Pressurized Thermal Shock Rule
Analysis Requirements and Acceptance Criteria,
Related Improvements in Analysis Methods and Data,
and Staff Plans to Revisit the Rule's Technical Basis

1. Introduction

The Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61 (Ref. 1), establishes agency requirements on the ability of the reactor vessel in pressurized water reactors (PWRs) to withstand events in which the vessel is both rapidly overcooled (thermally shocked) and pressurized (or repressurized). These accidents have the following characteristics:

- During fabrication of the pressure vessel, cracks and other flaws are created near the vessel's inner surface;

- During routine operation at normal power levels, decreases of the pressure vessel's toughness (against brittle fracture) occur in the vessel regions adjacent to the core by fast-neutron bombardment of irradiation-sensitive materials (e.g., trace amounts of copper in weld materials).

- Transient events leading to potential PTS scenarios may occur as a result of operational failures or human errors which result in all of the following: rapid cooling of the inside wall of the reactor pressure vessel; continuation of that cooling to a low vessel wall temperature; and maintenance of high reactor coolant system (RCS) pressure, or repressurization. To cause such conditions, serious PTS events would typically involve two or more of the following operational failures or errors: overfeed of one or more steam generators; colder than normal feedwater to one or more steam generators; higher than normal steam flow from one or more steam generators; excessive RCS pressurization by charging pumps and/or safety injection pumps; colder than normal primary system injection flow; or a small break in the RCS of such a size that significant RCS pressure could be maintained by the charging and/or safety injection pumps.

- Given the existence of all five of the above conditions to a sufficient degree, the pre-existing crack would extend within milliseconds (in an axial or circumferential direction, <u>and</u> deeper into the vessel's wall) due to the thermal stresses that result from the inner surface of the vessel being cooler than material deeper in the wall. These thermal stresses would be relieved as the crack grows deeper, and would not by themselves cause the crack to extend through the wall, i.e., thermal stresses would not by themselves fail the vessel. However, if the pressure inside the vessel were to be great enough, the resulting pressure stresses would cause the crack to extend through the vessel wall.

The crack would likely extend the full dimension of the plate or weld. The resulting rapid flow of RCS water out this opening may itself mechanically damage the core (with the potential for some core material to be dispersed out of the vessel), and the lack of subsequent cooling ability (the pressure vessel will likely no longer be able to hold water) would likely result in core overheating and melting.

- Such a failure of the reactor vessel would introduce a number of loads on the core, RCS, and the containment building. These would include dynamic loadings on the core and vessel internals, as well as on the reactor vessel, associated piping, and containment penetrations holding such piping, and containment pressure loadings as the vessel fails. The risk significance of these loadings is not well understood, nor is the degree to which the presence of abundant amounts of water in the containment, and the availability of systems such as the containment engineered safety features, would be effective in mitigating the consequences.

The rule establishes a series of steps which must be performed by PWR licensees in order to permit operation of the facility. The initial step involves a deterministic evaluation of materials properties, and a comparison of the vessel's RT_{PTS} with the screening criterion. If the RT_{PTS} value exceeds the screening criterion, a more general safety analysis or annealing of the vessel may be performed. Regulatory Guide 1.154 (Ref. 2), which includes the use of probabilistic methods, was written to provide one acceptable method for performing this safety analysis. These evaluations and related acceptance criteria are discussed in Section 2 below.

Since the rule was established, the staff has performed a considerable amount of research which has improved the capability to perform the evaluations required or permitted by the PTS rule. This research is summarized in Section 3. The staff's program to revisit the technical basis for the rule, using the results of this research and experience in rule implementation, is described in Section 4.

2. Analysis Requirements and Acceptance Criteria

The PTS Rule describes a process for determining the acceptability of operation of the reactor vessel. Specifically, the rule includes the following requirements:

- Paragraph (b)(1) requires an evaluation of RT_{PTS}, a measure of the materials strength of the vessel at the end of its licensed life. This evaluation is deterministic, with the calculational process described in Section (c) of the rule. Paragraph (b)(2) defines the screening criterion for this evaluation: 270°F for vessel plates, forgings, and axial weld materials, and 300°F for circumferential weld materials. The specification of the value for RT_{PTS} was based, in part, on judgments on the estimated through-wall crack frequency from PTS events, which was judged to be equivalent to vessel failure and core damage.

- If the RT_{PTS} of the limiting vessel material exceeds the screening criterion (at the projected end of licensed life), paragraph (b)(3) requires implementation of flux reduction programs that are "reasonably practicable" to avoid exceeding the screening criteria.

- If no reasonably practicable flux reduction program will prevent RT_{PTS} from exceeding the screening criterion, paragraph (b)(4) requires the licensee to submit a safety analysis to "determine what, if any, modifications to equipment, systems, and operations are necessary to prevent possible failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed." In this analysis, the licensee "may determine the properties of the reactor vessel materials based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques."

Regulatory Guide 1.154 describes one acceptable method for performing the plant-specific safety analysis described in paragraph (b)(4) of the rule. Some important characteristics of this analysis are:

- The analysis is basically a probabilistic analysis, but is more complex than "typical" PRAs in two areas - thermal hydraulics and fracture mechanics - to reflect the important interrelationships among vessel temperatures, reactor coolant system pressures, and the growth of flaws in the vessel.

- The result of the analysis is a frequency of a through-wall crack in the reactor vessel. With the limited understanding of the effects of a vessel failure via a through-wall crack, the staff has assumed that such a crack was equivalent to core damage.

- The analysis does not include consideration of containment performance during PTS-induced accidents or offsite consequences. Simple analyses performed during development of the rule, and discussed in SECY-82-465 (Ref. 3), indicated that containment performance was not expected to be seriously compromised (i.e., the probability of containment failure was assessed to be small), and thus offsite consequences were not a significant concern.

Regulatory Guide 1.154 also describes an acceptance guideline for this safety analysis. Specifically, the guide indicates that if the mean frequency of a through-wall crack is less than $5x10^{-6}$ per reactor year, then continued operation of the facility "would be acceptable to the staff."

It should be noted that experience in use of the guide (for Yankee Rowe) has shown that it is very difficult to use. Without significant revision, the staff does not believe that licensees will use the guide.

3. Improvements in Analysis Methods and Data

In late 1989 and early 1990, NRC staff and the licensee for the now decommissioned Yankee Rowe plant conducted an intensive evaluation of the pressure vessel for that plant (Ref. 4). The staff had identified previously a high level of embrittlement for the pressure vessel; both the licensee and staff turned to Regulatory Guide 1.154 to help determine what regulatory actions needed to be taken. During the course of that evaluation, the staff and industry identified a number of shortcomings and limitations in the regulatory guide method. Chief among these was the technical basis for the fabrication flaw distributions used in the probabilistic fracture mechanics analyses. The Yankee Rowe evaluation, as well as the earlier evaluations that had formed the basis for the rule and regulatory guide, demonstrated that the way flaws were modeled, using 1970's non-destructive examination (NDE) data, and the resulting predicted flaw distribution (Ref. 5), dominated the uncertainty in the calculated probability of vessel failure. Other variables were also shown to be important, including variables in the embrittlement estimation methods, the fracture toughness curves, and the pressure and temperature estimates obtained from thermal hydraulics calculations.

Using its experience from the Yankee Rowe analysis, the staff initiated a research program to specifically reassess the properties of reactor vessels and their impact on PTS risk. Key elements of this research are discussed below. These research results are being used to reconsider the PTS screening criterion, RT_{PTS}, discussed in paragraphs (b)(1) and (b)(2) of the rule, and the safety analysis method discussed in paragraph (b)(4) and in Regulatory Guide 1.154.

Flaw Size, Density, and Location Distributions

One of the results of the staff's technical work underlying the PTS rule (Ref. 6-8) was that the flaw related data (flaw size distribution, flaw location, and flaw density [number of flaws per unit volume of the material]) had the greatest level of uncertainty of the input data developed for these studies. Since the completion of the rule, NRC has supported research to establish a better technical basis for estimating the flaw distributions in the vessel beltline materials. The objective of this research has been to determine the number, location, and sizes of flaws in the vessel material. Key research in this has included:

- The dismantlement and examination of an actual unused PWR vessel in the Pressure Vessel Research User Facility (PVRUF), at Oak Ridge National Laboratory. Weld and adjacent base-plate portions of this vessel were subjected to extensive nondestructive examination (NDE) and selective destructive examination (DE) at Pacific Northwest National Laboratory (PNNL), under contract to NRC (Ref. 9). Similar portions of the Shoreham reactor vessel were also examined by PNNL (Ref. 9)   While a considerably higher number of flaws was found in PVRUF than had been postulated in previous staff analyses, all flaws were found to be sub-surface (embedded), i.e., no surface-breaking flaws were found. In all previous PTS fracture mechanics analyses, every flaw was assumed to be surface-breaking which results in a much higher crack driving force (stress intensity factor, K_I) and overly conservative predictions for the probability of vessel failure.

- The flaw size and density distributions work at PNNL for weld material is being further supplemented with NDE/DE data from River Bend-2 and Hope Creek-2 vessel welds and NDE of PVRUF plate material under NRC funding.

- NRC is now funding a major effort to develop generalized statistical distributions on flaw sizes, flaw locations and flaw densities in welds and base-metals (plates and forgings) of U.S. reactor vessels. These distributions will be developed using a formal expert elicitation process, involving over 15 experts in relevant areas, such as: reactor vessel fabrication, heavy-section steel welding, plates and forging manufacture, vessel inspection (NDE), metallurgy, ASME boiler and pressure vessels construction code, reliability of flawed welded structures, fracture mechanics and failure analysis.

EPRI has also performed NDE tests on some of these reactor vessel beltline materials. Their data are being further processed and will be made available soon for the staff's review.

Irradiation Embrittlement Correlations

Embrittlement correlations are used to predict the increased embrittlement over the life of the vessel due to neutron irradiation. Traditionally used correlations, described in Regulatory Guide 1.99, Revision 2 (Ref. 10), are based on analysis of Charpy v-notch impact-energy test data available in mid-1980's. Since then, a significantly larger body of additional Charpy surveillance data have become available, and the understanding of embrittlement mechanisms has advanced. Under NRC funding, the embrittlement correlations have been improved, and recently published by Modeling and Computing Services and the University of California at Santa Barbara (Ref. 11). Further refinement in the embrittlement correlations is now being performed under NRC funding to include more recent embrittlement data, effect of long irradiation exposure time at vessel normal operating temperatures, and statistical uncertainties in the predicted shift in RT_{NDT} (nil-ductility fracture-mode transition temperature).

Statistical Distributions for Material Fracture Toughness

In the presence of a crack, a material's resistance to fracture is represented by a property called fracture toughness. The toughness values of reactor vessel ferretic steel materials in the present Section XI of the ASME Boiler and Pressure Vessel Code (Ref. 12) are based on 1970's test data that were developed at various temperatures in the brittle-to-ductile fracture-mode transition temperature range (Ref. 13). These tests were conducted under predominantly brittle fracture conditions as per ASTM E-399 test standard (i.e., linear elastic fracture mechanics (LEFM) valid tests in which the loading induced crack-tip plastic zone is very small relative to the test specimen dimensions). To predict catastrophic, sudden brittle fracture (with very little or no plastic deformation) in reactor vessel beltline materials under PTS loading conditions, brittle crack-initiation toughness (K_{Ic}) and crack-arrest toughness (K_{Ia}) are used in performing fracture mechanics analysis. These fracture toughnesses are presented in the ASME Code as a function of normalized temperature ($T-RT_{NDT}$), and are deterministic lower-bound curves that are based on limited databases (171 data for K_{Ic}, and 50 for K_{Ia}, Ref. 13). Since the development of these ASME fracture toughness curves in the 1970's, additional ASTM E-399 standard based (LEFM-valid) test data have become available for vessel materials. Under NRC funding at Oak Ridge National Laboratory (Ref. 14), these additional test data have been identified to extend the original ASME fracture toughness databases (Ref. 13), and to develop rigorous statistical distributions for K_{Ic} and K_{Ia}. These statistical toughness models are presently being refined to decompose the uncertainties into epistemic (state of knowledge) and aleatory (randomness) components that can be used in overall uncertainty analysis to be performed in the PTS Rule re-evaluation. This additional work is being carried out at the University of Maryland under NRC funding for the probabilistic uncertainty aspects, and the micro-mechanical physical basis modeling under EPRI funding.

Statistical Distributions for Material Chemistry and Initial RT_{NDT}

Statistical distributions for plant-specific material chemistry (nickel, copper) and initial RT_{NDT} (RT_{NDTo}) need to be developed to represent the local variability of plate and weld materials used in determining the shift in RT_{NDT} due to irradiation embrittlement effects. This work is now being performed by NRC staff.

Beltline Vessel Fluence Calculations

Accurate calculation of fluence values in the reactor vessel beltline region is crucial for determining the effect of irradiation embrittlement on fracture toughness of the vessel materials. Fluence calculations and the uncertainties in the end of license fluence values for each of the plants that are being studied in the PTS Rule reevaluation will be based on up-to-date information of the plant's cycle-by-cycle fuel loading history and the draft regulatory guide DG-1053 proposed method (Ref. 15). This work is now being carried out at Brookhaven National Laboratory under NRC funding.

Improvements in Fracture Mechanics Methods

A new version of NRC's probabilistic fracture mechanics (P.M.) analysis computer code, FAVOR (Fracture Analysis of Vessels -- Oak Ridge), has been under development at Oak Ridge National Laboratory under NRC funding (Ref. 16) to investigate brittle fracture in PWR vessels under thermo-mechanical transient loading conditions, such as PTS . A number of significant improvements have been made in the code, and some others are presently being made, so that it can be used to perform the more realistic PFM analysis to be performed in the PTS rule reevaluation. Notable among these are:

- The effect of clad to base-metal differential thermal expansion induced residual stress is determined from more realistic, experimentally measured data.

- The residual stress distribution through the vessel has been modified to reflect more realistic information obtained from measurements on a non-operating PWR vessel.

- The stress intensity factor, K, solutions for semi-elliptical surface flaws have been determined for clad vessels using finite element computations in which the applied thermal and pressure induced stresses are represented by third-order polynomials through the vessel thickness.

- The stress intensity factor, K, solutions for elliptical sub-surface (embedded) flaws were determined using an ASME Section XI method (Ref. 11) which has been validated selectively by finite element computations.

4. Staff Program to Revisit Technical Basis

In 1999, the staff initiated a program to revisit and improve the realism of the technical basis of the PTS Rule, using the results of the research described above and experience in implementation of the rule. The key elements of this program, and dates for completion, are shown in Figure 1 and summarized below. As may be seen, this work is scheduled to be completed in early FY2002.

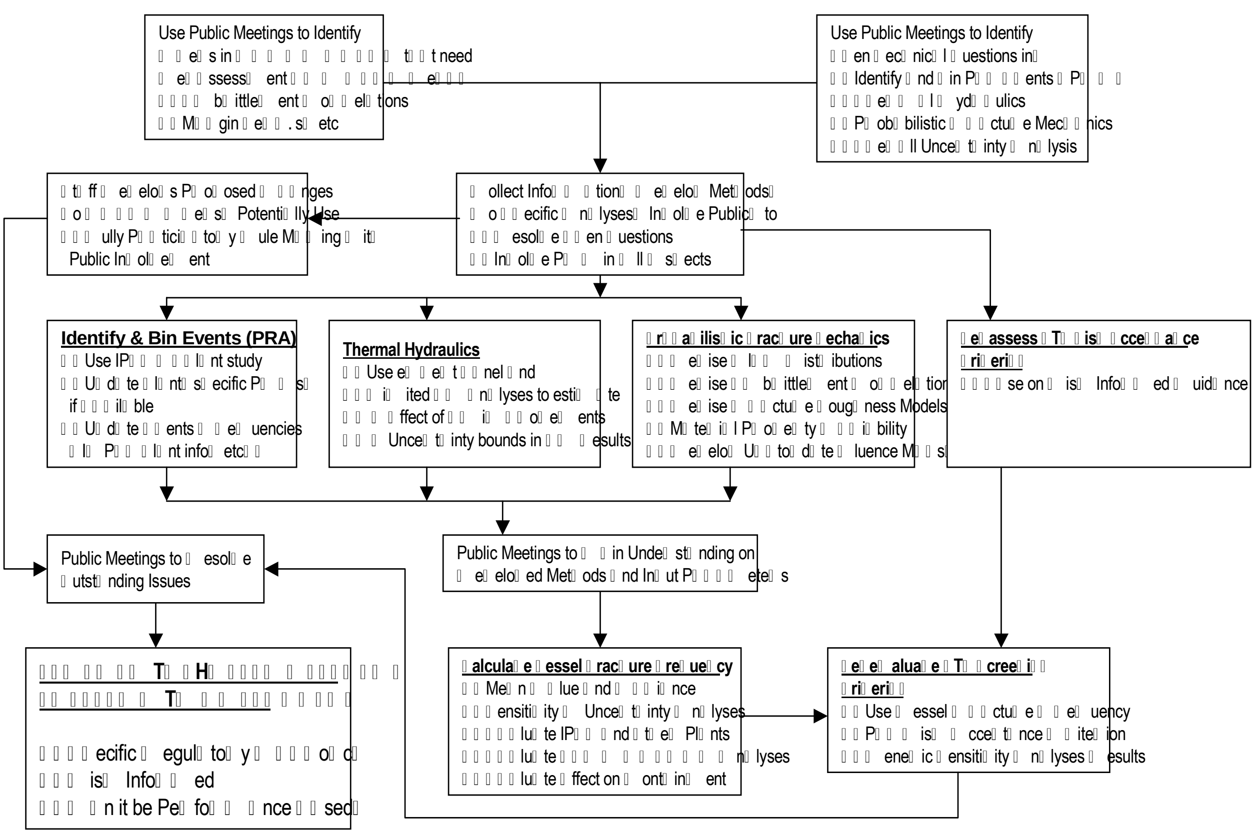
- Identify and Bin Events (PRA)

  The element provides information on the types of event sequences which could lead to PTS events, and the frequencies of these sequences. In this element, the staff will review previous PTS risk studies, review more recent PRAs and operational events to identify new sequences, provide an updated set of potentially challenging sequences, group these sequences into sets having similar thermal hydraulic characteristics, and estimate the frequencies (including estimates of uncertainties) of occurrence of these sets of sequences.

- Thermal Hydraulics

  The task of the thermal hydraulics work is to provide the reactor vessel down comer temperature and pressure boundary conditions for each potentially important group of event sequences, using state-of-technology computer models. The boundary conditions of interest are time-dependent system pressure, fluid temperature in the down comer, and the convective heat transfer coefficient from the fluid to the wall. Estimates of the uncertainties in these values will be provided.

Figure 1. Staff Process for Reevaluation of the PTS Rule Technical Basis



- Probabilistic Fracture Mechanics

  As discussed in Section 3 above, the models and data used in probabilistic fracture mechanics have been significantly improved in the past several years. In particular, the fracture mechanics models, the embrittlement database and embrittlement correlation, inputs for flaw distributions, and the probabilistic fracture mechanics (PFM) computer code have been refined. The principal focus of the probabilistic fracture mechanics element of the staff's work is to provide estimates of the probabilities of through-wall cracks for each of the sets of event sequences and thermal hydraulic conditions identified in previous elements including uncertainties. A major objective of this analysis is to determine the synergistic impact of these fracture-technology refinements together with updated PRA and TH systems analysis results on the probabilities of through-wall cracking failure of the reactor vessel.

- Reassess Probabilistic Aspects of PTS Screening Criterion

  In parallel with the development of revised technical information on PTS events and their frequencies and consequences, the staff is reassessing the basis for the "acceptable" frequency of such events.

- Calculate PTS Through-Wall Crack Frequency

  The frequency of a through-wall crack will be estimated for four selected plants, considering all event sequences and their frequencies, thermal hydraulic information, and PFM information. This frequency will be considered the same as the frequency of vessel failure and core damage. A simple analysis (involving less than six staff-months of effort and discussed in more detail below) of the impact of such vessel failures on containment performance during PTS events will also be performed as part of this element. Uncertainties in these frequencies will be estimated. The results from this work will be used to develop insights regarding the PTS risk for all plants potentially vulnerable to this event.

  As part of the integrated assessment of PTS, the staff intends to perform a scoping analysis to identify and assess the technical issues and risk implications of the impact of reactor vessel failure (due to PTS) on containment integrity. Consistent with the intent of the staff to use the SECY-00-0086 framework, this analysis would principally focus on the potential for PTS accidents to result in large early releases of radioactive material, including potential failures of penetrations due to PTS-induced motion of the reactor coolant system. The staff intends to make maximum use of available technology, including the results of the NRC severe accident research which resolved key containment integrity issues. A key aspect of the approach would be the development of a PTS containment event tree and the integrated analysis of vessel failure and concomitant blowdown conditions. This is the approach that the staff successfully used for demonstrating containment integrity under severe accident loading conditions that were originally thought could lead to an early containment failure, e.g., direct containment heating, α-mode (steam explosion-induced) containment failure, and containment liner meltthrough. Insights gained from these past efforts have shown that consistent treatment of the thermal-hydraulic and severe accident phenomena and containment structural response yields potential additional benefits in an integrated risk assessment. To the extent possible in this scoping study, improved analytical methods developed for thermal hydraulic and severe accident analysis will also be used. However, should the assessment of the timing and magnitude of fission product release become very resource and time intensive, alternative approaches to resolve this issue will be considered. One alternative approach could be to limit the frequency of vessel failure due to PTS to 1×10^{-6} per reactor year in order to meet established guidelines for large early release frequency.

- Re-evaluate PTS Screening Criterion

  The staff will develop recommendations for new values of the RT_{PTS}, using the estimates of through-wall crack frequency and the reassessment of the probabilistic aspects of the screening criterion, including containment performance.

- Propose Technical Basis for Revision to 10 CFR 50.61

  The information created and assembled in previous tasks will be integrated into a form which could support a new version of the rule. When completed, this material will be provided to the Commission with a recommendation on proceeding with rulemaking.

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