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TECHNICAL BASIS FOR REVISION OF THE PRESSURIZED THERMAL SHOCK (PTS) SCREENING LIMIT IN THE PTS RULE (10CFR50.61)

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ABSTRACT

One potentially significant challenge to the structural integrity of the reactor pressure vessel (RPV) in a pressurized water reactor (PWR) is posed by a PTS event wherein rapid cooling of the downcomer occurs, an event which may be followed by re-pressurization. The temperature drop produced by rapid depressurization coupled with the near-ambient temperature of the make-up water produces significant thermal stresses in the thick section steel wall of the RPV. For embrittled RPVs these stresses could be high enough to initiate a running crack, a crack that could propagate all the way through the vessel wall. Through-wall cracking of the RPV could precipitate core damage or, in rare cases, a large early release of radioactive material to the environment. This paper summarizes the results of a five year project conducted by the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (RES). The aim of this study was to develop the basis for a revision to the pressurized thermal shock (PTS) rule [10CFR50.61] consistent with current guidelines on risk informed regulation. Our findings suggest that there is a strong technical basis supporting a substantial relaxation of the current regulatory requirements on the level of fracture toughness needed to withstand PTS. Such a relaxation could be achieved without an increase in risk to the public. This paper includes our suggestions for quantitative revision of the materials-based screening limits in 10CFR50.61.

Keywords: Pressurized thermal shock, risk analysis, regulatory limits.

1 BACKGROUND

This paper summarizes the results of a five year project conducted by the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (RES). The aim of this study was to develop the basis for a revision to the PTS rule [10CFR50.61] consistent with current guidelines on risk informed regulation. NRC RES has recently finalized a number of reports providing that basis [EricksonKirk 05a, EricksonKirk 05b, Bessette 05, Whitehead 05, Siu 99, Simonen 03]. In this section we describe PTS, how it might occur, and what the potential consequences are for the vessel. This is followed by a summary of the current regulatory approach to PTS, which leads directly to a discussion of the motivations for conducting this project. The balance of the paper then addresses the following topics:

- Section 2 describes the approach we have taken to define an acceptable level of PTS risk and to estimate the level of PTS risk associated with PWR operation. In this discussion we address also our approach to treatment of uncertainties as well as the fundamental assumptions that underlie our calculations.
- Section 3 provides a brief summary of the results of our investigations.
- Section 4 uses the results from Section 3 to construct materials-based screening limits that ensure an acceptably low level of risk due to PTS.

Given the complexity of PTS, the information presented herein is of necessity a brief synopsis of our investigation. Interested parties are referred to other documents [EricksonKirk 05a, EricksonKirk 05b, Bessette 05, Whitehead 05, Siu 99, Simonen 03] for a fully detailed presentation of this information.

* Corresponding author. The views expressed herein are those of the authors and do not represent an official position of the United States Nuclear Regulatory Commission.

1.1 Description of PTS

One potentially significant challenge to the structural integrity of the RPV in a PWR is posed by a PTS event wherein rapid cooling of the downcomer occurs, an event which may be followed by re-pressurization. A number of abnormal events and postulated accidents have the potential to thermally shock the vessel (either with or without significant internal pressure); some of these include a pipe break in the primary pressure circuit, a stuck open valve in the primary pressure circuit, the break of the main steam line, and so on. During these events, both the water level and the temperature drops due to rapid depressurization. In events involving a break in the primary pressure circuit additional water level drop occurs due to leakage from the break. Automatic systems and operators must provide make-up water in the primary system to prevent overheating of the fuel in the core. The make-up water is much colder than that held in the primary system.

The temperature drop produced by rapid depressurization coupled with the near-ambient temperature of the make-up water produces significant thermal stresses in the thick section steel wall of the RPV. For embrittled RPVs these stresses could be high enough to initiate a running crack, a crack that could propagate all the way through the vessel wall. Through wall cracking of the RPV could precipitate core damage or, in rare cases, a large early release of radioactive material to the environment.

1.2 Current PTS Regulations in the United States

10CFR50.61 requires that licensees monitor the embrittlement of their RPVs using a 10CFR50 Appendix H qualified surveillance program [10CFR50H]. The results of surveillance are used together with the formulae and tables in 10CFR50.61 to estimate the fracture toughness transition temperature (RT_{NDT}) of the steels in the vessel's beltline and how these transition temperatures increase due to irradiation damage throughout the operational life of the vessel. For licensing purposes, 10CFR50.61 provides instructions on how to use these estimates of the effect of irradiation damage on RT_{NDT} to estimate the value of RT_{NDT} that will occur at end of license (EOL), a value called RT_{PTS}. 10CFR50.61 also provides "screening limits," or maximum values of RT_{NDT}, permitted during the operating life of the plant of +270°F (for axial welds, plates, and forgings) and +300°F (for circumferential welds). These screening limits correspond to a limit of $5x10^{-6}$ per reactor year (ry) on the through-wall cracking frequency, or *TWCF* [RG 1.154]. Should RT_{PTS} exceed these screening limits, 10CFR50.61 requires that the licensee either take actions to keep RT_{PTS} below the screening limit (either by implementing "reasonably practicable" flux reductions to reduce the embrittlement rate or by de-embrittling the vessel by annealing [RG 1.162]) or perform plant-specific analysis to demonstrate that operating the plant beyond the screening limit does not pose an undue risk to the public [RG 1.154].

While no PWR currently operating in the United States has an RT_{PTS} value that exceeds these screening limits before EOL, several plants are close to the limit (3 are within 2°F while 10 are within 20°F). These plants are likely to exceed the screening limits during the 20-year license renewal period that is currently being sought by many licensees. Moreover, some plants maintain their RT_{PTS} values below the screening limits by flux reduction (low leakage cores, ultra-low leakage cores), fuel management strategies that can be economically deleterious in a de-regulated marketplace. Thus, the 10CFR50.61 screening limits can restrict the licensable and the economic lifetime of PWRs.

1.3 Motivation for this Project

It is now widely recognized that state of knowledge and data limitations in the early 1980's necessitated a conservative treatment of several key parameters and models used in the probabilistic calculations that provide the technical basis of the current PTS rule [SECY-82-465, ORNL 85a, 85b, 86]. The most prominent of these conservatisms include the following:

- The highly simplified treatment of plant transients (very coarse grouping of many operational sequences (order of 10^5) into very few groups (≈10) necessitated by limitations in the computational resources needed to perform multiple thermal hydraulic calculations;
- The lack of any significant credit for operator action;
- Characterization of fracture toughness using RT_{NDT}, which has an intentional conservative bias;
- The use of a flaw distribution that placed *all* of the flaws on the interior surface of the RPV, and, in general, contains larger flaws than those usually detected in service;
- The modeling approach that treated the RPV as if it were made entirely from the most brittle of its constituent materials (welds, plates, or forgings); and
- The modeling approach that assessed RPV embrittlement using the peak fluence over the entire interior surface of the RPV.

These factors indicate the high likelihood that the current PTS screening limits are unnecessarily conservative. Consequently, it was believed that a re- examination of the technical basis for these screening

limits based on a modern understanding of all the factors that influence PTS would most likely provide strong justification for substantial relaxation of these limits. For these reasons the NRC undertook the *"PTS Re-Evaluation Project"* with the objective of developing the technical basis to support a risk-informed revision of the PTS rule and the associated PTS screening limit.

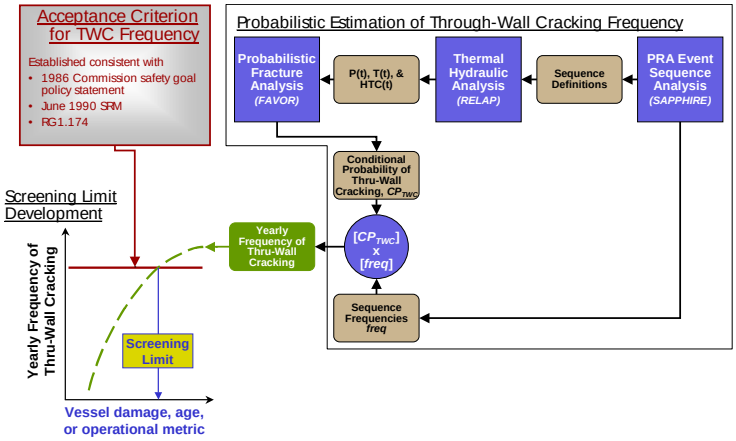

Figure 1. Schematic showing how a probabilistic estimate of TWCF is combined with a TWCF acceptance criterion to arrive at a PTS screening limit.

2 APPROACH

In this section we discuss the model used to estimate PTS risk and our approach to treatment of uncertainties in this model. Additionally we summarize the fundamental assumptions that underlie our calculations.

2.1 Model Used to Evaluate a Revised PTS Screening limit

2.1.1 Restrictions on Model

The desired outcome of this research effort is the establishment of the technical basis for a new PTS screening limit. To enable all operators of commercial PWRs to assess the state of their RPV relative to new screening limits without the need to make new material property measurements, the fracture toughness properties of the RPV steels need to be estimated using only information that is currently available (i.e., RT_{NDT} values, upper shelf energy values, and the chemical composition of the beltline materials), information summarized in the NRC's reactor vessel integrity database [RVID2].

2.1.2 Overall Structure of Model

Our overall model involves three major components, which are illustrated (along with their interactions), in Figure 1:

Component 1. *Probabilistic Evaluation of Through-Wall Cracking Frequency*: Estimate frequency of through-wall cracking as a result of a PTS event given the operating, design, and material conditions in a particular plant.

Component 2. *Acceptance Criterion for Through-wall Cracking Frequency*: Establish a value of reactor vessel failure frequency (*RVFF*) consistent with current guidance on risk-informed decision-making.

Component 3. *Screening Limit Development*: Compare the results of the two preceding steps to determine if some simple, materials-based screening limit for PTS can be established. Conceptually, plants falling below the screening limit would be deemed adequately resistant to a PTS challenge and would not require further analysis. Conversely, more detailed, plant-specific, analysis would be needed to assess the safety of plant operation beyond the screening limit.

In the remainder of this section we describe our approach to Component 1 in greater detail. Further discussion of Components 2 and 3 appear in Sections 3.4 and 4, respectively

As illustrated in Figure 1 there are three main models (shown as solid blue squares) that, together, permit estimation of the *TWCF* in an RPV:

- A probabilistic risk assessment (PRA) event sequence analysis
- A thermal hydraulic (TH) analysis
- A probabilistic fracture mechanics (PFM) analysis

Here we first describe the sequential execution of these three models to highlight the interrelationships and interfaces between them. Second we describe our iterative execution of all three models to establish the final mathematical representation PTS. Third we discuss the three plants analyzed. We conclude with a

discussion of the steps taken to ensure that conclusions based on these three analyses apply to United States PWRs *in general*.

Sequential Execution of Sub-Models to Estimate *TWCF*: First a PRA analysis is performed to define the sequences of events that are likely to produce a PTS challenge to RPV integrity and to estimate the frequency with which such sequences occur. These definitions are passed to a TH model that estimates the temporal variation of temperature, pressure, and heat-transfer coefficient in the RPV downcomer characteristic of each sequence. These time histories are then passed to a PFM model. The PFM model uses this along with other information concerning plant design and materials of construction to estimate the time dependent driving force to fracture produced by a particular event sequence. The PFM model compares this estimate of fracture driving force to the fracture resistance of the RPV steel. This comparison permits estimation of the probability that a particular sequence of events will drive a crack through the RPV wall were that sequence of events actually to occur. The final step involves a matrix multiplication of the probability of through-wall cracking (from PFM) with the frequency of occurrence of a particular event sequence (from PRA). This establishes an estimate of the *TWCF* for a particular sequence in a particular plant after a particular period of operation. The sequence *TWCF*s are then summed for all event sequences to estimate the total *TWCF* for the vessel. Performance of such analyses for various operating lifetimes provides an estimate of how the *TWCF* increases over the lifetime of the plant.

Iterative Process used to Establish Plant-Specific Models: The transients that represent a plant are identified using a PRA analysis, wherein many thousands of different initiating event sequences are "binned" together into groups of sequences believed to produce similar thermal hydraulic outcomes. Judgments regarding what transients to put into what bin are guided by such characteristics as similarity of break size, similarity of operator action, etc. From the many event sequences in each bin a single sequence is selected for analysis by the TH code RELAP to define the variation of pressure, temperature, and heat transfer coefficient *vs*. time used to represent that bin to the PFM analysis performed using the FAVOR code. FAVOR estimates the conditional probability of through wall cracking (*CPTWC*) for each transient. When multiplied by the initiating event frequency estimates from PRA, these *CPTWC* values become *TWCF* values, values that when rank ordered estimate the degree to which each bin contributes to the total *TWCF* of the vessel. At this stage many bins are found to contribute very little or nothing at all to the *TWCF*, and so receive little further scrutiny. However, some bins dominate the *TWCF* estimate. These bins are further subdivided by partitioning the initiating event frequency of the bin, and by selecting a TH transient to represent each part of the original bin. This refined model is then re-analyzed by FAVOR and the bins that provide significant contributions to *TWCF* are again examined. This process of bin partitioning and selection of a TH transient to represent each partitioned bin continues until the total *TWCF* estimated for the plant no longer changes significantly.

Plant Specific Analyses: We performed detailed calculations for three operating PWRs (Oconee 1, Beaver Valley 1, and Palisades), which together sample of a wide range of design and construction methods, and they contain some of the most embrittled RPVs in the United States operating fleet.

Generalization to all U.S. PWRs: Since the objective of this project is to develop a revision to the PTS screening limit that applies *in general* to all PWRs, we must understand to what extent these three analyses address adequately (in either a representative or in a bounding sense) the range of conditions experienced by PWRs in the United States in general. To achieve this goal:

- We perform sensitivity studies on both the TH and PFM models to address the effect of credible changes to these models. The results of these studies provide insights regarding the robustness of our conclusions to the general PWR population.
- We examine the plant design and operational characteristics of five additional high embrittlement plants. Our aim is to determine if the important design and operational features from our three plant specific analyses vary significantly enough in the PWR population to question the generality of our results.
- In our three plant-specific analyses we assume that the only possible origins of PTS events are *internal* to the plant. However, *external* events such as fires, floods, earthquakes, and so on can also be PTS pre-cursors, so we determine if external initiating events create significant additional risk.

2.2 Uncertainty Treatment

In 1999 Siu reviewed the NRC's then-current approach to PTS modeling, focusing on how uncertainties were propagated through the analysis, and how that approach compared with the NRC's guidelines on work supporting risk-informed regulation [Siu 99]. This review established a framework for model development and uncertainty treatment. Here we review this recommended framework and discuss its actual implementation.

2.2.1 Recommended Framework

Probabilistic calculations are performed to establish the technical basis for a revised PTS rule [Woods 01]. Our approach considers a broad range of factors that influence the likelihood of vessel failure during a PTS event

while accounting for uncertainties in these factors across a breadth of technical disciplines [Siu 99]. Two central features of this approach are a focus on the use of realistic input values and models (wherever possible), and an *explicit* treatment of uncertainties. Thus, our current approach improves upon that employed in the development of SECY-82-465 wherein intentional and unquantified conservatisms were included in the many aspects of the analysis, and where uncertainties were treated *implicitly* by incorporating them into the models.

Our probabilistic models distinguish between two types of uncertainties: aleatory and epistemic. Aleatory uncertainties arise due to the randomness inherent to a physical or human process, whereas epistemic uncertainties occur due to a limited in state of knowledge of that process. A practical way to distinguish between uncertainty types is that epistemic uncertainties can, in principle, be reduced by an increased state of knowledge. Conversely, because aleatory uncertainties arise due to randomness at a level below which a particular process is modeled, they are fundamentally irreducible. The distinction of uncertainty type is important because different mathematical procedures are used to represent different uncertainty types.

2.2.2 Implementation

We examine and characterize uncertainties are at two different times: early in the model development stage and later in the project once a complete set of analysis of the three study plants is complete. During model development we first establish a credible mathematical representation (or model) of the physical process of interest (in this case PTS in commercial PWRs operating in the United States). This overall model contains many sub-models and parameter inputs, all of which can (in principle) be subject to uncertainty. These uncertainties are (preferably) quantified or (at a minimum) addressed, as follows:

- Quantification of uncertainties
  - If some physical understanding of the underlying physical process is available it establishes a metric of truth independent of the credible model adopted for calculations. This metric can be used to assess the uncertainty associated with the calculation model.
  - Absent physical understanding, data is used to quantify uncertainties. In this situation it must also be demonstrated that available data characterizes adequately all conditions of interest.
- "Addressing" uncertainties: If physical models or adequate data is not available, it may not be possible to quantify uncertainties even though they exist. In these situations uncertainties can be addressed by several different means:
  - Sometimes uncertainties can be ignored without degrading the robustness of the model. For example, some uncertainties are very small on an absolute basis, and therefore have negligible effects on the estimated risk values. Other uncertainties may not be negligibly small, but are so much smaller than uncertainties represented explicitly in the calculations that they can be justifiably ignored because such omission degrades neither the robustness nor the precision of the calculation.
  - To account for uncertainties that cannot be quantified, intentionally conservative sub-models or inputs can be adopted.

Once a complete set of analyses of the three study plants is completed uncertainties are again examined. First sensitivity studies are performed based on credible perturbations to the baseline model. The need to modify the baseline model and then re-run the analysis is assessed based on the magnitude by which these credible model perturbations produced changes to the *TWCF* values. If these model perturbations produce only small changes to the *TWCF* then all of the residual uncertainties in the analysis are listed, and their magnitude assessed. The analysis is considered complete when both the magnitude the residual uncertainties is small, and when on balance there is judged to be more residual conservatism than residual non-conservatism in the model.

2.3 Fundamental Assumptions / Idealizations

Any mathematical model of a physical system incorporates some level of assumption and/or idealization to enable tractable estimation of the parameters of interest within the resource constraints on the project. As discussed in greater detail elsewhere, each of the PRA, TH, and PFM models involve a large number of sub-models and, thus, a large number of idealizations. Idealizations that occur within each of the PRA, TH, and PFM models are described in detail in elsewhere [Whitehead 05, Bessette 05, EricksonKirk 05b]. The following list summarizes the *fundamental* idealizations that define the PRA, TH, and PFM models *as a whole*.

- PRA
  - Insights from early analyses (Oconee) are used to simplify later analyses (Beaver & Palisades).
  - Operator actions modeled when they significantly alter event severity. For example, operator actions are modeled for stuck open valves but are not modeled for primary pipe breaks.
  - In the absence of good / enough data, systematically conservative judgments are made.
  - Timing of human actions are assumed to be the worst possible.
  - Reduction of ≈10,000 initiating event sequences into ≈100 bins is performed conservatively.

- TH
  - Plant behavior is resolved adequately from the number of thermal hydraulic calculations and corresponding PRA bins.
  - RELAP5 is able to adequately predict downcomer fluid temperature, pressure and heat transfer coefficient.
  - Adequate fluid mixing makes 3D effects in the beltline region negligible.
- PFM
  - A linear elastic fracture mechanics model is appropriate.
  - Sub-critical crack growth is negligible in both the stainless steel cladding and in the ferritic pressure vessel steel due to both environmental mechanisms and due to cyclic loading (fatigue). Consequently, it is appropriate to base risk calculations on a distribution of flaw sizes characteristic of initial fabrication defects.
  - The contribution of certain (small) flaws and (low severity or rare) transients to *TWCF* can be ignored *a priori*.
  - The fracture toughness of the stainless steel cladding is sufficiently high (even after irradiation exposure) that the potential for failure of the cladding by fracture does not need to be considered.

3 SUMMARY OF RESULTS

Here we summarize the results of this multi-year research effort, as follows:

- For the three study plants (Oconee Unit 1, Beaver Valley Unit 1, and Palisades) we describe,
  - Section 3.1: The values of frequency of crack initiation (FCI) and *TWCF* and the characteristics of the distributions from which these values are derived.
  - Section 3.2: We discuss the material features that contribute most significantly to the magnitude of *FCI* and *TWCF*. This leads to a methodology to quantify the embrittlement level of different RPVs on an equivalent basis.
  - Section 3.3: We discuss the classes of transients that contribute most significantly to the PTS risk at a particular plant. This information coupled with our understanding of how to normalize embrittlement levels between different plants permits assessment of the need to consider plant-specific factors when assessing the level of challenge posed to different plants by different transient classes. We examine further the question of the general applicability of these three analyses to all PWRs by considering the characteristics of five other high embrittlement PWRs.
- Section 3.4: We discuss the selection of a vessel failure metric and the establishment of an acceptable limit on the frequency with which vessel failures occur.
- Section 3.5: We assess the balance of conservatisms *vs*. non-conservatisms remaining in our model.

3.1 Estimated *FCI* and *TWCF* Values for the Three Study Plants

Table 1 presents FAVOR Version 04.1 estimates of the mean yearly *FCI* and the mean yearly *TWCF* for Oconee Unit 1, Beaver Valley Unit 1, and Palisades at 32 and 60 EFPY. To estimate values of these metrics close to the *TWCF* limit of 10^{-6} events/year proposed in Section 3.4 it is necessary to increase the amount of irradiation damage beyond that likely during operational lifetimes currently considered possible. To do this we performed analyses for some very long operating lifetimes (designated as Ext-A and Ext-B in the Table 1) thereby increasing the fluence and, consequently, the irradiation damage. The range of irradiation exposures examined includes conditions both below and above the current RT_{PTS} screening limits.

The results in Table 1 demonstrate that even at the end of license extension (EOLE: 60 operational years, or 48 EFPY at an 80% capacity factor) the mean *TWCF* does not exceed $2\text{x}10^{-8}$/year. Considering that the Beaver Valley and Palisades RPVs are constructed from some of the most irradiation sensitive materials in commercial reactor service today, these results suggest that, provided operating practices do not change dramatically in the future, the operating reactor fleet is in little danger of exceeding the *TWCF* acceptance criterion of $5\text{x}10^{-6}$/yr expressed by Regulatory Guide 1.154 [RG 1.154], even after license extension.

In Table 1 mean values of *FCI* and *TWCF* represent the underlying distributions. As illustrated in Figure 2, the *TWCF* distributions are both very broad and highly skewed toward zero. As a consequence of this skewness the mean values of *FCI* and *TWCF* in Table 1 actually bound their underlying distributions at the 90th percentile or greater (see also Figure 2). Both the skewness and the spread in the *TWCF* results arise as a direct consequence of the physical features of cleavage fracture. The absolute lower bounds associated with both the crack initiation fracture toughness (K_{Ic}) and crack arrest fracture toughness (K_{Ia}) distributions causes a large number of the Monte Carlo simulations to produce a through-wall cracking probability that is, *by definition*, zero (exactly zero, not just a very small number). However, on rare occasions the tails of many distributions are sampled, resulting in a larger crack being simulated to occur in a lower toughness material. This combined

possibility of zero and higher probabilities of *TWCF* leads to *TWCF* distributions that are broad and skewed. As also illustrated in Figure 2, the *TWCF* distributions tend to compress as the plants age because the more embrittled materials in these plants are less likely to produce *TWCF* that are either very low, or zero.

Table 1. Mean crack initiation and through-wall cracking frequencies estimated for Oconee Unit 1, Beaver Valley Unit 1, using FAVOR Version 04.1.

Plant	EFPY[1]	RT_{PTS} [°F][2]	Maximum Reference Temperatures [°F]			Mean Yearly Cracking Frequencies	
			for Axial Welds (RT_{MAX-AW})	for Plates (RT_{MAX-PL})	for Circ Welds (RT_{MAX-AW})	Crack Initiation (FCI)	Through-Wall Cracking (*TWCF*)
Oconee	32	221	152	79	175	1.29E-10	2.30E-11
	60	250	171	89	193	1.02E-09	6.47E-11
	Ext-Oa	323	232	136	251	1.01E-07	1.30E-09
	Ext-Ob	329	263	170	281	5.24E-07	1.16E-08
Beaver Valley	32	280	192	243	243	1.32E-07	8.89E-10
	60	299	210	272	272	5.19E-07	4.84E-09
	Ext-Ba	308	225	301	301	1.71E-06	2.02E-08
	Ext-Bb	312	250	354	354	8.87E-06	3.00E-07
Palisades	32	283	212	189	201	5.22E-08	4.90E-09
	60	311	230	205	215	1.23E-07	1.55E-08
	Ext-Pa	358	277	259	259	7.46E-07	1.88E-07
	Ext-Pb	372	333	335	335	4.47E-06	1.26E-06

1. All plants were analyzed for operational durations of 32 and 60 EFPY (or 40 and 75 operational years, respectively, at an 80% capacity factor. Each plant was also analyzed at two extended embrittlement levels (Ext-Oa and Ext-Ob for Oconee, for example) with the aim of obtaining mean through-wall cracking frequency values closer to a *TWCF* value of $1x10^{-6}$/year.
2. RT_{PTS} is defined as per the equations and procedures of 10CFR50.61 based on information in [RVID2].

3.2 The Most Significant Material Factors

When assessing the ability of a structure containing flaws to withstand loading without failure, the location of the flaw or flaws being assessed needs to be known (along with many other factors) to permit estimation of the resistance to fracture of the material at the flaw location. The situation in this study differs somewhat from a routine flaw assessment because the flaws are simulated, and because the effect of hundreds upon thousands of flaws are being assessed (not just one). Nevertheless, in order to correlate and/or predict the *TWCF*, some measure of the resistance of the materials in the vessel to fracture at the location of these many flaws is needed. A reference temperature (*RT*) establishes the resistance of a ferritic steel to fracture, the aleatory uncertainty in this resistance, and how this resistance varies with temperature. A *RT* is commonly thought of as positioning the cleavage fracture toughness transition curve on the temperature axis. However, because relationships exist that establish the position of the arrest transition curve and the upper shelf curve with respect to the cleavage transition curve in a consistent manner for all ferritic steels the temperature dependency of and the aleatory uncertainty in fracture toughness can be descried fully by a single *RT* [Kirk 02, EricksonKirk 04, EricksonKirk 05b]. Since *RT* values can be estimated from information on vessel materials available in the RVID database [RVID2] they provide a way to estimate the resistance of vessel materials to fracture and how this resistance diminishes with increased neutron irradiation.

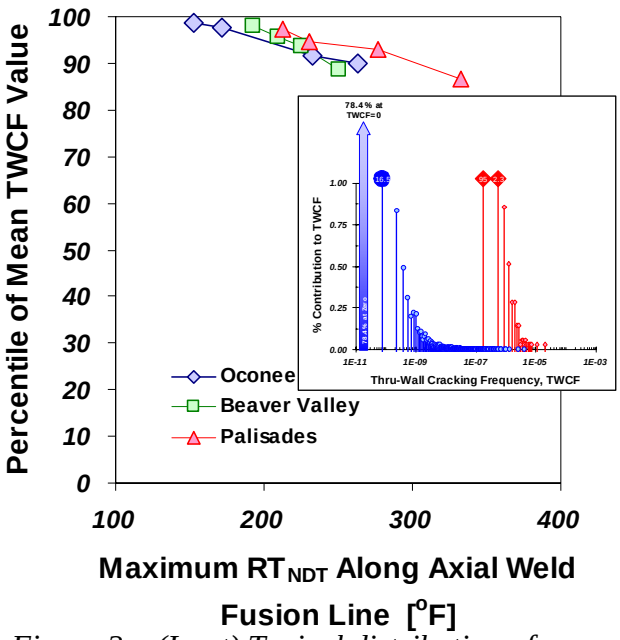

Figure 2. (Inset) Typical distribution of TWCF (as calculated for Beaver Valley at 32 EFPY (blue circles) and for extended embrittlement conditions (red diamonds)). (Main Plot) TWCF distribution percentile corresponding to the mean value.

Having established that a single *RT* value characterizes fracture toughness, the following need to be established:

- The location of the simulated flaws,
- Their size,
- Their orientation with respect to the vessel (axial or circumferential),
- If the flaws break the inner diameter surface of the vessel or are buried under the surface,
- The material properties at the flaw location, and
- The irradiation loading (fluence) at the flaw location.

Figure 3 illustrates the location and orientation of the flaws simulated to exist in the RPV, and the relationship between these flaw locations and the azimuthal and axial variations of fluence [Simonen 03, EricksonKirk 05b]. The information in Figure 3 is summarized as follows:

- Embedded Axial Weld Flaws**:** The overwhelming majority of flaws in axial welds are lack of fusion defects, and (so) occur along the fusion lines and must be axially oriented. The crack initiation and propagation behavior of these flaws is controlled by the less tough of the plate or weld that lie on either side of the flaws. As illustrated in Figure 3, the axial fluence variation is relatively minor along most of the axial weld fusion line length. However, the large azimuthal variation can expose each axial weld fusion line to different fluences. The likelihood of vessel fracture from axial weld flaws therefore depends upon (a) the total number of axial weld flaws (which scales with fusion line area) and, (b) the fluence to which these flaws are subjected.
- Embedded Circumferential Weld Flaws**:** The overwhelming majority of flaws in circumferential welds are lack of fusion defects, and (so) occur along the fusion lines and must be circumferentially oriented. The crack initiation and propagation behavior of these flaws is controlled by the less tough of the plate or weld that lie on either side of the flaws. As illustrated in Figure 3, the azimuthal fluence variation ensures that these circumferential weld flaws will somewhere be subjected to the maximum fluence that occurs anywhere on the vessel ID. Flaws are equally likely to occur at any position around the circumference of the RPV, and the initiation / propagation of fracture from such flaws is more likely at higher fluences. The likelihood of vessel fracture from circumferential weld flaws therefore depends upon (a) the total number of axial weld flaws (which scales with fusion line area) and, (b) the maximum fluence on the vessel ID.

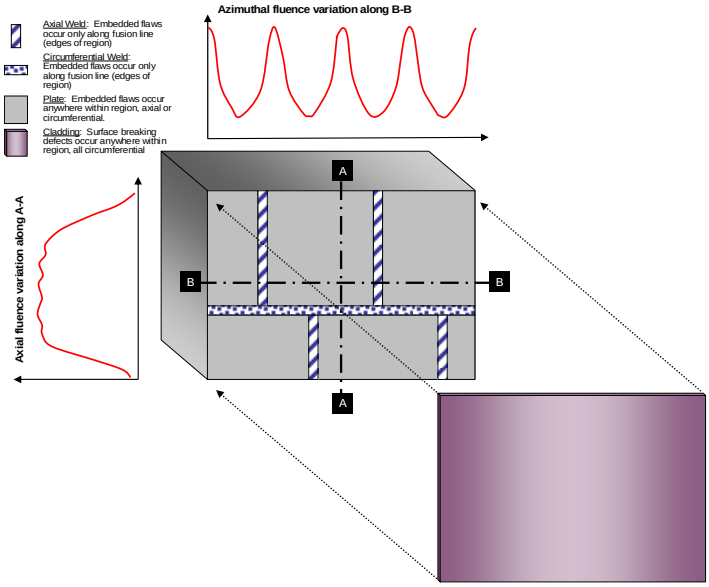

Figure 3. Location and orientation of flaws simulated by FAVOR to exist in different regions of the RPV beltline.

It should be noted that at an equivalent embrittlement level, the likelihood of a circumferential weld flaw leading to through wall cracking is much lower than for an axial weld flaw. Even though circumferential and axial weld flaws drawn from the same size distribution, the variation of crack driving force through the wall of a cylindrical RPV differs considerably for circumferential and for axial flaws. The driving force peak that occurs at approximately ¼-thickness from the vessel ID for circumferential cracks provides a natural crack arrest mechanism that occurs in all RPVs because of their cylindrical geometry. Conversely, the applied driving force for axial flaws continues to increase as their depth increases, which leads directly to the ability of axial flaws to propagate all the way through the RPV wall.

- Embedded Plate Flaws**:** Flaws in plates occur predominantly due to non-metallic inclusions. These can occur anywhere within the plate; they have no preferred orientation (i.e., they are equally likely to be axial or circumferential). As illustrated in Figure 3, the azimuthal fluence variation makes it certain that every plate will somewhere be subjected to the maximum fluence occurring on the vessel ID. Plate flaws are equally likely to occur at any position in the plate, so initiation / propagation of fracture from such flaws is more likely at higher fluences. The likelihood of vessel fracture from plate flaws therefore depends upon (a) the total number of plate flaws (which scales with plate volume) and, (b) the maximum fluence on the vessel ID.

It should be noted that at an equivalent embrittlement level the likelihood of a plate flaw leading to through wall cracking of the vessel is much lower than for an axial weld flaw for two reasons. First, half of

all simulated plate flaws are oriented circumferentially, which reduces their through-wall driving force relative to axial flaws. Additionally, plate flaws are generally much smaller than weld flaws. However, the azimuthal variation of fluence makes it virtually certain that some region of the plates will be subjected to a higher fluence (often a much higher fluence) than will the axial weld fusion lines. At some point this added embrittlement to which the plate flaws are subjected will overcome the smaller plate flaw driving force caused by their smaller size (*vs*. axial weld flaws), causing the fracture of plate flaws to become more likely than the fracture of axial weld flaws.

- Surface Breaking Flaws in the Stainless Steel Cladding**:** The only flaws simulated to break the inner diameter surface of the RPV occur because of lack of inner-run fusion between adjacent beads of weld-deposited stainless steel cladding. Since this cladding is always deposited circumferentially these flaws are always oriented circumferentially; they can occur anywhere over the entire ID surface of the vessel. All of the simulated flaws have a crack depth equal to the thickness of the cladding layer, so the toughness properties that control the crack initiation and propagation of these flaws are those of the axial weld, circumferential weld, or plate region that lie under the simulated location of the surface flaw. FAVOR reports the contribution of these flaws to *FCI* and *TWCF* along with the contribution of the underlying axial weld, circumferential weld, or plate region. Thus the contribution of these flaws to *FCI* and *TWCF* is addressed by the *RT* metrics for these regions, making an independent reference temperature metric for flaws in cladding unnecessary. Furthermore, the circumferential orientation of these flaws makes their contribution to *FCI* and *TWCF* very small.

Because of these differences in flaw size, orientation, material condition, and neutron exposure characteristic of different flaw population it is fundamentally impossible for a single *RT* to represent accurately the resistance of the entire RPV to fracture. Consequently, we developed both weighted average and maximum *RT* metrics to describe the vessel failure probability associated with cracks in each of the axial weld, circumferential weld, and plate regions. The weighted average *RT* metrics were expected to (and indeed did) correlate *TWCF* values better than did the maximum *RT* metrics because the weighted average metrics account more accurately for factors known to effect vessel failure probability. However, to simplify discussions in this short paper and to enable assessment of other PWRs based solely on information available in [RVID2] we focus on the following maximum *RT* metrics:

$$RT_{MAX-AW} \equiv MAX\left\{\left(RT_{NDT(u)}^{plate} + \Delta T_{30}^{plate}\left(\phi t_{FL}\right)\right), \left(RT_{NDT(u)}^{axialweld} + \Delta T_{30}^{axialweld}\left(\phi t_{FL}\right)\right)\right\} \quad (1a)$$

$$RT_{MAX-CW} \equiv MAX\left\{\left(RT_{NDT(u)}^{plate} + \Delta T_{30}^{plate}\left(\phi t_{MAX}\right)\right), \left(RT_{NDT(u)}^{circweld} + \Delta T_{30}^{circweld}\left(\phi t_{MAX}\right)\right)\right\} \quad (1b)$$

$$RT_{MAX-PL} \equiv RT_{NDT(u)}^{plate} + \Delta T_{30}^{plate}\left(\phi t_{MAX}\right) \quad (1c)$$

In eq. (1), $RT_{NDT(u)}$ refers to the RT_{NDT} of the material before irradiation, ΔT_{30} is the shift in the Charpy V-Notch 30 ft-lb energy produced by irradiation, ϕt_{FL} refers to the maximum fluence occurring along a particular axial weld fusion line, and ϕt_{MAX} refers to the maximum fluence occurring over the ID in the vessel beltline region. The superscripts indicate what material region (axial weld, circumferential weld, plate) the parameter describes.

Figure 4 illustrates the *TWCF* due to flaws occurring in each material region (axial welds, circumferential welds, plates) plotted as a function of the eq. (1) *RT* metrics. As expected, axial weld flaws dominate the *TWCF* while plate flaws make a minor contribution (a factor of ≈100 times less than axial weld flaws). The contribution of circumferential weld flaws to *TWCF* (≈5000 times less than that of axial weld flaws) is, for all practical purposes, negligible. The close agreement of *TWCF* at equivalent embrittlement levels between the three plants analyzed suggests that the level of PTS challenge between different plants is in fact quite similar. The reasons for this are discussed in Section 3.3 in greater detail.

3.3 The Most Significant Transient Classes, and the General Applicability of these Results to other United States PWRs

The set of transients that represent a plant are identified using a PRA event tree / fault tree approach, wherein many thousands of different initiating event sequences are "binned" together into groups of transients believed to produce similar thermal hydraulic outcomes. Judgments regarding what transients to put into what bin were guided by such characteristics as similarity of break size, similarity of operator action, etc. Between 30 and 60 bins are used to represent the PTS challenge for a particular plant. The types of transients within these bins fall into the following categories:

Primary Side Faults

- LOCA Pipe breaks of any diameter (1- to 22-in.) on the primary side
- SO-1 Stuck open valves (that may later re-close) on the primary side
- F&B Feed & bleed "LOCA"

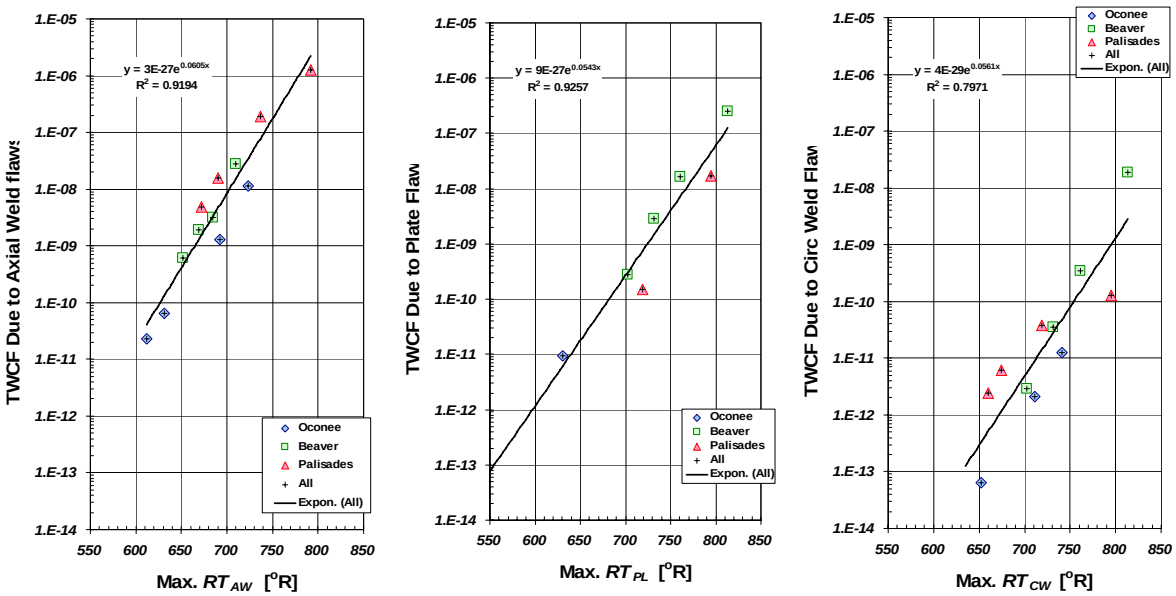

Figure 4. Correlation of through wall cracking frequencies with maximum reference temperature metrics for the three study plants (°R = °F + 459.69).

Secondary Side Faults

- MSLB Large diameter (or "main") steam line break
- SO-2 Smaller diameter secondary side breaks, including stuck open valves
- SGTR Steam generator tube rupture
- OVR Overfeed

Combination of Primary and Secondary Faults

- MIX Mixed primary and secondary faults

The degree to which transients within a particular class contribute to the total PTS risk depends on the likelihood of the transients in the class occurring and on their severity if they occur. The severity of the transients depends on a number of factors, including the cooling rate (rapid cooling rates produce higher thermal stresses than lower cooling rates), the minimum temperature (lower temperatures produce lower fracture toughness), and the existence, or not, of significant pressure loading. Table 2 compares these likelihood and severity factors qualitatively for these different transient classes. Our analysis, which quantitatively accounts for all of these factors, shows the following transient classes of dominate risk:

- Stuck open primary valves that later re-close (Major contribution): For the lower levels of embrittlement currently anticipated for service lifetimes between 40 and 60 operating years, valves on the primary side that stick open and later in the transient re-close (resulting in a late-stage re-pressurization) dominate the *TWCF*. The main factor that make this class of transients dominant is the late-stage re-pressurization associated with valve re-closure. This repressurization comes at a time when the cool-down has only recently stopped (so the thermal stresses are still high) and the temperature in the primary is reduced by the continued injection of cold water due to the stuck-open valve (so fracture resistance is low).
- Medium and large diameter pipe breaks, primary side (Major contribution): At higher levels of embrittlement (not foreseen for currently anticipated operating lifetimes and irradiation damage mechanisms) the high thermal stresses generated by the rapid cool-down of the primary system following a medium or large diameter pipe break can be sufficient to drive a crack all the way through the vessel even though the pressure is low. At high embrittlement levels the *TWCF* of these transients can equal or exceed that caused by stuck open valves that may later re-close.
- Main steam line break (Minor contribution): The break of a main steam line causes very rapid depressurization of the secondary system, and also of the primary due to the coupling between the two. This depressurization causes shrinkage and rapid cooling of the primary. However, these transients make only a minor contribution to the total PTS risk because for secondary side faults the temperature in the primary cannot fall below the boiling point of water at the pressure inside of containment (normally about 250°F). While secondary side faults can be coupled with safety injection directly into the primary, the volume of water injected is miniscule relative to the large heat sink area provided by the

unaffected steam generators. Thus, the temperature of the primary tracks that of the broken generator (between 212 and 250°F). At these temperatures the fracture toughness is sufficient to resist crack initiation and vessel fracture in most cases.

The contribution of all other transient classes to *TWCF* is, for practical purposes, negligible.

Figure 5 compares the contribution of these three transient classes to the total *TWCF*, and how this contribution varies with embrittlement level (as quantified by *RT*). These data show remarkable plant to plant consistency when the *TWCF* results for the three plants analyzed are compared at equivalent embrittlement levels. This consistency is not coincidental. The three classes of transients described above that dominate PTS risk are consistent from plant to plant because either the features of the plant that control transient severity are similar or the transients are little effected by the differences in design, operating procedures, and training that occur in different plants. Certainly plant-specific features can influence how some transients progress, and therefore the level of PTS risk associated with these transients. However, plant-specific factors only have a significant effect on transients that are either low likelihood or low challenge or both, making the effect of plant-specific factors on the total PTS risk negligible. The following paragraphs detail some of the reasons why the most risk-significant classes of transients produce reasonably consistent risk contributions at different plants.

Table 2. Qualitative assessment of the relative contribution of various classes of PTS challenge to overall risk of developing a through wall crack in the RPV.

Transient Class		Transient Likelihood	Transient Severity		
			Cooling Rate	Minimum Temperature	Pressure
Primary Side Pipe Breaks	Large Diameter	Moderate	Fast	Low	Low
	Medium Diameter	Moderate	Moderate	Low	Low
	Small Diameter	High	Slow	High	Moderate
Primary Stuck-Open Valves	Valve Re-closes	Moderate	Slow	Moderate	High
	Valve Remains Open	Moderate	Slow	Moderate	Low
Main Steam Line Break		Low	Fast	Moderate	High
Stuck Open Valve(s), Secondary Side		Low	Moderate	High	High
Feed and Bleed		Low	Slow	Low	Low
Steam Generator Tube Rupture		Low	Slow	High	Moderate
Mixed Primary & Secondary Initiators		Very Low	Slow	Mixed	

Color Key	Increases *TWCF* Contribution	Intermediate	Diminishes *TWCF* Contribution

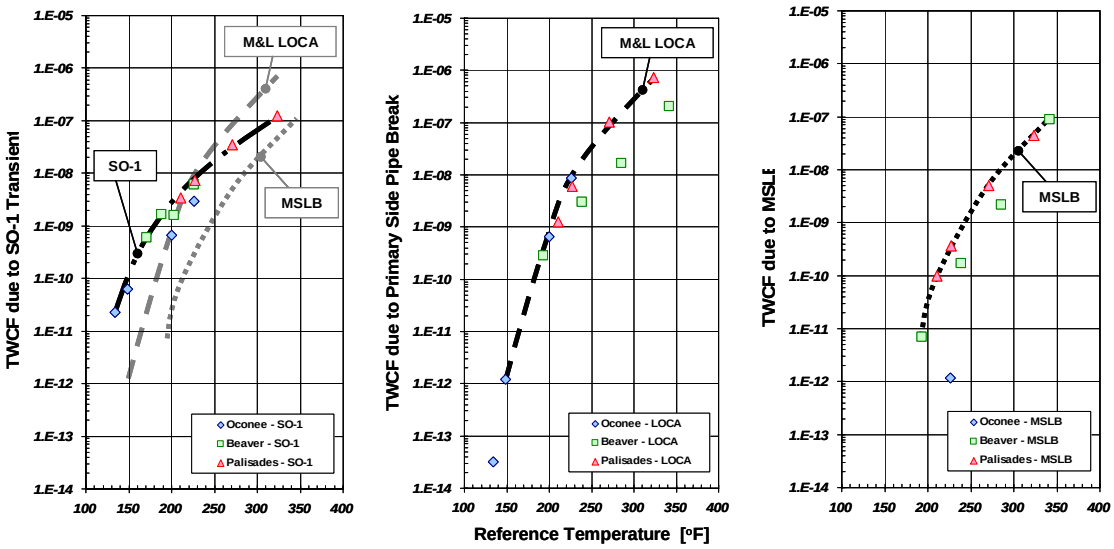

Figure 5. TWCF for different transient classes (left: stuck open primary system valves, middle: medium and large diameter primary system pipe breaks, right: main steam line breaks). On each graph an upper-bound curve is hand drawn to the data. On the left hand graph all three upper-bound curves are placed together for easy comparison.

- Stuck open primary valves that later re-close (Major contribution): As discussed earlier, the cooling rate associated with these transients is modest because the equivalent pipe diameter associated with primary system safety values is in the 2-in. range. An additional consequence of this small "break" size is that the minimum temperature of the primary system is also moderate (in the 100 to 200°F range); it is not nearly as low as that associated with larger breaks where the temperature in the primary can approach that of the emergency core cooling water, which can be as low as 40°F because it is stored in external tanks. These factors do not suggest a high-risk transient. Rather, it is the potential for late-stage re-pressurization that can occur when the valve re-closes that makes this transient class high consequence, and thereby high risk. The actions taken by operators to throttle high pressure injection (HPI) once the valve re-closes can prevent re-pressurization, thereby transforming these transients from being a dominant contributor to risk to being negligible. Thus, it would seem that (in principal) operator action should be an important factor in assessing this transient class. Since operator actions depend on training and on standard operating procedures, and since these factors can vary significantly depending upon both vessel manufacturer and plant management practices, it would therefore seem that (again in principal) the contribution of this transient class to total risk should vary significantly from plant to plant. Such is not the case due to the combined influence of two factors. Operator actions taken to throttle HPI will prevent re-pressurization provided ***both*** of the following conditions are met:
  1. Action to throttle occurs rapidly, i.e. within one minute of the time that the throttling criteria are met. In our analysis of Oconee Unit 1, of Beaver Valley Unit 1, and of Palisades operators were credited for being able to throttle within one minute 68%, 40%, and 0% of the time (respectively). Note that credit was not given to Palisades operators for reasons of computational convenience (i.e., this level of model refinement was not deemed necessary), not because Palisades operators are less competent than those at other plants.
  2. The stuck-open valve transient occurs under hot zero power (HZP) conditions, which in our models was assumed to occur 20% of the time. Under full-power conditions even rapid action to throttle can only delay re-pressurization, it cannot prevent it. This effect of power level on the re-pressurization response occurs because for HZP there is less heat in the system initially, and because the system is colder at the time of valve re-closure. Pressure and temperature are linked, so the need to heat up the colder water and having less heat to do so inhibits the sudden repressurization for HZP conditions.

  Combining these two factors shows that operator actions (which can vary from plant to plant) exert an ≈14%, ≈8%, and ≡0% influence on the *TWCF* at Oconee Unit 1, at Beaver Valley Unit 1, and at Palisades, respectively. These small factors do not influence the risk-significance of this transient class relative to other transient classes. More importantly, these factors do not change the total PTS risk, suggesting the general applicability of these results to all PWRs operating in the United States.
- Medium and large diameter pipe breaks, primary side (Major contribution): Once the diameter of the break exceeds ≈5-in. the rate of cooling of the vessel is limited by the thermal conductivity of the RPV steel (which, being a physical property, varies little from RPV to RPV), ***not*** by the cooling rate of the liquid in the primary. Thus the many factors that can influence the cooling rate of the primary system (seasonal conditions, break location, etc.) have no influence on the cooling rate experienced by the vessel and, thus, no influence on the risk of vessel failure provided the break diameter exceeds ≈5-in. Also, breaks this large cannot be isolated. Because the make-up water systems are inadequate to replace water lost out of the break no operator actions are possible: all the operator can do is continue injection in an effort to keep the core covered.
- Main steam line break (Minor contribution): Because of the high primary system temperatures associated with secondary side breaks, only a ***very*** large breach of the secondary pressure circuit will produce any challenge to the integrity of the RPV. The diameter of the main steam line is well in excess of the ≈5-in. lower-limit on conduction limited conditions just discussed. Thus, while the diameter of the main steam line can vary somewhat from plant to plant, all plants have main steam lines well in excess of 5-in. diameter. For reasons just discussed it is impossible for these variations to have any effect on the cooling rate of the RPV. Also, while there are many actions that an operator can take in response to a secondary side challenge, our calculations show that in the rare circumstances where RPV failure is predicted the failure occurs very quickly (i.e., within 10-15 minutes after transient initiation). Operator actions within this short timeframe are unlikely and, therefore, are not credited by our analysis. Thus, there is no effect of operator action on our estimated risk values.

This examination of the results of our detailed analyses of three plants suggests strongly that plant specific factors do not influence significantly the total PTS risk. To further validate this idea we examined plant design, operation, and training factors at five other plants having high projected levels of embrittlement at EOL (Salem 1,

Three Mile Island 1, Ft. Calhoun, Diablo Canyon 1, and Sequoyah 1). These examinations demonstrate that the great majority of design, operational, and training factors at these plants can be considered to be either well represented or conservatively bounded by the conditions modeled in our detailed analyses. In a few situations a design, operational, or training factor at these five plants was not either well represented or conservatively bounded by the conditions modeled in our detailed analyses. In these cases the results of sensitivity studies demonstrated that these factors do not influence significantly the total PTS risk. Overall, this examination of five additional plants confirms the idea that the results of our detailed analyses of three plants can be used to establish *RT*-based screening limits on PTS that apply to PWRs *in general*.

3.4 Vessel Failure Metric Selection and Acceptable Limits on Failure Frequency

The NRC has established considerable guidance on the use of risk information in regulation since it issued SECY-82-465 and published the original PTS Rule. In light of this more recent guidance, we have identified and assessed options for a risk-informed criterion for the *RVFF* associated with PTS (currently specified in RG 1.154 in terms of a *TWCF*). In this Section we first review current NRC guidance on risk-informed regulations, focusing on quantitative risk metrics. We then summarize the results of a semi-quantitative study concerning containment performance in the aftermath of a PTS accident that produces a through-wall crack because the performance of containment has implications for the specification of the *RVFF* acceptance criterion.

3.4.1 Guidance on Quantitative Risk Metrics

Key documents published since the issuance of the original PTS rule include the Commission's Safety Goal Policy Statement (issued in 1986); a June 1990 Staff Requirements Memorandum (SRM); and RG 1.174 and the associated revision to Standard Review Plan (SRP) Chapter 19.

- The Safety Goal Policy Statement [NRC FR 86] defines quantitative health objectives (QHOs) for the acceptable risk of nuclear power plant operations. The QHOs address the prompt fatality risk to individuals, and the cancer fatality risk to society. For both the individual and societal risks, the QHOs are defined to ensure that the public health and safety risk arising from nuclear power plant operations is a very small (0.1 percent or less) fraction of the total risk to the public.
- The June 1990 SRM [NRC 90] discusses subsequent Commission decisions with respect to the policy statement. Of particular interest, the SRM establishes a subsidiary core damage frequency (CDF) goal of 10^{-4}/ry. At the time it was developed, this subsidiary goal, as well as the qualitative safety goals and QHOs, was intended for use in generic agency decisions such as rulemakings. It was not aimed at plant-specific applications.
- Regulatory Guide 1.174 [RG 1.174] and SRP Chapter 19 [NRC 98b] describe a risk-informed process by which licensee-proposed amendments that act to change regulatory requirements can be submitted, reviewed, and, if appropriate, approved. RG 1.174 applies specifically to voluntary changes to a plant's licensing basis. However, it also provides a general template for improving consistency in regulatory decisions in areas in which the results of risk analyses are used to help justify regulatory action. Of greatest importance to this project, RG 1.174 provides acceptance guidelines for changes in both CDF and large early release frequency (LERF). These guidelines provide assurance that proposed increases in CDF and LERF are small, consistent with the intent of the Safety Goal Policy Statement. According to RG 1.174, if the baseline risk can be shown to be acceptable (meaning that CDF < 10^{-4}/ry and LERF < 10^{-5}/ry) then applications for plant changes leading to small increases in mean CDF (up to 10^{-5}/ry) and mean LERF (up to 10^{-6}/ry) will be considered for regulatory approval. Information in Appendix A of [Pratt 99] links these limits on LERF to a prompt fatality QHO limit (currently around $5\text{x}10^{-7}$/yr) consistent with the Safety Goal Policy Statement.

3.4.2 Semi-Quantitative Assessment of Containment Performance following Through-Wall Cracking

The current limit on *TWCF* of $5\text{x}10^{-6}$/ry provided in RG 1.154 was established to ensure that the risk associated with PTS is a small fraction of the acceptable level of risk established by the Safety Goals, and is consistent with the philosophy of distributing risk among accident types. However, the relationship between the RG 1.154 criterion and the CDF and LERF guidelines established in RG 1.174 is not clear because there is currently an incomplete understanding regarding the progression of an accident following a postulated PTS-induced RPV failure. To gain insight regarding the post-vessel failure consequences of PTS (that is, is core damage likely?, is large early release likely?, etc.) we conducted a scoping study. The study involved the structured identification of technical issues underlying the assessment of the margins to core damage and large early release following PTS-induced RPV failure, and the collection and evaluation of currently available information relevant to these issues. Of particular interest was the identification of PTS-unique physical mechanisms that could lead to dependent failures of accident mitigation features (e.g., containment sprays). To better inform the evaluation, a small number of limited-scope TH and structural calculations were performed.

The scoping study addressed differences between post-PTS induced RPV failure accident progression and accident progression associated with non-PTS core damage events. Thus, issues associated with the development and characteristics of the postulated opening of the RPV due to the through-wall crack, the resulting blowdown forces, the effect on key structural components (e.g., the RPV, containment penetrations, etc.), the potential for damaging missiles, and the potential for increased release fractions of ruthenium due to air oxidation of the fuel were all considered.

To support the systematic identification and semi-quantitative treatment of the post-vessel failure consequences of PTS, an accident progression event tree (APET) was developed. This assessment, and the considerations / analyses that arose from it, led to the following conclusions:

1. *RVFF* can be defined as being equivalent to *TWCF* (i.e., for PTS considerations, RPV "failure" can be defined as an occurrence of a through-wall crack.) *TWCF* is the most appropriate failure metric because it is a more direct measure than crack initiation frequency of the likelihood of events with potentially significant public health consequences to which RG 1.174 requirements are tied, a linkage desirable from a risk-informed decision-making perspective. Additionally, the uncertainties associated with the prediction of a through-wall crack are only slightly larger than those associated with the prediction of crack initiation, and this slight increase in uncertainty is a natural and expected consequence of a cleavage failure mechanism and does not reflect a state of knowledge limitation regarding crack arrest [Kirk 02].
2. An acceptance criterion for *RVFF* of 10^{-6}/ry is consistent with current NRC guidance on risk-informed decisionmaking. This is based on the following observations:
   a. The conditional probability of an un-scrubbed, large early release with a large air-oxidation source term (given a PTS-induced RPV failure) appears to be very small. It is particularly small for plants where water in the reactor cavity (following a PTS-induced RPV failure) will cover the fuel. For plants with larger cavities, the low probability of the scenario is largely due to the independence and reliability of containment sprays.
   b. The assessment underlying the above observation does not account for potential dependencies associated with PTS-events initiated by "external events" (e.g., earthquakes) or internal fires.
   c. For plants with cavities such that fuel cooling is not assured following a PTS-induced RPV failure, our analysis identifies the most probable scenarios where limited fuel damage might occur, even if ECCS operates as designed.

   Observation (a), taken in isolation, supports adoption of a *RVFF* criterion based on considerations of core damage consistent with those RG 1.174 guidance of 10^{-5}/ry. However, Observation (b) identifies a potentially significant uncertainty regarding the margin between PTS-induced RPV failure and large early release, and Observation (c) raises a potential concern regarding defense-in-depth. Therefore RG 1.174 guidelines on CDF supporting an acceptable *RVFF* of 10^{-5}/ry may not have sufficient justification, whereas the scoping study developed for RG 1.174 guidelines on LERF is currently more defensible. This rationale supports use of 10^{-6}/ry as an acceptable maximum for *RVFF*.
3. When assessing the acceptability of the PTS-associated risk at a given plant, the mean value of the plant's PTS-induced *TWCF* should be compared with the 10^{-6}/ry limit.

3.5 Balance of Conservatisms

An explicit aim in this investigation is the use of "best estimates" whenever possible. Nevertheless, for various reasons (lack of knowledge, practical expedience, etc.) certain aspects of our modeling cannot be reasonably represented as "best estimates." As detailed in Table 3, on balance there is a conservative bias to these non-best estimate aspects of our analysis. This, the fact that the *TWCF* values reported here represent 90th percentile values or greater (see Figure 2), and the fact that the information in Section 3.3 demonstrates that these findings can be expected to apply to United States PWRs *in general* suggests that *RT*-based screening limits derived from these results (see Section 4) can be used without the need to apply additional margin.

4 MATERIALS-BASED PTS SCREENING LIMITS

The information presented in Section 3 demonstrated that the challenge to the structural integrity of the RPV posed by the dominant transient classes is approximately equal (at equivalent levels of embrittlement) for the three plants we have modeled in detail. We also identified why the structural integrity challenge posed by these dominant transients are *not expected* to vary from plant to plant, and are *not expected* to be influenced by factors that may differ between the three study plants and the general population of PWRs. Overall, this information supports the use of the *TWCF* values presented in Table 1 together with the *RVFF* acceptance criterion of 10^{-6}/ry proposed in Section 3.4 to develop *RT*-based screening limits that can be used to assess the risk posed by PTS in any commercial PWR operating in the United States. Here we develop such a limits making use of the

maximum *RT* metrics also found in Table 1. As discussed earlier, a *RT* establishes the resistance of a material to fracture, the variability in this resistance, and how this resistance changes with temperature. Since *RT* values can be estimated based on information on vessel materials available in [RVID2] they provide a way to estimate the resistance of vessel materials to fracture and how this resistance diminishes with increased neutron irradiation. Figure 3 showed relationships between $RT_{MAX\text{-}AW}$, $RT_{MAX\text{-}PL}$, and $RT_{MAX\text{-}CW}$ (see eq. (1) for definitions) and the *TWCF* arising due to flaws in axial welds, in plates, and in circumferential welds (respectively). The fits shown in Figure 3 are combined to develop the following formula for estimating *TWCF* based on $RT_{MAX\text{-}AW}$, $RT_{MAX\text{-}PL}$, and $RT_{MAX\text{-}CW}$ values that can be determined solely based on information in [RVID2]:

Table 3. Residual conservatisms and non-conservatisms.

Situation in Service	Potential Conservatism in the Analytical Model
If the vessel fails, what happens next?	The model assumes that all failures produce a large early release, but most sequences lead only to core damage.
	An initiated axial crack is assumed to instantly propagate to infinite length. In reality their length will be finite and limited to the length of a single shell course because the cracks will most likely arrest when they encounter higher toughness materials in either the adjacent circ. welds or plates.
	An initiated circumferential crack is assumed to instantly 360° around the vessel ID. In reality their length is limited because the azimuthal fluence variation places strips of tougher material in the path of the extending crack.
How the many possible PTS initiators are binned, and how TH transient are selected to represent each bin to the PFM analysis.	When uncertainty regarding how to bin existed, consistently conservative decisions were made
Characterization of secondary side failures	The minimum temperature of main steam line break inside containment is modeled as ~50°F colder than it can be because containment pressurizes due to the steam escaping from the break.
	Stuck open valves on the secondary side conservatively modeled in Palisades.
Through-wall attenuation of neutron damage	Attenuation is assumed to be less significant than measured in experiments.
Model of material un-irradiated toughness and chemical composition variability.	The statistical distributions sampled produce more uncertainty than could ever occur in a specific weld, plate, or forging.
Correction for systematic conservative bias in RT_{NDT}	Model corrects for mean bias, but over represents uncertainty in RT_{NDT}.
Embrittlement shift model	Model used produces systematically higher *TWCF* than that estimated by the embrittlement shift model adopted by ASTM.
Flaw model	All defects found assumed to be planar.
	Systematically conservative judgments made when developing flaw distribution model.
Uncertainty in chemistry	The uncertainty in Cu, Ni, and P sampled represents a larger uncertainty than has been demonstrated to exist in any particular weld, plate, or forging.
Interdependency of between initiation toughness and arrest toughness.	Model employed allows all initiated flaws a chance to propagate into the vessel.
Extrapolation of irradiation damage	Most conservative approach taken (increasing time, *vs*. increasing unirradiated RT_{NDT})

Situation in Service	Potential Non-Conservatism in the Analytical Model
If the vessel fails, what happens next?	The potential for air oxidation has been ignored.
Through-wall chemistry layering	Model assumes that the mean level op copper can change 4 times through the vessel wall thickness. If copper layering is not present the *TWCF* would increase.

$$TWCF_{TOTAL} = \alpha_{AW} \cdot TWCF_{AXIAL-WELD} + \alpha_{PL} \cdot TWCF_{PLATE} + TWCF_{CIRC-WELD} \tag{2}$$

where

$$\alpha_{PL} = 1.6,\ TWCF_{AXIAL-WELD} = 3\text{x}10^{-27} \cdot \exp\{0.0605 \cdot (RT_{MAX-AW} + 459.69)\} \tag{2a}$$

$$\alpha_{PL} = 1.7,\ TWCF_{PLATE} = 9\text{x}10^{-27} \cdot \exp\{0.0543 \cdot (RT_{MAX-PL} + 459.69)\} \tag{2b}$$

$$TWCF_{CIRC-WELD} = 4\text{x}10^{-29} \cdot \exp\{0.0561 \cdot (RT_{MAX-CW} + 459.69)\} \tag{2c}$$

(see Section 3.2 for the definitions of $RT_{MAX\text{-}AW}$, $RT_{MAX\text{-}PL}$, and $RT_{MAX\text{-}CW}$)

In eq. (2) the *RT* values are expressed in °F; the formula converts Fahrenheit to Rankine to prevent the introduction of negative numbers to the exponential terms. The *TWCF* attributable to axial weld flaws and to plate flaws are multiplied by factors of 1.6 and 1.7 (respectively) to prevent a systematic under estimation of the *TWCF* results of Palisades and of Beaver Valley (respectively). Averaged across all embrittlement levels, eq. (2) over-predicts the Oconee, Beaver Valley, and Palisades results by 278%, 1%, and 2% (respectively).

A *RT*-based screening limit is established by setting $TWCF_{TOTAL}$ in eq. (2) equal to the 10^{-6}/ry limit. Here we develop a *RT*-based screening limit for plate vessels; similarly based limits for forged vessels are described

elsewhere [EricksonKirk 05a]. Plate vessels are made up of axial welds, plates, and circumferential welds, so in principal flaws in all of these regions contribute to the *TWCF*. However, our results show (see Figure 3) that the contribution of flaws in circumferential welds to *TWCF* is negligible relative to that of flaws in axial welds and in plates. A *RT*-based screening limit for PTS can therefore be derived from eq. (2) by (a) setting $RT_{MAX\text{-}CW}$ to a fixed value, (b) setting $TWCF_{TOTAL}$ to the $1\text{x}10^{-6}$ value developed in Section 3.4, and (c) solving eq. (2) to establish ($RT_{MAX\text{-}AW}$, $RT_{MAX\text{-}PL}$) pairs that satisfy equality. As shown in Figure 6 this procedure establishes a locus of ($RT_{MAX\text{-}AW}$, $RT_{MAX\text{-}PL}$) pairs (**red locus**). In the region of the graph between the **red locus** and the origin the *TWCF* is below the 10^{-6}/ry limit, so these combinations of $RT_{MAX\text{-}AW}$ and $RT_{MAX\text{-}PL}$ would be considered acceptable and require no further analysis. In the region of the graph outside of the **red locus** the *TWCF* is above the 10^{-6}/ry limit, indicating the need for additional analysis or other measures to justify continued plant operation. In Figure 6 these loci are used to assess the proximity of currently operating plate-welded PWRs in the United States to these *RT*-based screening limits. We assess the condition of operating PWRs at EOL (40 years, or 32 EFPY) and at EOLE (60 years of operation, or 48 EFPY). The ID fluence at EOLE is assumed to be 1½ times the value reported in RVID at EOL. The results of these calculations appear as individual assessment points (each point represents a single PWR) on Figure 6. At EOL at least 70°F (and up to 290°F) separate operating PWRs from the proposed screening limit; these values reduce by between 10 and 20°F at EOLE. The wide separation of operating plants at EOL from these *RT* screening limits contrasts sharply with the current regulatory situation where some operating plants lie within less than a single degree Fahrenheit of current 10CFR50.61 RT_{PTS} screening limits. This increase in estimated "distance" from a *RT* screening limit occurs due to the more accurate models and explicit treatment of uncertainties used throughout this investigation.

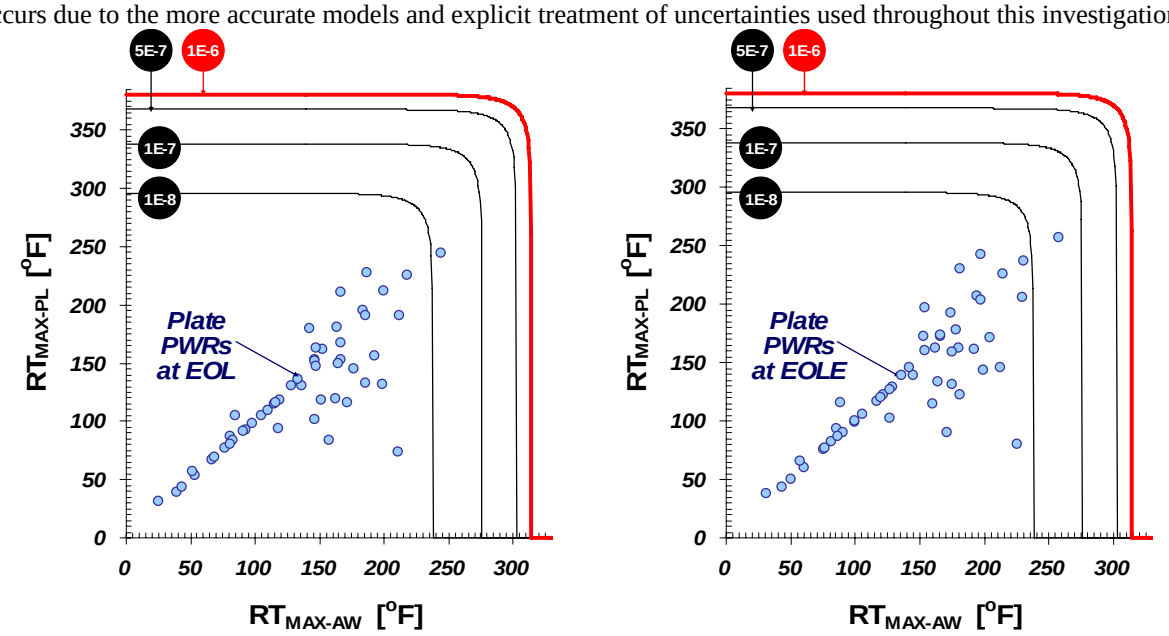

Figure 6. Comparison of a RT_{MAX}-based screening limit for plate vessels (red locus) with assessment points for operating plate-welded PWRs in the United States. Left: at EOL; Right: at EOLE. $RT_{MAX\text{-}CW}$ is 300 °F for both graphs, which exceeds all $RT_{MAX\text{-}CW}$ values at EOL or EOLE.

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