-evaluating pressure-temperature limit curves and RT_{PTS} values. The WCAP will be revised to include this requirement.

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

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

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