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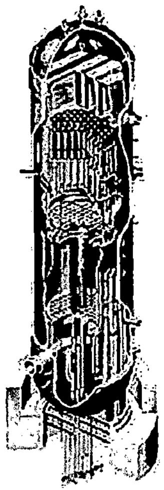
2019 TECHNICAL REPORT

BWRVIP-329NP: BWR Vessel and Internals Project
Updated Probabilistic Fracture Mechanics Analyses for BWR RPV Welds to Address Extended Operations



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BWRVIP-329NP: BWR Vessel and Internals Program

Updated Probabilistic Fracture Mechanics Analyses for BWR RPV Welds to Address Extended Operations

3002015930NP

Final Report, August 2019

EPRI Project Manager
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All or a portion of the requirements of the EPRI Nuclear Quality Assurance Program apply to this product.

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PRODUCT DESCRIPTION

In this current work, the BWRVIP objective is to provide an alternative probabilistic fracture mechanics (PFM) evaluation of boiling water reactor (BWR) reactor pressure vessel (RPV) welds that supports a technical basis for continued relief from the ASME Section XI examination requirements. The BWRVIP conducted bounding PFM analyses to identify the combinations of beltline material conditions that ensure regulatory safety goals are satisfied for a postulated low temperature isothermal pressure transient. This analysis can be used to assess compliance with the safety goals for axial and circumferential welds at the end of any specified operating interval.

Background

In September 1995, the BWRVIP published report BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendation". The purpose of this report was to provide a technical basis to justify a reduction in the required number of ASME Section XI examinations of axial and circumferential welds in the RPV. The results from PFM analyses were used to provide justification for the reduced examination scope. The PFM evaluation was performed for a postulated, low temperature isothermal over-pressure transient event, which had been determined to dominate the BWR RPV failure frequency.

Subsequent Safety Evaluation Reports (SERs) for BWRVIP-05 published by the Nuclear Regulatory Commission (NRC) staff defined the material and irradiation conditions, and the associated failure frequencies necessary to provide relief from examination of circumferential welds and to assess axial weld integrity. These requirements have been incorporated into BWRVIP-74-A, "BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal" and NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants".

The purpose of the work described in this report is to use NRC safety goals and PFM analysis procedures that have been developed since the publication of BWRVIP-05 to update the evaluation procedure and acceptance criteria specified in BWRVIP-74-A for providing relief from examination of circumferential welds and assessing axial weld integrity.

Objectives

- To evaluate the safety significance of a postulated, low temperature isothermal pressure transient in BWR reactor pressure vessels.
- To identify the combinations of beltline material conditions that will ensure regulatory safety goals are satisfied for the postulated transient.
- To determine if there are adequate margins against vessel failure during the postulated transient for an 80-year operating interval.

Approach

The BWR Vessel and Internals Program (BWRVIP) conducted PFM analyses to identify the combinations of RPV beltline material conditions that will ensure regulatory safety goals are satisfied for RPVs for a postulated, low temperature isothermal pressure transient. The approach for these analyses is consistent with previous industry and regulatory evaluations for the postulated BWR low temperature pressure transient, and with more recent analysis procedures employed by NRC staff in the development of the Alternate PTS Rule, 10CFR50.61a.

Results

The results identify the combinations of beltline material conditions for the BWR fleet that will ensure regulatory safety goals are satisfied for the postulated transient. The results from this work are used to demonstrate that reactor pressure vessels in the BWR fleet have margins against failure that satisfy regulatory criteria through at least an 80-year operating interval for the postulated, low temperature isothermal pressure transient.

Applications, Value, and Use

Results of this work provide the capability for BWR owners to demonstrate that reactor pressure vessels have margins against failure that satisfy regulatory safety goals for the postulated, low temperature isothermal pressure transient. Application of these results provide BWR owners with a continuing basis to justify relief from examination of circumferential welds, and the capability to demonstrate acceptable integrity for axial welds for RPVs.

Keywords

BWR reactor pressure vessels
Low temperature pressure transients
Probabilistic fracture mechanics
Examination requirements

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EXECUTIVE SUMMARY

Deliverable Number: 3002015930NP

Product Type: Technical Report

Product Title: BWRVIP-329NP: BWR Vessel and Internals Program, Updated Probabilistic Fracture Mechanics Analyses for BWR RPV Welds to Address Extended Operations

PRIMARY AUDIENCE: Engineers addressing issues related to allowable values of mean inner surface reference temperature for axial welds in boiling water reactor (BWR) reactor pressure vessels (RPV)

KEY RESEARCH QUESTION

A March 2000 Supplemental Safety Evaluation Report (SER) for BWRVIP-05, and an October 2001 SER for BWRVIP-74 specify that the mean inner surface reference temperature, RT_{MAX}, for RPV axial welds should not exceed 114°F. The results from materials surveillance tests indicate that the projected values of RT_{MAX} for some axial welds in the beltline region of RPVs may exceed 114°F prior to the end of a 60-year operating period. The bases for the conclusions contained in BWRVIP-05 need to be revisited for these RPVs, and for when operation beyond 60 years is considered.

RESEARCH OVERVIEW

The BWRVIP conducted Probabilistic Fracture Mechanics (PFM) analyses to evaluate the safety significance of a postulated, low temperature isothermal pressure transient in BWR RPVs, and to identify the combinations of beltline material conditions that will ensure regulatory safety goals are satisfied for the postulated transient. The bounding assessment included the limiting vessel beltline configuration (based on available data) and pressure and temperature conditions that bound values previously used to evaluate isothermal pressure transients (i.e., values used within BWRVIP-05).

The approach for these analyses is consistent with previous industry and regulatory evaluations for the postulated BWR low temperature pressure transients, and with more recent analysis procedures employed by the Nuclear Regulatory Commission (NRC) staff in the development of the Alternate Pressurized Thermal Shock (PTS Rule), 10CFR50.61a, for pressurized water reactors.

KEY FINDINGS

- The results from this work provide an alternative evaluation procedure that can be used to demonstrate compliance with the allowable RT_{MAX} specified for BWR axial beltline welds in paragraph 4.2.3.1.5 of SRP-LR, NUREG-1800, Revision 2.
- Application of the evaluation procedure indicates compliance with the regulatory risk goal of TWCF < 1E-6 yr^{-1} during the postulated isothermal pressure transient for the combinations of values of RT_{MAX} in plate, and axial and circumferential welds in the U.S. BWR fleet for an 80-year operating interval.
- The results from this work indicate that there are substantial margins against failure for the postulated isothermal pressure transient through at least 80 years of operation.

WHY THIS MATTERS

Recent results from BWR reactor pressure vessel material surveillance tests indicate that the RT_{MAX} for irradiated beltline weld material, in some instances, may be higher than initially predicted using the guidelines in Regulatory Guide 1.99, Revision 2, and higher than the limits specified in paragraph 4.2.3.1.5 of the SRP-

LR, NUREG-1800, Revision 2. In these cases, there is a need for an alternative that addresses the plant's initial license renewal commitment. This research provides an evaluation procedure to assess compliance with the regulatory criteria in paragraph 4.2.3.1.5 of the SRP-LR, and demonstrates that acceptable levels of BWR RPV integrity will be maintained during a postulated, low temperature isothermal pressure for axial weld RT_{MAX} values well above 114 °F.

There is also a need to comprehensively revisit the technical bases for examination relief for BWR RPV circumferential welds for operation beyond 60 years. The Standard Review Plan for Subsequent License Renewal (NUREG-2192) identifies this issue but does not include any recommendation on how plants should resolve this issue. The results from this evaluation provide a continuing basis to justify relief from the examination of RPV circumferential welds based on end-of-life RT_{MAX}, regardless of plant operating life.

HOW TO APPLY RESULTS

This report provides U.S. utilities with a technical basis supporting BWR continued relief from examination requirements for RPV welds, either requirements imposed by ASME Section XI, or by other regulatory codes and standards.

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PROGRAM: Boiling Water Reactor Vessel and Internals Program (BWRVIP) (P41.01.03)

IMPLEMENTATION CATEGORY: CATEGORY 1

ACRONYMS AND ABBREVIATIONS

ASME	American Society of Mechanical Engineers
AW	Axial Weld
BWR	Boiling water reactor
BWRVIP	EPRI Boiling Water Reactor Vessel and Internals Project
CPF	Conditional Probability of Failure
CW	Circumferential Weld
EFPY	Effective Full Power Years
NDT	Nil-Ductility Transition
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PFM	Probabilistic Fracture Mechanics
PTS	Pressurized Thermal Shock
RPV	Reactor Pressure Vessel
RT	Reference Temperature
SER	Safety Evaluation Report
TWCF	Through-Wall Cracking Frequency

CONVERSION FACTORS

Length:	1 in (inch) = 2.54 cm
Pressure:	1 MPa = 145.04 psi
Temperature:	°C = (°F - 32)(5/9)

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1 INTRODUCTION AND OBJECTIVES

Introduction

In September 1995, the BWRVIP published report BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendation" [1]. The purpose of this report was to provide a technical basis to justify eliminating the required ASME Section XI [17] examinations of circumferential welds and to reduce the number of required axial weld examinations from one-hundred percent to fifty percent in BWR reactor pressure vessels (RPVs). The results from the PFM analyses were used to provide justification for the reduced examination scope. The PFM evaluation was performed for a postulated low temperature isothermal over-pressure transient event which had been determined to be the dominant transient contributing to BWR RPV failure frequency.

Subsequently, the BWRVIP submitted report BWRVIP-05 to the NRC for review and approval. An initial Safety Evaluation Report (SER) was issued in July 1998 [2] and in March 2000, the NRC published a Supplemental SER to document the results of their review [3]. The NRC generally agreed with the conclusions in BWRVIP-05 but specified that:

- examination of one-hundred percent of the axial welds in the RPV would be required,
- relief would be granted from examination of circumferential welds in RPVs provided a plant specific assessment of the chemistry of the limiting weld and the fluence at the end of the license renewal term indicated that the failure frequency of the circumferential weld was within acceptable limits at the end of the license renewal term. The specified acceptable conditional failure probabilities ranged from 1.8 E-5 to 4.3E-4, depending on the vessel fabricator, and
- the inspection program could only be implemented for RPVs where the mean inner surface RT_{NDT} (referred to in this report as RT_{MAX}) for any axial weld did not exceed 114°F (45.6°C), and that plant-specific treatment of BWR RPV axial welds would be required for license renewal applications.

Recent information [4, 5] suggests that, prior to 60-years of operation, axial welds in some RPVs in the U.S. BWR fleet will have RT_{MAX} values that exceed the 114°F (45.6°C) allowable limit specified in the NRC March, 2000 SER. More importantly, when extended operations (operation beyond 80 years) is considered, it is observed that the many plants will no longer be able to apply the results from BWRVIP-05. Fortunately, since development of BWRVIP-05, substantial progress has been made with regard to application of PFM to RPVs. Specifically, updated and industry-accepted computational methods for RPV analysis are available in the FAVOR software code. This software was developed for the NRC at Oak Ridge National Laboratory (ORNL) for use in PWR PTS evaluations and has been subsequently applied to a significant number of RPV evaluations.

Report Objective

Generally, regulatory evaluations of the integrity of the RPV and other pressurized components in nuclear safety systems include the conservative assumption that propagation of a flaw through the vessel wall will produce component failure and potentially lead to reactor core damage. Acceptable margins against failure of pressurized components in nuclear safety systems must be maintained throughout their service life to ensure core integrity for all operational and postulated transient loading events.

The objective of this report is to apply the FAVOR Code to identify the limiting combinations of RT_{MAX} for plates and axial and circumferential welds that ensure total through-wall cracking frequency (TWCF) remains ≤ 1E-6 yr^{-1} for the RPV beltline and TWCF ≤ 1E-7 yr^{-1} for the circumferential welds during a postulated, low temperature isothermal pressure transient in BWR RPVs. Compliance with these safety goals provides a reasonable technical basis for relief from examination of the circumferential welds, while also demonstrating that axial welds have adequate margins against failure for the range of RPV conditions evaluated.

Report Scope

Section 2 of this report provides an overview of the industry and regulatory activities and regulatory requirements for assessing BWR RPV integrity for a low temperature, isothermal pressure transient. Section 3 describes the PFM analysis procedure and inputs that were used to calculate TWCF and determine the combinations of limiting beltline material conditions that meet the defined safety goals. Section 4 provides the computational results and presents the combinations of beltline material conditions that ensure compliance with the safety goals for BWR reactor pressure vessels during the postulated, transient. Section 5 provides a summary of the results and conclusions as well as implementation guidance.

Appendix A presents the results from a sensitivity study that was performed to assess the limiting beltline configuration. This configuration was used in the PFM analysis to identify the allowable combinations of RT_{MAX} for plates and axial and circumferential welds that will ensure adequate vessel integrity is maintained during a postulated, low temperature isothermal pressure transient.

Appendix B presents the results from a survey to assess plate and axial and circumferential weld limiting RT_{MAX} values for RPVs in the U.S. BWR fleet at 80-years of operation. This survey provides reasonable confidence that the results of the study are sufficient to support fleet operational goals related to extended operations.

Appendix C provides a description of the embedded and surface flaw distributions used in the PFM analyses.

Appendix D summarizes the major input variables used to perform the PFM analyses and the results from sensitivity calculations for several of these variables.

Appendix E presents the results from sensitivity calculations performed to demonstrate that the allowable RT_{MAX} limits defined using the limiting beltline configuration ensure that the total TWCF ≤ 1E-6 yr^{-1} for the RPV beltline and TWCF ≤ 1E-7 yr^{-1} for the RPV circumferential welds for a range of RPV dimensions that is enveloping of all U.S. and many international BWRs.

NEI 03-08 Implementation Requirements

The results documented in this report are intended to be used by individual utilities as a technical basis for obtaining relief from RPV circumferential weld examination requirements, while also ensuring the integrity of the axial welds. The implementation requirements of Nuclear Energy Institute (NEI) 03-08, Guideline for the Management of Materials Issues, are not applicable.

2
BACKGROUND

In September 1995, the BWRVIP published BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations" [1]. The purpose of this report was to provide a technical basis for relief from the ASME Section XI examination requirements for examination of BWR RPV circumferential and axial welds.

The results from the PFM analyses showed that the RPV failure frequency for circumferential welds was negligible, while the failure frequency for axial welds was less than 5×10^{-6} yr^{-1} for the RPV with the highest mean adjusted reference temperature at the vessel inner surface (RT_{MAX}) for axial welds in the U.S. BWR fleet [1]. These results were based on the evaluation of an unanticipated, isothermal high pressure transient postulated to occur at low temperature. This postulated transient had been determined to be the dominant transient contributing to BWR RPV failure frequency. Based on these results, BWRVIP-05 proposed elimination of all circumferential weld examinations and a 50% reduction in the number of required axial welds requiring examination.

Subsequently, the BWRVIP submitted BWRVIP-05 to the NRC for their review and approval. An initial Safety Evaluation Report (SER) was issued in July 1998 [2] and in March 2000, the NRC published a Supplemental SER to document the results from their review [3]. The NRC generally agreed with the conclusions in BWRVIP-05 but specified that:

- examination of one-hundred percent of the axial welds in the RPV would be required,
- relief would be granted from examination of circumferential welds in RPVs provided a plant specific assessment of the chemistry of the limiting weld and the fluence at the end of the license renewal term indicated that the failure frequency of the circumferential weld was within acceptable limits at the end of the license renewal term. The specified acceptable conditional failure probabilities ranged from 1.8 E-5 to 4.3E-4, depending on the vessel fabricator, and
- the inspection program could only be implemented for RPVs where the RT_{MAX} for any axial weld did not exceed 114°F (45.6°C), and that plant-specific treatment of BWR RPV axial welds would be required for license renewal applications.

Subsequently, the BWRVIP developed BWRVIP-74 to address aging management of the RPV for license renewal. This report was submitted to NRC for review and approval. In approving BWRVIP-74, the staff noted that:

> "As indicated in the March 2000 letter, an applicant shall monitor the axial beltline weld embrittlement"

This was identified as applicant action item 12 in the NRC acceptance of BWRVIP-74 for demonstrating compliance with the license renewal rule (documented as Appendix C to BWRVIP-74-A [6]).

Later, the NRC confirmed the plant specific action from BWRVIP-74-A regarding monitoring of axial weld embrittlement in a license renewal period in paragraph 4.2.3.1.5 of the Standard Review Plan for Review of License Renewal Applications (SRP-LR), NUREG-1800, Revision 2 [7]. The applicant action item is stated in Revision 2 of the SRP-LR as follows:

> "To demonstrate that the vessel has not been embrittled beyond the basis for the staff and BWRVIP analyses, the applicant should provide: (1) a comparison of the neutron fluence, initial RT_{NDT}, chemistry factor, amounts of copper and nickel, delta RT_{NDT}, and mean RT_{NDT} of the limiting axial weld at the end of the license renewal period to the reference case in the BWRVIP and staff analyses, and (2) an estimate of conditional failure probability of the RPV at the end of the license renewal term based on the comparison of the mean RT_{NDT} for the limiting axial welds and the reference case. **If this comparison does not indicate that the RPV failure frequency for axial welds is less than 5 x 10^{-6} per reactor year, the applicant should provide a probabilistic analysis to determine the RPV failure frequency for axial welds. (emphasis added)** Consistent with the staff's SER on Topical Report BWRVIP-05, dated March 7, 2000, the staff should ensure that the applicant's plant is bounded by the BWRVIP-05 analysis or that the applicant has committed to a program to monitor axial weld embrittlement relative to the values specified by the staff in its March 7, 2000, SER."

Recent information [4, 5], indicates that, prior to the end of a 60-year period of operation, axial welds in some RPVs in the U.S. BWR fleet are projected to have RT_{MAX} values that exceed the 114°F (45.6°C) allowable limit specified in the NRC March 7, 2000 SER. Consequently, for these plants, an alternative is needed to demonstrate acceptable failure frequency. Subsequently, NRC issued a Standard Review Plan for Subsequent License Renewal (SRP-SLR), NUREG-2192 [16]. The SRP-SLR does not provide any recommendations regarding an approach for demonstrating an acceptable failure frequency in the SLR period.

In this current work, the BWRVIP objective is to provide an alternative PFM evaluation of BWR RPV welds that provides a technical basis for continued relief from the ASME Section XI examination requirements for RPV circumferential and axial welds. To develop this methodology the BWRVIP conducted bounding PFM analyses to identify the combinations of beltline material conditions that ensure regulatory safety goals are satisfied for the postulated, low temperature isothermal pressure transient in BWR reactor pressure vessels. This methodology can be used to assess compliance with the safety goals for axial and circumferential welds at the end of any specified extended operating interval.

3 PFM ANALYSIS APPROACH AND INPUT

Introduction

This evaluation was performed using a parametric approach to determine the combinations of values of RT_{MAX} for beltline plates and axial and circumferential welds that satisfy current regulatory safety goals during the postulated low temperature isothermal pressure transient.

The safety goals used in this study are:

1. TWCF $\leq$ 1E-6 yr^{-1} for the entire RPV beltline and
2. TWCF $\leq$ 1E-7 yr^{-1} for the RPV circumferential welds only.

These safety goals were selected to maintain consistency with the safety goals used by the NRC to define the Alternate Fracture Toughness Requirements for Protection Against Thermal Shock Events (Alternate PTS Rule), 10CFR50.61a [8], and previous work for assessing BWR RPV integrity during an unanticipated, low temperature pressure transient [1,2,3]. Compliance with these safety goals provides the basis to obtain relief from the examinations of the circumferential welds and demonstrate that axial welds have adequate margins against failure.

Approach

Probabilistic Fracture Mechanics Analysis Methodology

Much has changed since the safety goals and associated maximum allowable values of RT_{MAX} for axial and circumferential welds were originally defined during development and review of BWRVIP-05 for a postulated, low temperature isothermal pressure transient in BWR vessels [1,2,3]. The major changes include the failure frequency safety goals, postulated flaw distributions, and the embrittlement damage and fracture toughness models. Most of these changes are based on the extensive work performed by the NRC during the development of the PWR pressurized thermal shock (PTS) screening criterion in the original PTS Rule,10CFR50.61, [9] and later in the Alternate PTS Rule, 10CFR50.61a [8].

Safety Goals

Included in the initial NRC SER for BWRVIP-05 [2] were the results from PFM analyses performed by the NRC staff to determine the CPF for axial and circumferential welds during the postulated, low temperature isothermal pressure transient at fluence levels corresponding to the end of license renewal at 64 EFPY [2]. The results from these analyses indicated that the CPF was approximately 5E-4 for the limiting circumferential weld in the BWR fleet, and together with an event frequency of 1E-3 yr^{-1}, the failure frequency or TWCF for the limiting circumferential weld, $TWCF_{CW}$, at the end of license renewal was approximately

$$TWCF_{CW}=5E\text{-}7\ yr^{-1} \qquad \textbf{Eq. 3-1}$$

The NRC staff concluded that the $TWCF_{CW}$ could increase significantly during operation beyond license renewal and that each plant requesting license renewal would be requested to perform a plant-specific assessment based on the chemistry of its limiting weld and the neutron fluence at the end of the license renewal term [2].

The results from the NRC staff PFM analyses [2] also indicated that the conditional probability of failure (CPF) for the limiting axial weld in the BWR fleet was approximately 8E-1. Subsequently, the NRC performed additional PFM analyses for axial welds using more realistic assumptions for the flaw density, size and location distributions and reported that RT_{MAX} = 114°F (45.6°C) for the limiting axial weld at the end of the initial 40-year license period and that the corresponding failure frequency was less than 5E-6 [3], or

$$TWCF_{AW}<5E\text{-}6\ yr^{-1} \qquad \textbf{Eq. 3-2}$$

Based on these results, the NRC noted that the results apply only for the initial 40-year license period of BWR plants, and that consideration of BWR axial welds for license renewal would require a plant-specific treatment by the license renewal applicant [3].

Equations 3-1 and 3-2 defined the bases for providing relief from the examination of circumferential welds and evaluating the integrity of axial welds.

Subsequently, during the NRC PTS evaluation, the safety goal for the allowable TWCF used for RPV PFM evaluations was revised [10]. The revised safety goal is:

$$TWCF=(CPF_{AW}+CPF_{CW}+CPF_{P} \bullet F\leq 1E\text{-}6yr^{-1}),\ \text{where} \qquad \textbf{Eq. 3-3}$$

TWCF is the through-wall cracking frequency, yr^{-1}, for all the plates and welds in the vessel beltline region, F is the frequency of the postulated event per operating year, yr^{-1}, and CPF_{AW}, CPF_{CW}, and CPF_{P} are the contributions to CPF for axial welds, circumferential welds, and plate, respectively. The safety goal in Equation 3-3 is the sum of contributions to the CPF for plate, axial welds and circumferential welds. In this case, no requirements were defined for individual contributions from flaws in plate or axial welds and circumferential welds.

Equation 3-2 was defined assuming that the CPF_{CW} and the CPF_{P} were negligible compared to the CPF_{AW} [2, 3]. If this assumption is made in Equation 3-3, then Equation 3-2 and Equation 3-3 would be the same except that the safety goal in Equation 3-3 is one-fifth of the safety goal in Equation 3-2. To update the previously defined safety goals [2, 3] shown in Equations 3-1 and 3-2 with the recent safety goal specified in Equation 3-3 [9, 10], Equation 3-2 is replaced by Equation 3-3, and the safety goal specified in Equation 3-1 for flaws in circumferential welds is divided by 5, or

$$TWCF_{CW}=CPF_{CW} \bullet F\leq 1E\text{-}7\ yr^{-1} \qquad \textbf{Eq. 3-4}$$

Equations 3-3 and 3-4 provide the basis for establishing the TWCF safety goals for ensuring overall vessel integrity and granting relief from examination of circumferential welds in vessels in the BWR fleet. These goals satisfy the previous safety goals [2,3] and have additional margin consistent with more recent safety goals [8,9,10]. Equations 3-3 and 3-4 can be applied to assess the integrity of any set of material and irradiation conditions in any RPV in the BWR fleet and are independent of operating period and fluence level.

Flaw Distributions

The flaw distribution in the original BWR evaluation used axial inner surface flaws in axial welds and circumferential inner surface flaws in circumferential welds [1]. Because the axial pressure stress is half the longitudinal pressure stress, the failure probability for the circumferential flaws was significantly less than the failure probability for the axial flaws. The flaw distribution in the plate contained significantly more flaws compared to the welds, but the flaws were smaller and the failure probability for flaws in the plate was significantly less than the failure probability for the axial flaws.

The flaw orientations and locations defined for the PWR PTS evaluations and currently used in the U.S for PFM analyses generally [11] are summarized in Table 3-1.

Table 3-1
Flaw Orientation and Location for Plates and Welds [11]

Material	Internal Surface Flaw Orientation		External Surface Flaw Orientation		Embedded Flaw Orientation	
	Axial	Circumferential	Axial	Circumferential	Axial	Circumferential
Axial weld	No	Yes	Yes	No	Yes	No
Circumferential weld	No	Yes	No	Yes	No	Yes
Plate or forging	No	Yes	Yes	Yes	Yes	Yes

In general, all the flaws identified in Table 3-1 must be considered in the PFM analysis to calculate the total CPF for the RPV, although some flaws and flaw locations may have little contribution to CPF for certain loading conditions. As indicated in the table, there are no axial inner surface flaws, and the only axial flaws are external surface flaws and embedded flaws.

Embrittlement Model and Fracture Toughness

Also, the earlier PFM analyses [1,2,3] used the embrittlement damage model in RG 1.99, Revision 2 [12]. The embrittlement damage model used in the PWR PTS evaluation and more recent PFM analyses is presented in the Alternate PTS Rule [8]. Coupled with the new embrittlement model is a revised material fracture toughness distribution [11]. This toughness distribution was defined using a more detailed statistical evaluation of the available fracture toughness data base.

In the original BWR axial weld evaluation the allowable RT_{MAX} for the welds was determined using the weld material properties [1,2,3]. However, during the PTS evaluation it was determined that flaws that may result from the welding process are located along the weld fusion line between the weld and base metal rather than in the interior of the weld. Consequently, the current PFM analysis methodology for the weld flaws uses the larger of the RT_{MAX} for the weld or attached plate to determine the fracture toughness for the material along the weld fusion line [11] and whether flaw extension will occur in the material with the larger RT_{MAX}.

These recent revised computational relationships are included in the software, FAVOR, which was developed for the NRC at Oak Ridge National Laboratory (ORNL) for use in the PWR PTS evaluation. FAVOR, v16.1 [11], which is the most recent publicly available version of the

software, was used to calculate the CPF for a postulated, low temperature isothermal pressure transient in BWR reactor pressure vessels.

PFM Evaluation Strategy

This effort involved several steps to define and obtain the data and computational tools needed to complete the risk assessment, including:

- Define the frequency and pressure and temperature conditions for the postulated low temperature isothermal, pressure transient used in the PFM analysis.
- Perform a sensitivity study to determine the limiting RPV beltline geometry based on information available from the U.S. BWR fleet that will be used as inputs to the PFM analysis.
- Construct a beltline model that includes plates and axial and circumferential welds representative of the limiting RPV beltline geometry in the BWR fleet.
- Perform iterative PFM analyses to determine the combinations of RT_{MAX} for plates and axial and circumferential welds that satisfy the criterion total TWCF = 1E-6 yr^{-1} for the limiting RPV beltline geometry and postulated, low temperature isothermal pressure transient.
- Ensure that the total TWCF ≤ 1E-6 for the combinations of RT_{MAX} for plates and axial and circumferential welds in the RPV beltline and $TWCF_{CW}$ ≤ 1E-7 yr^{-1} for circumferential welds in the RPV beltline.
- Perform the PFM analyses to maintain consistency with the methodology previously used by the NRC to define the Alternate PTS Rule, 10CFR50.61a [10, 13, 14] using the latest publicly released NRC software, FAVOR v16.1 [11].
- Apply the embedded and surface flaw distributions consistent with those previously used by the NRC to define the Alternate PTS Rule, 10CFR50.61a [10, 13, 14].
- Estimate the maximum RT_{MAX} at the end of the 80-year operating interval for the population of beltline plates and axial and circumferential welds in the U.S. BWR fleet of reactor pressure vessels as a means of ensuring that the results meet the objective of providing a technical basis for relief from the ASME Section XI examination requirements through at least 80 years of operation.
- Account for uncertainties arising from the use of generic analyses and the associated selection of inputs based on sensitivity studies.

Analysis Input

Postulated, Low Temperature Isothermal Pressure Transient

The BWRVIP and NRC previously defined the transient conditions associated with an unanticipated, low temperature, isothermal high pressure transient as 1,200 psi (8.27 MPa) at 100°F (37.8°C) [1] and 1,150 psi (7.93 MPa) at 88°F (31.1°C) [2], respectively; both evaluations used a transient frequency of 1E-3 yr^{-1}.

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Eq. 3-5

and

Eq. 3-6

RPV Limiting Beltline Configuration

The RPV beltline shell configuration used in the PFM analysis includes three shell courses fabricated from nine plates, nine axial welds and connected by two circumferential welds as illustrated in Figure 3-1. Content Deleted - EPRI Proprietary Information

The limiting beltline configuration was defined using the results from sensitivity studies that were performed to identify the combination of variables that would produce the highest total CPF for the postulated isothermal pressure transient. The variables included in the sensitivity study are vessel thickness and radius, surface and embedded flaw depths, cladding thickness and cladding stress free temperature. The sensitivity study scope and procedure and the detailed results are presented in Appendix A. The limiting vessel beltline and flaw dimensions that were defined in the sensitivity study, and were used in the PFM analyses, are presented in Table 3-2. Content Deleted - EPRI Proprietary Information

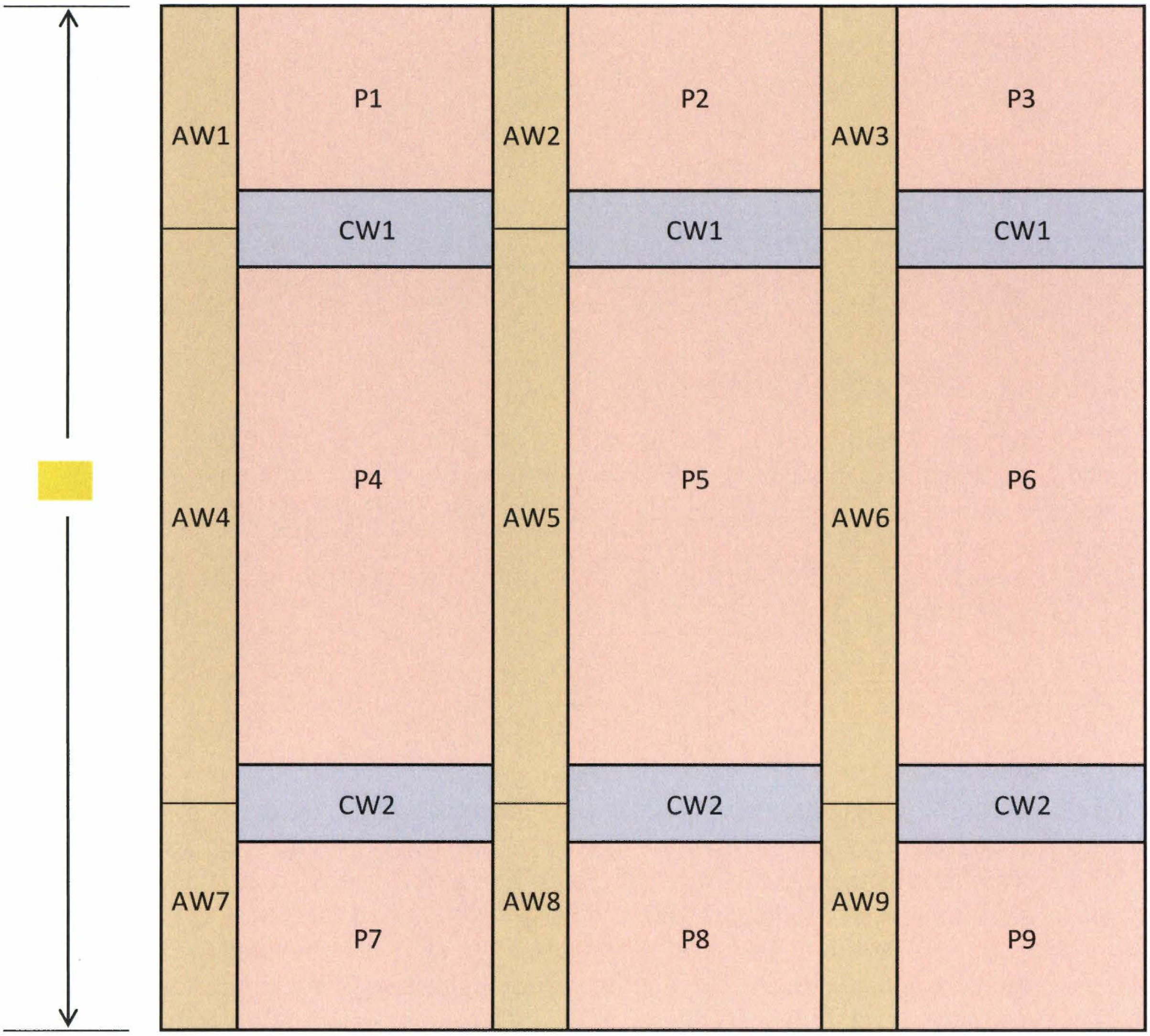

Figure 3-1
Illustration of the BWR Beltline Model

Table 3-2
Vessel, Cladding and Flaw Dimensions used for the PFM Evaluation of Isothermal Pressure Transients in BWR RPVs

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RPV Limiting Beltline Materials

A practical procedure to define the maximum allowable RT_{MAX} for axial welds in the vessels in the BWR fleet is to use an iterative process, where initially the CPF is calculated using the limiting RT_{MAX} values for the plates and axial and circumferential welds at 80 operating years. If the total CPF for the initial limiting values of RT_{MAX} for the plates and axial and circumferential welds is less (or greater) than , then RT_{MAX} for the axial welds is increased (or decreased) until CPF equals approximately .

Because there is a wide range of vessel specific values of RT_{MAX} for plates and welds in the BWR fleet, the iterative process to determine the axial weld RT_{MAX} that satisfies Equation 3-5 is repeated for a range of RT_{MAX} values for plate and circumferential welds. The results from these iterative calculations provide a graphical means to determine the allowable axial weld RT_{MAX} that meets the criteria specified in Equation 3-5 at any specified value of RT_{MAX} for plates and circumferential welds. The RT_{MAX} values determined using Equation 3-5 then were used to verify compliance with Equation 3-6 for the circumferential welds.

The iterative procedure requires changes in the material properties input to the FAVOR software to achieve the desired values of RT_{MAX} for each material. The changes were made to the Cu and Ni contents and initial RT_{NDT} values as needed to achieve the desired RT_{MAX} values. To be able to cover the range of RT_{MAX} values in the BWR fleet several pairs of Cu and Ni contents were used to produce high, medium and low values of RT_{MAX}. The values of $RT_{NDT(u)}$ in each of these ranges were then adjusted to reach the target value of RT_{MAX}. The values of $RT_{NDT(u)}$ generally were restricted to values for vessels in the BWR fleet (see Appendix B), or approximately -40°F $\leq RT_{NDT(u)} \leq$ +40°F for plates and welds. As a computational convenience, the same fluence is used for each beltline material, and the conservative assumption is made that the fluence is uniformly distributed throughout the entire beltline region.

Embedded and Surface Flaw Distributions

The surface and embedded flaw distributions used in the PFM analyses for plates and welds in the vessel include all the flaw locations and orientations listed in Table 3-1 and are consistent with those used by the NRC during the development of Alternate PTS Rule in 10CFR50.61a [8]. Embedded and inner and outer surface flaws are used in the PFM analyses, and because isothermal pressure loads are used in this study, the flaws are uniformly distributed through the vessel wall [11].

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Additional information related to the embedded and surface flaw distributions used in PFM analyses is presented in Appendix C.

Additional Analysis Input

The remaining variable values and distributions used in the PFM analyses are presented in Appendix D. Appendix D also includes sensitivity analyses to assess the effect of changes in these variables on CPF.

Summary

Table 3-3 presents a brief summary of the changes to the evaluation methodology, approach and analysis input used in this work compared to those used in BWRVIP-05 [1]. In general, most of the changes have been made to be consistent with the methodology employed by the NRC in the development of the Alternate PTS Rule [8].

Table 3-3
Summary of the Changes to the Evaluation Methodology, Approach and Analysis Input in This Work Compared to BWRVIP-05

Variable	Changes Compared to BWRVIP-05
Safety goal for all beltline materials	Content Deleted - EPRI Proprietary Information
Safety goal for circumferential beltline welds	Content Deleted - EPRI Proprietary Information
Event temperature and pressure	Content Deleted - EPRI Proprietary Information
Flaw Distributions	Content Deleted - EPRI Proprietary Information
Embrittlement model	Content Deleted - EPRI Proprietary Information

Table 3-3 (continued)
Summary of the Changes to the Evaluation Methodology, Approach and Analysis Input in This Work Compared to BWRVIP-05

Variable	Changes Compared to BWRVIP-05
Fracture Toughness	Content Deleted - EPRI Proprietary Information
RT_{MAX}	Content Deleted - EPRI Proprietary Information
Computational Software	Content Deleted - EPRI Proprietary Information
Approach	Content Deleted - EPRI Proprietary Information
Flaw Initiation	Content Deleted - EPRI Proprietary Information
Flaw Growth	Content Deleted - EPRI Proprietary Information

4

PFM ANALYSIS RESULTS

Parametric CPF Calculations for the Limiting Vessel Configuration

Because there are a large number of combinations of RT_{MAX} for plates and welds in vessels, a series of parametric CPF calculations were performed to define the combinations of RT_{MAX} for plates and welds that would correspond to total and satisfy Equation 3-5 for the limiting vessel configuration represented by Figure 3-1 and Table 3-2.

The parametric CPF calculations are performed by first selecting the same RT_{MAX} value for all nine plates and two circumferential welds and then selecting a RT_{MAX} value for all nine axial welds and calculating CPF. If the CPF is less (greater) than the allowable value , then the RT_{MAX} for the axial weld is increased (decreased) by some amount and CPF is recalculated. This process is continued until the axial weld RT_{MAX} is found where CPF satisfies the criterion in Equation 3-5. The iterative process to determine the axial weld RT_{MAX} that satisfies Equation 3-5 is repeated for a range of RT_{MAX} values for plate and circumferential welds; these results are used to construct a curve for the allowable RT_{MAX} for axial welds as a function of RT_{MAX} for plate and circumferential welds.

The iterative procedure requires changes in the material properties input to the FAVOR software relative to those for the limiting plate and welds to achieve the desired values of RT_{MAX} for each material. The changes were made to the Cu and Ni contents and initial RT_{NDT} values as needed to achieve the desired RT_{MAX} values. As a computational convenience, the same fluence was used for each beltline material, and the conservative assumption was made that the fluence is uniformly distributed at the vessel inner surface in the beltline region.

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Figure 4-1 Content Deleted - EPRI Proprietary Information

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The information presented in Figure 4-1 was used to determine the combinations of RT_{MAX} for plates and welds that correspond to for beltline materials in the BWR fleet; these combinations are presented in Table 4-1.

Table 4-1

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Table 4-2

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Confirmation of Conservatism for the BWR RPV Fleet

To confirm that the results generally meet needs associated with extended operations and are generically applicable to different vessel configurations, a review against available data from the U.S. BWR fleet was performed.

A review of the information in Appendix B indicates that the largest RT_{MAX} values for plate and axial and circumferential welds in the U.S. BWR fleet at the end of an 80-year operating interval are less than the allowable RT_{MAX} values corresponding to in Table 4-1 and Table 4-2. Content Deleted - EPRI Proprietary Information

The results from the calculations are summarized in Table 4-3 and include Content Deleted - EPRI Proprietary Information complete list of the computational conditions and results represented in Table 4-3 is presented in Appendix E.

Table 4-3 Content Deleted - EPRI Proprietary Information

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RT_{MAX} Acceptance Limit Curves

The information from Table 4-1 was used to construct a graphical relationship showing the combinations of plate and weld RT_{MAX} that correspond to total . In addition, the data in Figure 4-1 were used to determine the allowable RT_{MAX} plate and axial and circumferential weld combinations corresponding to ; these curves provide additional insight for assessing the additional margin against failure for the RPV RT_{MAX} values compared to the allowable values at . Figure 4-2 presents the allowable axial weld RT_{MAX} as a function of RT_{MAX} for circumferential weld and plate at the three CPF levels. As indicated in Table 4-2 and Table 4-3 application of the allowable RT_{MAX} plate and axial and circumferential weld combinations shown in Figure 4-2 ensures compliance with the safety goal specified in Equation 3-6.

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Figure 4-2
Combinations of Plate and Weld RT_{MAX} Corresponding to Several CPF Levels

Figure 4-2 can be used to determine the allowable value of axial weld RT_{MAX} for the limiting value of RT_{MAX} for circumferential welds or plates in the vessel. Or, Figure 4-2 can be used to estimate the CPF for the combinations of limiting RT_{MAX} for axial welds and plate and circumferential welds in the vessel at the end of any specified operating interval.

The allowable value of RT_{MAX} for the circumferential weld is determined using the larger of the RT_{MAX} for the plate or circumferential welds in the vessel. If the plate has the limiting RT_{MAX} relative to the circumferential weld, then the curves shown in Figure 4-2 accurately represents the overall contributions to CPF from the circumferential weld and plate. However, if the RT_{MAX} for the circumferential weld is larger than the RT_{MAX} for plate then the contribution to CPF from the plate will be overestimated and the allowable RT_{MAX} estimated for the axial weld from the curves in Figure 4-2 will be conservative. This conservatism will be negligible along the nearly horizontal portions of the curves in Figure 4-2, where the axial weld has the limiting CPF.

Comparison of Acceptance Limits with Vessels in the U.S. BWR Fleet at an 80-year Operating Interval

Figure 4-3 shows a comparison of the allowable limits shown in Figure 4-2 with the individual vessels in the U.S. BWR fleet at the end of an 80-year operating interval. The data pairs identified for each RPV in Figure 4-3 were obtained as follows:

- the value of RT_{MAX} for plate, when plate is the limiting beltline material, was obtained from Appendix B, Table B-1 and paired with the maximum axial weld RT_{MAX} in the same RPV
- the value of RT_{MAX} for the axial weld, when the axial weld is the limiting beltline material, was obtained from Appendix B, Table B-2 and paired with the larger of the maximum RT_{MAX} for the plate or circumferential weld in the same RPV
- the value of RT_{MAX} for the circumferential weld, when the circumferential weld is the limiting beltline material, was obtained from Appendix B, Table B-3 and paired with the maximum axial weld RT_{MAX} in the same RPV.

There are a number of vessels where the maximum value of RT_{MAX} is less than 50°F (10.0°C) in a beltline plate or an axial and circumferential weld. In these instances, the RT_{MAX} values that are less than 50°F (10.0°C) have been set equal to 50°F (10.0°C) on the plot in Figure 4-3 to maintain resolution for the graph.

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Figure 4-3
Comparison of BWR RPVs at the end of an 80-year Operating Interval with RT_{MAX} Acceptance Limits at Several CPF Levels

5 SUMMARY AND CONCLUSIONS

1. The BWRVIP conducted bounding probabilistic fracture mechanics (PFM) analyses to: evaluate the safety significance of a postulated, low temperature isothermal pressure transient in BWR reactor pressure vessels; identify the combinations of beltline material conditions that will ensure regulatory safety goals are satisfied for the postulated transient; and determine if there are adequate margins against vessel failure during the postulated transient at the end of an 80-year operating interval.
2. The PFM analysis procedures and input, and the risk goals used to define the limiting combinations of RT_{MAX} for plate and axial and circumferential welds, are consistent with previous industry and NRC evaluations [1, 2, 3] and the methodology used by the NRC staff to define the Alternate PTS Rule, 10CFR50.61a [8].
3. The results from the PFM analyses were used to define the combinations of RT_{MAX} for plate and axial and circumferential welds that ensure the total TWCF $\leq$ 1E-6 yr^{-1} for the RPV beltline and the $TWCF_{CW} \leq$ 1E-7 yr^{-1} for circumferential beltline welds in BWR reactor pressure vessels during a postulated, low temperature isothermal pressure transient. Application of these results provide BWR owners with a continuing basis to justify relief from examination of circumferential welds, and the capability to demonstrate acceptable integrity for axial welds for RPVs in the U.S BWR fleet.
4. The results from this work indicate that all the vessels in the U.S. BWR fleet have TWCF $\leq$ 1E-6 yr^{-1} with substantial margin at the end of an 80-year operating interval; Content Deleted - EPRI Proprietary Information
5. The PFM results presented in this work have been obtained using a number of conservative assumptions, including:
   - a transient pressure and temperature that bound values used in previous industry and NRC evaluations [1,2, 3]
   - a beltline geometry that bounds the vessels in the BWR fleet
   - assuming that all axial beltline welds in the vessel have the maximum allowable RT_{MAX} for the axial weld
   - assuming that all plates and circumferential welds in the vessel have the larger of the maximum allowable RT_{MAX} for the plate circumferential weld
   - using calculated values of CPF that are conservative and account for uncertainties in the computational procedure.

These conservative assumptions provide added assurance that there are substantial margins against RPV failure during the postulated isothermal pressure transient.

Implementation

The analytical approach involves defining the envelope of conditions for which the safety goals remain satisfied:

- TWCF $\leq$ 1E-6 yr^{-1} for the entire RPV beltline (sum of contributions from the limiting beltline plate, circumferential weld, and axial weld)
- $TWCF_{CW} \leq$ 1E-7 yr^{-1} for the limiting circumferential weld

Table 5-1 and Table 5-2 provide templates that can be used to verify that the PFM analyses in this report are applicable to an individual plant and that the safety goals defined in Section 4 of this report are met for the interval being evaluated.

Table 5-1 defines the envelope of RPV dimensions that are enveloped by the PFM evaluation. Because the parametric evaluation approach explicitly considered the RPV dimensions of all U.S. BWRs, these plants are by definition enveloped by this study. For non-U.S. BWRs relying on this study as a technical basis, if RPV dimensions are greater than or equal to (≥) the lower limits AND less than or equal to (≤) the upper limits specified in Table 5-1, the RPV dimensions are enveloped by the PFM evaluation.

Table 5-1
Template for Verification of Vessel Dimensions

Dimension	Lower Limit	Upper Limit
Reactor Vessel Inside Radius to CBMI, in. (mm)	Content Deleted - EPRI Proprietary Information	
Base Metal Wall Thickness, in. (mm)		
Radius / thickness		
Cladding Thickness, in. (mm)		

Table 5-2 provides a template that can be used by plants to demonstrate that end-of-interval RT_{MAX} values are within the envelope of limiting RT_{MAX} values for which the safety goals remain satisfied.

Table 5-2
Template for Verification of Acceptable RT_{MAX}

Parameter	Limiting Plate	Limiting Circumferential Weld	Limiting Axial Weld
Heat / Lot Identification Number			
Copper Content (wt. %)			
Nickel Content (wt. %)			
Chemistry Factor (CF) (°F) [1]			
EOI Neutron Fluence (f) (n/cm^2) [2]			
$RT_{NDT(U)}$ (°F) [3]			
EOI ΔRT_{NDT} (°F) [4]			
EOI RT_{MAX} (°F) [5]			
Limiting RT_{MAX} (°F) [6]	Content Deleted - EPRI Proprietary Information		
EOI RT_{MAX} < Limiting RT_{MAX}? [7]	Y / N	Y / N	Y / N

[1] Regulatory Guide 1.99 Chemistry Factor: Determined per Position 1.1 using Table 1 for Welds and Table 2 for Plates when less than two points of surveillance data are available or determined per Position 2.1 when two or more points of surveillance data are available.

[2] The end-of-interval (EOI) peak neutron fluence (E > 1.0 MeV) at the RPV inner surface for the limiting weld or plate being evaluated.

[3] Unirradiated (initial) reference temperature

[4] Increase in reference temperature due to irradiation at end of the interval for which the analysis is to be applied: $\Delta RT_{NDT} = (CF)\ f^{(0.28 - 0.10 \log f)}$, where fluence (f) is expressed in units of 10^{19} n/cm^2 (E > 1.0 MeV)

[5] RT_{MAX} at the end of the interval for which the analysis is to be applied: $RT_{MAX} = RT_{NDT(U)} + \Delta RT_{NDT}$

[6] Bounding RT_{MAX} values that satisfy risk goals (from Figure 4-2).

[7] If the EOI RT_{MAX} values for the limiting plate, circumferential weld, and axial weld are ALL less than the corresponding limiting RT_{MAX} values, the safety goals defined by this study remain satisfied.

6 REFERENCES

1. *BWRVIP-05: BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendation.* EPRI, Palo Alto, CA: 1995. TR-105697.
2. NRC Letter to Carl Terry, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report, July 28, 1998.
3. NRC Letter to Carl Terry, "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report, March 7, 2000, NRC (ADAMS Accession No. ML031430372).
4. LSCS Response to RAI4.2.5-1, Table 4.2.5-1, "Comparison of NRC Axial Weld Failure Probability Assessment for CE RPV (Mod 2 Variant) to LSCS Unit 1 54 EFPY Axial Weld Failure Probability Assessment".
5. *BWRVIP-135, Revision 3: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluation.* EPRI, Palo Alto, CA: 2014. 3002003144.
6. *BWRVIP-74-A: BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal.* EPRI, Palo Alto, CA: 2003. 1008872.
7. "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," NUREG-1800, Revision 2, Paragraph 4.2.3.1.5, December 2010, NRC (ADAMS Accession No. ML103490036).
8. Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, U.S. Nuclear Regulatory Commission, Federal Register/Vol. 75, No. 1/Monday, January 4, 2010/Rules and Regulations, pp. 13-29.
9. Code of Federal Regulations, 10 CFR Part 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events."
10. Recommended Screening Limits for Pressurized Thermal Shock (PTS), NUREG-1874, U.S. Nuclear Regulatory Commission, March 2010.
11. P.T. Williams, T.L. Dickson, B.R. Bass, and H.B. Klasky, Fracture Analysis of Vessels – Oak Ridge FAVOR, v16.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations, ORNL/TM-2016/309, Oak Ridge National Laboratory, Oak Ridge, TN, September, 2016.
12. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.
13. Technical Basis for the Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61) NUREG-1806, Vol. 1, August 2007.

14. Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61), NUREG-1806, Vol. 2- Appendix A, U.S. Nuclear Regulatory Commission, August 2007.

15. Reactor Vessel Integrity Database, RVID2, Nuclear Regulatory Commission, Version 2.1.0, 9/30/03. Available from http://www.nrc.gov/reactors/operating/ops-experience/reactor-vessel-integrity/database-overview.html#download.

16. "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants," NUREG-2192, Paragraph 4.2.2.1.6, July 2017, NRC.

17. Rules for Inservice Inspection of Nuclear Power Plant Components," ASME Boiler and Pressure Vessel Code, Section XI, Winter 1975 Addenda through the 2017 Edition, American Society of Mechanical Engineers (ASME).

A
LIMITING VESSEL GEOMETRY

Vessel Beltline Dimensions in the U.S. BWR Fleet

Table A-1 presents a list of the reactor pressure vessels in the U.S. BWR fleet and their beltline dimensions. Content Deleted - EPRI Proprietary Information

Limiting Vessel Geometry for Isothermal Pressure Transient Loads

The limiting vessel geometry for the isothermal pressure transient was determined from sensitivity studies where the CPF was calculated for different vessel wall thicknesses and associated vessel radii, and various flaw depths, cladding thicknesses and stress free temperatures. The vessel conditions included in the sensitivity study and the calculated CPF values are listed in Table A-2. The values of fluence, Cu, Ni, P, Mn and initial RT_{NDT} ($RT_{NDT(u)}$) used in the sensitivity study for the plates and axial and circumferential welds are presented in Table A-3.

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A more detailed discussion of the embedded and surface flaws is presented in Appendix C.

The difference between the thermal expansion coefficients for the ferritic and stainless steels can produce a significant residual stress gradient at the interface between the stainless steel cladding and the ferritic steel base metal, commonly referred to as the cladding to base metal interface (CBMI). The stress free temperature (SFT) is the temperature at which the residual stress at the CBMI is zero. As the operating temperature decreases below the SFT, the value of the applied stress intensity, K_I, for a small inner surface flaw increases, and at temperatures near room temperature, the K_I values can produce a substantial increase in the potential for the extension of small inner surface flaws near the CBMI.

The range of the SFT values used in the sensitivity study was determined from previous work Content Deleted - EPRI Proprietary Information. The results from an early study indicated that the SFT is about 400°F (204°C) [A-1]. Later, the results from [A-2] indicated that the SFT is 468°F (242°C) for an axial flaw and 364°F (184°C) for a circumferential flaw [A-3]. The SFT recommended in the FAVOR software is based on the work in [A-2] and is 488°F (253°C) [A-4], which is the value for the axial flaw adjusted for temperature dependent mechanical and physical material properties. Content Deleted - EPRI Proprietary Information

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Table A-1
Beltline Dimensions for Vessels in the U.S. BWR Fleet

Vessel	Base Metal Wall Thickness t inch	Inner Radius to CBMI R inch	R/t
Duane Arnold	4.5	91.5	20.3
Monticello	5.1	102.5	20.1
Brunswick 1	5.2	109.0	21.0
Hatch 2	5.3	109.0	20.6
Brunswick 2	5.4	109.0	20.2
River Bend	5.4	110.2	20.4
Hatch 1	5.4	110.4	20.3
Cooper	5.5	109.0	19.8
Clinton	5.6	109.0	19.5
Pilgrim	5.7	112.0	19.6
Peach Bottom 3	6.0	125.5	20.9
Perry	6.1	119.0	19.5
Quad Cities 2	6.1	125.5	20.6
Dresden 2	6.1	125.5	20.6
Peach Bottom 2	6.1	125.0	20.5
Susquehanna 2	6.1	125.5	20.6
Fermi 2	6.1	125.5	20.5
LaSalle1	6.1	127.0	20.7
Brown's Ferry 2	6.2	125.5	20.4
Brown's Ferry 3	6.2	125.5	20.4
Hope Creek	6.2	125.5	20.2
LaSalle 2	6.2	126.5	20.4

Table A-1 (continued)
Beltline Dimensions for Vessels in the U.S. BWR Fleet

Vessel	Base Metal Wall Thickness t inch	Inner Radius to CBMI R inch	R/t
Limerick 1	6.2	125.5	20.2
Limerick 2	6.2	125.5	20.2
Columbia - LIS	6.2	125.5	20.2
Grand Gulf 1	6.2	126.5	20.3
Dresden 3	6.3	125.5	20.0
Quad Cities 1	6.3	125.5	20.0
Susquehanna 1	6.3	125.5	19.9
Brown's Ferry 1	6.4	125.5	19.7
Nine Mile Point 2	6.5	126.5	19.5
Fitzpatrick	6.5	109.0	16.8
Nine Mile Point 1	7.1	106.5	15.0
Oyster Creek	7.1	106.5	15.0
Columbia - LS	9.5	125.5	13.2

Table A-2
Sensitivity Study Analysis Input and Results Used to Define the Limiting BWR Vessel Beltline Geometry

ID	Inner Radius (to clad) Inch	Base Metal Thickness Inch	Cladding Thickness inch	R/t	Surface Flaw Depth Inch	Flaw Depth to Thickness Ratio %	Flaw Depth into Base Metal inch	SFT °F	CPF
1	Content Deleted - EPRI Proprietary Information								
2									
3									
4									
5									
6									
7									
8									

Table A-2 (continued)
Sensitivity Study Analysis Input and Results Used to Define the Limiting BWR Vessel Beltline Geometry

ID	Inner Radius (to clad)	Base Metal Thickness	Cladding Thickness	R/t	Surface Flaw Depth	Flaw Depth to Thickness Ratio	Flaw Depth into Base Metal	SFT	CPF
	Inch	Inch	inch		Inch	%	inch	°F	
9	Content Deleted - EPRI Proprietary Information								
10									
11									
12									
13									
14									
15									

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Table A-3
Sensitivity Study PFM Analysis Input for Fluence, Cu, Ni, P, Mn and Initial RT_{NDT}

Variable	Axial Welds	Circumferential Welds	Plates
ID Fluence, 1E19 n/cm^2	Content Deleted - EPRI Proprietary Information		
Cu, wt.%			
Ni, wt.%			
P, wt.%			
MN, wt.%			
Initial RT_{NDT}, °F			

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Figure A-1
Sensitivity Study CPF Results to Define Limiting BWR Vessel Beltline Geometry in the BWR Fleet for an Isothermal Pressure Transient

References

A-1 Ganta, B.R., Ayres, D.J., and Hijeck, P. J., "Cladding Stresses in a Pressurized Water Reactor Vessel Following application of the Stainless Steel Cladding, Heat Treatment and Initial Service," ASME Pressure Vessel and Piping, Pressure Vessel Integrity, Vol. 213, pp. 245-252, 1991.

A-2 Dickson, T., et al., Evaluation of Margins in the ASME Rules for Defining the P-T Curve for a RPV, Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting NUREG/CP-0166, Vol. 1, October 26-28, 1998, (ADAMS Public Documents, Accession Number ML042230476).

A-3 Dickson, T., and Bass, R., Impact of Using a Stress Free Temperature of 364°F on Shallow Flaw Issue, NRC Public Meeting on Reactor Pressure Vessel Issues, January 19, 2016, Rockville, Maryland, USA. (ADAMS Accession No. ML16021A008).

A-4 P.T. Williams, T.L. Dickson, B.R. Bass, and H.B. Klasky, Fracture Analysis of Vessels – Oak Ridge FAVOR, v16.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations, ORNL/TM-2016/309, Oak Ridge National Laboratory, Oak Ridge, TN, September, 2016.

A-5 Assessment of a Stress-Free Temperature Model for Residual Stresses in Surface Cladding of a Reactor Pressure Vessel, Proceedings of the ASME Pressure Vessels and Piping Conference, July 16-20, 2017, PVP2017-65255.

B
BWR BELTLINE MATERIALS SURVEY

Limiting RPV Beltline Plates and Welds in the U.S. BWR Fleet

Table B-1, Table B-2 and Table B-3 list the vessels where a plate, axial weld or circumferential weld, respectively, is the beltline material with the highest value of RT_{MAX} in the vessel beltline. Table B-1 has two entries for the Columbia Generating Station (Columbia) RPV. The first is the lower intermediate shell (LIS) where the wall thickness is 6.2-inch (157 mm), and the second is the lower shell (LS) where the wall thickness is 9.5-inch (241 mm). The data shown in the tables were obtained using References [B-1], [B-2] and [B-3]. **Although it is recognized that the data used within Table B-1 through Table B-3, may not represent the latest licensing values, use of these data is reasonable for the general purpose of determining if the results are likely to address 80-year operations. The data contained in this appendix were not used in the derivation of the RT_{MAX} limits and do not have any impact on the conclusions presented in Section 5.**

Within Table B-1, Table B-2 and Table B-3, RT_{MAX} is the mean reference temperature at the vessel inner surface, and is determined from the relationship

$$RT_{MAX}=RT_{NDT(u)}+\Delta RT_{NDT}, \text{ where} \qquad \textbf{Eq. B-1}$$

$RT_{NDT(u)}$ is the unirradiated portion of the reference temperature (initial RT_{NDT}), which is determined using the procedures in the ASME Code [B-4] or other methods acceptable to the regulatory authorities, for example [B-5]. $\Delta RT_{NDT} = FF \cdot CF$ is the irradiated portion of the reference temperature at the vessel inner surface. The fluence factor, FF, is determined for the fluence at the vessel inner surface using the guidelines in Regulatory Guide 1.99, Revision 2 [B-6]. The chemistry factor, CF, is determined from the guidelines in [B-6], and when credible surveillance data are available the information and procedures in [B-2] are used to update the value of CF.

The vessels materials for which the CF values in Table B-1, Table B-2 and Table B-3 have been updated based on surveillance test results [B-2] are identified by the shaded cells in the tables. The materials included in BWR surveillance programs and the associated values of CF are summarized in Table B-4 for plates and Table B-5 for welds. The updated CFs in the fifth column of Table B-4 and Table B-5 are based on either a "fitted" CF, which is a weighted average obtained from credible surveillance data [B-2], or application of the CF Tables in RG 1.99 using best estimate Cu or Ni contents reported in [B-2]. The sixth column in Table B-4 and Table B-5 indicates if the material heat with the updated CF is the limiting material in the vessel beltline.

The information in Table B-1, Table B-2 and Table B-3 indicate that the highest values of RT_{MAX} in the BWR fleet at 80 operating years are 199°F (92.8°C) for plate, 195°F (90.6°C) for

axial welds and 186°F (85.6°C) for circumferential welds. The CPF evaluation was performed to encompass the maximum RT_{MAX} values in Table B-1, Table B-2 and Table B-3.

Table B-1
List of BWR Vessels where a Plate is the Limiting Beltline Material at 80 Operating Years

Vessel	Base Metal Wall Thickness	ID Fluence @ 80 Years	Fluence Factor @ ID	Chemistry Factor, CF	ΔRT_{NDT}	$RT_{NDT(u)}$	RT_{MAX}, ΔRT_{NDT}+ $RT_{NDT(u)}$
	inch	n/cm²	n/a	°F	°F	°F	°F
Columbia-LIS	6.2	1.70E+18	0.532	88.0	46.8	-8	39
Grand Gulf 1	6.2	5.63E+18	0.839		40.2	0	40
Peach Bottom 2	6.1	1.98E+18	0.567	82.4	46.7	-6	41
Susquehanna 2	6.1	1.44E+18	0.494	65.0	32.1	10	42
Nine Mile Point 2	6.5	2.03E+18	0.572	74.5	42.6	0	43
Susquehanna 1	6.3	1.71E+18	0.533		35.5	18	54
Hatch 2	5.3	3.13E+18	0.681	510	34.7	24	59
Columbia-LS	9.5	6.80E+17	0.344	110.0	37.9	28	66
Peach Bottom 3	6.0	1.78E+18	0.542	104.0	56.3	10	66
Hope Creek	6.2	1.11E+18	0.438	112.8	49.4	19	68
LaSalle 2	6.2	1.44E+18	0.494	81.2	40.1	32	72
Brunswick 2	5.4	2.29E+18	0.603	106.7	64.3	10	74
Limerick 1	6.2	4.23E+18	0.761	73.0	55.6	20	76
Dresden 2	6.1	8.10E+17	0.376	143.0	53.8	30	84
Brunswick 1	5.2	3.05E+18	0.675	139.8	94.3	10	104
Limerick 2	6.2	4.23E+18	0.761	101.3	77.1	40	117
Duane Arnold	4.5	8.10E+18	0.941		138.8	10	149
Hatch 1	5.4	4.37E+18	0.770		170.5	-20	150
Cooper	5.5	3.60E+18	0.718		185.5	-20	166
Oyster Creek	7.1	1.00E+19	1.001	138.2	138.3	31	169
Nine Mile Point 1	7.1	4.52E+18	0.779	173.9	135.5	40	175
Monticello	5.1	8.59E+18	0.957		172.3	27	199

Note: CF values in shaded cells were obtained from Table B-4.

Table B-2
List of BWR Vessels Where an Axial Weld is the Limiting Beltline Material at 80 Operating Years

Vessel	Base Metal Wall Thickness	ID Fluence @ 80 Years	Fluence Factor @ ID	Chemistry Factor, CF	ΔRT_{NDT}	$RT_{NDT(u)}$	RT_{MAX}, ΔRT_{NDT}+ $RT_{NDT(u)}$
	inch	n/cm^2	n/a	°F	°F	°F	°F
Perry	6.1	1.10E+19	1.027	82.0	84.2	-30	54
River Bend	5.4	1.49E+19	1.110		129.7	-50	80
Quad Cities 2	6.1	1.10E+18	0.437	140.6	61.4	23.1	84
Brown's Ferry 3	6.2	2.48E+18	0.621	140.6	87.4	23.1	110
Fermi 2	6.1	1.46E+18	0.497		160.9	-44	117
Clinton	5.6	1.55E+19	1.122	135.0	151.4	-30	121
Pilgrim	5.7	3.11E+18	0.679		203.9	-48	156
LaSalle 1	6.1	1.16E+18	0.446		195.0	-30	165
Brown's Ferry 2	6.2	2.48E+18	0.621		152.0	23.1	175
Fitzpatrick	6.5	4.07E+18	0.751		243.0	-48	195

Note: CF values in shaded cells were obtained from Table B-5.

Table B-3
List of BWR Vessels Where a Circumferential Weld is the Limiting Beltline Material at 80 Operating Years

Vessel	Base Metal Wall Thickness	ID Fluence @ 80 Years	Fluence Factor @ ID	Chemistry Factor, CF	ΔRT_{NDT}	$RT_{NDT(u)}$	RT_{MAX}, ΔRT_{NDT}+ $RT_{NDT(u)}$
	inch	n/cm^2	n/a	°F	°F	°F	°F
Dresden 3	6.3	1.15E+18	0.445	220.6	98.1	-5	93
Quad Cities 1	6.3	7.88E+17	0.371		115.4	-5	110
Brown's Ferry 1	6.37	1.71E+18	0.533		166.0	20	186

Note: CF values in shaded cells were obtained from Table B-5.

Table B-4
Updated Chemistry Factors from Surveillance Test Results for Plate [B-2, B-3]

Appendix [B-2]	Vessel	Material	Heat	CF (°F)	Limiting Material
A-1	Browns Ferry 2	Plate	A0981-1		An axial weld is limiting
A-2	Cooper	Plate	C2307-2		Limiting beltline material
A-3	Duane Arnold	Plate	B0673-1		Limiting beltline material
A-4	Hatch 1	Plate	C4114-2		Limiting beltline material
A-5	Hatch 2	Plate	C8554	RG-1.99=51.0	This plate is not limiting
A-6	Hope Creek	Plate	5K3238/1	RG-1.99=58.0	This plate is not limiting
A-7	LaSalle 1	Plate	C6345-1		An axial weld is limiting
A-8	Monticello	Plate	C2220		Limiting beltline material
A-9	Peach Bottom 2	Plate	C2761-2	RG-1.99=65.0	This plate is not limiting
A-10	Perry	Plate	C2557-1		An axial weld is limiting
A-11	River Bend	Plate	C3054-2	RG-1.99=51.0	An axial weld is limiting
A-12	SSP (Grand Gulf)	Plate	A1224-1		Limiting beltline material
A-13	SSP/Cooper (Millstone)	Plate	C1079-1		Limiting beltline material
A-14	SSP/Cooper (Quad Cities 1)	Plate	A0610-1		A circ weld is limiting
A-15	Susquehanna 1	Plate	C2433-1		Limiting beltline material
A-16	Nine Mile Point 1	Plate	P2130-2		This plate is not limiting
A-17	Hatch 1	Plate	C3985-2	RG-1.99=74	This plate is not limiting
A-18	Fitzpatrick	Plate	C3278-2		An axial weld is limiting
A-19	Cooper	Plate	C2331-2		This plate is not limiting

Table B-5
Updated Chemistry Factors from Surveillance Test Results for Welds [B-2, B-3]

Appendix [B-2]	Vessel	Material	Heat	CF (°F)	Limiting Material
B-15	Browns Ferry 1	Circ Weld	406L44		Limiting beltline material
B-1	Browns Ferry 2	Axial Weld	BF2-ESW		A plate is limiting
B-11	Clinton	Weld	5P6756		This weld is not limiting
B-11	Columbia	Circ Weld	5P6756		A plate is limiting
B-13	Fermi 2	Axial Weld	13253		Limiting beltline material
B-13	Fitzpatrick	Axial Weld	13253		Limiting beltline material
B-9	Grand Gulf	Axial Weld	5P6214B		A plate is limiting
Ref [B-3]	LaSalle 1	Axial Weld	1P3571		Limiting beltline material
B-11	Limerick 1	Circ Weld	5P6756		A plate is limiting
B-9	NMP 2	Axial Weld	5P6214B		A plate is limiting
B-9	Perry	Axial Weld	5P6214B		This weld is not limiting
B-12	Pilgrim	Axial Weld	27204		Limiting beltline material
B-15	Quad Cities 1	Circ Weld	406L44		Limiting beltline material
B-11	River Bend	Axial Weld	5P6756		Limiting beltline material
B-14	Susquehanna 1	Axial Weld	402K9171,411L3071		A plate is limiting

References

B-1 Reactor Vessel Integrity Database, RVID2, Nuclear Regulatory Commission, Version 2.1.0, 9/30/03. Available from http://www.nrc.gov/reactors/operating/ops-experience/reactor-vessel-integrity/database-overview.html#download.

B-2 *BWRVIP-135, Revision 3: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluation*. EPRI, Palo Alto, CA: 2014. 3002003144.

B-3 LSCS Response to RAI4.2.5-1, Table 4.2.5-1, "Comparison of NRC Axial Weld Failure Probability Assessment for CE RPV (Mod 2 Variant) to LSCS Unit 1 54 EFPY Axial Weld Failure Probability Assessment."

B-4 Rules for Construction of Nuclear Facility Components," ASME Boiler and Pressure Vessel Code, Section III, Subparagraph NB-2331, Summer 1972 Addenda through the 2017 Edition, American Society of Mechanical Engineers (ASME).

B-5 U.S. Nuclear Regulatory Commission, Standard Review Plan, NUREG-0800, Branch Technical Position 5-3, Rev. 2, March 2007.

B-6 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

C
EMBEDDED AND SURFACE FLAW DISTRIBUTIONS

Background

Distributions for the number, size and shape of embedded and surface flaws used in PFM analyses of RPVs have been defined previously based on an extensive study performed by the NRC as part of the development of the Alternate PTS Rule in 10CFR50.61a [C-1, C-2]. The previous study [C-2] defined embedded flaw distributions for plate and both shop and field fabricated RPVs, including seam welds and repair welds. In general, submerged arc welds (SAW) are used in shop fabrication and shielded metal arc welds (SMAW) are used in field fabrication; SMAW is used for weld repairs. The embedded flaw distributions used in the development of the Alternate PTS Rule in 10CFR50.61a [C-1] are based on a blend of the embedded flaw distributions for field and shop welds (approximately 97% SAW and 1% SMAW), and include repair welds, which constitute 2% of the total embedded flaws in the beltline welds. The distribution of surface flaws used in the PTS study included relatively few flaws (1-2 flaws per vessel) with total flaw depth from the inner surface of about 3% of the total wall thickness, including the cladding thickness.

Reference [C-3] reports the results from a recent study where the CPF was calculated for BWR RPVs during a pressure transient with maximum pressure = 1,150 psi (7.93 MPa) and a transient temperature that increased from 79°F (26.1°C) to 89°F (31.7°C) during the event. The time history for the temperature change from 79°F (26.1°C) to 89°F (31.7°C) was not reported. This study evaluated RPV weld and plate materials for a range of Cu and Ni contents, fluence levels and associated values of RT_{NDT}. The CPF at the end of 72 EFPY operating period was calculated for a vessel with a 220-inch vessel diameter. The vessel and cladding thicknesses used in the evaluation were not reported. The FAVOR software was used for the CPF calculations, but the specific version of the software used for the calculations was not reported.

Reference [C-3] includes a detailed discussion of the fabrication conditions and flaw distributions in welds and plate in BWR RPVs. Most BWR RPVs are shop fabricated and each vessel typically has 93-97% SAWs, 2-5% SMAWs and 2% repair welds. Eight RPVs in the U.S. BWR fleet were reported to be field fabricated with an estimated 98% SMAWs and 2% repair welds. In nine vessels the axial welds were fabricated using the electroslag welding (ESW) process. Typically, a shop fabricated vessel will have 13,000 to 15,000 embedded flaws and 6-7 surface flaws. Field fabricated vessels have approximately twice the number of embedded flaws and the same number of surface flaws compared to shop fabricated vessels. Most of the embedded flaws are relatively small; the largest embedded flaws are in repair welds, where the maximum through-wall flaw depth is 2-inch (51 mm) [C-1]. The number and size of the embedded flaws from repair welds are the same for shop and field fabricated vessels. The size of the surface flaws was not reported, but the surface flaw depth is likely to be in the range from about 3-5% of the total wall thickness, including cladding.

Reference [C-3] includes sensitivity studies to determine the CPF for the range of material properties and flaw conditions representative of vessels the U.S. BWR fleet at the end of a 72 EFPY operating period. The results from the CPF calculations indicate that:

- When the axial weld RT_{NDT} is relatively large compared to the RT_{NDT} for the plate the contribution to CPF is dominated by the larger embedded repair weld flaws in axial welds. Because the number and size of the embedded repair weld flaws are about the same in shop and field welds, there is not much difference in CPF for shop and field welds.
- When RT_{NDT} for the plate is relatively large compared to the RT_{NDT} for the axial welds the largest contribution to CPF is from the surface flaws in plate.
- Overall, the CPF < 1E-4 for all the different combinations of flaw and material conditions evaluated in the study [C-3.]

Current Evaluation

Because there are both shop and field fabricated RPVs in the U.S. BWR fleet, sensitivity calculations were performed to determine the difference in CPF for three different fabrication and associated flaw conditions, including: (1) a shop fabricated vessel with 98% SAW and 2% repair weld, (2) a field fabricated vessel with 98% SMAW and 2% repair weld, and (3) a blend of 97% SAW, 1% SMAW and 2% repair weld. Fabrication Conditions 1 and 2 represent the minimum and maximum number of embedded flaws, respectively, in a BWR vessel, while Condition 3 represents a blend similar to that used in the development of the Alternate PTS Rule [C-1] for PWRs. Each vessel has the same number and size of inner and outer surface flaws. The CPF was determined using each of the indicated flaw distributions and the limiting vessel configuration and dimensions described in Figure 3-1 and Table 3-2, and all the flaw locations and orientations listed in Table 3-1. Because isothermal pressure loads are used in this study the embedded flaws are uniformly distributed through the vessel wall and both inner and outer surface flaws are used in the PFM analyses.

The CPF computational results are summarized in Table C-1 for each of the three fabrication conditions and associated flaw distributions. The CPF values in Table C-1 are the maximum values obtained from the three cladding/surface flaw conditions listed in Table 3-2.

Table C-1
CPF Results for Three Fabrication Conditions for the Postulated Low Temperature Isothermal Pressure Transient

Fabrication Condition	Content Deleted - EPRI Proprietary Information
Shop: 98% SAW, 2% repair	
Field: 98% SMAW, 2% repair	
Blend: 97% SAW, 1% SMAW, 2% repair	

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The number of sampled flaws in a vessel depends on the volume and surface area of the vessel wall. Table C-2 lists the total number of embedded and surface flaws for the blended weld condition in each trial vessel for each of the nominal vessel wall thicknesses in the U.S. BWR fleet.

Table C-2
Total Number of Embedded and Surface Flaws in Each Trial Vessel for PFM Analyses, Blended Weld Condition for Embedded Flaws

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Table C-3
Size Distribution for Flaws in Welds for Trial Vessels, Blended Weld Condition

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The weld flaws are uniformly distributed through the vessel wall, so the numbers of embedded weld flaws in the inner and outer halves of the vessel are approximately equal. Similarly, the number of surface weld flaws at the vessel inner and outer surfaces is approximately equal.

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Table C-4
Size Distribution for Flaws in Plates for Trial Vessels

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The weld and plate materials in each vessel beltline geometry listed in Table C-2 have embedded and surface flaw size distributions similar to those shown in Table C-3 and Table C-4. The specific number of flaws and associated flaw sizes will depend on the volume and surface area of the vessel beltline and the cladding thickness and the vessel base metal wall thickness.

References

C-1 Technical Basis for the Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61) NUREG-1806, Vol. 1, August 2007.

C-2 Simonen, F.A., et al., "Reactor Pressure Vessel Failure Probability Following Through-Wall Cracks Due to Pressurized Thermal Shock Events", NUREG/CR-4483, U.S. Nuclear Regulatory Commission, March 1986.

C-3 Purtscher, P., et al., Analysis of Circumferential Welds in BWRs for Life Beyond 60, Proceedings of the ASME Pressure Vessels and Piping Conference, July 19-23, 2015, PVP2015-45836.

D
INPUT TO FAVOR SOFTWARE

Input Files

There are three sets of input files for the FAVOR software, including flaw input files, a load input file and a pfm file. The flaw input to FAVOR has been discussed in Appendix C. This appendix provides a summary of the major input variables in the FAVOR load and pfm files and includes sensitivity calculations to assess the sensitivity of CPF to changes in some of the major variables in the pfm file.

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Table D-1
Summary of the Major Input Variables for the FAVOR Load Input File for a Thick RPV

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Table D-2 summarizes the major input variables and their values for the FAVOR pfm file. In general, the variables in the pfm file are used to calculate the material fracture toughness (resistance to fracture), K_{IC}, and typically are sampled variables. A detailed description of these variables is contained in Ref. [D-2], and many of the variable values used in this work are the same as those used by the NRC staff in the development of the Alternate PTS Rule [D-3].

Table D-2
Summary of the Major Input Variables for the FAVOR pfm Input File

Variable	Variable Value	Used in the RT_{MAX} Calculation
Number trial vessels	Content Deleted - EPRI Proprietary Information	
Flaw population model		
Warm prestress option		
Embrittlement correlation model		
RPV coolant temperature at normal full power operation, °F		
Effective Full Power Years		
Flow stress, ksi		
K_{Ia} model		
Layer option		
Failure criterion		
Mean ID surface fluence, n/cm^2		
Fluence multiplier		
Mean Cu, Ni, P, Mn contents		
Standard deviations for Cu, Ni, P, Mn in welds and plates		
Weld residual stress		
Weld		
Plate		
Initial RT_{NDT}, °F		
Initial RT_{NDT} standard deviation, welds and plates, °F		

FAVOR uses a Monte Carlo sampling methodology to obtain the variable values from the variable distributions, and the accuracy of the results will depend on the number of simulated vessels. The accuracy of the calculated CPF is assessed in sensitivity calculations, which are presented later in this appendix. Content Deleted - EPRI Proprietary Information

The flaw population model used in this work contains embedded flaws that are uniformly distributed through the total RPV thickness, and an equal number of inner and outer surface

flaws with equal total flaw depths [D-2]. This distribution is appropriate when pressure is the dominant loading [D-2].

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The irradiation degradation related variables (Cu, Ni, Mn, P, fluence) are sampled by the FAVOR software during the PFM analysis. The standard deviations used for sampling Cu, Ni, Mn, P and fluence are the default values in the FAVOR software [D-2]. The mean values of Cu and Ni and are used in sensitivity calculations described later in this appendix.

It is assumed that there are no measured values for the initial RT_{NDT} for welds and plate. In this instance, the standard deviation for initial RT_{NDT} for the beltline welds is 17°F (9.4°C) [D-4]. The standard deviation for the beltline plates is 15°F (8.3°C), consistent with values determined for plate material when data correlations are used to define initial RT_{NDT} [D-5].

Sensitivity Calculations

Convergence of the CPF Computational Results

FAVOR uses a Monte Carlo sampling methodology to compute CPF. This methodology calculates the CPF for a specified number of trial vessels each of which contains a population of deterministic and distributed variables. Because each trial vessel has a different population of sampled variable values, such as fluence, element content, and flaw size and location, a large number of trial vessels must be evaluated to reach a convergent value of CPF and achieve an accurate estimate of the CPF. Content Deleted - EPRI Proprietary Information

Figure D-1 shows plots CPF as a function of the number of trial vessels for the following conditions:

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Figure D-1
CPF as a Function of Number of Vessel Trials at Nominal CPF Values of
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Figure D-2 shows plots CPF as a function of the number of trial vessels for the following conditions: Content Deleted - EPRI Proprietary Information

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Figure D-2
CPF as a Function of Number of Vessel Trials at RT_{MAX} Combination
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Figure D-3 shows plots CPF as a function of the number of trial vessels for the following conditions: Content Deleted - EPRI Proprietary Information

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Figure D-3
CPF as a Function of Number of Vessel Trials at RT_{MAX} Combination

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Sensitivity of CPF to Changes in Variables Used to Compute RT_{MAX}

Any combination of values of the variables marked "Yes" in Table D-1 that results in the same RT_{MAX} should provide a reasonable approximation of the CPF, all other things being equal. This means that application of the allowable RT_{MAX} limits shown in Figure 4-2 and Figure 4-3 is not restricted by the variable values used to compute RT_{MAX}, such as those marked "Yes" in Table D-1.

This section provides the results of calculations performed to assess the sensitivity of CPF to changes in the deterministic values of RPV coolant temperature and EFPY and the sampled variables, Cu and Ni contents and mean $RT_{NDT(U)}$. The results from the sensitivity calculations demonstrate that any combination of values of the variables marked "Yes" in Table D-1 that results in the same RT_{MAX} will provide approximately the same CPF, and consequently that the application of the allowable RT_{MAX} limits shown in Figure 4-2 and Figure 4-3 is not restricted by the variable values used to compute RT_{MAX} and construct Figure 4-2 and Figure 4-3.

Because there is a wide range of vessel specific values of RT_{MAX} for plates and welds in the BWR fleet, no single set of element contents can provide the range of target values of RT_{MAX} needed to determine CPF over the entire range of interest in the BWR fleet. Consequently, several sets of element contents were used for each material to obtain high, medium and low values of ΔT_{30}. Then a value of $RT_{NDT(u)}$ is selected to provide the desired value of RT_{MAX}. The

specific set of elements selected for use in the PFM analyses is determined so that the target RT_{MAX} values can be obtained by Content Deleted - EPRI Proprietary Information.

Table D-3 presents the Cu, Ni, P and Mn contents used in the PFM analyses. Also shown in the table are the values of ΔT_{30}, which were calculated using the indicated fluence, chemical contents, RPV coolant temperature and EFPY shown in Table D-2.

Table D-3
Mean Cu, Ni, P, and Mn Contents used in the PFM Analyses Reach Target Values of RT_{MAX}

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Table D-4
Sensitivity Calculations for CPF for RT_{MAX} Combination for Changes in RPV Coolant Temperature, Axial Weld Nickel Content, and $RT_{NDT(U)}$

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Table D-5
Sensitivity Calculations for CPF for RT_{MAX} Combination for Changes in RPV Coolant Temperature, Axial Weld Nickel Content, and $RT_{NDT(U)}$

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Table D-6
Sensitivity Calculations for CPF for RT_{MAX} Combination for Changes in RPV Coolant Temperature, Axial Weld Nickel Content, and $RT_{NDT(U)}$

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Table D-7
Sensitivity Calculations for CPF for RT_{MAX} Combination for Changes in Axial Weld Cu and Nickel Contents, and $RT_{NDT(U)}$

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References

D-1 Assessment of a Stress-Free Temperature Model for Residual Stresses in Surface Cladding of a Reactor Pressure Vessel, Proceedings of the ASME Pressure Vessels and Piping Conference, July 16-20, 2017, PVP2017-65255.

D-2 P.T. Williams, T.L. Dickson, B.R. Bass, and H.B. Klasky, Fracture Analysis of Vessels – Oak Ridge FAVOR, v16.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations, ORNL/TM-2016/309, Oak Ridge National Laboratory, Oak Ridge, TN, September, 2016.

D-3 Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, U.S. Nuclear Regulatory Commission, Federal Register/Vol. 75, No. 1/Monday, January 4, 2010/Rules and Regulations, pp. 13-29.

D-4 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

D-5 *Assessment of the use of NUREG-800 Branch Technical Position 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels (MRP-401 and BWRVIP-287)*. EPRI, Palo Alto, CA: 2015. 3002005348.

E

CPF RESULTS FOR WALL THICKNESS/CLADDING THICKNESS/SURFACE FLAW DEPTH COMBINATIONS

Appendix A presents the results from sensitivity calculations that were performed to identify the vessel geometry in the U.S. BWR fleet that would provide the highest calculated CPF for the postulated, isothermal low temperature pressure transient. Content Deleted - EPRI Proprietary Information

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Table E-1
Sensitivity Calculations to Demonstrate RT_{MAX} Limits Ensure

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Table E-1 (continued)
Sensitivity Calculations to Demonstrate RT_{MAX} Limits Ensure

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Table E-1 (continued)
Sensitivity Calculations to Demonstrate RT_{MAX} Limits Ensure

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Table E-2
Results from the FAVOR pfm Output File, Content Deleted - EPRI Proprietary Information

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