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Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)

Summary Report

U.S. Nuclear Regulatory Commission
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Abstract

During plant operation, the walls of reactor pressure vessels (RPVs) are exposed to neutron radiation, resulting in localized embrittlement of the vessel steel and weld materials in the core area. If an embrittled RPV had a flaw of critical size and certain severe system transients were to occur, the flaw could very rapidly propagate through the vessel, resulting in a through-wall crack and challenging the integrity of the RPV. The severe transients of concern, known as pressurized thermal shock (PTS), are characterized by a rapid cooling of the internal RPV surface in combination with repressurization of the RPV. Advancements in our understanding and knowledge of materials behavior, our ability to realistically model plant systems and operational characteristics, and our ability to better evaluate PTS transients to estimate loads on vessel walls led the NRC to realize that the earlier analysis, conducted in the course of developing the PTS Rule in the 1980s, contained significant conservatisms.

This report summarizes 21 supporting documents that describe the procedures used and results obtained in the probabilistic risk assessment, thermal hydraulic, and probabilistic fracture mechanics studies conducted in support of this investigation. Recommendations on toughness-based screening criteria for PTS are provided.

Foreword

The reactor pressure vessel is exposed to neutron radiation during normal operation. Over time, the vessel steel becomes progressively more brittle in the region adjacent to the core. If a vessel had a preexisting flaw of critical size and certain severe system transients occurred, this flaw could propagate rapidly through the vessel, resulting in a through-wall crack. The severe transients of concern, known as pressurized thermal shock (PTS), are characterized by rapid cooling (i.e., thermal shock) of the internal reactor pressure vessel surface that may be combined with repressurization. The simultaneous occurrence of critical-size flaws, embrittled vessel, and a severe PTS transient is a very low probability event. The current study shows that U.S. pressurized-water reactors do not approach the levels of embrittlement to make them susceptible to PTS failure, even during extended operation well beyond the original 40-year design life.

Advancements in our understanding and knowledge of materials behavior, our ability to realistically model plant systems and operational characteristics, and our ability to better evaluate PTS transients to estimate loads on vessel walls have shown that earlier analyses, performed some 20 years ago as part of the development of the PTS rule, were overly conservative, based on the tools available at the time. Consistent with the NRC's Strategic Plan to use best-estimate analyses combined with uncertainty assessments to resolve safety-related issues, the NRC's Office of Nuclear Regulatory Research undertook a project in 1999 to develop a technical basis to support a risk-informed revision of the existing PTS Rule, set forth in Title 10, Section 50.61, of the Code of Federal Regulations (10 CFR 50.61).

Two central features of the current research approach were a focus on the use of realistic input values and models and an explicit treatment of uncertainties (using currently available uncertainty analysis tools and techniques). This approach improved significantly upon that employed in the past to establish the existing 10 CFR 50.61 embrittlement limits. The previous approach included unquantified conservatisms in many aspects of the analysis, and uncertainties were treated implicitly by incorporating them into the models.

This report summarizes a series of 21 reports that provide the technical basis that the staff will consider in a potential revision of 10 CFR 50.61; it includes a description of analysis procedures and a detailed discussion of findings. The risk from PTS was determined from the integrated results of the Fifth Version of the Reactor Excursion Leak Analysis Program (RELAP5) thermal-hydraulic analyses, fracture mechanics analyses, and probabilistic risk assessment. These calculations demonstrate that, even through the period of license extension, the likelihood of vessel failure attributable to PTS is extremely low ($\approx 10^{-8}$/year) for all domestic pressurized water reactors. Limited analyses are continuing to further evaluate this finding. Should the $\approx 10^{-8}$/year value be confirmed, this would provide a basis for significant relaxation, or perhaps elimination, of the embrittlement limit established in 10 CFR 50.61. Such changes would reduce unnecessary conservatism without affecting safety because the operating reactor fleet has little probability of exceeding the limits on the frequency of reactor vessel failure established from NRC guidelines on core damage frequency and large early release frequency through the period of license extension.

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Contents

Appendices

Figures

Tables

Executive Summary

This report summarizes the results of a 5-year study conducted by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research (RES). The aim of this study was to develop the technical basis for revision of the Pressurized Thermal Shock (PTS) Rule, as set forth in Title 10, Section 50.61, of the *Code of Federal Regulations* (10 CFR 50.61), "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," consistent with the NRC's current guidelines on risk-informed regulation. This report, together with other supporting reports documenting the study details and results, provides this basis.

This executive summary begins with a description of PTS, how it might occur, and its potential consequences for the reactor pressure vessel (RPV). 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. Following this introductory information, we describe the approach used to conduct the study, and summarize our key findings and recommendations, which include a proposal for revision of the PTS screening limits. We then conclude the executive summary with a discussion of the potential impact of this proposal on regulations other than 10 CFR 50.61.

Description of PTS

During the operation of a nuclear power plant, the RPV walls are exposed to neutron radiation, resulting in localized embrittlement of the vessel steel and weld materials in the area of the reactor core. If an embrittled RPV had an existing flaw of critical size and certain severe system transients were to occur, the flaw could propagate very rapidly through the vessel, resulting in a through-wall crack and challenging the integrity of the RPV. The severe transients of concern, known as pressurized thermal shock (PTS), are characterized by a rapid cooling (i.e., thermal shock) of the internal RPV surface and downcomer, which may be followed by repressurization of the RPV. Thus, a PTS event poses a potentially significant challenge to the structural integrity of the RPV in a pressurized-water reactor (PWR).

A number of abnormal events and postulated accidents have the potential to thermally shock the vessel (either with or without significant internal pressure). These events include a pipe break or stuck-open valve in the primary pressure circuit, a break of the main steam line, etc. During such events, the water level in the core drops as a result of the contraction produced by rapid depressurization. In events involving a break in the primary pressure circuit, an additional drop in water level occurs as a result of leakage from the break. Automatic systems and operators must provide makeup water in the primary system to prevent overheating of the fuel in the core. However, the makeup water is much colder than that held in the primary system. As a result, the temperature drop produced by rapid depressurization coupled with the near-ambient temperature of the makeup water produces significant thermal stresses in the thick section steel wall of the RPV. For embrittled RPVs, these stresses could be sufficient to initiate a running crack, which could propagate all the way through the vessel wall. Such through-wall cracking of the RPV could precipitate core damage or, in rare cases, a large early release of radioactive material to the environment. Fortunately, the coincident occurrence of critical-size flaws, embrittled vessel steel and weld material, and a severe PTS transient is a very low-probability event. In fact, only a few currently operating PWRs are projected to closely approach the current statutory limit on the level of embrittlement during their planned operational life.

Current Regulatory Approach to PTS

As set forth in 10 CFR 50.61, the PTS Rule requires licensees to monitor the embrittlement of their RPVs using a reactor vessel material surveillance program qualified under Appendix H to 10 CFR Part 50, "Reactor Vessel Material Survellience Program Requirements." The surveillance results are then used together with the formulae and tables in 10 CFR 50.61 to estimate the fracture toughness transition temperature (RT_{NDT}) of the steels in the vessel's beltline and how those transition temperatures increase as a result of irradiation damage throughout the operational life of the vessel. For licensing purposes, 10 CFR 50.61 provides instructions on how to use these estimates of the effect of irradiation damage to estimate the value of RT_{NDT} that will occur at end of license (EOL), a value called RT_{PTS}. 10 CFR 50.61 also provides "screening limits" (maximum values of RT_{NDT} permitted during the plant's operational life) of +270°F (132°C) for axial welds, plates, and forgings, and +300°F (149°C) for circumferential welds. These screening limits correspond to a limit of 5×10^{-6} events/year on the annual probability of developing a through-wall crack [RG 1.154]. Should RT_{PTS} exceed these screening limits, 10 CFR 50.61 requires the licensee to either take actions to keep RT_{PTS} below the screening limit (by implementing "reasonably practicable" flux reductions to reduce the embrittlement rate, or by deembrittling the vessel by annealing [RG 1.162]), or perform plant-specific analyses to demonstrate that operating the plant beyond the 10 CFR 50.61 screening limit does not pose an undue risk to the public [RG 1.154].

While no currently operating PWR has an RT_{PTS} value that exceeds the 10 CFR 50.61screening limit before EOL, several plants are close to the limit (3 are within 2°F, while 10 are within 20°F). Those plants are likely to exceed the screening limit during the 20-year license renewal period that is currently being sought by many operators. Moreover, some plants maintain their RT_{PTS} values below the 10 CFR 50.61 screening limits by implementing flux reductions (low-leakage cores, ultra-low-leakage cores), which are fuel management strategies that can be economically deleterious in a deregulated marketplace. Thus, the 10 CFR 50.61 screening limits can restrict both the licensable and economic lifetime of PWRs.

Motivation for this Project

It is now widely recognized that the state of knowledge and data limitations in the early 1980s necessitated conservative treatment of several key parameters and models used in the probabilistic calculations that provided the technical basis for the current PTS Rule. The most prominent of these conservatisms include the following factors:

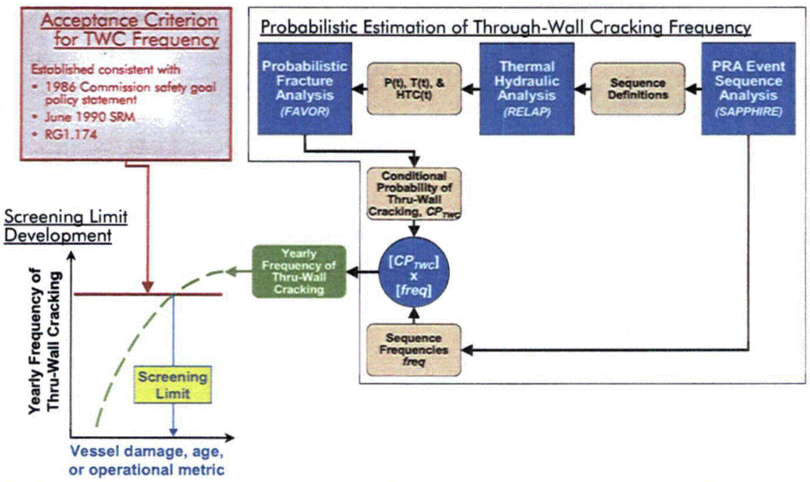
- highly simplified treatment of plant transients (very coarse grouping of many operational sequences (on the order of 10^5) into very few groups (≈ 10), necessitated by limitations in the computational resources needed to perform multiple thermal-hydraulic calculations)
- lack of any significant credit for operator action
- characterization of fracture toughness using RT_{NDT}, which has an intentional conservative bias
- use of a flaw distribution that places *all* flaws on the interior surface of the RPV, and, in general, contains larger flaws than those usually detected in service
- a modeling approach that treated the RPV as if it were made entirely from the most brittle of its constituent materials (welds, plates, or forgings)
- a 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 10 CFR 50.61 PTS screening limits are unnecessarily conservative. Consequently, the NRC staff believed that reexamining 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 substantially relaxing these limits. For these reasons, RES undertook this study with the objective of developing the technical basis to support a risk-informed revision of the PTS Rule and the associated PTS screening limit.

Approach

As illustrated in the following figure, three main models (shown as solid blue squares), taken together, allow us to estimate the annual frequency of through-wall cracking in an RPV:

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



Schematic showing how a probabilistic estimate of through-wall cracking frequency (TWCF) is combined with a TWCF acceptance criterion to arrive at a proposed revision of the PTS screening limit

First, a PRA event sequence analysis is performed to define the sequences of events that are likely to cause a PTS challenge to RPV integrity, and estimate the frequency with which such sequences can be expected to occur. The event sequence definitions are then passed to a TH model that estimates the temporal variation of temperature, pressure, and heat-transfer coefficient in the RPV downcomer, which is characteristic of each sequence definition. These temperature, pressure, and heat-transfer coefficient histories are then passed to a PFM model that uses the TH output, along with other information concerning plant design and construction materials, to estimate the time-dependent "driving force to fracture" produced by a particular event sequence. The PFM model then compares this estimate of fracture driving force to the fracture toughness, or fracture resistance, of the RPV steel. This comparison allows us to estimate the probability that a crack could grow to sufficient size that it would penetrate all the way through the RPV wall if that particular sequence of events actually occured. The final step in the analysis involves a simple matrix multiplication of the probability of through-wall cracking (from the PFM analysis) with the frequency at which a particular event sequence is expected to occur (as defined by the event-tree analysis). This product establishes an estimate of the annual frequency of through-wall cracking that can be expected for a particular plant after a particular period of operation when subjected to a particular sequence of events. The

annual frequency of through-wall cracking is then summed for all event sequences to estimate the total annual frequency of through-wall cracking for the vessel. Performance of such analyses for various operating lifetimes provides an estimate of how the annual frequency of through-wall cracking can be expected to vary over the lifetime of the plant.

The probabilistic calculations just described are performed to establish the technical basis for a revised PTS Rule within an integrated systems analysis framework. 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. 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 (using currently available uncertainty analysis tools and techniques). Thus, our current approach improves upon that employed in developing SECY-82-465, which included intentional and unquantified conservatisms in many aspects of the analysis, and treated uncertainties *implicitly* by incorporating them into the models.

Key Findings

The findings from this study are divided into the following five topical areas: (1) the expected magnitude of the through-wall cracking frequency (TWCF) for currently anticipated operational lifetimes, (2) the material factors that dominate PTS risk, (3) the transient classes that dominate PTS risk, (4) the applicability of these findings (based on detailed analyses of three PWRs) to PWRs *in general*, and (5) the annual limit on TWCF established consistent with current guidelines on risk-informed regulation. In this summary, ***the conclusions are presented in boldface italic***, while the supporting information is shown in regular type.

TWCF Magnitude for Currently Anticipated Operational Lifetimes

- ***The degree of PTS challenge is low for currently anticipated lifetimes and operating conditions.***
  - Even at the end of license extension (60 operational years, or 48 effective full-power years (EFPY) at an 80% capacity factor), the mean estimated TWCF does not exceed $2x10^{-8}$/year for the plants analyzed. Considering that the RPVs at the Beaver Valley Power Station and Palisades Nuclear Power Plant are constructed from some of the most irradiation-sensitive materials in commercial reactor service today, these results suggest that, provided that operating practices do not change dramatically in the future, the operating reactor fleet is in little danger of exceeding either the TWCF limit of $5x10^{-6}$/yr expressed by Regulatory Guide 1.154 [RG 1.154] or the value of $1x10^{-6}$/yr recommended in Chapter 10 of this report — even after license extension.

Material Factors and their Contributions to PTS Risk

- ***Axial flaws, and the toughness properties that can be associated with such flaws, control nearly all of the TWCF.***
  - Axial flaws are much more likely than circumferential flaws to propagate through the RPV wall because the applied fracture driving force increases continuously with increasing crack depth for an axial flaw. Conversely, circumferentially oriented flaws experience a driving force peak mid-wall, providing a natural crack arrest mechanism. It should be noted that crack initiation from circumferentially oriented flaws is likely; it is only their through-wall propagation that is much less likely (relative to axially oriented flaws).
  - It is, therefore, the toughness properties that can be associated with axial flaws that control nearly all of the TWCF. These include the toughness properties of plates and axial welds at the flaw locations. Conversely, the toughness properties of both circumferential welds and forgings have little effect on the TWCF because these can be associated only with circumferentially oriented flaws.

Transients and their Contributions to PTS Risk

- ***Transients involving primary side faults are the dominant contributors to TWCF, while transients involving secondary side faults play a much smaller role.***
    - The severity of a transient is controlled by a combination of three factors:
        - initial cooling rate, which controls the thermal stress in the RPV wall
        - minimum temperature of the transient, which controls the resistance of the vessel to fracture
        - pressure retained in the primary system, which controls the pressure stress in the RPV wall
    - The significance of a transient (i.e., how much it contributes to PTS risk) depends on these three factors and the likelihood that the transient will occur.
    - Our analysis considered transients in the following classes (as shown in the following table):
        - primary side pipe breaks
        - stuck-open valves on the primary side
        - main steam line breaks
        - stuck-open valves on the secondary side
        - feed-and-bleed
        - steam generator tube rupture
        - mixed primary and secondary initiators

Factors contributing to the severity and risk-dominance of various transient classes

Transient Class		Transient Severity			Transient Likelihood	TWCF Contribution
		Cooling Rate	Minimum Temperature	Pressure		
Primary Side Pipe Breaks	Large-Diameter	Fast	Low	Low	Low	Large
	Medium-Diameter	Moderate	Low	Low	Moderate	Large
	Small-Diameter	Slow	High	Moderate	High	~0
Stuck-Open Valves, Primary Side	Valve Recloses	Slow	Moderate	High	High	Large
	Valve Remains Open	Slow	Moderate	Low	High	~0
Main Steam Line Break		Fast	Moderate	High	High	Small
Stuck-Open Valve(s), Secondary Side		Moderate	High	High	High	~0
Feed-and- Bleed		Slow	Low	Low	Low	~0
Steam Generator Tube Rupture		Slow	High	Moderate	Low	~0
Mixed Primary & Secondary Initiators		Slow	Mixed		Very Low	~0

Color Key	Enhances TWCF Contribution	Intermediate	Diminishes TWCF Contribution

-
    - The table above provides a qualitative summary our results for these transient classes in terms of both transient severity and the likelihood that the transient will occur. The color-coding of table entries indicates the contribution (or lack thereof) of these factors to the TWCF of the various classes of transients. This summary indicates that the risk-dominant transients (medium- and large-diameter primary side pipe breaks, and stuck-open primary side valves that later reclose) all have multiple factors that, in combination, result in their significant contributions to TWCF.
        - For medium- to large-diameter primary side pipe breaks, the fast to moderate cooling rates and low downcomer temperatures (generated by rapid depressurization and emergency injection of low-temperature makeup water directly to the primary) combine to produce a high-severity transient. Despite the moderate to low likelihood that these transients will occur, their severity (if they do occur) makes them significant contributors to the total TWCF.

- For stuck-open primary side valves that later reclose, the repressurization associated with valve reclosure coupled with low temperatures in the primary combine to produce a high-severity transient. This, coupled with a high likelihood of transient occurrence, makes stuck-open primary side valves that later reclose significant contributors to the total TWCF.
- The small or negligible contribution of all secondary side transients (main steam line break, stuck-open secondary valves) results directly from the lack of low temperatures in the primary system. For these transients, the minimum temperature of the primary for times of relevance is controlled by the boiling point of water in the secondary (212°F (100°C) or above). At these temperatures, the fracture toughness of the RPV steel is sufficiently high to resist vessel failure in most cases.

Applicability of These Findings to PWRs in General

- ***Credits for operator action, while included in our analysis, do not influence these findings in any significant way.*** Operator action credits can dramatically influence the risk-significance of ***individual*** transients. Therefore, appropriate credits for operator action need to be included as part of a "best estimate" analysis because there is no way to establish *a priori* if a particular transient will make a large contribution to the total risk. Nonetheless, the results of our analyses demonstrate that these operator action credits have a small ***overall effect*** on a plant's ***total TWCF***, for reasons detailed below.
  - Medium- and Large-Diameter Primary Side Pipe Breaks: No operator actions are modeled for any break diameter because, for these events, the safety injection systems do not fully refill the upper regions of the reactor coolant system (RCS). Consequently, operators would never take action to shut off the pumps.
  - Stuck-Open Primary Side Valves that May Later Reclose: Reasonable and appropriate credit for operator actions (throttling of the high-pressure injection (HPI) system) has been included in the PRA model. However, these credits have a small influence on the estimated values of vessel failure probability attributable to transients caused by a stuck-open valve in the primary pressure circuit (SO-1 transients) because the credited operator actions only prevent repressurization when SO-1 transients initiate from Hot Zero Power (HZP) conditions and when the operators act promptly (within 1 minute) to throttle the HPI. Complete removal of operator action credits from the model only slightly increases the total risk associated with SO-1 transients.
  - Main Steam Line Breaks: For the overwhelming majority of transients caused by a main steam line break (MSLB), vessel failure is predicted to occur between 10 and 15 minutes after transient initiation because the thermal stresses associated with the rapid cooldown reach their maximum within this timeframe. Thus, all of the long-term effects (isolation of feedwater flow, timing of HPSI control) that can be influenced by operator actions have no effect on vessel failure probability because such factors influence the progression of the transient *after failure has occurred* (if it occurs at all). Only factors affecting the initial cooling rate (i.e., plant power level at time of transient initiation, break location inside or outside of containment) can influence the conditional probability of through-wall cracking (CPTWC), and operator actions do not influence such factors in any way.
- ***Because the severity of the most significant transients in the dominant transient classes is controlled by factors that are common to PWRs in general, the TWCF results presented herein can be used with confidence to develop revised PTS screening criteria that apply to the entire fleet of operating PWRs.***
  - Medium- and Large-Diameter Primary Side Pipe Breaks: For these break diameters, the fluid in the primary cools faster than the wall of the RPV. In this situation, ***only*** the thermal conductivity of the steel and the thickness of the RPV wall control the thermal stresses and, thus, the severity of the fracture challenge. Perturbations in the fluid cooldown rate controlled by break diameter, break location, and season of the year do not play a role. Thermal conductivity is a physical property,

so it is very consistent for all RPV steels, and the thicknesses of the three RPVs analyzed are typical of PWRs. Consequently, the TWCF contribution of medium- to large-diameter primary side pipe breaks is expected to be consistent from plant-to-plant and can be well represented for all PWRs by the analyses reported herein.

  - Stuck-Open Primary Side Valves that May Later Reclose: A major contributor to the risk-significance of SO-1 transients is the return to full system pressure once the valve recloses. The operating and safety relief valve pressures of all PWRs are similar. Additionally, as previously noted, operator action credits only slightly affect the total risk associated with this transient class.

  - Main Steam Line Breaks: Since MSLBs fail early (within 10–15 minutes after transient initiation), only factors affecting the initial cooling rate can have any influence on the CPTWC values. These factors, which include the plant power level at event initiation and the location of the break (inside or outside of containment), are not influenced by operator actions in any way.

- ***Sensitivity studies performed on the TH and PFM models to investigate the effect of credible model variations on the predicted TWCF values revealed no effects significant enough to recommend changes to the baseline RELAP and FAVOR models, or to recommend cautions regarding the robustness of those models.***

- ***An investigation of design and operational characteristics for five additional PWRs revealed no differences in sequence progression, sequence frequency, or plant thermal-hydraulic response significant enough to call into question the applicability of the TWCF results from the three detailed plant analyses to PWRs in general.***

- ***An investigation of potential external initiating events (e.g., fires, earthquakes, floods) revealed that the contribution of those events to the total TWCF can be regarded as negligible.***

Annual Limit on TWCF

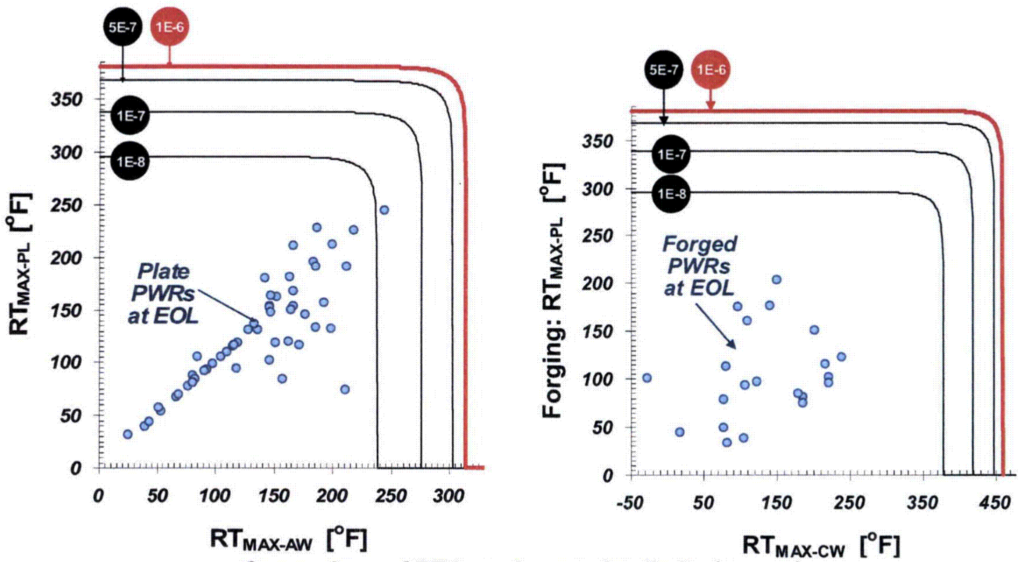
- ***The current guidance provided by Regulatory Guide 1.174 [RG 1.174] for large early release is appropriately applied to setting an acceptable annual TWCF limit of $1x10^{-6}$ events/year.***

  - While many post-PTS accident progressions led only to core damage (which suggests a TWCF limit of $1x10^{-5}$ events/year limit in accordance with Regulatory Guide 1.174), uncertainties in the accident progression analysis led to our recommendation to adopt the more conservative limit of $1x10^{-6}$ events/year based on LERF.

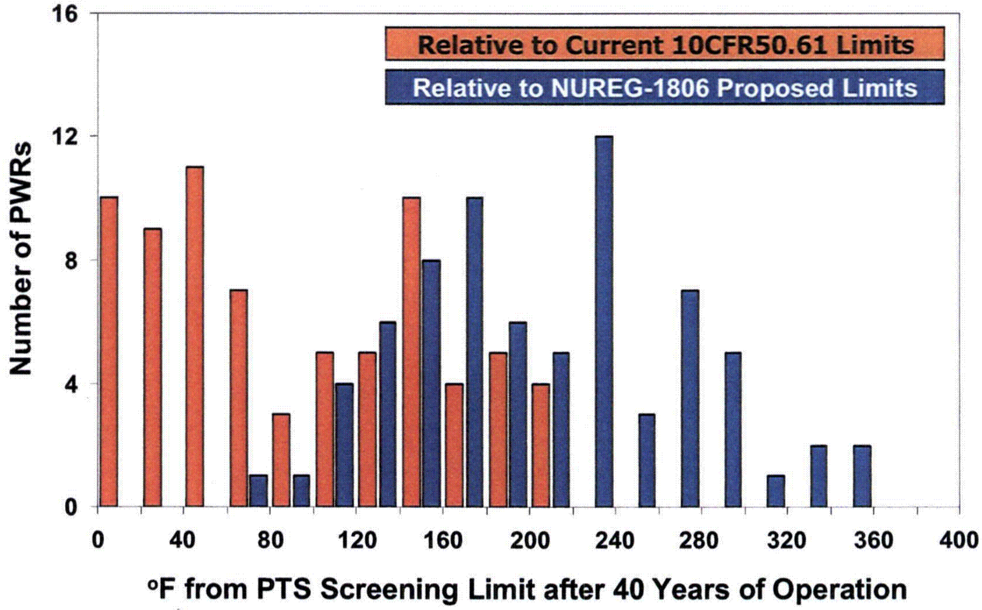
Recommended Revision of the PTS Screening Limits

We recommend using different reference temperature (*RT*) metrics to characterize an RPV's resistance to fractures initiating from different flaws at different locations in the vessel. Specifically, we recommend a reference temperature for flaws occurring along axial weld fusion lines (RT_{AW} or $RT_{AW\text{-}MAX}$), another for flaws occurring in plates or in forgings (RT_{PL} or $RT_{PL\text{-}MAX}$), and a third for flaws occurring along circumferential weld fusion lines (RT_{CW} or $RT_{CW\text{-}MAX}$). In each of these reference temperature pairs, the first metric is a weighted value that accounts for the differences between plants in weld fusion line area or plate volume, while the second metric is a maximum value that can be estimated based only on the information in the NRC's Reactor Vessel Integrity Database (RVID). We also recommend using different *RT* values together to characterize the fracture resistance of the vessel's beltline region, in recognition of the fact that the probability of vessel fracture initiating from different flaw populations varies considerably in response to factors that are both understood and predictable. Correlations between these *RT* metrics and the TWCF attributable to axial weld flaws, plate flaws, and circumferential weld flaws show little plant-to-plant variability because of the general similarity of PTS challenges among plants.

RT-based screening limits were established by setting the total TWCF (i.e., that attributable to axial weld flaws and plate flaws and circumferential weld flaws) equal to the reactor vessel failure frequency acceptance criterion of 1×10^{-6} events per year. The following figures graphically represent these screening limits (for the maximum *RT* metrics), along with an assessment of all operating PWRs relative to these limits. In these figures, the region of the graphs between the red locus and the origin has TWCF values below the 1×10^{-6} acceptance criterion, so these combinations of reference temperatures would be considered acceptable and require no further analysis. By contrast, the region of the graph outside of the red locus has TWCF values above the 1×10^{-6} acceptance criterion, indicating the need for additional analysis or other measures to justify continued plant operation. Clearly, operating PWRs do not closely approach the 1×10^{-6}/year limit. At EOL, at least 70°F, and up to 290°F, (39 to 161°C) separate plate-welded PWRs from the proposed screening limit; this separation between plant-specific values and the proposed screening limit reduces by 10–20°F (5.5 to 11°C) at end of license extension (EOLE, defined as 60 operating years or 48 EFPY). Additionally, no forged plant is anywhere close to the limit of 1×10^{-6} events per year at either EOL or EOLE. This separation of operating plants from the screening limit contrasts markedly with the current situation, where the most embrittled plants are within 1°F (0.5°C) of the screening limit set forth in 10 CFR 50.61. These differences in the "proximity" of operating plants to the current (10 CFR 50.61) and proposed screening limits are illustrated by the bar graph on the next page.



Comparison of RT-based screening limits (curves) with assessment points for operating PWRs at EOL (Left: plate vessels, Right: forged vessels)



Difference between the proximity of operating PWRs to the current RT_{PTS} screening limits and to the screening limits proposed based on the work presented in this report.

These *RT*-based screening limits (and similar limits described in the text for application to weighted *RT* values) apply to PWRs in general, subject only to the following provisos:

- When assessing a forged vessel where the forging has a very high reference temperature (RT_{PL} above 225°F (107°C)) ***and*** the forging is believed to be susceptible to subclad cracking, a plant-specific analysis of the TWCF produced by the subclad cracks should be performed. However, no forging is projected to reach this level of embrittlement, even at EOLE.
- When assessing an RPV having a wall thickness of 7-in. (18-cm) or less (7 vessels), the proposed *RT* limits are conservative.
- When assessing an RPV having a wall thickness of 11-in. (28-cm) or greater, the proposed *RT* limits may be nonconservative. For the three plants meeting this criterion, either the *RT* limits would need to be reduced or known conservatisms in the current analysis would have to be removed to demonstrate compliance with the TWCF limit of 1×10^{-6} event/year. However, because these three plants are Units 1, 2, and 3 of the Palo Verde Nuclear Generating Station, which have vessels with very low embrittlement projected at EOL and EOLE, there is little practical need for such plant-specific analysis.

Aside from relying on different *RT* metrics than 10 CFR 50.61, this proposed revision of the PTS screening limit differs from the current screening limit in the absence of a "margin term." Use of a margin term is appropriate to account (at least approximately) for factors that occur in application but were not considered in the analysis upon which the screening limit is based. For example, the 10 CFR 50.61 margin term accounts for uncertainty in copper, nickel, and initial RT_{NDT}. However, our model explicitly considers uncertainty in all of these variables, and represents these uncertainties as being larger (a conservative representation) than would be appropriate in any plant-specific application of the proposed screening limit. Consequently, use of the 10 CFR 50.61 margin term with the new screening limits is inappropriate. In general, the following additional reasons suggest that use of *any* margin term with the proposed screening limits is inappropriate:

(1) The *TWCF* values used to establish the screening limit represent 90th percentile values or greater.

(2) The results from our three plant-specific analyses apply to PWRs *in general*, as demonstrated in Chapters 8 and 9 of this report.

(3) Certain aspects of our modeling cannot reasonably be represented as "best estimates." On balance, there is a conservative bias to these non-best-estimate aspects of our analysis because residual conservatisms in the model far outweigh residual nonconservatisms.

Abbreviations

¼-T FLAW	Surface-breaking flaw defined by ASME Boiler and Pressure Vessel Code as having a depth equal to one-quarter of the vessel wall thickness and a length equal to six times the flaw depth
1D	One-Dimensional
ABAQUS	Commercial finite element code developed by Hibbett, Karlsson, and Sorenson in Pawtucket, Rhode Island
ACRS	Advisory Committee on Reactor Safety (NRC)
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
APET	Accident Progression Event Tree
APEX	Advanced Plant Experiment
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient without Scram
B&W	Babcock and Wilcox
BWOG	Babcock and Wilcox Owners' Group
BCC	Body-Centered Cubic
BWR	Boiling-Water Reactor
CDF	Core Damage Frequency
CE	Combustion Engineering
CEOG	Combustion Engineering Owners' Group
CFD	Computational Fluid Dynamics
CL	Cold Leg
CFR	*Code of Federal Regulations*
CFT	Core Flood Tank
CPI	Conditional Probability of Crack Initiation
CPTWC	Conditional Probability of Through-Wall Cracking
CSAU	Code Scaling, Applicability, and Uncertainty Methodology
CSNI	Committee on the Safety of Nuclear Installations
CST	Condensate Storage Tank
CVN	Charpy V-Notch
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EFPY	Effective Full-Power Years
EFW	Emergency Feedwater
EOL	End of License (40 operating years, 32 EFPY)
EOLE	End of License Extension (60 operating years, 48 EFPY)

EPRI	Electric Power Research Institute
ESFAS	Engineered Safety Features Actuation System
F&B	Feed-and-Bleed
FAVOR	Fracture Analysis of Vessels, Oak Ridge
FCI	Frequency of Crack Initiation
GMAW	Gas Metal Arc Weld
H2TS	Hierarchical, Two-Tiered Scaling
HCLPF	High Confidence of Low Probability of Failure
HEP	Human Error Probability
HFE	Human Failure Event
HPI	High-Pressure Injection
HPSI	High-Pressure Safety Injection
HRA	Human Reliability Analysis
HSSI	Heavy Section Steel Irradiation (Project)
HZP	Hot Zero Power
IAEA	International Atomic Energy Agency
ID	Inner Diameter
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IPTS	Integrated Pressurized Thermal Shock
ISLOCA	Interfacing Systems Loss-of-Coolant Accident
ITV	Intermediate Test Vessel
IVO	Imatran Voima Oy
LAS	Low-Alloy Steel
LBLOCA	Large-Break Loss-of-Coolant Accident (pipe diameters above ~8-in. (~20-cm))
LEFM	Linear Elastic Fracture Mechanics
LER	Licensee Event Report
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LOF	Lack of Inter-Run Fusion
LOFT	Loss-of-Fluid Test facility
LPI	Low-Pressure Injection
LPSI	Low-Pressure Safety Injection
MBLOCA	Medium-Break Loss-of-Coolant Accident (pipe diameters of ~4 to 8-in. (~10 to 20-cm))
MFIV	Main Feedwater Isolation Valve
MFW	Main Feedwater
MIST	Multi-loop Integral System Test
MRJ	Materials Reliability Project
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break

NDT	Nil-Ductility Temperature
NEA	Nuclear Energy Agency (OECD)
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
NUREG/CR	NRC Technical Report Designator (Contractor-prepared Report published by the U.S. Nuclear Regulatory Commission)
OD	Outer Diameter
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
PFM	Probabilistic Fracture Mechanics
PIRT	Phenomena Identification and Ranking Table
PNNL	Pacific Northwest National Laboratories
PORV	Power-Operated Relief Valve
Ppb	Parts per Billion
PRA	Probabilistic Risk Assessment
PRODIGAL	Probability of Defect Initiation and Growth Analysis
PTS	Pressurized Thermal Shock
PTSE	Pressurized Thermal Shock Experiment
PVRUF	Pressure Vessel Research Users' Facility
PWR	Pressurized-Water Reactor
QHO	Quantitative Health Objective, as defined by the Commission's Safety Goal Policy Statement [NRC FR 86]
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RELAP	Reactor Leak and Power excursion code
REMIX	a computer program used to determine the temperature of a plume in the downcomer when the flow in the loops is stagnant
RES	Office of Nuclear Regulatory Research (NRC)
RG	Regulatory Guide
RLE	Review-Level Earthquake
ROSA	Rig of Safety Assessment
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RT	Reference Temperature
RVFF	Reactor Vessel Failure Frequency
RVID	Reactor Vessel Integrity Database
RWST	Refueling Water Storage Tank
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SAW	Submerged Arc Weld
SBLOCA	Small-Break Loss-of-Coolant Accident (pipe diameters below ~4-in. (~10-cm))
SCC	Stress Corrosion Cracking
SECY	Secretary of the (U.S. Nuclear Regulatory) Commission

SEMISCALE	a 1:1705 scaled experimental facility that simulates the primary system of a 4-loop PWR plant
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIAS	Safety Injection Actuation Signal
SIT	Safety Injection Tank
SMAW	Submerged Metal Arc Weld
SO-1	Stuck-open valve in the primary pressure circuit
SO-2	Stuck-open valve in the secondary pressure circuit
SQA	Software Quality Assurance
SRM	Staff Requirements Memorandum
SRV	Safety/Relief Valve
SSC	System, Structure, or Component
SSE	Safe-Shutdown Earthquake
SSRV	Secondary System Relief Valve
TBV	Turbine Bypass Valve
TH	Thermal-Hydraulics
TMI	Three Mile Island
TSE	Thermal Shock Experiment
TWCF	Through-Wall Cracking Frequency
UMD	University of Maryland
UPTF	Upper Plenum Test Facility
USE	Charpy V-Notch Upper-Shelf Energy
V&V	Verification and Validation
VCIF	Vessel Crack Initiation Frequency
(W)	Westinghouse
WOG	Westinghouse Owners' Group
WPS	Warm Pre-Stress

Nomenclature

Symbols Used in Thermal-Hydraulics

α	thermal diffusivity, m^2/s
β	bulk coefficient of expansion, 1/C
μ	viscosity, kg/m-s
ν	kinematic viscosity, m^2/s
ρ	density, kg/m^3
σ	stress, kg/s^2
τ	characteristic time
C_p	heat capacity, m^2/s^2-C
g	gravitational acceleration, m/s^2
Gr	Grashof Number
h	convective heat transfer coefficient, W/m^2-C
D	diameter, m
J	joules, kg-m^2/s^2
k	conductivity, W/m-C
l	length, m
Nu	Nusselt Number
Pr	Prandtl Number
P	pressure, kg/m-s^2
q	heat flux, W/m^2
Re	Reynolds Number
Ri	Richardson Number
s	seconds
t	thickness, m
t	time, s
u	velocity, m/s
T	temperature, C
W	watts, kg-m^2/s^3

Symbols Used in Fracture Mechanics

$2a$	Flaw depth measured through the vessel wall thickness
$2c$	Flaw length measured parallel to the axial or circumferential direction of the vessel
Cu	Copper content, weight%
J_{Ic}	A fracture toughness measure defined by ASTM E1820, which quantifies the resistance of metals to crack initiation by the initiation, growth, and coalescence of microvoids
J-R	A fracture toughness measure defined by ASTM E1820, which quantifies the resistance of metals to ductile tearing
K_{Jc}	A fracture toughness measure defined by ASTM E1921, which quantifies the resistance of metals to crack initiation by cleavage mechanisms
K_{Ia}	A fracture toughness measure defined by ASTM E1221, which quantifies the ability of metals to arrest (stop) a running cleavage crack
K_{Ic}	A fracture toughness measure defined by ASTM E399, which quantifies the resistance of metals to crack initiation under plane strain conditions
$K_{Ic(min)}$	The minimum K_{Ic} fracture toughness possible at a particular temperature
$K_{APPLIED}$	Linear elastic crack driving force
$\mathcal{L}$	For a buried defect, distance from the wetted clad surface on the vessel ID to the inner crack tip
ℓ	The length of the fusion line of an axial weld
Ni	Nickel content, weight%
P	Phosphorus content, weight%
RT_{AW}	A fracture toughness reference temperature, which characterizes the RPV's resistance to fractures initiating from flaws found along the axial weld fusion lines. It corresponds to the maximum RT_{NDT} of the plates/welds that lie to either side of the weld fusion lines, and is weighted to account for differences in weld fusion line length (and, therefore, number of simulated flaws) between vessel courses.
RT_{PL}	A fracture toughness reference temperature, which characterizes the RPV's resistance to fractures initiating from flaws found in plates that are not associated with welds. It corresponds to the maximum RT_{NDT} occurring anywhere in the plate.
RT_{CW}	A fracture toughness reference temperature, which characterizes the RPV's resistance to fractures initiating from flaws found along the circumferential weld fusion lines. It corresponds to the maximum RT_{NDT} of the plates/welds that lie to either side of the weld fusion lines.
RT_{NDT}	Transition fracture toughness reference temperature defined by ASME NB-2331
$RT_{NDT(u)}$	Unirradiated value of RT_{NDT}
RT_{PTS}	RT_{NDT} projected end of license to account for the effects of irradiation (defined in 10 CFR 50.61)
t_{WALL}	Vessel wall thickness
t_{CLAD}	Stainless steel cladding thickness

T_{30}	The temperature at which the mean CVN energy is 30 ft-lbs (41J)
$T_{35/50}$	Charpy V-notch energy transition temperature defined as the temperature at which the CVN energy is at least 50 ft-lbs (68J) and the lateral expansion of the specimen is at least 0.035-in. (0.89-mm) [See the definition on page 2-7]
T_{NDT}	Nil-ductility temperature defined by ASTM E-208
ΔT_{30}	The shift in the CVN 30 ft-lb (41J) transition temperature produced by radiation damage
σ_{flow}	Flow strength, average of tensile yield and tensile ultimate strength
ϕt	Fluence

Glossary

Terms Used in Probabilistic Risk Assessment

Abnormal operating procedure	A procedure (i.e., list of actions) used to address unique or special plant circumstances identified while using emergency operating procedures (EOPs). These abnormal operating procedures are usually called by EOPs, but may be indicated directly by some plant conditions.
Accident progression event tree	The event tree used to model the part of the accident sequence that follows the onset of core damage, including containment response to severe accident conditions, equipment availability, and operator performance.
Binning	The process of taking a large number of sequences and combining then into a smaller number of groups, that are expected to have similar characteristics (e.g., TH conditions), to allow effective utilization of limited resources.
Core damage	Uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated and involving enough of the core to cause a significant release.
Dominant scenario	An accident sequence (scenario) that is usually represented by the top 10 or 20 events or groups of events modeled in a PRA, which accounts for a large fraction of the specified end state.
Emergency operating procedure	The primary procedure (i.e., list of actions) used to respond to a plant disturbance resulting from an initiating event.
Event tree	A logic diagram that begins with an initiating event or condition and progresses through a series of branches that represent expected system or operator performance that either succeeds or fails and arrives at either a successful or failed end state.
Fault tree	A deductive logic diagram that depicts how a particular undesired event can occur as a logical combination of other undesired events.
Large Early Release	The rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions, such that there is a potential for early health effects.
Latin Hypercube sampling	A stratified sampling technique, in which the random variable distributions are divided into equal probability intervals, and probabilities are then randomly selected from within each interval.
Mitigating equipment	Systems or components, used to respond to an initiating event, of which successful operation prevents the occurrence of an undesired event or state.
Pre-initiator human failure event	Human failure events that represent the impact of human errors committed during actions performed prior to the initiation of an accident (e.g., during maintenance or the use of calibration procedures).
Post-initiator human failure event	Human failure events that represent the impact of human errors committed during actions performed in response to an accident initiator.

Prompt fatality	A fatality that results from substantial radiation exposures incurred during short time periods (usually within weeks, though up to 1 year for pulmonary effects).
PTS bin	A group of sequences that are expected to have similar TH characteristics and are represented by one unique set of TH characteristics during a FAVOR calculation.
Risk-informed	An approach to analyzing and evaluating activities, which bases decisions on the results of traditional engineering evaluations, supported by insights derived from the use of PRA methods.
Scenario	See Sequence.
Screening	The process of eliminating items from further consideration based on their negligible contribution to the probability of an undesired end state or its consequences.
Sequence	A representation in terms of an initiating event followed by a sequence of failures or successes of events (i.e., system, function, or operator performance) that can lead to undesired consequences, with a specified end state (e.g., potential for PTS).

Terms Used in Thermal-Hydraulics

Blowdown	Rapid depressurization of a system in response to a break.
Break flow	Flow of water (liquid and vapor) out a pipe break or a valve.
Break energy	Energy content of the fluid flow out a break.
Bottom-up	To break up a complex system into its subsystems, and then break up each subsystem into its components, examine individual local phenomena and processes that most affect each component, and build up the total complex system from these individual pieces (like manufacturing a car).
Coast down	Time required for a pump to stop rotating once power is shut off due to inertia.
Decay heat	Heat generated from radioactive decay of fission products.
Enthalpy	Sum of internal energy and volume multiplied by pressure.
Flash	Change of phase from saturated liquid to vapor resulting from decrease in pressure.
Flow quality	Mass fraction of flow stream that is steam. Higher quality flow would have a high mass fraction of steam.
Forced flow	Flow driven by a pump.
Inventory	Mass of water.
Loop flow	Mass flow rate of coolant in a circuit.
Makeup water	Water reservoir available for inventory control.
Natural circulation	Flow driven by buoyancy (gravity).
Pressure drop	Change in pressure due to conversion of mechanical energy to internal energy.
Protection system	Electrical controls to actuate engineering safety features.
Quality	Mass fraction of steam in a two-phase steam-water mixture.

Saturation temperature	A temperature corresponding to phase change from liquid to vapor.
Sensible heat	The product of specific heat and temperature change of subcooled liquid.
Subcooled	A system is *subcooled* if it exists entirely in a liquid state. The *degree of subcooling* is the number of degrees that the temperature of the system would have to be raised to cause boiling.
Throttled	Operation of a control valve to regulate flow.
Top-down	To characterize a complex system by establishing the governing behavior, or phenomenon, that is most important, and then proceed from that starting point to successive lower levels, by identifying the processes that have the greatest influence on the top-level phenomenon.
Trip	A "trip" occurs when a breaker opens in response to its trip mechanism (an arm that holds the breaker closed moves to allow the breaker to open). When a reactor trips, all of the breakers that provide power to the rod control system open, causing the rods to be inserted in the core and stopping the nuclear reaction. When a pump trips, the breaker opens, thereby disconnecting power and causing the pump to stop.
Water solid	A situation in which there is no steam in the system (i.e., it is all liquid). A "water solid" system is *subcooled.*

Terms Used in Fracture Mechanics

Brittle	Fracture occurring without noticeable macroscopic plastic deformation (stretching) of the material.
Cleavage fracture	Microscopically, cleavage is a fracture mode that occurs preferentially along certain atomic planes through the grains of the material. Cleavage can only occur in ferritic steels (i.e., steels having a body-centered cubic lattice structure). Macroscopically, cleavage fracture is often called "brittle" fracture because little noticeable plastic deformation (stretching) of the material occurs. (Note, however, that plastic flow at the micro-scale is a necessary precursor to cleavage.) Macroscopically, cleavage fracture is also characterized as being a sudden event, with cracks of very large dimensions developing over durations measured in fractional seconds. A useful, although inexact, analogue for cleavage fracture in common experience is the breaking of glass.
Ductile fracture	Microscopically, ductile fracture occurs through the initiation, growth, and eventual coalescence of micro-voids in the material into a macroscopic crack. These micro-voids tend to initiate at local heterogeneities in the material (e.g., inclusions, carbides, clusters of dislocations). Macroscopically, ductile fracture is associated with considerable plastic deformation (stretching) of the material. Relative to cleavage fracture, ductile fracture occurs very slowly, with crack growth rates measured in seconds rather than in micro-seconds (for cleavage).
Fracture toughness	A general term referring to a material's resistance to fracture. The term may be modified to refer to fractures by different mechanisms: <u>Arrest fracture toughness</u> measures a material's ability to stop a running cleavage crack. <u>Cleavage fracture toughness</u> measures a material's ability to resist crack initiation in cleavage. <u>Ductile fracture toughness</u> measures a material's ability to resist crack initiation attributable to ductile mechanisms on the upper shelf.

Lower shelf	At low temperatures, the toughness behavior of steels occurs by transgranular cleavage and is said to be on the lower shelf. On the lower shelf, a fracture is unstable, and is often referred to as a "brittle" fracture.
Reference temperature	A characteristic temperature used to locate the transition curve of a ferritic steel on the temperature axis.
Transition (or transition curve)	Between lower shelf and upper shelf temperatures, the fracture behavior of a ferritic material is said to be in "transition." At low temperatures in transition, fracture occurs by cleavage. As temperature increases through the transition regime, fracture occurs by ductile crack initiation and growth, a process which is terminated by cleavage. At still higher temperatures, cleavage cannot occur, and upper shelf conditions exist.
Upper shelf	At high temperatures, the toughness behavior of steels occurs by ductile mechanisms (micro-void initiation, growth, and coalescence) and is said to be on the upper shelf. On the upper shelf, afracture is stable and dissipates considerable amounts of energy.

Terms Used in Uncertainty Analysis

Aleatory	Aleatory uncertainties arise as a result of the randomness inherent in a physical or human process. Consequently, aleatory uncertainties are fundamentally irreducible. If the uncertainty in a variable is characterized as being aleatory, the entire distribution of the variable is carried through each simulation run.
Epistemic	Epistemic uncertainties are caused by limitations in our current state of knowledge (or understanding) of a given process. Epistemic uncertainties can, in principle, be reduced by an increased state of knowledge. If the uncertainty in a variable is characterized as being epistemic in a probabilistic simulation, individual values of the variable are randomly selected from a distribution and propagated through the calculation. This procedure models the understanding that the "correct" value of the variable is knowable, at least in principal. Thus, for epistemic uncertainties, individual simulation runs are deterministic, while the totality of all simulation runs captures the uncertainty characteristic of the epistemic variable.

1 Motivation for and Objective of this Study

1.1 Description of Pressurized Thermal Shock

During the operation of a nuclear power plant, the walls of the reactor pressure vessel (RPV) are exposed to neutron radiation, resulting in localized embrittlement of the vessel steel and weld materials in the area of the reactor core. If an embrittled RPV had an existing flaw of critical size and certain severe system transients were to occur, the flaw could very rapidly propagate through the vessel, resulting in a through-wall crack and challenging the integrity of the RPV. The severe transients of concern, known as pressurized thermal shock (PTS), are characterized by a rapid cooling (i.e., thermal shock) of the internal RPV surface and downcomer, which may be followed by repressurization of the RPV. Thus, a PTS event poses a potentially significant challenge to the structural integrity of the RPV in a pressurized-water reactor (PWR).

A number of abnormal events and postulated accidents have the potential to thermally shock the vessel (either with or without significant internal pressure). These events include a pipe break or stuck-open valve in the primary pressure circuit, or a break of the main steam line, among others. During such events, the water level in the core drops as a result of the contraction produced by rapid depressurization. In events involving a break in the primary pressure circuit, an additional drop in water level occurs as a result of leakage from the break. Automatic systems and operators must provide makeup water in the primary system to prevent overheating of the fuel in the core. However, the makeup water is much colder than that held in the primary system. As a result, the temperature drop produced by rapid depressurization coupled with the near-ambient temperature of the makeup water produces significant thermal stresses in the thick section steel wall of the RPV. For embrittled RPVs, these stresses could be sufficient to initiate a running crack, which could propagate all the way through the vessel wall. Such through-wall cracking of the RPV could precipitate core damage or, in rare cases, a large early release of radioactive material to the environment. Fortunately, the coincident occurrence of critical-size flaws, embrittled vessel steel and weld material, and a severe PTS transient is a very low-probability event.

1.2 PTS Limits on the Licensable Life of a Commercial Pressurized Water Reactor

In the early 1980s, attention was focused on the possibility that PTS events could challenge the integrity of the RPV wall for two reasons:

- Operational experience suggested that overcooling events, while not common, did in fact occur.
- The results of in-reactor materials surveillance programs suggested that the steels used in RPV construction were prone to loss-of-toughness over time as a result of neutron irradiation-induced embrittlement.

This possibility of accident loading combined with degraded material conditions motivated investigations aimed at assessing the risk of vessel failure posed by PTS for the purpose of establishing the operational limits needed to ensure that the likelihood of RPV failures caused by PTS transients is kept sufficiently low. These efforts led to the publication of a Commission paper [SECY-82-465], which provided the technical basis for subsequent development of what has come to be known as the "PTS Rule," as set forth in Title 10, Section 50.61, of the *Code of Federal Regulations* (10 CFR 50.61), "Fracture Toughness

Requirements for Protection Against Pressurized Thermal Shock Events" [10 CFR 50.61].

Currently, 10 CFR 50.61 requires licensees to monitor the embrittlement of their RPVs using a reactor vessel material surveillance program qualified under Appendix H to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The surveillance results are used together with the formulae and tables in 10 CFR 50.61 to estimate the fracture toughness transition temperature (RT_{NDT}[†]) of the steels in the vessel's beltline and how those transition temperatures increase as a result of irradiation damage throughout the operational life of the vessel. For licensing purposes, 10 CFR 50.61 provides instructions on how to use these estimates of the effect of irradiation damage to estimate the value of RT_{NDT} that will occur at end of license (EOL), a value called RT_{PTS}. 10 CFR 50.61 also provides "screening limits" (maximum values of RT_{NDT} permitted during the plant's operational life) of +270°F (132°C) for axial welds, plates, and forgings, and +300°F (149°C) for circumferential welds. These screening limits correspond to a limit of 5×10^{-6} events/year on the annual probability of developing a through-wall crack [RG 1.154]. Should RT_{PTS} exceed these screening limits, 10 CFR 50.61 requires that the licensee to either take actions to keep RT_{PTS} below the screening limit (by implementing "reasonably practicable" flux reductions to reduce the embrittlement rate, or by deembrittling the vessel by annealing [RG 1.162]), or perform plant-specific analyses to demonstrate that operating the plant beyond the 10 CFR 50.61 screening limit does not pose an undue risk to the public [RG 1.154]. While no currently operating PWR has an RT_{PTS} value that exceeds the 10 CFR 50.61screening limit before EOL, several plants are close to the limit (3 are within 2°F while 10 are within 20°F, see Figure 1.1). Those plants that are close to are likely to exceed the screening limit during the 20-year license renewal period that is currently being sought by many operators. Moreover, some plants maintain their RT_{PTS} values below the 10 CFR 50.61 screening limits by implementing flux reductions (low-leakage cores, ultra-low-leakage cores), which are fuel management strategies that can be economically deleterious in a deregulated marketplace. Thus, the 10 CFR 50.61 screening limits can restrict the licensable and the economic lifetime of PWRs. As detailed in the next section, there is considerable reason to believe that these restrictions are not necessary to ensure public safety and, in fact, place unnecessary burden on licensees.

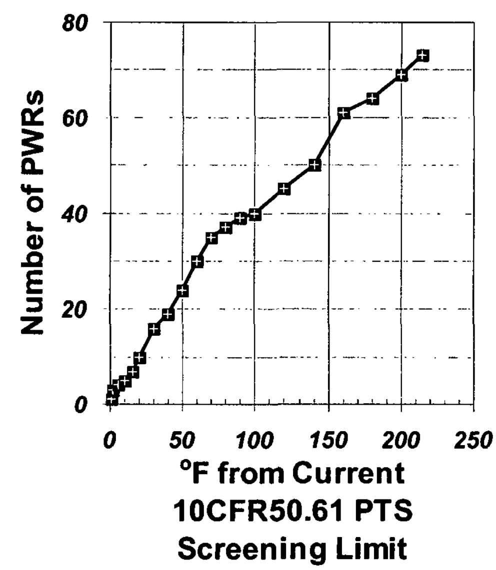


Figure 1.1. Proximity of currently operating PWRs to the 10 CFR 50.61 screening limit for PTS

† The RT_{NDT} index temperature was intended to correlate with the fracture toughness transition temperature of the material. Fracture toughness, and how it is reduced by neutron irradiation embrittlement, are key parameters controlling the RPV's resistance to any loading challenge. For a more detailed description of RT_{NDT} (in specific) and fracture toughness (in general), see [*EricksonKirk PFM*].

1.3 Technical Factors Suggesting Conservatism of the Current PTS Rule

It is now widely recognized that state of knowledge and data limitations in the early 1980s necessitated conservative treatment of several key parameters and models used in the probabilistic calculations that provided the technical basis [SECY-82-465] of the current PTS Rule [10 CFR 50.61]. The most prominent of these conservatisms include the following factors:

- highly simplified treatment of plant transients (very coarse grouping of many operational sequences (on the order of 10^5) into very few groups (≈ 10), necessitated by limitations in the computational resources needed to perform multiple thermal-hydraulic calculations)
- lack of any significant credit for operator action
- characterization of fracture toughness using RT_{NDT}, which has an intentional conservative bias [ASME NB2331]
- use of a flaw distribution that places *all* flaws on the interior surface of the RPV, and, in general, contains larger flaws than those usually detected in service
- a modeling approach that treated the RPV as if it were made entirely from the most brittle of its constituent materials (welds, plates, or forgings)
- a 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 10 CFR 50.61 PTS screening limits are unnecessarily conservative. Consequently, the staff of the U.S. Nuclear Regulatory Commission (NRC) believed that reexamining 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 substantially relaxing these limits.

1.4 Statement of Objective

For the reasons stated in Section 1.3, the NRC's Office of Nuclear Regulatory Research (RES) undertook this study with the objective of developing the technical basis to support a risk-informed revision of the PTS Rule and the associated PTS screening limit, and thereby providing the basis for potential rulemaking.

1.5 Guide to this Report

As discussed in Chapter 4, this report summarizes a much larger documentary package, and updates a previous report [***Kirk 12-02***]. We begin in Chapter 2 by describing PTS, its potential precursors, and the historical occurrence of PTS in reactor operations. We also summarize the key findings on which the current rule is based, and we detail the provisions of the current rule. Chapter 3 describes this study in detail and discusses its guiding principles, our investigative approach, and our fundamental assumptions. Additionally, we detail the many organizations and individuals that have contributed to this project. Chapter 4 provides a "map" to the documents from which this summary report was drawn and describes the information available in each of those detailed reports. In Chapters 5, 6, and 7, we synopsize the key features of our probabilistic risk assessment (PRA) and human reliability analysis (HRA), our thermal-hydraulic (TH) analysis, and our probabilistic fracture mechanics (PFM) analysis, respectively. Chapters 6 and 7 also address experimental validation of our TH and PFM methodologies. Chapter 8 presents the results of our baseline plant-specific analysis of three PWRs, while Chapter 9 examines the general applicability of these results to the larger population of all commercial PWRs. In Chapter 10, we develop a limit on the risk posed by PTS that is consistent with current regulatory guidance. Chapter 11combines the information from Chapters 8 through 10 to develop a proposed revision to the 10 CFR 50.61 screening limit. Finally, Chapter 12 summarizes our findings and discusses some considerations for rulemaking.

2 Pressurized Thermal Shock Background

In this chapter, we provide background information on PTS. We begin with a general description of the progression of a PTS event, including its precursors and their effect on both the primary pressure circuit and the RPV itself (Section 2.1). We then discuss the historical incidence of PTS (Section 2.2). Finally, we summarize the findings of SECY-82-465, which provide the technical basis for the current PTS Rule (Section 2.3), and we review the rule's provisions (Section 2.4).

2.1 General Description of the Progression of a PTS Event

In the following sections, we describe the event sequence that can give rise to a PTS event (Section 2.1.1), the effect of those events on both the primary and secondary pressure circuits (Section 2.1.2), and the challenge that these transients can pose to the structural integrity of the RPV (Section 2.1.3).

2.1.1 Precursors

Normally, the RPV is very hot because of the high temperature of the water it contains (600°F 315°C)). Several types of malfunctions or accidents can cause the vessel to suddenly fill with cool water or cause the reactor coolant water temperature to decrease rapidly. Such rapid cooling causes the vessel to experience thermal shock. If the RPV is then subjected to high pressure, the phenomenon is referred to as PTS.

Most significant PTS scenarios fall into one of the following two categories:

- breaks in the primary side of the reactor coolant system (RCS) (see Sections 2.1.2.1 and 2.1.2.2)
- breaks in the secondary system (see Section 2.1.2.3).

Here, we use the term "break" to refer to pipe breaks (e.g., hot leg break, cold leg break, main steam line break, steam generator tube rupture), as well as stuck-open valves. Initially, a stuck-open valve transient behaves much like a transient initiated by a pipe break. However, later in stuck-open valve transients the valve can unstick (and, therefore, reclose), so repressurization is possible. Our analysis considers potential failures of the following valves:

- Primary Side/RCS: Power-operated relief valves (PORVs) and safety relief valves (SRVs)
- Secondary Side: Main steam isolation valves (MSIVs), steam generator atmospheric dump valves (ADVs), main steam safety valves (MSSVs), and so on.

Figure 2.1 illustrates the arrangement of the major components of both the primary and secondary systems in a PWR.

2.1.2 Thermal-Hydraulic Response of the Vessel

2.1.2.1 Pipe Breaks in the Primary System

When a break occurs in the primary system, mass is lost from the RCS. The level of water in the pressurizer (PZR) decreases, thereby causing a decrease in RCS pressure. If the RCS pressure or PZR level decreases too far, the reactor protection system (RPS) will generate a reactor trip signal, which in turn will insert the control rods and stop the fission process. Additionally, the engineered safety features actuation system (ESFAS) will generate a safety injection actuation signal (SIAS). The SIAS will start the high-pressure injection (HPI) and low-pressure injection (LPI) pumps, which will start the supply of emergency core cooling (ECC) water to the RCS, as pressure allows.

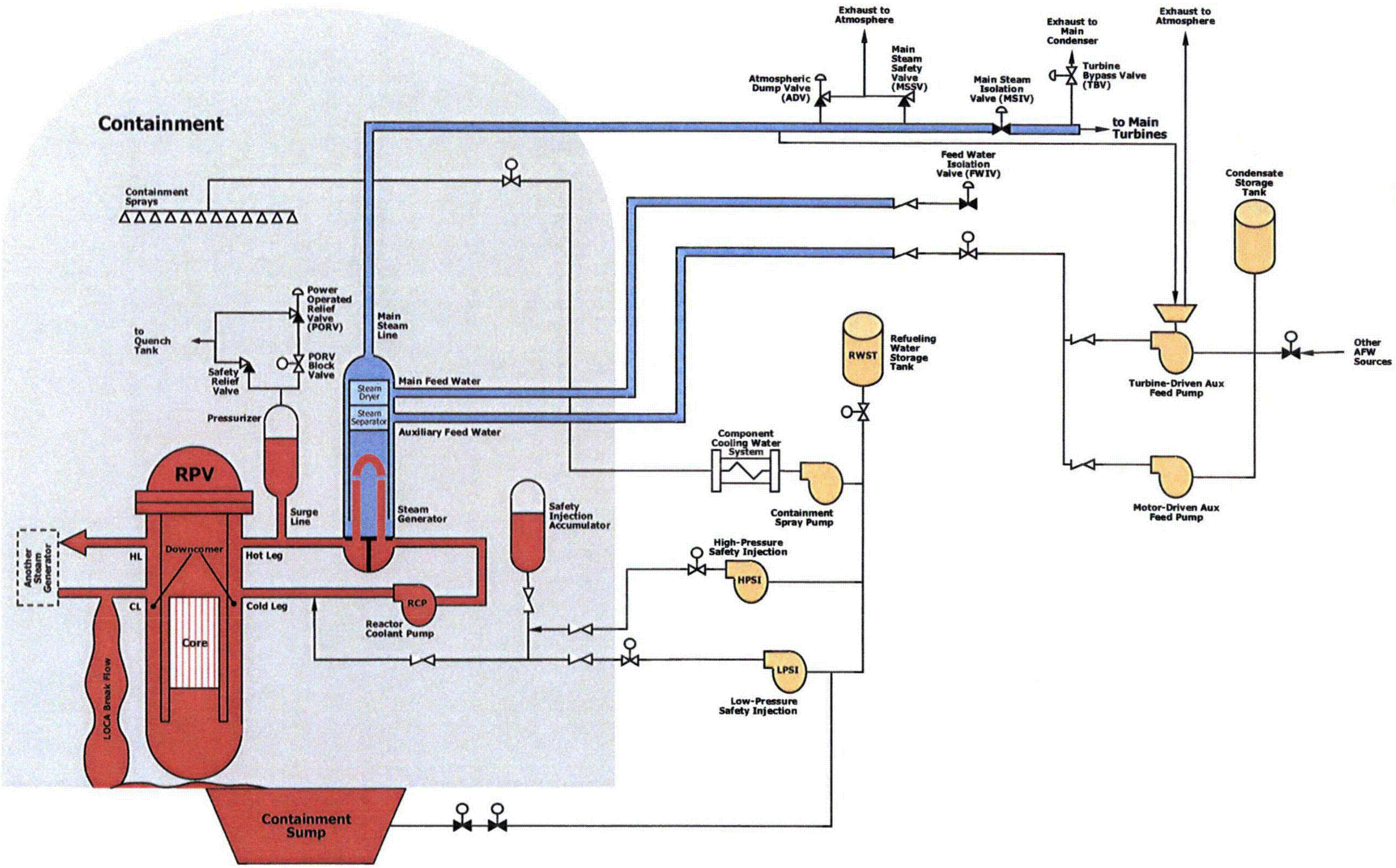


Figure 2.1. Schematic illustration of the main components of the primary (red) and secondary (blue) systems in a pressurized-water reactor

Further progress of the transient depends upon the ability of HPI to make up for the mass lost through the break.

For very small breaks (less than ~1.4-in. (3.5-cm) in diameter), HPI is sufficient to make up the lost mass and, thereby, maintain RCS mass and pressure control. For larger breaks, HPI is insufficient to replace the lost mass, so PZR level and RCS pressure continue to fall. As the pressure continues to fall, the safety injection tanks (SITs) discharge and, eventually, the LPI begins injecting colder water into the RCS. The ECC injection rates are substantially greater for large breaks (compared to small breaks), and result in much greater cooldown of the primary system and subsequently the RPV wall.

The controlling feature of such loss-of-coolant accidents (LOCAs) is the size of the break. The larger the break, the faster the transient proceeds, and the more severe the cooldown. Larger breaks cause a greater pressure decrease, which results in larger ECC flows. This causes the downcomer temperature to drop rapidly. The rate of temperature decrease (which is controlled by the break size) and the minimum temperature achieved (which is controlled by the temperature of the ECC water and whether ECC water is recirculated from the sump) are the dominant TH factors that influence the severity of the transient. Since RCS pressure is low, it not a significant factor.

2.1.2.2 Stuck-Open Primary Safety Relief Valves

During stuck-open valve transients, a primary SRV is assumed to stick open as a result of mechanical binding or other possible causes. This binding of the valve may release later in the transient, resulting in valve reclosure. Phenomenologically, stuck-open valve scenarios are similar to small hot leg break LOCAs, until the valve recloses. The "break" areas of stuck-open valve scenarios are typically in the small-break LOCA range of ≈1.5 to 2+-in. (≈3.8 to 5+-cm).

Following valve reclosure, HPI will gradually fill the RCS. As the RCS fills, the primary system pressure will increase above the saturation pressure of the coolant, thereby reestablishing subcooling in the loops. When operating procedures allow, the HPI can be controlled to avoid overfilling the RCS. If, however, the operator fails to attend to HPI control in a timely manner, the RCS can continue to fill until the primary system is water solid, meaning there is no longer any steam in the system. Since water is nearly incompressible, the RCS pressure rises very rapidly, and the pressure created by HPI will reopen the SRVs. RCS pressure will then remain at the SRV setpoint of 17.25 MPa (2500 psi) for as long as the HPI remains on.

The controlling features of stuck-open valve scenarios are the length of time the valve is open, and the repressurization associated with the primary system becoming water solid. The longer the SRV stays open, the cooler the downcomer temperature becomes. Timely operator control of HPI is an important factor influencing transient severity because it determines the maximum pressure achieved in the primary system. Thus, for these scenarios, both downcomer temperature and RCS pressure are important.

2.1.2.3 Breaks in the Secondary System

Secondary side breaks can include both actual breaks of the steam line and the sticking-open of one or more of the numerous control and safety valves in the steam system. A stuck-open valve is, therefore, also referred to as a "break," consistent with the terminology adopted to describe stuck-open valves in the primary system Break sizes can range from a single valve sticking open to a complete main steam line break (MSLB). Similar to LOCAs, *time* and *break size* are directly related. The larger the break, the faster the transient proceeds, and the more severe the cooldown.

Following break initiation, the response of engineered safety features systems may result in safety injection actuation, actuation of main feedwater isolation valves (MFIVs) and/or MSIVs, as well as automatic control of auxiliary feedwater (i.e., through isolation of a turbine-driven pump). If the break is downstream of the MSIVs, the steam line break will be terminated

when the MSIV shuts. For larger MSLBs, MSIV closure is automatic and occurs rapidly after break initiation. For smaller steam line breaks, however, the operators will isolate the affected steam generator by securing flow to the generator and manually isolating the MSIVs. Therefore, downstream breaks are not PTS-significant. However, the main steam SRVs and ADVs are upstream of the MSIVs and, consequently, are not affected by an MSIV closure. For breaks occurring upstream of the MSIVs, steam will continue to blowdown until the affected steam generator is completely depressurized.

Steam generator depressurization causes cooling of the primary system. As steam continues leaking out of the break, the secondary side pressure continues to decrease, and the water in the generator remains saturated. Consequently, as the secondary side pressure decreases, the secondary side temperature also decreases. The primary side and secondary side remain "thermally coupled" during secondary side break scenarios, meaning that the primary system temperature will track the temperature of the affected steam generator. This primary system cooling increases the density of the primary water, so the volume of the water in the RCS shrinks. For sufficiently large secondary breaks, the shrinkage may be sufficient to actuate the emergency core cooling system (ECCS), causing direct injection of water at a temperature between that of external storage tanks (≈40°F (4.4°C)) and that of water recirculated from the sump (~120°F (49°C)). The flow of this colder water increases the cooling rate of the primary, thereby increasing transient severity. However, HPI does not reduce the minimum temperature of the primary below the boiling point of water in the secondary because the large heat transfer area in the affected steam generator (which is now at saturated conditions) is more than sufficient to bring the HPI water temperature up to the boiling point in the secondary system.

If the break is outside of containment, the lowest temperature expected would be 212°F (100°C) (saturation for atmospheric pressure); however, the final temperature could be higher (~250°F (120°C)) if the break is in containment. In this case, the minimum temperature of the primary system depends on the final containment pressure following blowdown of the steam generator.

The controlling features of steam line break scenarios are the size of the break, control of feedwater to the broken steam generator, proper steaming of the unaffected generator, and control of HPI if HPI is actuated. Large steam line breaks resemble large LOCAs in terms of the rate of downcomer cooling. The differences between large steam line breaks and primary system pipe breaks (LOCAs) are as follows:

- The downcomer does not get as cold during secondary side breaks. Temperatures typically range from 212°F to 250°F (100°C to 121°C) for secondary side breaks, depending on whether the break is outside or inside of containment and, if inside, what the containment pressure is. By contrast, temperatures for primary system pipe breaks (LOCAs) can be as low as ≈40°F (4.4°C) for LOCAs because the minimum temperature is controlled by the boiling point of water, rather than ambient outside temperatures.
- Natural circulation flow rates (characteristic of large steam line breaks) are higher than loop stagnation flow rates (characteristic of large LOCAs). This higher flow ensures thorough mixing in the downcomer of the reactor coolant and ECCS flow (provided that coolant flow is initiated). The need to consider thermal plumes or streaming effects can thus be eliminated *a priori* for breaks in the secondary system.

2.1.3 Response of the Vessel to PTS Loading

As detailed in Section 2.1.2, all PTS precursors cause rapid cooling of the primary system. Depending on the transient, this cooling may or may not be accompanied by significant pressure. Both of these factors (rapid cooling and pressure) produce stresses in the vessel wall. The thermal stresses are (approximately) equal both along the axis of the vessel and around its circumference because of its cylindrical geometry. At the

beginning of the cooldown, the thermal stresses are tensile on the inner diameter (ID) of the vessel, and are equilibrated by compressive stresses on its outer diameter (OD). As the transient continues, these thermal stresses reduce to zero to the extent that isothermal conditions are achieved in the vessel wall. The pressure stresses are twice as large in the circumferential direction as they are in the axial direction (again as a direct consequence of the cylindrical vessel geometry). Also, pressure stresses are constant through the thickness of the RPV wall.

The degree to which these stresses challenge the integrity of the vessel is controlled by a number of factors, the most important of which are as follows:

- the existence of a crack in the RPV, as well as its location, orientation, and size
- the material's resistance to cracking at the location of the flaw, which is measured by its "fracture toughness." Fracture toughness depends on a number of other factors (each defined at the flaw location), the most important of which are as follows:
    - temperature
    - irradiation damage (fluence)
    - chemical composition
    - fracture toughness before irradiation

Qualitatively, when high stresses occur in the presence of a large crack at a low temperature, and when the material at the location of the crack has low fracture toughness, initiation of the crack becomes more likely. The likelihood that this initiated crack will propagate all the way through the vessel wall (thereby producing a breach in the primary system and leading to a condition we have defined as "failure") again depends on the interplay between the applied stresses (and how they vary through the wall) with the material's ability to stop a running crack (known as "arrest fracture toughness"). The factors that influence fracture toughness (listed above) also influence arrest fracture toughness. While arrest is by no means certain, it is true that as the crack propagates into the vessel wall, arrest becomes progressively more likely for transients where the stresses are primarily thermal. For these "mostly thermal" transients (medium- and large-diameter primary side pipe breaks, for example) arrest becomes more likely as the crack progresses into the vessel wall because temperature tends to increase while irradiation damage tends to decrease. (Both of these changes increase arrest fracture toughness.) Conversely, the progression of a crack initiated by a transient that produces both thermal and pressure stresses is fundamentally different. In this situation, the stresses remain higher through the wall because of the contribution of the pressure stress. Additionally, the fracture driving force tends to increase as the crack travels through the wall, as a result of the effect of primary system pressure on the crack faces. For these reasons, in transients that produce both thermal and pressure stresses, almost all cracks that initiate also propagate all the way through the vessel wall.

2.2 Historical Incidence of PTS

In the technical basis document written to support the current PTS Rule [SECY-82-465], the staff summarized the operational events that had, to that date, presented PTS challenges to operating plants. These events are depicted in Figure 2.2, which illustrates the three key operational parameters influencing event severity. Specifically, those parameters are the final RCS temperature, severity of the thermal shock (*dT/dt*) caused by the event, and existence (or lack thereof) of high pressure. A more recent search of licensee event reports (LERs) submitted between 1980 and 2000 reveals the occurrence of 128 "potentially PTS-significant" events. Approximately half of those LERs report the minimum temperature reached by the RCS, and the data suggest that the most recent transients are nearly all benign, with all but one having minimum RCS temperatures above 500°F (260°C). Thus, while overcooling events have occurred, they have only rarely been severe enough to challenge the structural integrity of the RPV.

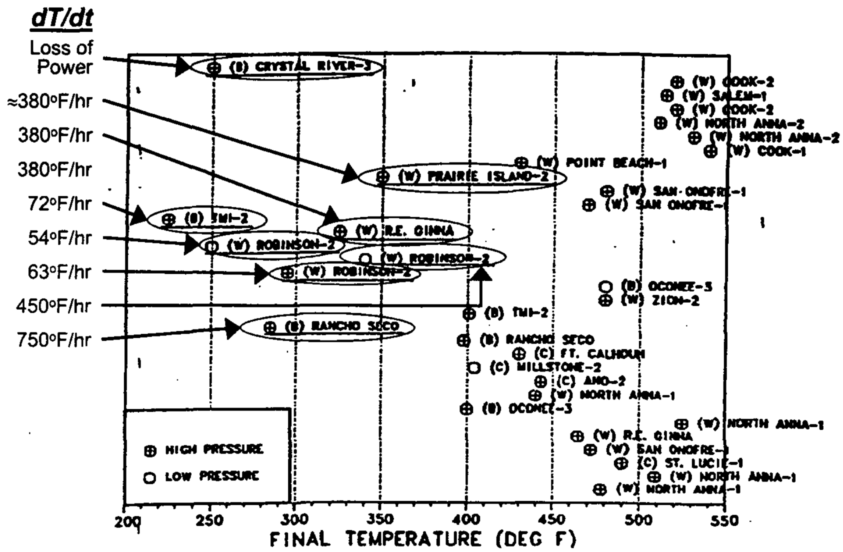


Figure 2.2. Figure 2.13 from SECY-82-465, depicting the final temperatures for 32 PTS precursor events experienced in commercial reactor service prior to 1982 (Cooling rates associated with the most significant transients have been superimposed on this graph.)

2.3 Summary of SECY-82-465 Findings

In the early 1980s, the nuclear industry and the NRC staff performed a number of investigations to assess the risk of vessel failure posed by PTS, and to establish the operational limits needed to ensure that the likelihood of RPV failures caused by PTS transients is maintained at an acceptably low level. These efforts led to the publication of a Commission paper [SECY-82-465] that provided the technical basis for subsequent development of what has come to be known as the "PTS Rule" [10 CFR 50.61]. The Commission paper included a number of probabilistic calculations performed to assess the influence of both contributory and mitigating factors (e.g., plant design, operator actions, operator training, material toughness, flaw population, and so on) on the outcome (vessel failure or non-failure) of a PTS event. The results of these calculations were used to develop a relationship between the probability of a through-wall crack developing in the RPV and the RT_{NDT} index temperature of the RPV. Regulatory Guide 1.154 [RG 1.154] later used this relationship, together with the judgment that an annual through-wall cracking frequency of $5x10^{-6}$ is acceptable, to establish "screening limits," or maximum values of RT_{NDT} permitted during the operating life of the plant. Specifically, the established limits were +270°F (132°C) for axial welds, plates, and forgings, and +300°F (149°C) for circumferential welds [10 CFR 50.61].

In the mid-1980s, the NRC conducted a number of follow-on studies concerning the risk associated with PTS events [ORNL 85a, 85b, 86]. These studies, featuring plant-specific analyses of H.B. Robinson Steam Electric Plant, Calvert Cliffs Nuclear Power Plant, and Oconee Nuclear Station, demonstrated that plants embrittled to the PTS screening limit of +270°F (132°C) had an annual probability of developing a through-wall crack below $5x10^{-6}$ events/reactor year. These plant-specific analyses demonstrate the conservatism of the generic analyses reported in SECY-82-465, which served as the basis for the provisions of 10 CFR 50.61.

2.4 Current Provisions of 10 CFR 50.61

As previously stated, 10 CFR 50.61 establishes "screening limits" (or maximum values of RT_{NDT} permitted during the operating life of the plant) of +270°F (132°C) for axial welds, plates, and forgings, and +300°F (149°C) for circumferential welds. Here, we discuss in greater detail the provisions of 10 CFR 50.61, as follows:

- Section 2.4.1: why an index-temperature approach is adopted to characterize transition fracture toughness in ferritic steels
- Section 2.4.2: the approach used to characterize irradiation effects on the index temperature
- Section 2.4.3: the specific provisions of 10 CFR 50.61
- Section 2.4.4: an evaluation of currently operating PWRs, relative to the PTS screening limits In 10 CFR 50.61

2.4.1 Index Temperature Approach to Characterize Transition Fracture Toughness in Ferritic Steels

"Fracture toughness" is a measure of a material's ability to deform without breaking in the presence of preexisting cracks. Physical evidence and numerous experimental observations demonstrate that the temperature dependence of the cleavage initiation fracture toughness of ferritic steels (K_{Ic} or K_{Jc}) does not depend on composition, heat treatment, material forming techniques (weld, plate, or forging), or irradiation conditions [Kirk 01a]. These factors influence only the position of the toughness transition curve on the temperature axis. This has led to widespread use of transition temperature approaches to characterize the cleavage fracture toughness of ferritic materials [WRC 175, Wallin 93a]. Such approaches employ empirical and/or physical evidence to establish the temperature dependence of fracture toughness that is common to all ferritic steels. Figure 2.3 shows the data used to establish the ASME K_{Ic} curve, one of the earliest transition temperature characterizations developed specifically using nuclear grade ferritic steels and weldments [WRC 175]. The formula for the curve in Figure 2.3 is as follows:

Eq. 2-1 $K_{Ic} = 33.2 + 2.806 \cdot \exp[0.02 \cdot (T - RT_{NDT} + 100)]$

where

RT_{NDT} is defined in accordance with ASME NB2331, as follows: $RT_{NDT} = MAX\{T_{NDT}, T_{35/50} - 60\}$.

T_{NDT} is the nil-ductility temperature (NDT) determined by testing specimens in accordance with ASTM E208.

$T_{35/50}$ is the transition temperature at which Charpy-V notch (CVN) specimens tested in accordance with ASTM E23 exhibit lateral expansion of at least 0.035-in. (0.89-mm) and absorbed energy of at least 50 ft-lbs (68J).

In Eq. 2-1, RT_{NDT} serves as an "index temperature" (i.e., a single value that characterizes the combined effects of alloying heat treatment, irradiation, etc. on fracture toughness)‡. Combining an index temperature with the (independently established) temperature dependence of fracture toughness (Eq. 2-1) defines the variation of toughness with temperature throughout the transition regime. The ease with which an index temperature can be experimentally established (relative to the much greater testing burden necessary to establish the complete toughness transition curve) makes transition temperature approaches attractive in applications where extensive material characterization is either economically infeasible or, for practical reasons, impossible.

‡ While RT_{NDT} is an index temperature that has customarily been used along with a fracture toughness transition curve (i.e., the ASME K_{Ic} curve), RT_{NDT} is ***not*** a *fracture toughness* index temperature. As specified by ASME NB-2331 (and as represented in Eq. 2-1), RT_{NDT} is defined based on non-fracture toughness tests that can, at best, be correlated with fracture toughness. [***EricksonKirk PFM***] provides a more detailed description of RT_{NDT}.

The monitoring of neutron irradiation embrittlement falls into both categories because of the expense associated with the testing of irradiated materials and the limited volumes of material that can be irradiated as part of a surveillance program.

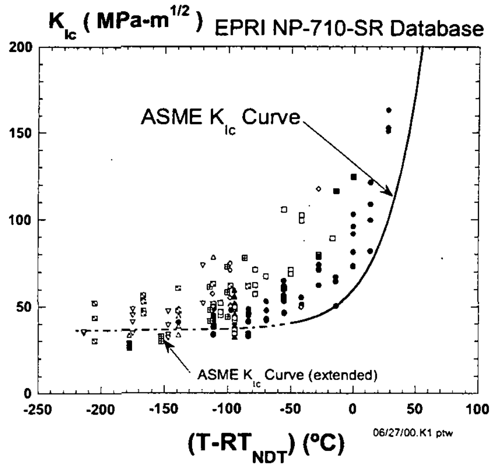


Figure 2.3. The empirical data used to establish the ASME K_{Ic} curve

2.4.2 Irradiation Effects on Index Temperature

As part of their required reactor vessel material surveillance programs qualified under Appendix H to 10 CFR Part 50, licensees attach surveillance capsules to the inner diameter and/or internal structures of the RPV. These capsules, which contain Charpy V-notch (CVN) specimens [ASTM E23], are removed over the lifetime of the RPV [ASTM E185], and the CVN specimens are tested to establish the index temperature T_{30} (the temperature at which the energy consumed in fracturing the CVN specimen is 30 ft-lbs). The difference between T_{30} after some amount of irradiation and T_{30} before irradiation begins is called ΔT_{30}, a metric which has long been used to assess the degree of irradiation damage imparted to the steel. (ΔT_{30} is closely related to the irradiation-induced shift in fracture toughness transition temperature. See [Kirk 01b] and [***EricksonKirk-PFM***].) These ΔT_{30} values from RPV surveillance programs provide the empirical basis to establish embrittlement trend curves that correlate the effect of irradiation exposure and chemical composition on ΔT_{30}. 10 CFR 50.61 adopts the following embrittlement trend curve [Randall 87]:

Eq. 2-2 $$\Delta T_{30} = (CF) f^{(0.28-0.1\log f)}$$

where

- CF is a "chemistry factor" that characterizes the irradiation sensitivity of the steel. CF depends on copper content, nickel content, and product form. 10 CFR 50.61 includes tables of CF values.
- f is the fast neutron fluence in neutrons per cm^2 (E>1Mev) divided by 10^{19}. For the purposes of 10 CFR 50.61, f is defined as the peak fluence at the clad-to-base metal interface at EOL.

2.4.3 Provisions of the Current Rule

10 CFR 50.61 uses the RT_{NDT} index temperature and the ΔT_{30} index temperature shift to estimate the effect of irradiation on RT_{NDT}, as follows:

Eq. 2-3 $$RT_{NDT(f)} = RT_{NDT(u)} + \Re \cdot \Delta T_{30} + M$$

where

- $RT_{NDT(f)}$ is the estimated RT_{NDT} of the vessel material after irradiation to the fluence f. Toughness is determined from $RT_{NDT(f)}$ through its use as an index temperature for the ASME K_{Ic} and K_{IR} curves.
- $RT_{NDT(u)}$ is the unirradiated RT_{NDT}. It can be determined based on either ASME NB2331 or other alternative techniques [NRC MEMO 82, NRC MTEB 5.2].
- $\Re$ is 1 if ΔT_{30} is calculated from chemistry and fluence using Eq. 2-2. If ΔT_{30} is evaluated based on surveillance CVN data, $\Re$ is the ratio of the CF value estimated from the chemistry of the surveillance capsule to the CF value estimated from the heat average chemistry.

ΔT_{30}	is defined by Eq. 2-2.
M	is defined by **Eq. 2-4**.

Eq. 2-4 $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$

where

σ_I	is the standard deviation in the value of $RT_{NDT(u)}$.
σ_Δ	is the standard deviation in the value of ΔT_{30}.

According to 10 CFR 50.61, a nuclear power plant licensee is required to estimate $RT_{NDT(f)}$ at EOL using **Eq. 2-2** for all materials in the vessel beltline. The highest of these values, defined as RT_{PTS}, is compared to the 10 CFR 50.61 PTS screening limit of +300°F (149°C) for circumferential welds and +270°F (132°C) for all other materials. If RT_{PTS} exceeds the screening limit, the licensee is required to either (1) implement flux reduction techniques to keep RT_{PTS} below the screening limit, (2) anneal the vessel according to Regulatory Guide 1.162 [RG 1.162], or (3) submit a safety analysis to the NRC demonstrating that the plant is safe to operate beyond the screening limit.

2.4.4 Evaluation of Operating Plants Relative to the Current PTS Screening Limits

Figure 1.1 compares RT_{NDT} values for all currently operating PWRs evaluated at the end of their originally licensed life (40 years) using Eq. 2-3 with the current 10 CFR 50.61 PTS screening limits. A number of points should be noted:

- No plants currently exceed the screening limit. However, since the operators of the Yankee Rowe Nuclear Power Plant failed to persuade the staff to permit operation in excess of the screening limit using the probabilistic procedures outlined in Regulatory Guide 1.154 [RG 1.154], all plants that have predicted that they would exceed the screening limit before EOL have elected to remain in statutory compliance by (1) implementing flux reduction, (2) pursuing new technological approaches coupled with exemption requests, or (3) using a combination of the two approaches.
- Currently, 10 plants project an RT_{NDT} at EOL ($\equiv RT_{PTS}$) within 20°F of the screening limit.
- While the most embrittled region in one-third of the operating PWRs is the circumferential weld, less than half of those plants (11 of 23) have their operation limited by the circumferential weld because of the higher PTS screening limit currently used to assess these plants (+300°F (149°C) *vs.* the +270°F (132°C) value used for axial welds, plates, and forgings).

3 PTS Reevaluation Project

This chapter describes the PTS Reevaluation Project, which the NRC's Office of Nuclear Regulatory Research (RES) initiated in 1999. The chapter is structured as follows:

- Model Structure: Section 3.1 provides an overview of the model used to evaluate the technical basis for a revised PTS screening limit.
- Uncertainty Treatment: Since the objective of this study is to develop the technical basis for a risk-informed revision of 10 CFR 50.61, a systematic treatment of uncertainties is a central feature of this project. Section 3.2 describes our framework for uncertainty treatment and propagation and provides an overview of how uncertainties were addressed in the PRA, TH, and PFM.
- Assumptions: Section 3.3 summarizes the fundamental assumptions made in developing our model.
- Contributors: Section 3.4 describes the organizations and individuals that have made key contributions over the course of this project.
- Peer Review: Given the complex interdisciplinary nature of this project, and at the request of the NRC's Advisory Committee on Reactor Safeguards (ACRS), the staff convened a panel of external experts to review the study's methodologies, findings, and recommendations. Section 3.5 describes the conduct of this review group and the staff's approach to addressing their comments.

3.1 Model Used to Evaluate a Revised PTS Screening Limit

3.1.1 Restrictions on the Model

The desired outcome of this study is to establish the technical basis for a new PTS screening limit. To enable all commercial PWR licensees to assess the state of their RPVs relative to such a new criterion 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 chemical composition of beltline materials). All of this information is summarized in the NRC's Reactor Vessel Integrity Database [RVID2].

3.1.2 Overall Structure of the Model

Our overall model involves three major components, which are illustrated (along with their interactions) in Figure 3-1:

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 PTS screening limit 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