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# REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

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# FORMAT AND CONTENT OF PLANT-SPECIFIC PRESSURIZED THERMAL SHOCK SAFETY ANALYSIS REPORTS FOR PRESSURIZED WATER REACTORS

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# INTRODUCTION

## Background and Purpose of This Guide

The pressurized thermal shock (PTS) rule, § 50.61 of 10 CFR Part 50 issued on July 23, 1985 (50 FR 29937), establishes a screening criterion based on reactor vessel nil-ductility-transition temperature ($RT_{NDT}$). The screening criterion was established after extensive industry and NRC analyses regarding the likelihood of vessel failure due to PTS events in pressurized water reactors (PWRs). The analyses were applied generically and contained conservative assumptions to make the results bounding for any PWR. Based on the results, the NRC concluded that the risk due to PTS events is acceptable at any plant so long as the $RT_{PTS}$* of the reactor pressure vessel remains below the screening criterion.

Extensive safety analyses are required by the rule for any plant that wishes to operate with $RT_{PTS}$ values above the screening criterion. The recommended methods to be used in performing the analyses are outlined in this guide. The purpose of the analyses is to assess the risk due to PTS events for proposed operation of the plant with reactor vessel $RT_{PTS}$ above the screening criterion. Effective 1 year after the publication of this regulatory guide, Section 50.61 requires that these analyses be completed 3 years before the screening criterion would be exceeded to allow adequate time for implementation on the plant of any corrective actions assumed in the analyses before the plant operates above the screening criterion.

This regulatory guide describes a format and content acceptable to the NRC staff for these plant-specific PTS safety analyses and describes acceptance criteria that the NRC staff will use in evaluating licensee analyses and proposed corrective measures.

The references listed in this guide include a set of analyses sponsored by the NRC that, taken together, constitute an example of the analyses described by this guide. The staff recommends that these references be extensively used, along with this guide, by those performing the plant-specific PTS analyses required by the PTS rule, § 50.61. References 1, 2, and 3, for example, each represent an analysis by the Oak Ridge National Laboratory (ORNL) predicting through-wall crack frequency for one PWR. These references will provide guidance through the analyses. Reference 3 (analysis of H. B. Robinson) should be most helpful because it was the last one performed and includes the experience gained in performing the two earlier analyses.

## Objectives of Plant-Specific PTS Safety Analysis Reports

Paragraph 50.61(b)(4) requires that a licensee whose plant will exceed the screening criterion before expiration of the operating license submit safety analyses to determine what, if any, modifications to equipment, systems, and

*To avoid confusion among several (preexisting) slightly different definitions of $RT_{NDT}$, § 50.61 contains its own definition of an $RT_{NDT}$ (called $RT_{PTS}$) to be used when comparing plant-specific vessel material properties with the PTS screening criterion.

operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. These analyses must include the effects of all corrective actions the licensee believes necessary to achieve an acceptable PTS-related risk for continued operation of the plant. The final objective of the plant-specific PTS study, therefore, is to justify continued operation of the plant by demonstrating that the likelihood of a through-wall crack during continued operation is acceptably low. The study must include calculation, for the remainder of plant life, of the expected frequency of through-wall cracks due to PTS.

In calculating these results, it will be necessary to:

- Identify the dominant accident sequences.
- Identify operator actions, control actions, and plant features important to PTS.
- Estimate the effectiveness of potential corrective actions in reducing the expected frequency of through-wall cracks.
- Identify the sources and approximate magnitude of the major uncertainties and their effects on the conclusions.
- Present and justify the licensee's proposed program for corrective measures.
- Present and justify the licensee's proposed basis for continued operation at embrittlement levels above the screening criterion. This must include comparison with the acceptance criteria described below of the PTS-related through-wall crack frequency with corrective actions implemented as necessary.

Staff Review of Plant-Specific PTS Safety Analysis Reports and Acceptance Criteria for Continued Operation

The PTS rule specifies a screening criterion based on $RT_{NDT}$ (called $RT_{PTS}$ for use as defined within the rule) of 270°F for axial weld and plate materials and 300°F for circumferential weld materials. As detailed in SECY-82-465 (Ref. 4), these values were selected based on generic studies of the expected frequency and character of a wide spectrum of transients and accidents that could cause pressurized overcooling of the reactor vessel (PTS events) and on operating experience data. The risk due to PTS events was assessed in terms of probabilistic fracture mechanics calculations of the expected frequency of through-wall crack penetration of the pressure vessel due to the PTS events. In selecting the screening criterion based on those calculations, the conservative assumption was made that any through-wall crack could result in severe core degradation or melt. Core melt itself was viewed as an event to be avoided even though risk to the public due to such an event in terms of person-rems and early and late fatalities was not calculated with any certainty. The estimated through-wall crack frequency developed as a function of $RT_{NDT}$ for axial welds (Fig. 8.3 of Ref. 4) is shown in Figure 1.

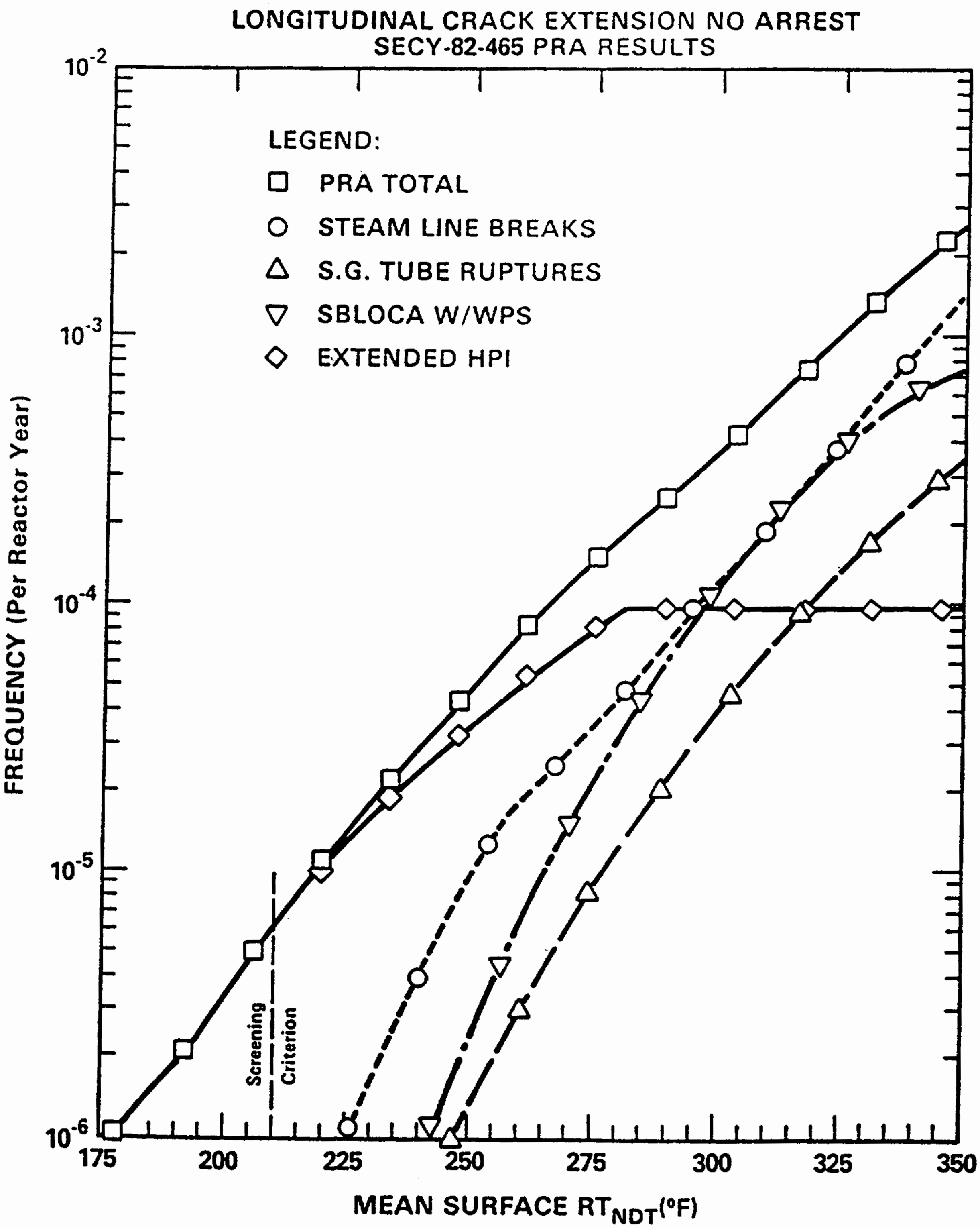
LONGITUDINAL CRACK EXTENSION NO ARREST
SECY-82-465 PRA RESULTS
LEGEND:
PRA TOTAL
STEAM LINE BREAKS
S.G. TUBE RUPTURES
SBLOCA W/WPS
EXTENDED HPI
Screening Criterion
FREQUENCY (Per Reactor Year)
$10^{-2}$
$10^{-3}$
$10^{-4}$
$10^{-5}$
$10^{-6}$
175
200
225
250
275
300
325
350
MEAN SURFACE $RT_{NDT}$(°F)

Figure 1

The $RT_{PTS}$ screening criterion selected by the staff corresponds to a mean (or average) "best estimate" surface $RT_{PTS}$ of 210°F. The staff used a "2-sigma" value (spread between "best estimate" and "upper limit") of 60°F;* thus the screening criterion expressed in terms of $RT_{PTS}$, which, by definition, is this upper limit value, was selected at 210 + 60 = 270°F. For axial weld and plate materials, Figure 1 gives a through-wall crack frequency of about $5 \times 10^{-6}$ per reactor year at 210°F, which corresponds with an $RT_{PTS}$ of 270°F. For circumferential welds, the same frequency is believed to be bounded by an $RT_{PTS}$ of approximately 300°F (Ref. 4). The Commission concluded that the PTS-related risk at any PWR is acceptable so long as the $RT_{PTS}$ values remain below the specified screening criterion.

It was realized that there are many unknowns and uncertainties inherent in the probabilistic calculations; thus it was with deliberate intent that conservative assumptions such as those stated above were made. The expectation was that the true risk at any plant due to PTS events would in all likelihood be considerably below that derived from Figure 1 and would therefore be acceptable. Also contributing to the belief that the real PTS risk at any given plant was lower than that resulting from the analysis in Reference 4 was the belief that many of the generic plant assumptions made in Reference 4 (e.g., material properties, system performance, crack distribution) would prove to be overconservative for analysis of a specific plant and that the resulting plant-specific analysis, when performed, is likely to result in a reduced prediction of PTS risk.

If the plant-specific PTS analyses submitted by licensees in accordance with § 50.61 using the methodology described in this guide (or acceptable equivalent methodology) predict that the PTS-related, through-wall crack penetration mean frequency will remain less than $5 \times 10^{-6}$ per reactor year for the requested period of continued operation, such operation would be acceptable to the staff.

In all the analyses performed, the licensee must justify that the important input values used are valid for the remaining life of the plant.

Recommended Format

The recommended content of plant-specific PTS safety analyses is presented in Chapters 1 through 10 of this guide. Use of this format by licensees will help ensure the completeness of the information provided, will assist the NRC staff in locating the information, and will aid in shortening the time needed for the review process. If the licensee chooses to adopt this format, the numbering system of this guide should be followed at least down to the section level. Certain sections may be omitted if they are clearly unnecessary to provide for comprehension of the analysis or if they are repetitive.

*Based on preliminary $RT_{NDT}$ data from many plants (see Table P.1 of Enclosure A to Ref. 4).

Additional guidance on style, composition, and specifications of safety analysis reports is provided in the Introduction of Revision 3 to Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)."

The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the issuance of this regulatory guide.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 50, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 50 have been cleared under OMB Clearance No. 3150-0011.

## 1. OVERALL APPROACH, SCOPE OF ANALYSIS, AND REPORT ORGANIZATION

This chapter is to describe the overall approach to the analysis and outline the individual tasks in terms of the nature and source of input, the methods used for analysis, and the nature and subsequent use of the output. The interrelationship of the tasks should be described and should be illustrated by a flow chart. How the analysis tasks are integrated to achieve the results and conclusions is to be described.

Major emphasis should be placed on analyzing event sequences leading to vessel through-wall cracking and corrective actions to prevent this from occurring.

The report should include both probabilistic and deterministic fracture mechanics analyses. The probabilistic analyses should be used to determine the statistical likelihood of vessel through-wall crack penetration assuming a crack size distribution appropriately justified for the vessel being analyzed and appropriate uncertainties and distribution of the significant input parameter such as material properties. The deterministic analyses should be used to evaluate the critical time interval in the transient during which mitigating action can be effective. The deterministic analyses should be carried out using the two sigma upper and lower bounding values of the appropriate parameters such as fluence, copper content, nickel content, fracture initiation toughness, fracture arrest toughness, and ductile fracture toughness.

The input to the probabilistic analysis should be best estimates based on appropriate assumptions. Uncertainties and conservatisms should be explicitly presented in the decision rationale for the licensee's proposed corrective measures and basis for continued operation.

The analysis should include effects of operator actions, control system interactions, and support systems such as electric power, instrument air, and service water cooling.

The report should be organized by starting with a description in Chapter 1 of how the report chapters and supporting appendices are interrelated and what material is in the appendices.

The main report should describe the objectives and overall approach used in the study, outline the plant systems analyzed, describe the engineering analyses performed, present the results obtained and conclusions drawn, and present and justify the licensee's proposed program of corrective measures.

Appendices should contain data, detailed models, sample calculations, and detailed results needed to support the various chapters of the report. Appendices should contain little supporting text. Instead, the nature and relevance of material in the appendices should be described in the pertinent chapters of the main report.

Throughout the guide, wherever it is specified or suggested that detailed descriptive materials should be submitted as part of the licensee's analyses, these detailed materials may be provided by incorporation of reference material already submitted to the NRC (for example, in the final safety analysis report). It remains the responsibility of the licensee to provide a coherent, readable document that does not unduly burden a reviewer with collecting extensive references before proceeding with the review. Therefore, care should be exercised in limiting such material provided by reference to the reviewer who is conducting an extensive, detailed evaluation of the submitted work.

Certain details (noted in Chapter 1 and in Section 4.3 of this regulatory guide) that have not been previously submitted to the NRC may be made available for NRC inspection and may also be referenced by the submitted analyses.

## 2. PLANT DATA

This chapter is to briefly describe plant systems and operations pertinent to PTS. Chapter 2 of Reference 3 (the H. B. Robinson analysis by ORNL) provides a good example. Supporting appendices or references are to present the design and operating data used in the analysis or needed to understand the analysis. References to other documents (e.g., the final safety analysis report (FSAR)) should indicate specific sections. (Reliability data, however, are to be in Section 3.4, "Sequence Quantification," or its supporting appendices and references.)

### 2.1 Systems Pertinent to PTS

Summarize design and operating features of systems pertinent to PTS. Illustrate each system with a simplified process and instrumentation diagram or a single line diagram. Identify on each illustration any interfaces with other systems. For each system, include a table summarizing key design and operating data. Give the maximum, minimum, and nominal values for those cases in which design data may vary with time (for example, high-pressure injection (HPI) water temperature may vary with season). Such values used in the analysis should be identified and justified. Refer to appendices or other documents (e.g., specific sections of the FSAR) as necessary for more details. Systems to be considered should include pertinent portions of:

- Reactor cooling system
- Condensate and main feedwater systems
- Steam system
- Auxiliary feedwater system
- Reactor protection system
- Chemical and volume control system
- Emergency core cooling systems
- Instrumentation and control systems
- Support systems
  - Electric power
  - Instrument air
  - Service cooling water

### 2.2 Reactor Vessel

Summarize the reactor vessel construction and its material properties. Use tables, drawings, or graphs to show:

- Vessel design (including weld locations and hot leg and cold leg penetrations).

- Vessel materials and chemical composition in the beltline region (including both base and weld material properties).

- Vessel fabrication procedures, particularly welding and cladding.

- Vessel properties (e.g., $RT_{NDT}$, initial $RT_{NDT}$, appropriate fracture toughness data, including the upper-shelf regime, residual stresses, flaw density distribution, etc.). Describe and justify methods used to calculate or otherwise determine properties.

Available information on the vessel properties should be reexamined in detail to fill any gaps in the supporting data for making an estimate of $RT_{NDT}$ and to support resolution of any disagreements about the validity of values used.

Few data are currently available and validated to support the selection of a value for the initial $RT_{NDT}$. The confidence that can be placed in estimates of the initial $RT_{NDT}$ depends not only on material tests but also on the accurate documentation of welding technique, weld wire used, and weld flux used. The credibility of such estimates could be enhanced by performing more tests on archival material, by discovering previously unreported test results on weld specimens from the particular plant, or by evaluating properties of welds considered typical of the plant-specific weld.

## 2.3 Fluence

Present (or incorporate by reference to a submitted report) the current and projected fluence on the vessel using benchmarked computer programs and methodology and information from neutron flux surveillance dosimetry. Use the weld locations and fluence values to identify the critical welds. Show how the fluence varies along the length and depth of the critical welds. Describe the basis for these estimates and their uncertainty. These fluence values should be benchmarked, for example, through use of ENDF/B-IV or V cross sections, to quantify the error.

## 2.4 Inservice Inspection Results

To the extent pertinent to the probabilistic analysis and proposed corrective actions, summarize:

- Results - The number, size, depth, and location of any flaws found should be well defined and described.

- Methods used - The method used to perform the inspection should be well described with documentation of any validation information.

Note: Only those inservice inspections (ISIs) that have actually been performed should be discussed in this section. Improved ISI programs as proposed by the licensee should be described under corrective measures in Chapter 8, "Effect of Corrective Actions on Vessel Through-Wall Crack Frequency."

## 2.5 Plant Operating Experience

Summarize overcooling transients that have occurred at this station and similar stations. Also, summarize lessons learned from these and other transients, and indicate actions taken to prevent recurrence or minimize severity of overcooling transients.

## 2.6 Operating Procedures

This section provides procedural data, e.g., what the operator is supposed to do and when. This section, for example, should present and describe the important operator actions as defined by existing procedures associated with potential overcooling transients. Also emphasize how the procedures were evaluated and optimized in light of any competing risks that might arise from events other than PTS events to ensure that overall plant safety is appropriately balanced. The conditions under which the operator takes each action, the expected time for performing the action, and how the time was derived should be identified. Some examples of these operator actions are:

- Trip reactor coolant pumps.
- Throttle/terminate emergency core coolant.
- Throttle/terminate main and emergency feedwater.
- Restore main and emergency feedwater.
- Isolate break (primary or secondary).

Supply a summary of training materials associated with overcooling events in general and with respect to principal initiators. In addition, a summary of simulator exercises associated with potential overcooling events should be provided.

Note: Proposed improvements in procedures, diagnostic instrumentation, display systems, and operator training should be presented in Section 8.2 under the licensee's program of corrective measures.

# 3. DETERMINATION OF DETAILED PTS SEQUENCES FOR ANALYSES

This chapter is to present the methods and analyses used to identify those transient sequences that could contribute significantly to the PTS risk. A good example is presented in Chapter 3 of Reference 3. The scope includes identifying initiating events, developing event trees, modeling and quantifying the reliability of relevant systems and operator actions, and collapsing the event trees to identify specific relevant sequences. Detailed models, data, and sample calculations should be included in appendices or referenced. However, the logic of the analysis, criteria used, results, and insights gained are to be described in the main report.

## 3.1 Approach Used

Describe how the material presented in this chapter fits into the overall PTS study. Provide a general description of the process used to identify PTS sequences. It should be made clear how the approach used will result in completeness of identification of all classes of events that could contribute significantly to PTS risk, how specific events are selected for more detailed analysis to represent each class, and finally how the events so analysed are used to determine total PTS risk at the plant.

## 3.2 Sequence Delineation

Identify potential overcooling transients in a well-defined manner, and document them in such a way that it is clear to a reviewer that all important potential overcooling conditions have been considered. Classes of initiators should be developed, important variations of initiators within each class should be identified, and potential transients resulting from these initiators should be defined.

Operating experience at the specific plant and at similar plants should be carefully examined to aid in the identification of potentially significant PTS initiators, contributing failures, and potential corrective actions. The ORNL contribution to Systematic Evaluation Program reviews (Ref. 5, for example) is a technique that can be used for this purpose.

### 3.2.1 Development of Classes of Initiators

Any class of transients that could lead to overcooling of the reactor vessel should be considered in the analysis. It should, however, be appropriate to use logical arguments to eliminate classes of transients as actual PTS initiators whenever justifiable. Examples of initiators that should be included are:

- Loss-of-coolant accidents (LOCAs), including steam generator tube rupture accidents.
- Steam line breaks.
- Overfeeds.
- Combinations of these, including possible return to criticality.

### 3.2.2 Identification of Important Initiator Variations

After the classes of potential initiators have been identified, it is important to consider variations within any individual class. These variations should include:

1. Decay heat level - The decay heat level, determined by recent operating history of the plant, can have a major impact on the potential consequences of a given event. Thus, various decay heat conditions should be considered. Clearly, decay heat associated with a reactor trip from full power (assuming operation at full power for some considerable time) should be examined. Zero decay heat represents the opposite extreme but for all practical purposes occurs only once at the beginning of life for the plant when PTS is not important. Therefore, the analyst may choose to use some other level of decay heat that would cover potential decay heat conditions after the initial startup of the plant. The reasons for choosing particular decay heat levels for analysis should be documented. Each identified initiator should be examined at all decay heat levels defined whenever appropriate.

2. Power level - Power level may be important since certain equipment conditions or configurations may only exist at certain power levels, e.g., hot standby. As in the case of decay heat level identification, the reasons for the selection of specific power levels for analysis purposes should be stated. It should be noted that under certain conditions a reactor system may be at a high power level with a low decay heat condition.

3. Location of event - In many instances the location of the event is defined. For example, an event consisting of a failed open turbine bypass valve has the location defined since it is a specific valve failure. However, for some events such as pipe breaks, the location is not defined and could have an impact on the progression of the event. In the case in which location is not defined, all locations that could be significant should be considered. Each location should then be eliminated by logical argument, bounded by consequences associated with another location, or treated as a separate event.

4. Magnitude of event - Many of the initiators can occur to various degrees. For example, a LOCA can range from a very small break to a full guillotine pipe break. Break sizes should be examined to identify categories of sizes that lead to similar system conditions. In the case of the LOCA event, special consideration should be given to the identification of break sizes that could lead to loop flow stagnation. The larger-sized LOCAs typically do not contribute to PTS risk since the pressure cannot be maintained because of the large flow out of the break.

### 3.2.3 Definition of Potential Transients Resulting from Each Initiator

After the complete set of significant initiators has been defined, event trees are required to identify potential sequences resulting from each initiator. The development of the event tree headings and branches should be done in a consistent and logical manner. This was done in the ORNL studies (Refs. 1, 2, and 3) by using what have been called system state trees. These trees define the potential states of each plant system of interest conditional on specific thermal-hydraulic conditions. Initiator-specific event trees can then be developed by examining the system state trees with respect to each initiating event. A similar or equivalent approach should be used to ensure traceability of the event trees and to ensure that important sequences are not inadvertently eliminated.

Support system failures should also be presented within some type of event tree structure. If the event trees are developed as previously described, any support system failure would most likely lead to a sequence of events that is already mapped out on the event trees, but in many instances with a higher probability of occurrence. In other cases, it may be necessary to define event trees resulting from a support system failure. In either case, it is important that the support systems be examined to identify their potential impact on overcooling conditions. The results of this examination should be presented as a separate section with the identification of specific support system failure sequences of interest. The support system review should at least include:

- The electrical supply system.
- The compressed air instrument system.
- The component and service water systems.

### 3.3 Operator Effects

The operator effects are analyzed in two separate sections. In this section the potential operator actions are identified. These actions are further analyzed in Section 3.4 in which the probabilities associated with the performance of an operator action are developed.

The operator can improve, aggravate, or initiate an overcooling transient. All three of these categories should be discussed in this section.

1. Procedures and/or the operators' general knowledge can lead to actions that improve the conditions associated with an overcooling event. Explanation should be included as to why it is perceived that this action would be taken. Where appropriate, these operator actions should be either included directly on the event trees or presented as separate operator action trees that can later be coupled with the principal event trees.

2. Although the ORNL studies (Refs. 1, 2, and 3) did not include operator-initiated events or events aggravated by operator actions contrary to procedures, this category of events should also be examined as part of a plant-specific analysis.

3. The analyses should include a quantitative approximation of the PTS risk resulting from operator acts of commission. Also included should be the possibility that an operator could initiate or exacerbate some milder event into a more severe PTS-type event. Since there is no generally accepted way to perform such analyses, the approximation used by the licensee for this purpose should be discussed and justified for applicability to this particular plant. The "confusion matrix" approach (Ref. 6) used in human reliability analysis could provide an acceptable structure for identifying and analyzing these potential operator actions.

## 3.4 Sequence Quantification

Quantify the event trees by using identified initiating event frequencies, appropriate conditional probabilities associated with the success or failure of specific equipment operations, and success and failure probabilities associated with operator actions. Plant-specific data should be used whenever appropriate to define these probabilities, including appropriately adjusted simulator studies. This should be supplemented by vendor-specific or PWR-generic data bases when plant-specific data do not appear to provide an adequate data base. Reference 7 includes guidance about treatment of generic and plant-specific data. Its appendices include an updated generic data base that should be used.

Identify by specific reference or provide in appendices all the reliability data used as input to quantify the event sequences. An explanation should be supplied as to how the data were derived for each data point.

### 3.4.1 Initiating Events

Initiating event frequencies should be developed based on the number of observed events within selected periods of operation for similar plants under consideration. If no failures have been observed and no other information is available with which to estimate a probability, a standard statistical method such as the Poisson distribution can be used to determine a probability, or the technique described in Appendix B to Reference 3 for estimating plant-specific initiating event frequencies can be used. For some initiators, it may be necessary to estimate the frequency of events in a particular operating mode, e.g., hot zero power. The data should be researched to identify trends associated with the occurrence of the event and the operating mode. In addition, the initiator itself should be examined to identify physical conditions that might favor failure in one mode rather than another. If this examination reveals no evidence of correlation between frequency and operating mode, the fraction of time spent in each operating mode can be used as a weighting factor.

### 3.4.2 Equipment Failures

Following each initiating event, certain components are designed to perform in a defined manner. Failure of a component to perform its required function could lead to PTS considerations. Thus, it is necessary to assign a failure and successful operation probability for each component on a per-demand basis. These probabilities can be obtained by estimating the number of failures observed within a period of time, combined with an estimate of the number of demands expected within that same period, or by developing fault trees. If no failures have been observed and no other information is available with which to estimate a failure-on-demand probability, a standard statistical method can be used to develop a probability.

As with all event trees, the probability associated with a particular branch is conditional on the prior branches in the sequence. Questions of conditional probability should be carefully considered before a failure probability is assigned.

The potential for coupled or common cause failures within a system or between systems should be examined in the analysis. Careful consideration should be given to increasing the failure potential of a component, given the failure of one or more components of the same type in the same system or in other systems being subjected to the same environment or fault causes. As additional components of a particular type are postulated to fail, the probability for the next component of the same type to fail should increase. Based on the ORNL analysis, a simplified approach would be to assume that the failure probability of the second component, given that the first component has failed, might be as high as 0.1. The third component might be assumed to fail with a 0.3 probability, given the failure of two identical components. One could then assume that, after the failure of three components of the same type, all remaining components of that type in the same or in other systems being subjected to the same environment or fault causes would fail with a probability of 1.0. The licensee should discuss how these types of coupled failures are handled in the analysis.

Common cause failures of a different type may occur, as previously discussed, through the failure of a support system or a control signal. An analysis of these potential failures should be made and the branch probabilities should be adjusted whenever appropriate.

### 3.4.3 Operator Actions

Operator action probabilities are particularly difficult to determine because of the lack of a data base. The problem is further complicated when time becomes an important variable. The procedure outlined below represents one approach to quantifying operator actions. This procedure should be conservative for any operator action performed as required by procedures assuming that the equipment required is operational. For operator actions that might not be associated with procedural steps, it is not clear that this simplified approach would produce conservative frequencies. Therefore, the approach described would only be recommended for operator actions associated with procedural steps. Regardless of the method used, the human error probabilities used in these analyses should be supported by data validated for the plant being analyzed.

1. Identify operator actions - In this step the procedures associated with each initiator would be reviewed to identify those operator actions that would have an impact on downcomer temperature.

2. Identify time constraint - In the case of each operator action, the transient would be reviewed assuming no operator action to identify the timeframe available for successful completion of the operator action.

3. Assign screening failure probabilities - In this step a conservative value for the failure of the operator action would be identified. For operator actions required by procedures to be performed within the first 5 minutes of the transient, the time-reliability curve as presented in NUREG/CR-2815 (Ref. 7) could be used to identify a screening value. After 5 minutes, a value of 0.9 for success and 0.1 for failure would be assumed for all operator actions. The entire PTS analysis would then be completed using these screening values.

4. Identify dependency factors - In some instances, there may be coupled failures associated with operator actions just as there were coupled failures

associated with equipment failures. In many instances, the potential failure of an operator action may be linked, to various degrees, to the success or failure of a previous operator action. Thus, it is recommended that each operator action be reviewed with respect to dependency. This can be accomplished using the dependency tables as presented in the human reliability handbook (Ref. 8).

5. If any of the dominant sequences involve the failure of an operator action, a more comprehensive evaluation of the failure would be performed for that operator action. When necessary, the comprehensive evaluation should be performed using a human reliability methodology. The acceptability of this methodology for the purpose should be justified by the licensee (Refs. 9 through 13).

## 3.5 Event Tree Collapse

Collapse the event trees using a frequency screening criterion to form a list of specific sequences and a set of residual groups to be analyzed. This is important since the event trees may generate thousands of end states that cannot be individually analyzed. A screening value of 1.0E-7/reactor year is recommended. This value should ensure that important sequences are treated individually, and it should also help to keep the size of the residual small. This is particularly important since it may be necessary to treat the residual using a bounding consequence condition.

### 3.5.1 Specific Sequences

Those sequences that survive the frequency screening should be defined and their frequency noted. It is recommended that some identification be assigned to each sequence to enhance its traceability through the remainder of the analysis. Grouping and identifying each sequence with respect to initiator type may also prove helpful.

### 3.5.2 Residual Groups

Those sequences that do not survive the frequency screening must also be considered. They should be grouped together based on transient characteristics to form a set of residual groups. The residual groups should be reviewed to identify sequences that should be grouped with previously defined sequences because of transient similarity or should be specifically evaluated because of their severe consequence. It is important to attempt to reduce the size of each residual group since it will be necessary to assign a bounding consequence that would apply within each group. Each residual group should be defined and its frequency noted.

# 4. THERMAL-HYDRAULIC ANALYSIS

This chapter is to present the reactor coolant pressures, temperatures, and heat transfer coefficients at the vessel's interior surface in the beltline region for the set of overcooling sequences that envelops the plant's potential for experiencing a PTS event. A good example is presented in Chapter 4 of Reference 3. Also the chapter is to present the details of the analysis methods used to obtain these fluid conditions and is to include the following sections:

1. The thermal-hydraulic analysis plan and logic.

2. A description and evaluation of the thermal-hydraulic models.

3. A description of any simplified analysis methods used in the study.

4. A description of the methods used to evaluate the effects of thermal stratification and mixing.

5. Graphs of all the best-estimate thermal-hydraulic results with their associated uncertainties and a detailed explanation of the transient behavior observed.

## 4.1 Thermal-Hydraulic Analysis Plan

This section should outline the logic and identify the subtasks in the thermal-hydraulic analysis. Subtasks include detailed thermal-hydraulic systems analysis, simplified thermal-hydraulic systems analysis, and thermal stratification analysis. The logic should describe the sampling plan used to select sequences for detailed or simplified analysis. ORNL experience favors selecting detailed thermal-hydraulic analysis sequences, including at least a few severe examples of each type of postulated overcooling transient in order to understand and benchmark the plant behavior for subsequent simplified calculations. The order in which the scenarios are evaluated can result in a considerable reduction in expenditures. By first analyzing the scenarios that are expected to be the bounding cases (i.e., the most severe), calculations for an entire class of overcooling scenarios may be deemed unnecessary if the bounding case is not of PTS concern. Similarly, careful selection of the first set of scenarios to be evaluated can permit simple extrapolation or interpolation of the results to other scenarios that share common controlling thermal-hydraulic phenomena.

During the analysis, the sequence identification analyst and the thermal-hydraulic analyst should coordinate activities to ensure that pertinent details of the delineated sequences are thoroughly understood. Similarly, close coordination must be maintained between the thermal-hydraulic analyst and the fracture mechanics analyst so that the transient fluid conditions are calculated at the appropriate vessel locations.

## 4.2 Thermal-Hydraulic Models

This section and supporting appendices should present a detailed description of the thermal-hydraulic computer models used in this analysis. The models should include an accurate representation of the pertinent parts of the primary and secondary systems. This includes the condensate system, the main and auxiliary feedwater systems, and parts of the steam system. The model should include appropriate secondary-side metal heat capacity. Particular attention should be given to the modeling of control system logic and characteristics such as valve closure times and liquid level measurements. References 14 through 17 illustrate some of the modeling details included in such a study. The thermal-hydraulic models should be capable of predicting single and two-phase flow behavior and critical flow as required. The models should be capable of predicting plant behavior for LOCAs, steamline breaks, and steam generator tube ruptures. In general, a one-dimensional code is suitable for most overcooling transient calculations. However, if any of the control systems are dependent solely on the fluid conditions in a single loop (e.g., reactor coolant pump restart criteria), a method of estimating the three-dimensional effects in the downcomer may be necessary for some of the asymmetric cooldown scenarios encountered in the PTS study. Sensitivity of calculated results to the nodalization schemes used should be discussed. The thermal-hydraulic models should be coupled, where appropriate, with neutronic models that have the capability to analyze pressure surges resulting from any relevant sequences involving recriticality.

This section of the report must also present the results of benchmarking the computer models against suitable plant data or data from experimental facilities or incorporate this information by reference to an NRC-approved topical report. As a minimum, the plant data comparison should fully exercise the modeling features that are employed in the thermal-hydraulic computer programs such as the pressurizer (including heaters and sprays), feedwater heaters and liquid level controls, the steam generator liquid level controls, and the turbine bypass (i.e., steam dump) controls under steady-state and transient conditions. If overcooling transients have occurred at the plant or at a similar plant, they should be benchmarked against the computer models. The licensee is encouraged to use codes and methods accepted by the NRC at the time the calculation is performed.

The models should be capable of accurately predicting condensation at all steam-water interfaces in the primary system, especially in the pressurizer during the repressurization phase of an overcooling event or during refilling of the primary system with cold safety-injection water. The effects of noncondensible gases, if present, on system pressure and temperature calculations should be addressed.

All code input and modeling assumptions should be documented and available for NRC review during the analysis review period (normally starting 3 years before the plant exceeds the screening limit and continuing until the evaluation results and any requisite actions are approved by the Commission).

### 4.3 Simplified Analysis Methods

This section should present the technical bases for any simplified analysis methods that are applied in the study. This includes the grouping of similar sequences by controlling phenomena and any extrapolations used to modify existing calculations. If a simplified thermal-hydraulic plant model is used to predict portions of the plant transients, all the simplifying assumptions inherent to this model should be stated and justified. Reference 18 provides examples of how to group sequences and develop a simplified thermal-hydraulic model suitable for portions of the analysis.

## 4.4 Thermal Stratification Effects

Transient thermal-hydraulic computer programs available to analyze LWR response to overcooling scenarios do not model fluid behavior with sufficient detail to predict the onset of HPI thermal fluid stratification in the cold leg and the subsequent cold leg and downcomer behavior. As a result, additional analysis methods may be needed to determine which transients are affected by thermal stratification and the extent of such effects.

This section should describe and justify the thermal fluid mixing analysis methods that have been applied in the study. References 19 through 24 describe the results of recent mixing analyses and experiments. Reference 19 identifies a useful stratification criterion to determine which overcooling transients will require the additional mixing analysis. Particular attention should be given to scenarios that involve HPI under very low flow or stagnant loop conditions. When stagnation is partial (i.e., not all loops stagnate), stratification is expected only within the cold legs of the stagnant loops. However, scenarios involving complete loop stagnation will require the evaluation of a transient cooldown in the presence of stratified layers both in the cold legs and in a portion of the downcomer. The mixing model should include the effect of metal heating on the mixing behavior, particularly in a stagnant flow situation. Also, the effect of noncondensible gases, if present, should be included. References 19 through 23 describe tools that have been used for such an analysis.

This section should also document the heat transfer correlations applied in the mixing analysis. The research efforts described in References 18 through 23 indicated that the downcomer heat transfer coefficients generally exceeded 300 Btu/hr-ft$^2$-°F. These values of heat transfer coefficient were generally high enough to keep the vessel wall surface temperatures within a few degrees of the downcomer fluid temperature. Furthermore, because the vessel wall cooldown was controlled by conduction processes rather than convection processes, the vessel wall surface temperatures were insensitive to heat transfer coefficient variations due to changes in flow and heat transfer regimes.

## 4.5 Thermal-Hydraulic Analysis Results

This section should present graphs of the best-estimate downcomer pressures, fluid temperatures, and heat transfer coefficients and their associated uncertainty ranges as a function of time at the critical weld areas. This includes the results of the detailed thermal-hydraulic model, the simplified model, and mixing analysis calculations.

The duration assumed for each overcooling scenario should be justified. It is assumed that a scenario duration of 2 hours may be reasonable for many cases since the overcooling transient would probably be identified and mitigated prior to that time. However, there may be scenarios requiring lengthier evaluation periods because the controlling phenomena delay the scenario's evolution.

Also provide a discussion of the accuracy of the results, including a demonstration that nodalization and error estimation methods chosen are appropriate, and how the predicted plant behavior compared to plant history and operating experience. Time-dependent uncertainty estimates for the downcomer pressure, fluid temperature, and heat transfer coefficients at the critical welds should be provided for each scenario. These uncertainties are often limited by physical phenomena. For example, the pressurizer power-operated relief valve (PORV) setpoints will limit the system pressure for certain high-pressure scenarios. Therefore, the uncertainty is limited by PORV operating characteristics. References 16 and 18 describe some uncertainty analysis techniques.

## 5. FRACTURE MECHANICS ANALYSIS

For each sequence identified in Chapter 3, "Determination of Detailed PTS Sequences for Analyses," calculate (or for unimportant sequences, estimate using bounding conditions) the conditional probability of through-wall crack penetration given the occurrence of the event versus fluence or $RT_{NDT}$. (Although licensees were required to use the method of determining $RT_{NDT}$ ($RT_{PTS}$) specified in paragraph 50.61(b)(2) when evaluating their vessel properties with respect to the screening limits, in performing these plant-specific calculations, they are encouraged to use any alternative methods/data/correlations for which they provide justification of applicability to their specific plant.) Specific sequences identified in Section 3.5.1 should be calculated individually in detail. Less important events such as the residual groups identified in Section 3.5.2 may be conservatively bounded without a calculation for each sequence in the group. A good example is provided in Chapter 5 of Reference 3. Input for these calculations includes the primary system pressure, the temperature of the coolant in the reactor vessel downcomer, the fluid-film heat transfer coefficient adjacent to the vessel wall, all as a function of time, and the vessel properties. The calculations should be performed with a probabilistic fracture mechanics code such as OCA-P or VISA-II (Refs. 25 and 26).

An acceptable procedure to be followed in the fracture mechanics analysis is as follows: A one-dimensional thermal and stress analysis for the vessel wall should be performed. The effect of cladding should be accounted for in both the thermal and stress analyses. The fracture mechanics model can be based on linear elastic fracture mechanics with a specified maximum value of $K_{Ic}$ and $K_{Ia}$ to account for upper-shelf behavior. Plastic instability should be considered in the determination of failure. Warm prestress should not be assumed in evaluations of the postulated transients. Acceptable types of material properties are given in the study of the H. B. Robinson reactor (Ref. 3).

In the Monte Carlo portion of the analysis, as a minimum, each of the following should be assigned distribution functions:

$K_{Ic}$ = Static crack initiation fracture toughness

$K_{Ia}$ = Crack arrest fracture toughness

$RT_{NDT}$ = Nil-ductility reference temperature

Cu = Concentration of copper, wt-%

Ni = Concentration of nickel, wt-%

F = Fast neutron fluence

The functions used should be justified. Examples of these distributions are found in Reference 3.

The following additional information should be supplied:

1. Flaw density - The number of cracks per unit surface area should be established for use in the calculations and should be justified. A value of 0.2 flaw per square meter of 8-inch-thick material (one flaw/cubic meter) was selected in References 1, 2, and 3.

2. Flaw depth density function - The flaw depth density distribution should be established. The function to be used can be that specified in References 1, 2, and 3.

3. Flaw size, shape, and location - Axial flaws with depths less than 20 percent of the wall thickness and all circumferential flaws should be modeled in infinite length. Axial flaws with depths greater than 20 percent of the wall thickness may be modeled in infinite or finite length depending on the relative toughness of the weld regions and plate material. For instance, the length of an axial flaw in an axial weld that suffers severe radiation damage relative to the plate can be limited to the length of the weld. The flaws should be assumed to be located at the inner surface of the vessel and should extend through the cladding to the inner surface of the vessel.

Reference 20 provides a comprehensive discussion of recommendations for input distributions to be used in probabilistic fracture mechanics calculations.

4. All regions of the beltline should be considered. This includes axial and circumferential welds as well as the base material.

The following relationships are required:

$K_{Ic} = f(T, RT_{NDTo}, \underline{\Delta} RT_{NDT})$, and

$K_{Ia} = f(T, RT_{NDTo}, \underline{\Delta} RT_{NDT})$

where

$T$ = Wall temperature

$RT_{NDTo}$ = Initial nil-ductility reference temperature

$\underline{\Delta} RT_{NDT}$ = Increase in nil-ductility reference temperature due to radiation damage, f(Cu,Ni,fluence). If plant surveillance data meet the criteria for credibility given in Reference 27, they may be used as described therein.

Examples of these functions are described in References 3 and 27.

In reporting the results, the methods used for the probabilistic vessel-integrity analysis should be described, their limitations for this analysis identified, and the impact of uncertainties in the resulting vessel failure probabilities estimated. Discussion of the analysis should include a listing of the assumptions used, their bases, and a discussion of the sensitivity of the results to variations in the assumptions. Vessel dimensions and material properties used should be given.

For each transient of interest, a deterministic analysis that includes a set of critical crack-depth curves as functions of time (see Refs. 1, 2, and 3), i.e., a plot of crack depths corresponding to initiation and arrest events versus time, should be carried out. This plot should also have curves indicating the depth of crack at which upper-shelf toughness is effective. These results should correspond to minus two sigma values for $K_{Ic}$ and $K_{Ia}$, plus two sigma values for $RT_{NDT}$, and plus two sigma values for the copper and nickel contents as well as plus two sigma for the fluence value.

These curves, which graphically represent the worst-case condition for each transient of interest, will be used in the evaluation of the critical time interval from the initiation of the transient during which mitigating action can occur.

# 6. INTEGRATION OF ANALYSES

In this chapter, the event frequencies are coupled with the results of the fracture mechanics analysis to obtain an integrated frequency of vessel through-wall cracking due to PTS. An example of one acceptable method is presented in Chapter 6 of Reference 3. A table that supplies the following information for each specific sequence and residual group identified in Section 3.5 should be provided. These results are to be provided for the operating time at which the reactor will reach the PTS screening criterion and for any additional operation life being requested:

- Sequence identification.
- Type of initiator (small-break LOCA with low decay heat, large steamline break at full power, etc.).
- Estimated sequence frequency.
- Method used to determine conditional through-wall crack penetration probability.
- Sequence conditional through-wall crack penetration probability.*
- Frequency of through-wall cracking due to sequence obtained by the product of sequence frequency and sequence conditional through-wall crack penetration probability.

For each dominant sequence, a section or table should be provided that supplies (1) specific reference to the graph of temperature, pressure, and flow as provided in Chapter 4, "Thermal-Hydraulic Analysis"; (2) a time-line description of the accident sequence noting important operator actions, control actions, protection system actions, equipment faults, and vessel failure; and (3) frequency of through-wall crack penetration as a function of fluence or $RT_{NDT}$.

Results should then be summed within each initiator type to provide a frequency of through-wall crack penetration as a function of initiator type.

The discussion should explain why each initiator type is or is not important to PTS.

Finally, the results should be summed over all initiator types to provide an integrated frequency of through-wall cracking for the vessel. This integrated value should be reported as a function of fluence, or $RT_{NDT}$, and plotted with uncertainty values as determined in Chapter 7, "Sensitivity and Uncertainty Analyses of Through-Wall Crack Frequency," and included on the plot. The discussion should identify important operator actions, control actions, and plant features that can cause or prevent vessel failure.

*The conditional through-wall crack penetration probability is the probability of a through-wall crack as determined by the fracture mechanics analysis, given that the event occurs.

# 7. SENSITIVITY AND UNCERTAINTY ANALYSES OF THROUGH-WALL CRACK FREQUENCY

In order for the results of the probabilistic analysis to be useful for regulatory decisionmaking, the sensitivity of the results to input parameters and assumptions should be determined, the major sources of uncertainty should be identified, and the magnitude of the uncertainty should be estimated. In this chapter, the results and the procedures used to perform each of these processes are to be documented. A good example is given in Chapter 7 of Reference 3. Portions of that analysis, or other analyses, may be referenced in lieu of portions of the analysis described in this chapter, provided the licensee demonstrates the applicability of the referenced analyses to the specific plant.

## 7.1 Sensitivity Analysis

Perform a sensitivity analysis to estimate the change in the through-wall crack frequency for a known change of a single parameter. Parameters examined in the sensitivity analysis should include (1) the initiating event and event tree branch frequencies, (2) the thermal-hydraulic variables (temperature, pressure, etc.), and (3) the fracture mechanics variables (fluence, flaw density, etc.). Where appropriate, 68th percentile (1-sigma) values should be used to represent the change in the parameter. This should provide a sufficient change to illustrate the effects of the change, and the use of the 68th percentile value whenever possible will help to define the important variabilities. In the case of temperature and pressure, however, the 68th percentile values may vary from one sequence to another. In this case, it may be easier to identify a representative change in the parameter that could then be used for all sequences rather than to try to use the 68th percentile values.

Each variable examined in the sensitivity analysis should be listed along with the change in the variable. In the cases in which changes are represented by using 68th percentile values, some explanation should be provided to document the reasons the value is considered a 68th percentile value. In those cases in which something other than a 68th percentile value is chosen, discussion should center around the reasons for choosing the value used.

Sensitivity factors should be obtained by dividing the through-wall crack frequency obtained with the changed variable by the through-wall crack frequency obtained with each variable at its mean value. Supply the sensitivity factors obtained for both positive and negative changes in each of the variables. The sensitivity factors obtained for changes made in the PTS-adverse direction should be ranked according to magnitude and provided in table form.

## 7.2 Uncertainty Analysis

### 7.2.1 Parameter Uncertainties

Each step in the probabilistic analysis should include an uncertainty analysis. This should include uncertainty in frequency of occurrence of a sequence, uncertainty in temperatures and pressures reached during the sequence, including that resulting from the nodalization scheme chosen as discussed in Section 4.5, and uncertainty in the fracture mechanics model for vessel failure given the transients.

For the following reasons, a Monte Carlo simulation is appropriate for portions of the PTS uncertainty analysis.

- The temperature and pressure error distributions are not symmetric.
- The fracture mechanics results are nonlinear with respect to variations in input parameters, particularly the temperature and pressure time histories.
- The results of the Monte Carlo analysis can indicate the shape of the output distribution.

The Monte Carlo approach would involve four steps as described below:

1. Develop a statistical distribution for each variable used in the calculation - This step will involve the representation of each variable as a distribution with 5th and 95th percentiles as previously identified. The shapes of the distributions selected should be discussed.

2. Select a random value from each distribution - A random sampling code should be used to sample from each of the distributions.

3. Calculate a through-wall crack frequency estimate based on values obtained in the previous step - In this step, the through-wall crack frequency is obtained based on the randomly selected variables. This requires understanding the form of the relationship between each input variable and through-wall crack frequencies. For some variables such as initiating event and branch frequencies and flaw density, this is simple since the through-wall crack frequency is directly proportional to the value of these parameters over the range of variable values considered. Other variables such as temperature and pressure may require the development of an appropriate relationship. In such cases in which the effect of a variable change may be dependent on the value of another variable, response-surface techniques may be used to estimate important interaction effects.

4. Summarize the resulting estimates and approximate frequency distribution - Steps 2 and 3 are repeated until a statistically valid number of trials have been performed. A distribution of through-wall crack frequencies is then produced from the results of the trials. The 95th and 5th percentiles and the mean (expected value) of this distribution should be identified and discussed.

### 7.2.2 Modeling Uncertainties (Biases)

During the process of performing the PTS analysis, the analyst will make simplifying assumptions in order to make the analysis tractable. Such assumptions include decisions on thermal-hydraulic models, fracture mechanics models, grouping of sequences both for thermal-hydraulic analysis and fracture mechanics analysis, nodalization in the thermal-hydraulic models, etc. These assumptions can introduce conservative or nonconservative biases into the analysis. These biases should be identified and their potential impact on the results discussed. In this section, important assumptions made as part of the analysis should be listed. Each assumption should be identified as being either conservative or nonconservative. A discussion should be supplied for each assumption with respect to its impact on the overall value of through-wall crack frequency.

Whenever excess conservatism or nonconservatism is suspected to be present in an assumption, an alternative assumption should also be used in the full calculation procedure and the impacts on the overall result compared.

## 8. EFFECT OF CORRECTIVE ACTIONS ON VESSEL THROUGH-WALL CRACK FREQUENCY

This chapter is to summarize the licensee's program of corrective measures. Each corrective measure considered by the licensee should be presented and explained. In each case, the reasons for considering the action as a corrective measure are to be documented, and the estimated impact of the action with respect to through-wall crack frequency provided. Corrective actions that are to be considered include, but are not limited to, those discussed in the remaining sections of the chapter. An example can be found in Chapter 8 of Reference 3.

### 8.1 Flux Reduction Program

Early analysis and implementation of such flux reductions as are reasonably practicable to avoid reaching the screening criterion are already being required and accomplished in accordance with the PTS rule, § 50.61. Further flux reductions to critical areas of the vessel wall that would reduce the risk of continued operation beyond the screening criterion should be considered. If such additional flux reductions are needed, in view of the irreversibility of embrittlement, the licensee should consider early implementation before reaching the screening criterion. For licensees who are considering applications to extend the operating license beyond its present expiration date, it may be prudent to implement the reduction as early as possible to avoid the necessity of vessel annealing or replacement.

### 8.2 Operating Procedures and Training Program Improvements

Operator actions and associated plant response play a key role in the initiation and mitigation of PTS events. Therefore, ensure that the actions are based on approved technical guidelines that include an integrated evaluation of relevant technical considerations, including, but not limited to, PTS, core cooling, environmental releases, and containment integrity. The evaluation should address the following types of concerns:

- Frequent realistic "team" training should be conducted, exposing the operators to potential PTS transients and their precursor events. The training should give the operators actual practice in controlling reactor system pressure and cooldown rates during PTS situations. Specific training should include, but not be limited to, reactor coolant pump trip criteria, the HPI throttling criterion, control of natural circulation, recovery from inadequate core cooling, recovery from solid plant operations, and the use of PORVs to control primary overpressure.

- Instructions should be based on analyses that include consideration of system response delay times (e.g., loop transport time, thermal transport time).

- Whether or not there is a need for cooldown rate limits for periods shorter than 1 hour should be evaluated.

- Methods for controlling cooldown rates should be provided. Reference should be made to these methods with respect to the dominant PTS risk sequences whenever possible.

- Guidance should be provided for the operator if cooldown rates or pressure-temperature limits are exceeded. These guidelines should take into account potential core cooling, environmental release, or containment integrity problems that could exist as a result of responding to the abnormal cooldown rate. These guidelines should leave little doubt as to when PTS concerns are more important than other safety issues and when other safety issues assume primary importance over PTS concerns. It should be emphasized how the guidelines were evaluated and optimized in light of any competing risks that might arise from events other than PTS events to ensure that plant safety is appropriately balanced.

- The desired region of operation between the pressure-temperature limit and the limit determined by avoidance of saturation conditions should be evaluated to determine if it can be revised to minimize total risk due to plant operation from PTS plus non-PTS events.

- Instructions for controlling pressure following depressurization transients should be provided.

- Instructions should be available for the condition where natural circulation is lost and the primary system main circulation pumps are not available.

Portions of the above may be provided by incorporation by reference, for example, to the plant-specific Emergency Response Guidelines. However, a summary discussion relating the referenced material to the overall subject should be provided.

## 8.3 Inservice Inspection and Nondestructive Examination Program

The use of state-of-the-art nondestructive examination (NDE) techniques could provide an opportunity to decrease any conservatism that might exist in the flaw density value used in the analysis. This decrease in conservatism, however, may be less important than the decrease in uncertainty in the actual flaw density that may result from an examination of this type.

Existing inservice inspection programs should be reevaluated to consider incorporation of state-of-the-art examination techniques for inspecting the clad-base metal interface and the near-surface area. This includes plant-unique consideration of the clad surface conditions. Consideration should be given to increased frequency of inspections.

The reliability of the NDE method selected to detect small flaws should be documented.

## 8.4 Plant Modifications

All plant modifications should be evaluated and optimized in light of any competing risks that might arise from events other than PTS events to ensure that overall plant safety is appropriately balanced. Plant modifications that may be considered include the following:

1. Instrumentation, Controls, and Operation

   a. Reactor vessel downcomer water temperature monitor.
   b. Instantaneous and integrated reactor coolant system cooldown rate monitors.
   c. Steam dump interlock.
   d. Feedwater isolation/flow control logic.
   e. Reactor coolant system pressure and temperature monitors.
   f. Control system to prevent repressurization of the reactor primary coolant system during overcooling events.
   g. Monitor to measure margin between vessel inner-surface temperature and current $RT_{NDT}$ at that location.
   h. Diagnostic instrumentation and displays.
   i. Primary coolant system pump trip logic.
   j. Automatic isolation of auxiliary feedwater to broken steam lines/generators.

2. Increased Temperature of Emergency Core Cooling Water and Emergency Feedwater

If plant modifications are proposed to prevent overcooling, the report should include an evaluation of undesirable side effects (i.e., undercooling) and a discussion of steps planned to ensure that the modifications represent a net improvement in safety when PTS and non-PTS related events are considered.

## 8.5 In Situ Annealing

If in situ annealing is part of the licensee's program of corrective measures, the licensee should describe the program to ensure that annealing will achieve the planned increase in vessel toughness, the surveillance program to monitor vessel toughness after annealing, the program directed toward code requalification after annealing, and the program to ensure that annealing does not introduce other safety problems.

## 9. FURTHER ANALYSES

The PTS rule (§ 50.61 of 10 CFR Part 50) requires Commission approval for plant operation with $RT_{PTS}$ values above 270°F. This regulatory guide outlines the analyses that should be performed in support of any request to operate at $RT_{PTS}$ values in excess of 270°F, as required in paragraphs 50.61(b)(4) and 50.61(b)(5), and states that the staff's primary acceptance criterion will be licensee demonstration that expected through-wall crack frequency will be below $5 \times 10^{-6}$ per reactor year for such operation.

In the event that a licensee is unable to meet this primary acceptance criterion, he may request Commission approval for continued operation under the provisions of paragraph 50.61(b)(6), which allows the submittal of further analyses. The content of these further analyses would be determined by the licensee and might include topics such as overall plant risk analyses that are beyond the scope of the vessel failure analyses covered by this regulatory guide.

# 10. RESULTS AND CONCLUSIONS REGARDING PTS ANALYSES

This chapter is to summarize the models used and the results obtained and provide the conclusions reached with respect to continued operation of the plant.

## 10.1 Summary of Analysis

In this section the major findings of each aspect of the PTS analysis, as described in the previous chapters, should be presented. These should include:

- Expected (mean) value of frequency of reactor vessel through-wall crack penetration versus time, with uncertainty bound (95th percentile).

- Identification of dominant accident sequences.

- If sensitivity/uncertainty analysis shows that slightly different assumptions could lead to different dominant sequences, identification of these assumptions and discussion of the impact on results given the different assumptions.

- Identification of important operator actions, control actions, and plant features that can increase or decrease the frequency or severity of overcooling transients, and whether these have been appropriately balanced to ensure optimum overall plant safety.

- Major sources and magnitudes of uncertainty in the analysis.

- The relative effectiveness of potential alternative corrective measures in reducing the expected (mean) value of through-wall crack penetration.

- The program of planned corrective measures.

## 10.2 Basis for Continued Operation

Finally, as part of the plant-specific analysis package, the licensee should provide a basis for concluding whether or not continued plant operation is justified. The basis for continued operation should include comparison with NRC's PTS acceptance criteria given in the Introduction to this guide.

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26. F. A. Simonen et al., "VISA-II - A Computer Code for Predicting the Probability of Reactor Vessel Failure," Battelle Pacific Northwest Laboratories, USNRC Report NUREG/CR-4486, April 1986.

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## REGULATORY ANALYSIS

The pressurized thermal shock (PTS) rule, § 50.61 of 10 CFR Part 50 (July 23, 1985--50 FR 29937), requires collection and reporting of material properties data, analyses of flux reduction options, and detailed plant-specific PTS risk analyses for those plants that reach the screening criterion based on $RT_{NDT}$,* as specified in the rule, during the term of the operating license. The regulatory guide addresses the detailed plant-specific risk analysis requirement, providing recommendations regarding how licensees should perform and how the NRC staff should review those analyses.

Neither the PTS rule nor the regulatory guide requires specific corrective actions. The guide merely provides guidance for the performance of the analyses required by the rule to identify and select necessary corrective actions. Therefore, in accordance with the Commission's Regulatory Analysis Guidelines (NUREG/BR-0058, Revision 1), this regulatory analysis does not provide extensive and detailed assessment of required, specific corrective actions.

The background material, nature of the problem, objectives, and costs, etc., of the PTS rule's requirements are covered in the regulatory analysis prepared as part of the rulemaking proceeding (Enclosure B to SECY-83-288, Proposed Pressurized Thermal Shock (PTS) Rule, July 15, 1983, and Enclosure D to SECY-85-60, Final Pressurized Thermal Shock (PTS) Rule, February 20, 1985). This regulatory analysis therefore addresses only (1) the need for publishing guidance regarding how licensees should perform the required plant-specific analyses, (2) the appropriateness of this particular guidance, and (3) the basis for the NRC staff acceptance criteria provided in the subject guide.

### 1. Need for Guidance

The NRC staff has gained considerable experience concerning PTS risk analyses. This experience has come from performance of analyses by the staff, from prototype plant-specific analyses performed by national laboratories and sponsored by NRC, and from review of industry-sponsored analyses. The regulatory guide reflects the lessons learned from this experience and will aid licensees in performing analyses that will efficiently derive risk estimates in the form the NRC needs for use in evaluating their conformance with the regulations.

This need for guidance is particularly acute since the plant-specific PTS analyses should use a probabilistic risk analysis (PRA) approach, as opposed to the more traditional design basis accident (DBA) approach, as explained below.

The PTS risk is developed as the sum of the small risks resulting from each of a large number of possible (but unlikely) PTS events. The regulatory guide accordingly describes acceptable methods to identify as many as possible of the potential PTS events, group them, calculate the frequencies and consequences of each group, determine the risk due to each group by multiplying the predicted frequency by the calculated consequences, and then sum the results

*Reference Temperature for the Nil Ductility Transition, a measure of the temperature range in which the materials' ductility changes most rapidly with changes in temperature.

from all groups to obtain total PTS risk estimates that can be compared with the acceptance criteria given in the regulatory guide.

The DBA approach, on the other hand, would attempt to define a worst credible event (the "design basis accident") and then show that (1) consequences from that event are acceptable and (2) all other credible events are less severe and therefore acceptable. The staff has determined that this DBA approach is not appropriate for plant-specific PTS analyses because the total risk from all credible PTS events can be significant even though each event individually is less severe than the DBA. The NRC staff therefore believes that this guide will encourage licensees to use the acceptable PRA approach and not waste time and resources on the more traditional DBA approach.

## 2. Justification of This Particular Guidance

The NRC staff has performed prototype plant-specific analyses for three plants. They constitute the most detailed, thorough analyses performed to date, and the lessons learned in their performance are reflected in the guide. The NRC staff has incorporated into the guide descriptions of the best methods found regarding how to assemble details of a plant's design (and to what level those details should be included), how to use event tree methodologies to identify and group potential PTS events, how to calculate severity of the events, how to integrate the resulting risk, and many other subjects. The staff believes that the benefit of this experience is presented in this guide, and its use by licensees will enable them to avoid many of the false starts and errors made by the staff and their contractors in performing the prototype analyses, thereby saving time and resources.

## 3. Justification of Acceptance Criteria

The guide states that, in judging the acceptability of continued operation beyond the PTS screening criterion, the staff will accept any analyses performed with acceptable methods such as those described in the subject regulatory guide that predict a through-wall crack penetration frequency less than $5 \times 10^{-6}$ per reactor year.

The mean frequency of reactor vessel through-wall crack penetration is used as the principal acceptance criterion because the staff's analyses predict that there is a high likelihood of core damage in the event of such cracks. Core damage events have potential public health and safety consequences that are difficult to analyze with certainty. They would also have severe economic impacts upon the licensee and the public who will pay for cleanup and replacement power. For all these reasons, reactor vessel through-wall crack penetration frequency is used as the principal acceptance criterion. The particular value of $5 \times 10^{-6}$ mean frequency per reactor year was selected as an achievable, realistic goal that will result in an acceptable level of risk. It is believed that this value is acceptably low considering that pressure vessel failure is not part of the design basis of the plant and therefore must have a frequency low enough to be considered incredible. When the various (unquantifiable) biases that are inherent in the analyses are taken into account at least qualitatively, such as the implicit assumption that "core damage" is equivalent to "core melt," this value probably results in a core melt mean frequency close to one per million reactor years.

In the opinion of the NRC staff, there are no practical quantities on which to base the acceptance criteria other than reactor vessel through-wall cracks (i.e., vessel failure).

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