analysis would be needed to assess the safety of plant operation beyond the screening limit.

Each of these components is described in the following subsections.

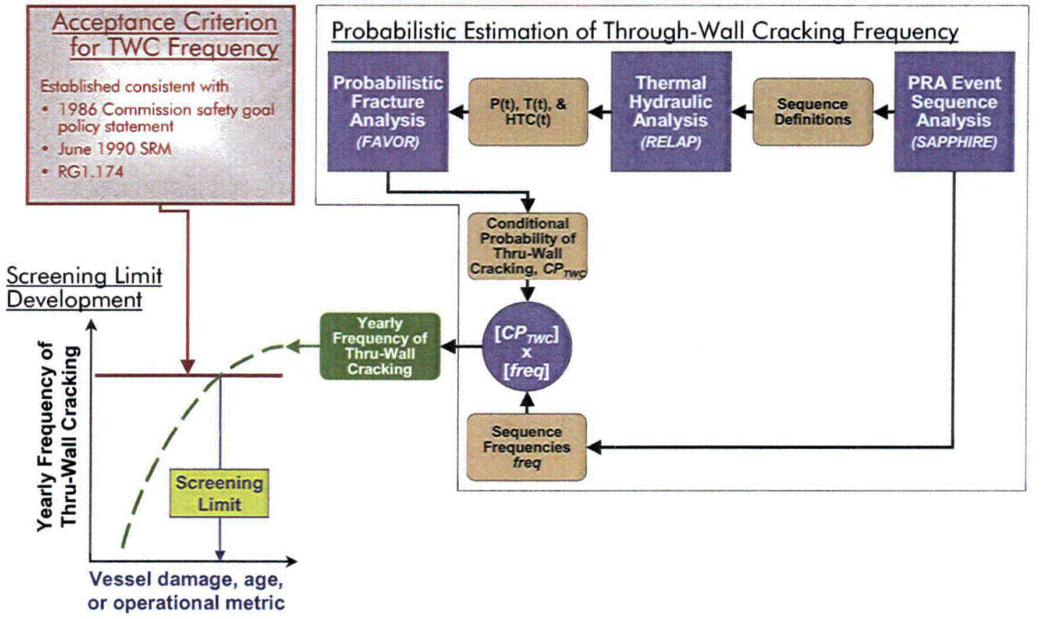


Figure 3-1. High-level schematic showing how a probabilistic estimate of through-wall cracking frequency (TWCF) is combined with a TWCF acceptance criterion to arrive at a proposed revision of the PTS screening limit.

3.1.2.1 Component 1: Probabilistic Estimation of Through-Wall Cracking Frequency

As illustrated in Figure 3-1, three main models (shown as solid blue squares), taken together, allow us to estimate the annual frequency of through-wall cracking in an RPV:

- PRA event sequence analysis
- TH analysis
- PFM analysis

In the following subsections, we first describe these three models and their sequential execution to give the reader an appreciation of their interrelationships and interfaces (Section 3.1.2.1.1). Secondly, we describe the iterative process we undertook, which involved repeated execution of all three models in sequence, to arrive at final models for each plant (Section 3.1.2.1.2). We then discuss the three specific plants we analyzed in detail (Section 3.1.2.1.3). Finally, we conclude with a discussion of the steps taken to ensure that our conclusions based on these three analyses apply to domestic PWRs *in general* (Section 3.1.2.1.4).

3.1.2.1.1 Sequential Description of How PRA, TH, and PFM Models are Used To Estimate TWCF

First, a PRA event sequence analysis is performed to define the sequences of events that are likely to cause a PTS challenge to RPV integrity, and estimate the frequency with which such sequences can be expected to occur. The event sequence definitions are then passed to a TH model, which estimates the temporal variation of temperature, pressure, and heat-transfer coefficient in the RPV downcomer characteristic of each sequence definition. These temperature,

pressure, and heat transfer coefficient histories are then passed to a PFM model that uses the TH output, along with other information concerning plant design and construction materials, to estimate the time-dependent "driving force to fracture" produced by a particular event sequence. The PFM model then compares this estimate of fracture driving force to the fracture toughness, or fracture resistance, of the RPV steel. This comparison allows us to estimate the probability that a crack would be created and would penetrate all the way through the RPV wall if that particular sequence of events actually occurred. The final step in the analysis involves a simple matrix multiplication of the probability of through-wall cracking (from the PFM analysis) with the frequency at which a particular event sequence is expected to occur (as defined by the event-tree analysis). This product establishes an estimate of the annual frequency of through-wall cracking that can be expected for a particular plant after a particular period of operation when subjected to a particular sequence of events. The annual frequency of through-wall cracking is then summed for all event sequences to estimate the total annual frequency of through-wall cracking for the vessel. Performance of such analyses for various operating lifetimes provides an estimate of how the annual through-wall cracking frequency can be expected to vary over the lifetime of the plant.

3.1.2.1.2 Iterative Process Used To Establish Plant-Specific Models

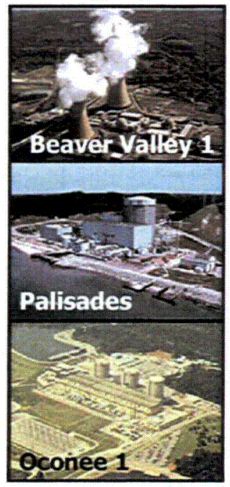
The set of transients used to represent a particular plant are identified using a PRA event-tree approach, in which many thousands of different overcooling sequences are "binned" together into groups of transients believed to produce similar thermal-hydraulic outcomes. Judgments regarding which transients to put into which bin were guided by such characteristics as similarity of break size, operator action, etc., and resulted in "bins" such as medium-break primary system LOCAs, MSLBs, etc. From each of the tens or hundreds of individual event sequences in each bin, a single sequence was then selected and programmed into the Reactor Leak and Power (RELAP) TH excursion code to define the variation of pressure, temperature, and heat transfer coefficient *vs.* time. These TH transient definitions were then passed to the Fracture Analysis of Vessels, Oak Ridge (FAVOR) PFM code, which estimated the conditional probability of through-wall cracking (CPTWC) for each transient. When multiplied by the bin frequency estimates from the PRA, these CPTWCs become TWCF values, which (when rank-ordered) estimate the degree to which each bin contributes to the total TWCF of the vessel. At this stage, many bins were found to contribute very little or nothing at all to the total TWCF, and so received little further scrutiny. However, some bins invariably dominated the TWCF estimate. These bins were further subdivided by partitioning the bin frequency, and selecting a TH transient to represent each part of the original bin. This refined model was then reanalyzed using FAVOR, and the bins that provide significant contributions to TWCF are again examined. This process of bin partitioning, and selection of a TH transient to represent each newly partitioned bin, continued until the total estimated TWCF for the plant no longer changes significantly.

3.1.2.1.3 Plant-Specific Analyses Performed

In this study, we performed detailed calculations for three operating PWRs (Oconee 1, Beaver Valley 1, and Palisades), as shown in Figure 3-2. Together, these three plants sample a wide range of design and construction methods, and they contain some of the most embrittled RPVs in the operating fleet.

3.1.2.1.4 Generalization to All Domestic PWRs

Since the objective of this study is to develop the technical basis for revision of the 10 CFR 50.61 PTS screening limit that applies *in general* to all PWRs, we must understand the extent to which the three plant-specific analyses adequately address (in either a representative or a bounding sense) the range of conditions experienced by domestic PWRs *in general.*



- High embrittlement plant
- Westinghouse design

- High embrittlement plant
- Combustion Engineering design

- Plant used in 1980s PTS study
- Babcox & Wilcox design

Figure 3-2. The three plants analyzed in detail in the PTS reevaluation effort.

To achieve this goal, we have took the following measures:

- We performed sensitivity studies on both the TH and PFM models to address the effect of credible changes to the models and/or their input parameters. The results of these studies provide insights regarding the robustness of our conclusions (based on three plants), when applied to the PWR population in general.
- We examined plant design and operational characteristics of five additional plants. In so doing, our aim was to determine whether the design and operational features identified as being important in our three plant-specific analyses vary significantly enough in the population of PWRs to question the generality of our results.
- In our three plant-specific analyses, we assumed that the only possible origins of PTS events are caused by events *internal* to the plant. However, the PRA categorized *external* events (such as fires, floods, and earthquakes), which can also be PTS precursors. We, therefore, examined the potential for external initiating events to create significant additional risk relative to the internal initiating events we already modeled in detail.

3.1.2.2 Component 2: Acceptance Criterion for Through-Wall Cracking Frequency

Since the issuance of SECY-82-465 and the original PTS Rule, the NRC has established a considerable amount of guidance on the use of risk metrics and risk information in regulation [e.g., NRC FR 86, and RG 1.174]. To ensure consistency with this guidance, the PTS Reevaluation Project staff identified and assessed options for a risk-informed criterion for RVFF (which Regulatory Guide 1.154 currently specifies in terms of TWCF).

As described in a May 2002 status report on risk metrics and criteria for PTS [SECY-02-0092], the options developed involved both qualitative concerns (the definition of RPV failure) and quantitative concerns (a numerical criterion for the reactor vessel failure frequency). These options reflected uncertainties in the margin between PTS-induced RPV failure, core damage, and large early release. The options also incorporated input received from the ACRS [NRC LTR 02], regarding concerns related to the potential for large-scale oxidation of reactor fuel in an air environment.

Our assessment of the options involved identifying technical issues unique to PTS accident scenario development, developing an accident progression event tree to structure consideration of the issues, performing a scoping study of containment performance during PTS accidents, and reviewing the options in light of this information. The scoping study involved collecting and evaluating available information, performing a few limited-scope thermal-hydraulic and structural calculations, and conducting a semi-quantitative analysis of the likelihood of various accident progression scenarios.

3.1.2.3 Component 3: Screening Limit Development

As illustrated schematically in the lower left corner of Figure 3-1, a screening limit for PTS can be established based on a simple comparison of *TWCF* estimates as a function of an appropriate measure of RPV embrittlement with the *TWCF* acceptance criterion (see Chapter 10). Beyond the work to establish both the *TWCF vs.* embrittlement curve and the limit value for *TWCF*, it is also necessary to establish a suitable vessel damage metric that, ideally, allows different conditions in different materials at different plants to be normalized. From a practical standpoint, "suitable" implies that the metric needs to be based on readily available information regarding plant operation and materials.

3.2 Uncertainty Treatment

At the outset of this project (1999), a staff member reviewed the NRC's existing approach for PRA modeling, focusing on how uncertainties should be treated, how they were propagated through the PRA, TH, and PFM models, and how that approach compared with the NRC's guidelines on work supporting risk-informed regulation [***Siu 99***]. This review established the general framework for model development and uncertainty treatment adopted in this study. In the following two sections, we first review this recommended framework (Section 3.2.1), and then discuss its actual implementation (Section 3.2.2). Section 3.2.2 provides an overview of the uncertainty treatment implemented in the PRA, TH, and PFM analyses and discusses how uncertainties are "passed" between these three main technical modules. Details of these implementations appear elsewhere in this report (Sections 5.2.6–5.2.7, 6.8.2, and 7.4, respectively) and in other documents [***Whitehead-PRA***, ***Chang***, and ***EricksonKirk-PFM***, respectively].

3.2.1 Recommended Framework

In this study, we performed probabilistic calculations to establish the technical basis for a revised PTS Rule within an integrated systems analysis framework [Woods 01]. Our approach considers a broad range of factors that influence the likelihood of vessel failure during a PTS event, while accounting for uncertainties in these factors across a breadth of technical disciplines [***Siu 99***]. Two central features of this approach are a focus on the use of realistic input values and models (wherever possible), and *explicit* treatment of uncertainties (using currently available uncertainty analysis tools and techniques). Thus, our current approach improves upon that employed in developing SECY-82-465, in which many aspects of the analysis included intentional and unquantified conservatisms, and uncertainties were treated *implicitly* by incorporating them into the models (RT_{NDT}, for example).

Our probabilistic models distinguish between aleatory and epistemic uncertainties. Aleatory uncertainties arise as a result of the randomness inherent in a physical or human process, whereas epistemic uncertainties are caused by limitations in our current state of knowledge (or understanding) of a given process. A practical way to distinguish between aleatory and epistemic uncertainties is that the latter can, in principle, be reduced by an increased state of knowledge. Conversely, because aleatory uncertainties arise as a result of randomness at a level below which a particular process is modeled, they are fundamentally irreducible. The distinction between aleatory and epistemic uncertainties is an important part of PTS analysis because different mathematical and/or modeling procedures are used to represent these different types of uncertainty.

3.2.2 Implementation

In this section, we describe our implementation of the uncertainty framework synopsized in Section 3.2.1, focusing specifically on the following aspects:

- How the framework was implemented in of the PRA, TH, and PFM analyses. Consistent with the framework, we systematically identify uncertainties and characterize their nature (as aleatory or epistemic). These uncertainties are then either quantified or addressed as part of the overall structure of the mathematical model.
- How uncertainties are "propagated" through the major components of the computational model used to estimate the TWCF illustrated in the upper right corner of Figure 3-1. This includes propagating uncertainties from PRA to TH, PRA to PFM, and TH to PFM.
- How the uncertainties considered in all three models (i.e., PRA, TH, and PFM) become manifest in the uncertainties in the estimated value of TWCF.

The first two points are described in Sections 3.2.2.1 through 3.2.2.3, for PRA, TH, and PFM, respectively. The final point is addressed in Section 3.2.2.4. Finally, Section 3.2.2.5 addresses the uncertainties associated with the potential "incompleteness" of our mathematical model relative to the physical reality we are trying to represent.

3.2.2.1 PRA

As illustrated in Figure 3-1, the PRA analysis has two major outputs:

- Bin Definition: the representation (or model) of the total PTS challenge using a finite number of event "bins," each of which represents an assortment of TH scenarios (that are believed to be similar)
- Bin Frequency: an estimate (central tendency and distribution) of the frequency with which the events represented by each bin are expected to occur

Each of these outputs has an associated uncertainty, as described in the following sections.

3.2.2.1.1 Bin Definition

Each bin represents an assortment of TH scenarios (i.e., PTS sequences) for the following reasons:

- Like most PRAs, ours includes the usual idealization that both equipment failures and operator actions are binary (i.e., a valve either sticks open or it does not, an operator either acts or fails to act)[§]. Clearly, reality is continuous; valves may stick open by various amounts and operators may act, but after some delay. This idealization leads to the situation where our mathematical representation (a single bin) represents a spectrum of potential outcomes, with each outcome having a distinct TH characteristic.
- Another common PRA feature that we adopt is to group "similar" transients together in a single bin. For example, all primary system pipe breaks having a break diameter of 8-in. (20-cm) and above are placed in a single bin called "large-break LOCAs." This approach is motivated by previous experience indicating that transients grouped in this manner have "similar" severity. Nonetheless, such "similarity" is an approximation. To continue with the large-break LOCA example, hot leg and cold leg breaks have different severities for the same break diameter, break diameter changes above 8-in. (20-cm) cause slightly different severities, and so on. All of these unmodeled effects occur for well-recognized physical reasons. Again, this idealization leads to the situation where our model (a single bin) represents a spectrum of potential outcomes, with each outcome having a distinct TH characteristic.

[§] As detailed in Section 5.2.6.1, this statement is not always true. When judged to be important, certain equipment failures and operator actions were further subdivided (e.g., 30% stuck-open valves, operator actions at 1 *vs.* 10 minutes, etc.). Nonetheless, the PRA model is still a discrete representation of a continuum, and each PRA bin still represents a spectrum of TH responses.

Thus, the structure of the PRA representation of the PTS challenge contains within it an uncertainty that is random and (hence) aleatory, having to do with all the ways that a PTS challenge could occur (i.e., PTS sequences). Discretizing the continuum of potential PTS sequences (which number in the tens or hundreds of thousands) into a tractable number of bins for detailed analysis (~hundreds) means that each bin contains many sequences, each of which can (in principle) produce a different TH response and, thereby, a different effect on the vessel.

As is often the case in PRA, only a portion of the entire aleatory uncertainty significantly affects the overall results of the analysis. The important part of the aleatory uncertainty is determined by the way the PRA model was developed and how the bins were defined. As described in Section 3.1.2.1.2, an initial PRA model is developed and individual TH sequences are selected to represent each bin. These TH definitions are then passed to the FAVOR PFM code, which estimates the CPTWC for each transient. When multiplied by the frequency estimates for each bin, these CPTWC values become TWCF values, which (when rank-ordered) estimate the degree to which each bin contributes to the total TWCF of the vessel. At this stage, many bins are found to contribute very little or nothing at all to the total TWCF. However, some bins invariably dominate the TWCF estimate. These bins are then further subdivided by partitioning the frequency of the bin, and selecting a TH transient to represent each part of the original bin. This refined model is then reanalyzed using FAVOR, and the bins that provide significant contributions to TWCF are again examined. This process of bin partitioning, and selection of a TH transient to represent each newly partitioned bin, continues until the total estimated TWCF for the plant no longer changes significantly. At this point, that portion of all possible PTS sequences (and, hence, the aleatory uncertainty) that significantly affects the overall results is determined and remains in the final model as representing the aleatory uncertainty associated with how a PTS challenge might occur.

3.2.2.1.2 Bin Frequency

For each bin, there is uncertainty regarding the true frequency of occurrence. The uncertainty in the frequency with which the events represented by each bin occurs depends upon the following three factors, each of which is also uncertain:

- uncertainty in the initiating event and its associated frequency
- uncertainty in the series of equipment successes and/or failures that may follow the initiating event, and the uncertainty in their associated probabilities
- uncertainty in the operator actions that may or may not be taken following the initiating event, and the uncertainty in their associated probabilities

Thus, the frequency of occurrence of each bin is a function of the frequencies and probabilities of these factors. (The bin frequency is estimated from the individual frequencies and probabilities using Latin Hypercube sampling techniques to develop the bin frequency histogram that is provided as input to the FAVOR post-processor (FAVPOST).) Each of these factors has an associated epistemic uncertainty, which is described by a distribution. These uncertainties are epistemic in nature because our belief as to the estimates of these frequencies and probabilities is influenced by our limited state of knowledge about these rare events; and better knowledge would clearly lead to reduced uncertainty.

3.2.2.2 TH

The approach used to address uncertainty in the TH analysis principally utilized sensitivity studies to quantify the effects of phenomenological and boundary condition uncertainties/variations on the severity of a TH sequence. The results of these studies were used in two ways:

(1) They were combined with probability estimates of the sensitivity parameters being evaluated to adjust the bin frequencies from the PRA analysis.

(2) They were used to justify further subdivision of the PRA bins. (See the discussion in Section 3.2.2.1.1.)

In this way, the TH uncertainty analysis accounts for certain parameters that can affect the thermal-hydraulic response of the plant, which were not explicitly considered in the PRA analysis (e.g., season of the year). Because the uncertainty analysis also produced insights regarding the effects of various system parameters and TH models on event severity, it also helped to identify the transient used to represent each PRA bin to the PFM analysis.

This method of accounting for TH uncertainty does not quantify the uncertainties associated with each TH sequence. Rather, it characterizes the uncertainties associated with each PRA bin. This is appropriate because, as illustrated in Figure 3.3, each TH sequence that is passed to the PFM analysis represents a much larger number of TH sequences that, together, constitute a PRA "bin." Provided the combined effects of the TH parameter and modeling uncertainties on the severity of this one representative sequence is small relative to both

- the uncertainty in the frequency of occurrence of all of sequences in the bin, and
- the variability in severity between the different sequences in the bin

then, the uncertainty associated with TH parameter and modeling uncertainties of the representative sequence can be considered negligible. The appropriateness of not accounting for these uncertainties because they are negligibly small is ensured by the iterative process used to define the PRA bins.

As described in Section 3.2.2.1.1, PRA bins that contribute significantly to the estimated TWCF were continually partitioned (including appropriate partitioning of the bin frequencies and selection of new TH sequences to represent each partitioned bin) until the total estimated TWCF for the plant did not change significantly with continued partitioning. Thus, any errors caused by not explicitly accounting for the TH parameter and modeling uncertainties associated with the TH sequence used to represent each PRA bin are not expected to influence the outcome of the analysis (i.e., the estimated values of TWCF).

3.2.2.3 PFM

Development of the PFM model featured a comprehensive review of all model components (both sub-models and parameters) with the aim of identifying, classifying, and quantifying the uncertainties in each [***EricksonKirk-PFM***]. In the great majority of cases, the best-estimate models (and associated uncertainties) were quantified, and these were propagated through the calculation. In some cases, inadequate empirical and/or physical evidence existed to support creation of a best-estimate with uncertainties. In these cases, conservative models and parameters were adopted [***EricksonKirk-SS***]). The judgment to include these conservatisms as part of the overall model is itself a treatment of uncertainty, not through quantification, but rather by influencing the structure of the overall PFM model.

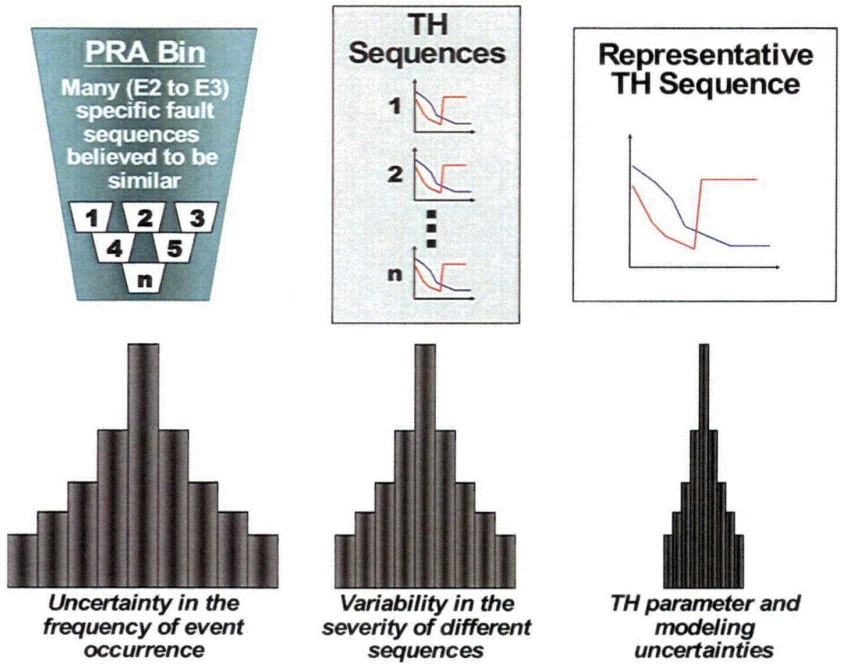


Figure 3.3. Characterization of TH uncertainties

The great majority of the parameters in the PFM model (e.g., RT_{NDT}, Cu, Ni, fluence, flaw parameters) were determined to have epistemic uncertainties. Statistical distributions were developed to characterize these uncertainties from representative data. In some cases, physical models guided these characterizations.

Conversely, the various fracture toughness parameters in the PFM model were all determined to have alleatory (irreducible) uncertainties. These alleatory uncertainties are a direct and natural consequence of the heterogeneity of the material at the same size scale as the crack-tip deformation fields. They also arise because the interaction of two factors (material resistance *vs.* applied loading) produces the measured parameter called fracture toughness (again, see [***EricksonKirk-PFM***] for full details).

The output of the PFM model is distributions quantifying the CPTWC for each transient analyzed. (This value is termed "conditional" because occurrence of the transient is assumed in the PFM calculation.) These distributions account for the uncertainties in the various toughness parameters, non-toughness parameters, and sub-models that together make up the PFM model.

3.2.2.4 What the Uncertainties in TWCF Represent

Sections 3.2.2.1, 3.2.2.2, and 3.2.2.3 described the uncertainties in the PRA, TH, and PFM models, respectively. Table 3.1 summarizes that discussion, and indicates that in each of these three areas, the important uncertainties have been either "accounted for" (in that they influenced the structure of the computational model) or "numerically quantified" (as part of the model). Thus, a description of what the uncertainties in the reported values of TWCF represent requires more than a strictly numerical answer. As described in a NUREG-series report on the theory and implementation of the FAVOR code [***Williams***], FAVPOST estimates the numerical value of TWCF by performing a matrix multiplication of the distribution of the frequency of each bin defined in the PRA analysis with the distribution of CPTWC estimated by the PFM analysis. However, these

uncertainties and their quantifiable distributions arise as a direct consequence of the particular model we have used to calculate them and, as indicated in Table 3.1, the structure of the model itself accounts for a number of uncertainties that have not been numerically quantified. Thus, the uncertainties in our reported TWCF values represent all of the uncertainties discussed in this section and in the detailed companion reports [***Whitehead-PRA***, ***Chang***, and ***EricksonKirk-PFM***].

Table 3.1. Summary of uncertainty treatment in the three major technical areas

Technical Area	Uncertainty Type	Uncertainties that were accounted for in the structure of the model	Uncertainties that were numerically quantified
PRA	Aleatory	Discretization of all of the ways a PTS challenge could occur into a finite number of "bins"	---
	Epistemic	---	Bin frequency
TH	Aleatory	Boundary condition uncertainties	The effects of certain boundary condition uncertainties are reflected in the frequencies assigned to certain PRA bins.
	Epistemic	Model uncertainties	---
PFM	Aleatory	---	Uncertainties in fracture toughness values (e.g., K_{Ic}, K_{Ia}, J_{Ic})
	Epistemic	Adoption of conservative models (e.g., RT_{NDT}, flaw distribution, fluence attenuation)	Uncertainties in non-toughness values (e.g., Cu, Ni)

3.2.2.5 Incompleteness Uncertainty

As with any attempt to represent a complex physical system using a mathematical model, the question of "incompleteness uncertainty" arises. That is, "What has been left out of the model and, as a result, how confident should a decision-maker be in using the results of the analysis?". It is fundamentally impossible to quantitatively address uncertainties arising from unknown factors. However, our process for model building, verification and validation (V&V) of our computational models, conservatisms known to remain in the models, the various reviews to which our work has been subjected, and the potential implementation of our results in future rules all provide qualitative assurance that any incompleteness in the model should have a negligible effects on the results. We discuss each of these factors below:

- Process for Model Building: The PRA, TH, and PFM models were developed and continually improved throughout this study. Licensees from the three study plants provided input and review of both the PRA and TH models. The commercial nuclear power industry, working under the auspices of the Electric Power Research Institute (EPRI) Materials Reliability Project, was involved in reviewing all three models from the inception of the study. Additionally, subject-matter experts from the industry played a key role in both developing and reviewing the PFM model. To address uncertainties in a manner consistent with the framework proposed by Siu and synopsized in Section 3.2.1, various new models of both flaws and fracture toughness behavior were created for the PFM model. These new models have been presented for review and comment in various public and international venues, and have been published in both peer-reviewed journals and conference proceedings [Kirk 01a, Kirk 02a, Natishan 01, EricksonKirk 04].
- Computational V&V: Calculations made in PRA, TH, and PFM are performed using computer codes referred to as SAPHIRE, RELAP, and FAVOR, respectively. SAPHIRE and RELAP are commercially available programs and have been subjected to extensive review and V&V. The FAVOR PFM code was developed by RES for the express purpose of performing probabilistic

simulations of PTS. Accordingly, we have performed and reported V&V of FAVOR according to the software quality assurance (SQA) guidance in NUREG/BR-0167 [***Malik***].

- Known Conservatisms: While we devoted considerable effort throughout this study to perform "best estimate" analyses, it is nonetheless true that a number of conservatisms remain. Primary among these is the decision to treat through-wall cracking of the RPV as *equivalent to* occurrence of a large early release of radioactivity to the atmosphere. Chapter 10 discusses the reason for, and conservatism implicit in, this assumption. Furthermore, throughout the development of all the PRA, TH, and PFM models, there has been a tendency to address uncertainties by adopting conservative models or input values when the weight of physical and empirical evidence was inadequate to construct a "best-estimate" model. These types of conservatisms are discussed throughout Chapters 5–7 and in the supporting detailed reports [***Whitehead-PRA, Bessette, EricksonKirk-PFM***], and are summarized in Section 11.4.4.
- Reviews: As described under *Process for Model Building* above, our models were subjected to both internal and external review during their development. Additionally, we solicited and received reviews of the entire project from three sources. In December 2002, we published an interim report summarizing the results of computations performed up to that time [***Kirk 12-02***]. This report was reviewed by both the commercial nuclear power industry and staff from the NRC's Office of Nuclear Reactor Regulation (NRR), with both groups providing written comments [NEI Comments, NRR Comments]. These reviews indicated the need for numerous minor revisions, remodeling of some Oconee transients, and (most significantly) a fundamental restructuring and expansion of the documentation to improve both clarity and completeness. Addressing these comments resulted in this document (and the supporting documents, see Section 4.1), which have been subjected to review by an international group of experts. The comments provided by this panel (see Section 3.5 and Appendix B) have, again, resulted in improvements in both our documentation and our mathematical models. It is important to recognize that the combined effect of all of these changes (i.e., changes made in response to NEI, NRR, and external review panel comments) to the TWCF results [***Kirk 12-02***] has been to reduce the total TWCF by, on average, approximately one-third. Thus, while the comments received from the review panels have improved both the clarity of our documentation and the overall completeness and correctness of our models, the changes have not substantially altered either the overall structure of the models or the TWCF results that could be used to establish a new numerical value for the PTS screening criterion.
- Potential Implementation: Should NRR elect to use the information presented in this and supporting documents as the basis for rulemaking to revise the requirements of 10 CFR 50.61, it must be remembered that it is only a *screening limit* that is being revised. Exceeding a screening limit does not suggest that failure is imminent (or even likely). It merely signals the need for the licensee to take additional actions (either analytical or mitigative) to assure NRR that plant operation beyond the screening limit does not unduly increase the risk to the public. Additionally, the current structure of 10 CFR 50.61 requires that these actions be taken three years before the limits are actually exceeded. It also requires continued surveillance (according to the requirements of Appendix H to 10 CFR Part 50) to ensure the continued validity of assumptions made during development of the screening limit regarding irradiation embrittlement mechanisms. Maintenance of this rule structure mitigates the practical impact on the overall public risk posed by PTS, as a result of any incompleteness uncertainties associated with the recommended numerical value of the screening limit.

3.3 Fundamental Assumptions and Idealizations

Any mathematical model of a physical system inherently involves some level of assumption and/or idealization to make estimates of the parameters of interest tractable within the practical constraints associated with the particular problem of interest. As discussed in greater detail in Chapters 5, 6, and 7, the PRA, TH, and PFM models each involve a large number of sub-models and, thus, a large number of possible assumptions and/or idealizations. Assumptions and idealizations that occur within each of the PRA, TH, and PFM sub-models are, therefore, addressed in Chapters 5, 6, and 7, respectively, or within their supporting reports. In the following subsections, we discuss the fundamental assumptions and idealizations that pertain to the PRA, TH, and PFM sub-models *as a whole*.

3.3.1 Probabilistic Risk Assessment

As with any PRA or HRA, the analysis team found it necessary to make assumptions in this study. In addition to the typical assumptions made as part of a PRA (e.g., actual plant system configuration is represented by the as-built as-operated documented information), the analysis team made various additional assumptions during the detailed PTS analyses. These assumptions are grouped into seven categories, as follows:

1. Project execution
   a. Lessons learned from the Oconee analysis and preliminary PFM calculations for Beaver Valley and Palisades were used to simplify the model construction for Beaver Valley and Palisades.
2. Possible PTS Initiating Events
   a. Scenarios initiated by an anticipated transient without scram (ATWS) were screened from the PTS analyses for two reasons. First, ATWS events generally begin with severe undercooling (i.e., there is too much power for the heat removal capability) and likely involve other failures to achieve an overcooling situation. Second, with typical ATWS frequency estimates in the range of 10^{-5}/yr to 10^{-6}/yr combined with the need for other failures to occur to *possibly* cause a continuing and serious overcooling situation, ATWS-initiated sequences should not be significant contributors to PTS risk when compared to other modeled scenarios with initiator frequencies commonly in the range of 1/yr to 10^{-3}/yr.
   b. Interfacing systems loss-of-coolant accidents (ISLOCAs) could involve overcooling from the start of the event. However, significant ISLOCAs often fail, or are assumed to fail, mitigating equipment in PRAs, which ultimately causes an undercooling event, rather than an overcooling event; thus, ISLOCAs were not analyzed. Also, similar to ATWS, frequency estimates for ISLOCAs of sufficient size to cause a sever cooldown are in the range of 10^{-5}/yr to 10^{-6}/yr. Therefore, ISLOCAs should not be significant contributors to PTS risk when compared to other modeled scenarios with initiator frequencies commonly in the range of 1/yr to 10^{-3}/yr.
   c. It was asssumed that the frequency of inadvertent reactor/turbine trips under hot zero power (HZP) conditions is 20% of that occuring under full power conditions. The basis of this 20% factor is as follows:
      i. The plant operates at HZP approximately 2% of the time.
      ii. Except for inadvertent reactor/turbine trips attributable to transient conditions that arise while purposely changing feedwater and steam conditions along with changing power and other parameters in the plant, a review of transients occurring while at HZP provided no evidence that initiators

are significantly more prone to occur at HZP than at full power. While no statistical treatment of this observation was attempted, engineering judgment was used to suggest that reactor/turbine trips seem more likely under HZP than under full power conditions because operators are often adjusting feedwater and steam conditions during HZP, factors that increase the likleyhood of tripping the plant. On this baisis, a factor of 10 increase in the likelihood of trips under HZP (vs. full power condtions) was assumed.

iii. 2% x 10 = 20%

3. Scenario development

   a. Medium- and large-break LOCAs were modeled as leading directly to a significant thermal transient for the reactor vessel without the need to consider the response of mitigating systems.

   b. The status of pressurizer PORVs and SRVs (i.e., whether they were open or closed) was assumed to be unimportant in the development of small LOCA scenarios. The basis for this assumption was that the pressure drop resulting from the LOCA initiating event should preclude the demand to open a primary side PORV or SRV.

   c. The PTS models excluded certain systems, structures, and components (SSCs) (e.g., pressurizer sprays and heaters) because they typically were found to have little impact on PTS risk.

   d. The functions of some SSCs were simply assumed for certain scenarios (e.g., accumulators were assumed to inject their inventory if conditions in the primary were such that injection should occur—failure of accumulator check valves was not modeled).

   e. The analysts recognized the importance of *when* an operator action occurred or *when* a piece of equipment changed state to the degree of overcooling experienced during a PTS scenario. To account for this, the scenarios incorporated a ***limited*** set of important operator actions (e.g., operator fails to throttle high-pressure injection) and equipment state changes (e.g., stuck-open pressurizer SRV recloses).

4. Systems analysis

   a. The impact of heating and ventilation failures on equipment performance can be ignored because of the relatively slow effects on PTS-relevant equipment (e.g., failure of a pump as a result of room cooling failure typically takes a few hours by which time the PTS event is most likely over).

5. Data

   a. Engineering judgment was used to estimate the failure probabilities for some SSCs. The numerical values provided by these judgments were typically conservative (i.e., the values were chosen such that potential PTS scenarios would not be inadvertently eliminated).

6. Human reliability analysis

   a. Pre-initiator human failure events (HFEs) were not explicitly modeled in the Oconee and Beaver Valley PTS PRAs. Such human events were assumed to be included in the industry-wide data that was used to model system unavailabilities. For the Palisades analysis, pre-initiator HFEs were left "as-is" (i.e., the existing pre-initiator HFEs in the Palisades PRA model used in the PTS analysis were not modified).

   b. The time at which operators perform an action is taken to be either the earliest the action can be performed or the latest the action can be performed, whichever exacerbates PTS conditions (e.g., if the action involves the operator successfully throttling a pump by 20 minutes, then

the action would be modeled as occurring at 20 minutes).

c. Given the uncertainty associated with the various plant conditions that could exist during hot shutdown, some human error probabilities (HEPs) were assumed to be greater than their corresponding full-power HEPs.

7. PTS bin development

a. The assignment of the large number of potential PTS scenarios (tens of thousands) to a more limited number of PTS TH bins (tens to over one hundred) involved the analysts' judgments as to how various combinations of equipment and operator successes affected the TH response of the plant when compared to a limited set of initial TH calculations. If the analysts judged that a scenario's response would be similar to an existing TH calculation, the scenario was "binned" into the existing calculation's bin. If the analysts judged that a scenario's response could be sufficiently different from the existing calculations, a new TH calculation was requested, thereby creating a new bin.

b. Typically, the analysts estimated the impact of the various equipment and operator combinations on two parameters (i.e., minimum downcomer temperature and primary pressure).

c. Minimum downcomer temperature was the most important parameter that the analysts used to decide whether an existing TH bin could represent a scenario, or whether a new TH bin should be created.

d. If the analysts determined that a PTS scenario could "fit" into more than one TH bin having similar characteristics (i.e., minimum downcomer temperatures approximately the same), they assigned the scenario to the bin believed to be more conservative (i.e., the scenario was assigned to the bin with the highest primary pressure).

3.3.2 Thermal-Hydraulics

The appropriateness of the RELAP TH analysis to assess PTS rests on the validity of the following fundamental assumptions:

- We assume that the TH methodology implemented in RELAP is appropriate to assess the conditions in the downcomer during a PTS event. RELAP estimates fluid temperatures and wall-to-fluid heat transfer coefficients that represent well-mixed conditions in the downcomer at the core elevation. This approach assumes that jets, thermal plumes, and thermal streaming are not significant factors for PTS-type loadings.
- We assume that it is appropriate to use the variation of pressure, temperature, and heat transfer coefficient with time characteristic of a *single* TH transient to represent an entire PRA bin (which may contain many tens or hundreds of transients).

In the following subsections, we discuss the appropriateness of each of these assumptions.

3.3.2.1 Appropriateness of the RELAP TH Model, in General

At the most basic level, a TH analysis requires calculation of conservation of mass and energy, from which pressure and temperature follow from the equation of state. From this information, the analysis then estimates the distribution of energy within the RCS. Within the downcomer, the interface between the thermal-hydraulic and fracture mechanics calculations is the heat flux between the downcomer fluid and the vessel wall. Heat flux quantifies the RCS energy distribution, which depends on both the temperature and heat transfer characteristics of the downcomer region. In this study, we used RELAP5/MOD3.2.2γ to estimate the heat flux and pressure boundary conditions. RELAP5 is a best-estimate systems code that models heat transfer and hydrodynamic processes without any intentional conservative or nonconservative modeling features. The code has been extensively documented [RELAP 01]. Our specific validation of RELAP5 addressed

its ability to accurately estimate pressure, downcomer fluid temperature, and wall-to-fluid heat transfer coefficients for PTS loading conditions [***Fletcher***]. In these validation studies, which are summarized in Section 6.2, we compared RELAP5/MOD3.2.2γ predictions of pressure and temperature to measurements made in the most ideally scaled integral systems test facilities. These comparisons demonstrate that RELAP5/MOD3.2.2γ predictions of pressure and temperature appropriately characterize PTS loading events.

3.3.2.2 Appropriateness of the TH Model

RELAP5 calculates fluid temperatures and wall-to-fluid heat transfer coefficients that are characteristic of a well-mixed downcomer (at the core elevation). Dickson evaluated the suitability of this assumption using a predecessor of FAVOR [Dickson 87]. In that study, the base-case calculation represented a hot leg break with a diameter of 2-in. (5-cm) and a "nominal" plume strength of 140°F (60°C). (Plume strength equals the temperature difference between the colder water below the cold legs and the balance of the downcomer.) It should be noted that this "nominal" plume strength greatly exceeds any plumes that have been measured, as detailed in the following paragraph, and this "nominal" plume had no discernable effect of (relative to no plume at all) on the probability of through-wall cracking estimated by FAVOR. Furthermore, a doubling of the nominal plume strength produces only a 30% increase in the estimated probability of through-wall cracking. This study provided an indication that the well-mixed downcomer assumptions made by both RELAP and FAVOR are appropriate.

More recently, we have performed additional work to establish the adequacy of the assumption of a one-dimensional (1D) temperature boundary condition, as follows:

- A new integral experimental program was conducted at the APEX-CE test facility at Oregon State University to study cold leg and downcomer mixing [***Reyes-APEX***].
- We reviewed existing experimental databases, including integral system tests in the Loss-of-Fluid Test (LOFT) facility and the Rig of Safety Assessment (ROSA), as well as full-scale tests in the Upper Plenum Test Facility (UPTF), and reduced-scale mixing tests at Creare, Purdue University, and Imatron Voimy Oy (Finland).
- We performed mixing calculations using the REMIX code and computational fluid dynamics (CFD) codes.

In thermal-hydraulic evaluations of PTS [***Bessette***], we compare these experimental data and RELAP5/MOD3.2.2γ predictions of pressure and temperature to establish the adequacy of the uniform temperature approximation. As seen consistently in the experimental data, the downcomer is well-mixed. In integral system test data, the temperature variations seen in the in the axial or azimuthal directions is on the order of 9°F (5°C). Large temperature gradients (i.e., on the order of 180°F, or 100°C) are often seen in the cold leg following loop flow stagnation. However, temperature gradients in the cold leg do not translate to corresponding temperature variations in the downcomer because of the large eddy mixing occurring in the downcomer.

In summary, the maximum plume measured in any integral test facility representation of a PTS transient is on the order of 9°F (5°C). Probabilistic fracture mechanics calculations show that much larger plumes (strengths of ≈ 216°F, or 120°C) are needed to have even small effects on the estimated probability of through-wall cracking [***Bessette***, Section 5.5]. For these reasons, the modeling approaches of both the RELAP and FAVOR codes with regard to temperature uniformity throughout the downcomer are viewed as both appropriate and non-biasing for this application.

3.3.2.3 Appropriateness of a Using a Single TH Transient To Represent an Entire PRA Bin

In Section 3.1.2.1.2, we described the iterative process used to establish the single TH transient that represents an entire PRA bin (which may contain many tens or hundreds of transients). This process includes continual partitioning of the PRA bins that contribute significantly to the estimated TWCF until the total estimated TWCF for the plant does not change significantly with continued partitioning. Given that process, the appropriateness of using a single TH transient to represent an entire bin (which may contain tens or hundreds of sequences that can produce, in principal, a like number of different TH responses) is not justified based on the exact agreement of the representative TH transient to all of the other transients in the bin (which is not, and cannot, be guaranteed). Rather, the appropriateness is justified by the procedure detailed in Section 3.1.2.1.2, which ensures that further subdivision of the bins would not result in significant changes to the TWCF (the desired output of the analysis).

3.3.3 Probabilistic Fracture Mechanics

The appropriateness of the FAVOR PFM analysis to assess PTS rests on the validity of the following four fundamental assumptions:

- We assume (in general) that linear elastic fracture mechanics (*LEFM*) is an appropriate methodology to use in assessing the structural integrity of RPVs subjected to PTS loadings, and (in particular) that FAVOR predictions of the fracture response of RPVs in response to PTS loading are accurate.
- We assume that the effect of crack growth by subcritical mechanisms (i.e., environmentally assisted cracking and/or fatigue) is negligible and, consequently, the flaw population of interest is that associated with initial vessel fabrication.
- We assume that the fracture toughness of the stainless steel cladding is adequately high, and remains so even after irradiation, so there is no possibility of cladding failure as a result of the loading imposed by PTS transients.
- We assume that stresses are sufficiently low at locations in the vessel wall between $3/8 \cdot t_{wall}$ from the vessel ID and the OD, so the probability of failure associated with postulated defects in this region does not have to be calculated because it is zero.
- We assume that if a particular transient does not achieve a temperature in the downcomer below 400°F (204°C), it does not contribute to the vessel failure probability.

In the following subsections, we discuss the appropriateness of each of these assumptions.

3.3.3.1 Use of Linear Elastic Fracture Mechanics

One fundamental assumption in constructing our PFM model is that a linear elastic stress analysis of the vessel, and consequent fracture integrity assessment using the techniques of linear elastic fracture mechanics (*LEFM*), are accurate. Evidence supporting the appropriateness of this assumption is available in the following areas:

(1) In Section 7.10, we summarize the results of studies aimed at experimentally validating the appropriateness of *LEFM* techniques when applied to assessing the integrity of RPVs under thermal shock and PTS experiments. The results of three experimental series performed on scaled pressure vessels at ORNL in the 1970s and 1980s demonstrate the accuracy of *LEFM* techniques in these applications.

(2) One fundamental requirement for LEFM validity is that the dimensions of the plastic zone at the tip of a loaded crack must be very small compared with the dimensions of the crack being assessed and the structure in which the crack resides [Rolfe]. Under these conditions, the error introduced by plastic flow (which is not accounted for within LEFM theories) is acceptably small. To assess plastic zone sizes characteristic of the PTS problem, we had the FAVOR probabilistic fracture mechanics code report

all of the applied driving force to fracture ($K_{applied}$) values from an analysis of Beaver Valley Unit 1 at 60 EFPY, which contribute to the *TWCF*, (i.e., those that have a conditional probability of crack initiation greater than 0). The top graph in Figure 3-4 shows these $K_{applied}$ values overlaid on the K_{Ic} transition curve, while the bottom graph shows these same values expressed in the form of a cumulative distribution function. The lower graph indicates that 90% of the $K_{applied}$ values that contribute to the *TWCF* estimate lie between 20 and 35 ksi√in (22 – 38.5 MPa√m). Using these stress intensity factor values together with Irwin's equation for the plastic zone size under plane strain conditions [Rolfe] indicates that the plastic zone radii characteristic of PTS loading range from ~0.03 to ~0.13-in. (~0.76 to ~3.30-mm) depending upon the value of $K_{applied}$ (here taken to range from 20 to 35 ksi√in, or 22 – 38.5 MPa√m), and the value of the yield strengths (here taken to be 70 ksi (on average) for unirradiated materials and 90 ksi (on average) for irradiated materials (483 and 621, respectively). These values of plastic zone radii are small compared with the thickness of a PWR reactor vessel, indicating the appropriateness of LEFM techniques. Moreover, it can be noted that as the vessel ages, irradiation damage causes the yield strength to increase. Thus, as vessels approach EOL and extended EOL conditions, LEFM techniques become, if anything, more appropriate.

3.3.3.2 Assumption of No Subcritical Crack Growth

3.3.3.2.1 Due to Environmental Effects on the Low-Alloy Pressure Vessel Steel

Stress corrosion cracking (SCC) requires the presence of three factors: an aggressive environment, a susceptible material, and a significant tensile stress. If these three factors exist and SCC can occur, growth of intrinsic surface flaws in a material is possible. Since an accurate PTS calculation for the low-alloy steel (LAS) pressure vessel should address realistic flaw sizes, the potential for crack growth in the reactor vessel LAS as a result of SCC needs to be analyzed, in principle. However, for the reasons detailed in the following paragraphs, SCC for LAS in PWR environments is highly unlikely and, therefore, is appropriately assumed not to occur for the purposes of the FAVOR calculations reported herein.

The first line of defense against SCC of LAS is the cladding that covers much of the LAS surface area of the reactor vessel and main coolant lines. This prevents the environment from contacting the LAS and, therefore, obviates any possibility of SCC of the pressure boundary.

Additionally, several test programs have been conducted over the past three decades, all of which show that SCC in LAS cannot occur in normal PWR or boiling-water reactor (BWR) operating environments. SCC of LAS in the reactor coolant environment is controlled by the electrochemical potential (often called the free corrosion potential). The main variable that controls the LAS electrochemical potential is the oxygen concentration in the coolant. During normal operation of a PWR, the oxygen concentration is below 5ppb. The electrochemical potential of LAS in this environment cannot reach the value necessary to cause SCC [IAEA 90, Hurst 85, Rippstein 89, Congleton 85]. During refueling conditions, the oxygen concentration in the reactor coolant does increase. However, the temperature during an outage is low, rendering SCC kinetically unfavorable. During refueling outage conditions with higher oxygen concentrations but lower temperatures, the electrochemical potential of the LAS would still not reach the values necessary for SCC to occur [Congleton 85].

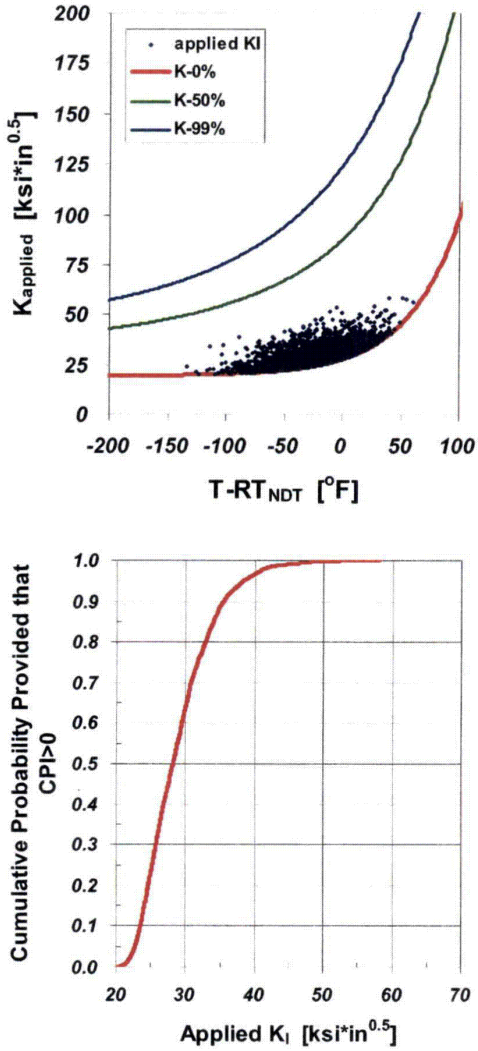


Figure 3-4. Illustration of the magnitude of $K_{applied}$ values that contribute to TWCF because they have a conditional probability of crack initiation > 0. The top graph shows all $K_{applied}$ values with $CPI > 0$ overlaid on the K_{Ic} transition curve from an analysis of Beaver Valley Unit 1 at 60 EFPY. The bottom graph shows these same results expressed in the form of a cumulative distribution function.

3.3.3.2.2 Due to Environmental Effects on the Austenitic Stainless Steel Cladding

As stated in Section 3.3.3.2.1, one assurance of the negligible effects of environmentally assisted crack growth on the low-alloy pressure vessel steel is the integrity of the austenitic stainless steel cladding that provides a corrosion-resistant barrier between the LAS and the primary system water. Under conditions of normal operation, the chemistry of the water in the primary pressure circuit is controlled with the express purpose of ensuring that SCC of the stainless steel cladding cannot occur. Even under chemical upset conditions (during which control of water chemistry is temporarily lost), the rate of crack growth in the cladding is exceedingly small. For example, Ruther et al. reported an upper bound crack growth rate of $\approx 10^{-5}$ mm/s ($\approx 4 \times 10^{-7}$ in/s) in poor-quality water (i.e., high oxygen) environments [Ruther 84]. The amount of crack extension that could occur during a chemical upset is therefore quite limited, and certainly not sufficient to compromise the integrity of the clad layer.

3.3.3.2.3 Due to Fatigue

Fatigue is a mechanism that initiates and propagates flaws under the influence of fluctuating or cyclic applied stress and can be separated into two broad stages: fatigue damage accumulation (potentially leading to crack initiation), and fatigue crack growth.

Fatigue is influenced by variables that include mean stress, stress range, environmental conditions, surface roughness, and temperature. Thermal fatigue can also occur as thermal stresses develop when a material is heated or cooled. Generally, fatigue failures occur at stresses having a maximum value less than the yield strength of the material. The process of fatigue damage accumulation, crack initiation, and crack growth closely relates to the phenomenon of slip attributable to static shear stress. Following a period of fatigue damage accumulation, crack initiation will occur by the progressive development and linking of intrusions along

slip bands or grain boundaries. Growth of these initiated cracks includes fracture deformation sequences, plastic blunting followed by resharpening of the crack tip, and alternate slip processes.

The PWR vessel is specifically designed so that all of its components satisfy the fatigue design requirements in Section III of the Boiler and Pressure Vessel Code promulgated by the American Society of Mechanical Engineers (ASME), or equivalent. Several studies have shown that the 60-year anticipated fatigue "usage" of the vessel beltline region attributable to normal plant operations, including plant heatup/cooldown, design-basis transients, etc. is low, so fatigue-initiated cracks will not occur. Similarly, fatigue loading of the vessel is considered insufficient to result in propagation of any existing fabrication defects [EPRI 94, Kasza 96, Khaleel 00].

3.3.3.3 Assumption that the Stainless Steel Cladding will not Fail as a Result of the Loads Applied by PTS

Stainless steel, even in the clad form, typically exhibits initiation fracture resistance (J_{Ic} and J-R) values that far exceed those of the ferritic steels from which the RPV wall is made (see [Bass 04] for cladding data, compared to [EricksonKirk 04] for ferritic steel data). This is especially true for the levels of embrittlement at which vessel failure becomes a (small) probability because, at the fluences characteristic of the vessel ID location, the fracture toughness of ferritic steels can be considerably degraded by neutron damage, while the fracture toughness of austenitic stainless steels are essentially unaffected by these same levels of irradiation damage [Chopra 06]. This high toughness of the stainless steel cladding coupled with the small characteristic size of defects found in the cladding [***Simonen***] justifies the assumption that the stainless steel cladding will not fail as a result of the loads applied by PTS.

3.3.3.4 Non-Contribution of Flaws Deep in the Vessel Wall to Vessel Failure Probability

In FAVOR, flaws simulated to exist further than $\frac{3}{8}\cdot t_{wall}$ from the inner diameter surface are eliminated, *a priori*, from further analysis. This screening criterion is justified based on deterministic fracture mechanics analyses, which demonstrate that for the embrittlement and loading conditions characteristic of PTS, such flaws have zero probability of crack initiation. As illustrated in Figure 3.5, in practice, crack initiation almost always occurs from flaws that having their inner crack tip located within $0.125\cdot t_{wall}$ of the inner diameter, further substantiating the appropriateness of eliminating cracks deeper than $\frac{3}{8}\cdot t_{wall}$ from further analysis.

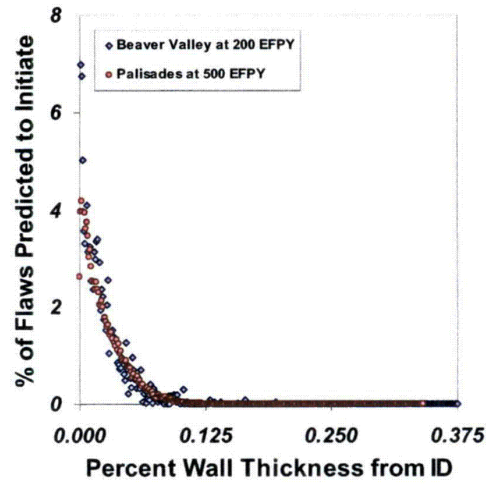


Figure 3.5. Distribution of crack initiating depths generated by FAVOR Version 03.1

3.3.3.5 Non-Contribution of Certain Transients to Vessel Failure Probability

When running a plant-specific analysis using FAVOR, we only calculated the CPTWC for TH transients that reach a minimum temperature at or below 400°F (204°C). This *a priori* elimination of transients is justified based on experience and deterministic calculations, both of which demonstrate that such transients lack

adequate severity to have non-zero values of CPTWC, even for very large flaws and very large degrees of embrittlement. Additionally, the results of our plant-specific analyses (reported in Chapter 8) show that a minimum transient temperature of 352°F (178°C) must be reached before CPTWC will rise above zero, validating that our elimination of transients with minimum temperatures above 400°F (204°C) does not influence our results in any way.

3.4 Participating Organizations

This study could not have succeeded without the cooperation of a large number of individuals both within and outside the NRC. From its inception, the commercial nuclear power industry, working under the auspices of EPRI, has been a key participant in this project. Table 3.2 summarizes the key organizations and individuals, and their contributions to this study.

3.5 External Review Panel

In response to a letter [Bonaca 03] from the Chairman of the ACRS, the NRC's Executive Director for Operations (EDO) [Travers 03] identified a need to conduct formal peer review of the technical basis developed for potential revision of the screening criteria in the PTS Rule (10 CFR 50.61). In response to the EDO's direction, RES solicited a panel of experts to perform an independent review of this report, and all supporting documentation that comprises the basis for our recommended revisions to the PTS Rule. Two peer reviewers were selected from each of the three key technical areas (PRA, TH, and PFM). Each peer reviewer was asked to provide his or her individual comments on the entire PTS technical basis without developing a consensus on a unified set of comments, to satisfy the requirement that this peer review panel must not constitute a Federal advisory committee. The following individuals served on the peer review panel.

- **Dr. Ivan Catton:** Professor at the University of California, Los Angeles. Prof. Catton is an internationally recognized expert in thermal-hydraulics, and has served as a member of the NRC's ACRS.
- **Dr. David Johnson:** Vice President of ABS Consulting Inc., Irvine, California. Dr. Johnson is an internationally recognized expert in PRA. He is involved in major risk studies and in using those studies to support decision-making.
- **Dr. Thomas E. Murley:** The chair of this peer review panel, Dr. Murley is a former Director of the NRR. Dr. Murley played a key role in regulating the operation of nuclear power plants for many years in comprehensive, high-level, broad-scope management of programs on water-cooled nuclear reactor power plants' safety and risk assessments.
- **Dr. Upendra Rohatgi:** A researcher at the U.S. Department of Energy's Brookhaven National Laboratory, Upton, NY. Dr. Rohatgi has been extensively involved in the development of thermal-hydraulic computer codes for nuclear power plant applications. In the mid-1980s, he reviewed the thermal-hydraulic analyses performed for two of the plants analyzed in developing the current PTS Rule.
- **Mr. Helmut Schulz:** Head of the Department of Structural Integrity of Components at Gesellschaft fuer Anlagen-und Reaktorsicherheit (GRS), Cologne, Germany. As a senior manager, Mr. Schulz has been involved in directing the development of PFM methodologies and managing various international cooperative research projects concerning fracture mechanics under the auspices of the Committee on the Safety of Nuclear Installations (CSNI) and the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD) in Europe.
- **Dr. Eric vanWalle:** Head of the Reactor Materials Research Department, Belgian Nuclear Research Center (SCK-CEN), Mol, Belgium. Dr. vanWalle is extensively involved in irradiation embrittlement characterization of RPV materials, and

various International Atomic Energy Agency (IAEA) and OECD/NEA cooperative research projects in fracture mechanics related to ensuring the structural integrity of nuclear power plants.

Appendix B provides more details of the peer review, including both the reviewers' comments regarding our technical basis and recommendations, and the staff's responses to those comments.

Table 3.2. Participating organizations

Sponsor/Organization		Individuals	Responsibilities
NRC	RES/DET/MEB	Mark EricksonKirk, Shah Malik, Tanny Santos, Debbie Jackson, Todd Mintz	Project management, materials, fracture mechanics
	RES/DRAA/PRAB	Roy Woods, Nathan Siu, Lance Kim, Mike Junge	PRA, human reliability analysis, event sequence analysis, risk goal
	RES/DSARE/SMSAB	Dave Bessette	Thermal-hydraulics analysis
	Oak Ridge National Laboratory	Terry Dickson, Richard Bass, Paul Williams	PFM Code FAVOR
	Pacific Northwest National Laboratory	Fred Simonen, Steve Doctor, George Schuster	Flaw distribution
	Brookhaven National Laboratory	John Carew	Fluence
	Sandia National Laboratories	Donnie Whitehead, John Forester, Vincent Dandini	PRA, human reliability analysis, event sequence analysis, external events analysis, generalization task
	SAIC	Alan Kolaczkowski, Susan Cooper, Dana Kelly	PRA, human reliability analysis, event sequence analysis, external events analysis, generalization task
	University of Maryland	Mohammad Modarres, Ali Mosleh, Fei Li, James Chang	Uncertainty analysis of PFM and TH
	The Wreathwood Group	John Wreathall	Human reliability analysis
	Buttonwood Consulting	Dennis Bley	Human reliability analysis
	INEEL	William Galyean	PRA, event sequence analysis
	ISL	Bill Arcieri, Robert Beaton, Don Fletcher	Thermal-hydraulic calculations using RELAP

Sponsor/Organization		Individuals	Responsibilities
EPRI	EPRI	Stan Rosinski	Program Management
	EPRI Materials Reliability Program (MRP) RPV Integrity Issue Task Group	Robert O. Hardies	ITG Chairman – Constellation Nuclear
	Westinghouse Electric Company	Ted Meyer, Bruce Bishop, Randy Lott, Steve Byrne, Robert Lutz, Barry Sloan, Eric Frantz	PRA, risk goal, PFM Code FAVOR, Fracture mechanics, materials, uncertainty analysis of PFM and TH
	Framatome ANP	Ken Yoon	Materials, fracture mechanics
	Sartrex Corporation	Ron Gamble	PRA, risk goal, PFM Code FAVOR
	Phoenix Engineering Associates	Marjorie EricksonKirk	Uncertainty analysis of PFM, fracture mechanics
	Constellation Nuclear – Calvert Cliffs	Robert O. Hardies	Plant-specific PTS
	First Energy – Beaver Valley	Dennis Weakland	Plant-specific PTS
	Duke Energy – Oconee	Jeff Gilreath, Steve Nadar	Plant-specific PTS
	Nuclear Management Company – Palisades	John Kneeland, Brian Brogan, Christer Dahlgren, Gary Pratt	Plant-specific PTS
	Applied Reliability Engineering	Dave Blanchard	Palisades PRA

4 Structure of this Report, and Changes Relative to Previous Reports

4.1 Report Structure

This report summarizes information found in a collection of other documents. As illustrated in Figure 4-1, various reports that concern either procedures or calculated results are available in each of three main technical areas (PRA, TH, and PFM). In this report, we do not attempt to provide a comprehensive summary of all aspects of the PFM, TH, and PRA procedures or results. Rather, in Chapters 5, 6, and 7, we focus on the key features of the PRA, TH, and PFM models, respectively, placing particular emphasis on changes between these models and those that were used to establish the technical basis for the current PTS Rule [10 CFR 50.61]. Chapter 8 goes on to detail the results of our "baseline" probabilistic calculations for Oconee Unit 1, Beaver Valley Unit 1, and Palisades. Chapter 9 summarizes various studies we have performed that collectively demonstrate the applicability of the results in Chapter 8 to PWRs in general, rather than just to the specific conditions analyzed herein. In Chapter 10, we discuss considerations associated with selecting an acceptable annual limit on TWCF, while in Chapter 11, we compare this limit to the results from Chapter 8 to establish a revision to the RT_{PTS} screening criteria currently expressed in 10 CFR 50.61.

4.2 Changes Relative to Previous Studies

To assist readers familiar with the details of calculations that form the basis for the current PTS rule [SECY-82-465] or the calculations previously reported from this effort [***Kirk 12-02***], we provide a guide to where our methodology and results differ from the previous studies, and provide pointers to locations in this and other documents where those changes are discussed in greater detail

4.2.1 Studies Providing the Technical Basis of the Current PTS Rule

As detailed in Section 3.2, one fundamental difference between our approach and that of SECY-82-465 is that here we consider all of the known factors that influence the likelihood of vessel failure during a PTS event, while accounting for uncertainties in these factors in a consistent manner across a breadth of technical disciplines (see [***Siu 99***] for details). Two central features of this approach are a focus on the use of realistic input values and models (wherever possible), and *explicit* treatment of uncertainties (using currently available uncertainty analysis tools and techniques). Thus, our current approach improves upon that employed in developing SECY-82-465, in which many aspects of the analysis included intentional and unquantified conservatisms, and uncertainties were treated *implicitly* by incorporating them into the models.

In addition to this overall change in modeling approach, the following specific changes were made in the three main technical areas:

<u>Modifications to PRA</u>

Table 5.1 (in Section 5.2.2) summarizes the differences between the current PRA and that used to support the current PTS Rule. These differences fall into the following three major categories:

(1) greater refinement and detail in the current PRA

(2) more realistic treatment of operator actions in the current PRA

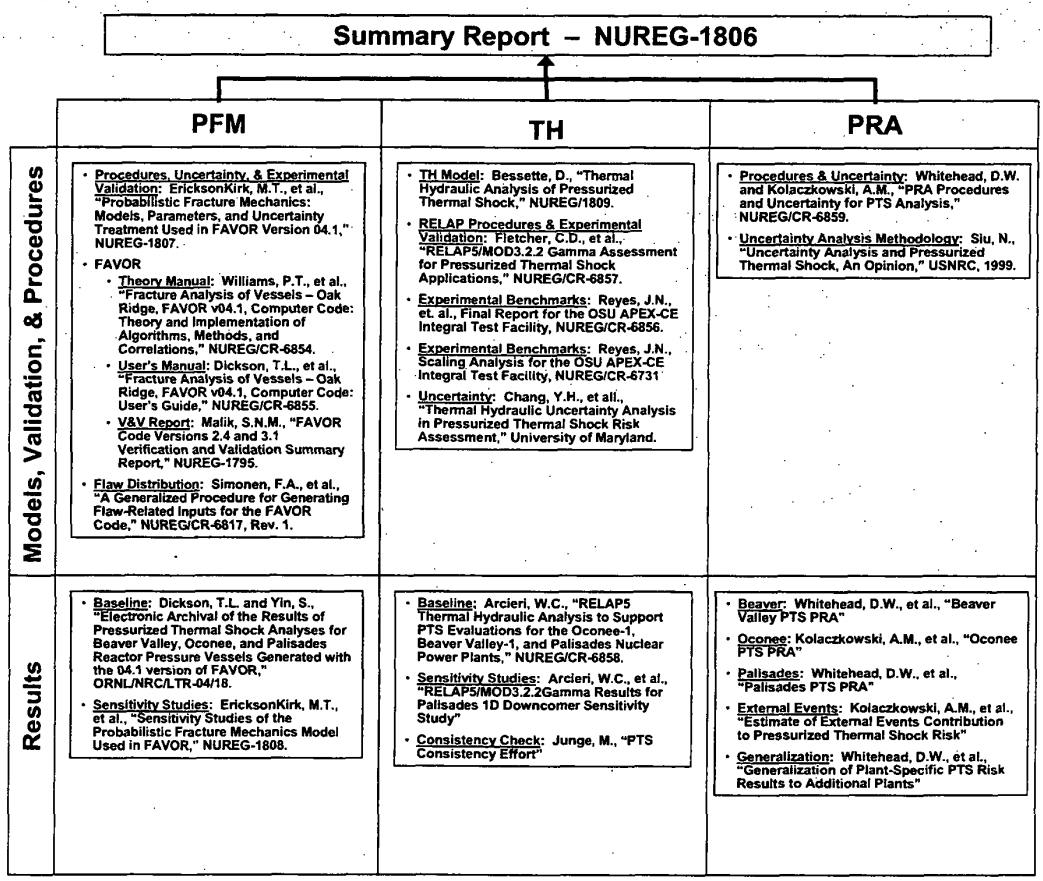


Figure 4-1. Structure of documentation summarized by this report. When these reports are cited in the text, the citation appears in *italicized boldface* to distinguish them from literature citations.

(3) Use of the latest available data on initiating event frequencies and equipment failure probabilities in the current PRA

As noted in the table, since these improvements were made with the intent of increasing both the accuracy and comprehensiveness of the PRA representation of the plants, they neither systematically increase nor reduce the estimated risk from PTS.

Modifications to TH

The first PTS study was performed during the early 1980s. In that study, TH calculations were performed for Oconee Unit 1 with RELAP5/MOD1.5 (circa 1982) and for H.B. Robinson Unit 2 with RELAP5/MOD1.6 (circa 1984). The results of those calculations were documented in [ORNL 86, ORNL 85b].

By contrast, the TH calculations performed in the current study employed RELAP5/MOD3.2.2Gamma, which was released in 1999 [RELAP 99]. The changes in the RELAP5 code in the intervening 20 years have been extensive [RELAP 99]:

- RELAP5/MOD3.2.2Gamma uses a revised treatment of non-equilibrium modeling, including wall heat transfer models and coupling of the wall heat transfer and vapor generation models.
- Interphase friction models were revised, and now incorporate a new interphase drag model for the vertical bubbly and slug flow regimes.
- A general cross-flow modeling capability was installed, allowing cross-flow connections between most types of components and among the cell faces on those components.

Other changes were implemented as a result of the code assessments related to the RELAP5 analysis for the AP600 advanced passive reactor:

- The Henry-Fauske critical flow model was added to provide a standard-reference critical flow model upon which code calculations are based.
- Changes were made in code numerics to greatly reduce recirculating flows within model regions nodalized with a multidimensional approach.
- A mechanistic interphase heat transfer model was implemented to include the effects of noncondensible gases. This change greatly improved the simulation of condensation, preventing erratic behavior and code execution failures. This change is particularly important for situations where the plant accumulators empty and nitrogen is discharged into the reactor coolant system (a situation that typically led to code execution failure at the time of the first PTS study).

In the current study, no major changes were made from the RELAP5 plant input modeling approach used in the first PTS study [ORNL 86, ORNL 85a]. With only a few exceptions, the plant input models use the same nodalization schemes as before. Those nodalization schemes reflect plant modeling recommendations and guidance for the general modeling of plant transients, which evolved over years of RELAP4 and RELAP5 experimental assessments and plant applications preceding the first PTS study.

However, the current study used capabilities in RELAP5/MOD3.2.2Gamma, including renodalization of the reactor vessel downcomer (using the general cross-flow modeling capability), conversion of the vessel/hot and cold leg connections and the hot leg-to-pressurizer surge line connection to the cross-flow format, and addition of junction hydraulic diameter input data as required by the conversion of the code to junction-based interphase drag. [***Bessette, Fletcher*** document how these RELAP5/MOD3.2.2Gamma capabilities influence the models used in the current study.]

Current computer calculation speeds and data storage capabilities exceed greatly those used during the first PTS study, allowing the number of transients that can be reasonably evaluated directly using RELAP5 to be expanded by more than an order of magnitude. In the first PTS study, budget and schedule considerations limited the number of transients evaluated per plant to about 10 to 15. By contrast, the current study used more than 500 RELAP5 transient calculations to characterize the risk of vessel failure.

Enormous advances in analysis tools (automated processes and plotting and data extraction routines) have also occurred. These tools lead to more comprehensive analyses, better communication and sharing of data, and more effective reporting of results.

<u>Modifications to PFM</u>

(1) A significant conservative bias in the unirradiated toughness index temperature (RT_{NDT}) model was removed. (See item 3 in Section 7.7.2.2 of this report and Section 3.2.2.3.1 of [***EricksonKirk-PFM***].)

(2) The spatial variation in fluence was recognized. (See item 1 in Section 7.7.2.2 of this report and Section 3.2.3.1 of [***EricksonKirk-PFM***].)

(3) Most flaws are now embedded, rather than on the surface, and are also smaller than before. (See Section 7.5 of this report and [***Simonen***].)

(4) Material region-dependent embrittlement properties were used. (See item 1 in Section 7.7.2.2 and Table 8.2 of this report.)

(5) Non-conservatisms in the crack arrest model were removed. (See item 2 in Section 7.8.2 of this report, Section 4.1 of [***EricksonKirk-PFM***], and [Kirk 02a].)

(6) Non-conservatisms in the embrittlement model were removed. (See Section 3.2.3 of [***EricksonKirk-PFM***]).

(7) The possibility of fracture on the upper shelf has been accounted for. (See item 1 in Section 7.8.2 of this report, Section 4.2 of [***EricksonKirk-PFM***], and [EricksonKirk 04].)

(8) The effect of warm pre-stress (WPS) has been accounted for. (See Section 7.7.1.1 of this report, Appendix B to [***EricksonKirk-PFM***])

(9) Uncertainties on chemical composition and $RT_{NDT(u)}$, which bound all known individual materials, have been included. (See Appendix D to [***EricksonKirk-PFM***].)

4.2.2 December 2002 Draft Report

In December 2002, we issued a draft report that detailed the results of plant-specific analyses performed on Oconee Unit 1, Beaver Valley Unit 1, and Palisades [***Kirk 12-02***]. Since that report was issued, we have made the following significant changes to our model:

Modifications to PRA

No significant changes were made to the PRA/HRA models since [***Kirk 12-02***].

Modifications to TH

The RELAP5 Oconee model was revised to incorporate comments received from Duke Power [***Arcieri-Base***]. In addition, momentum flux modeling in the downcomer was changed to avoid the erroneous prediction of recirculating flows in the downcomer that, for a small number of cases, were unphysically high. When erroneous predictions of recirculating flows occurred, the high liquid velocity resulted in correspondingly high calculations of the heat transfer coefficient. The entire set of Oconee cases was rerun.

Modifications to PFM

We revised the FAVOR PFM code. The information presented in [***Kirk 12-02***] was generated with FAVOR Version 02.4, whereas the information presented herein was generated with FAVOR Version 04.1. We made the following significant changes to FAVOR between these versions:

(1) As part of our V&V effort, we identified a bug in how FAVOR associated material properties with cracks that lie on the fusion line of welds. This bug was most significant when the toughness properties of the plates adjacent to the weld are lower and, thus, control the fracture response, as is the case with Beaver Valley Unit 1. Details of this bug fix can be found [***Malik***].

(2) FAVOR now considers the possibility of failure occurring by ductile tearing on the upper shelf. Section 7.8 of this report describes the upper-shelf model we used and our rationale for its introduction, while Section 4.2 of [***EricksonKirk-PFM***] and [***Williams***] provide full details of the FAVOR Version 04.1 upper-shelf model.

(3) FAVOR now models the effects of crack face pressure loading, as described in [***Williams***].

(4) FAVOR now accounts for the temperature dependence of thermal-elastic material properties, as described in [***Williams***].

5 Probabilistic Risk Assessment and Human Reliability Analysis

5.1 Introduction

This section describes the analysis activities associated with performing the PRA and HRA portions of the PTS reanalysis project.
As depicted in Figure 5.1, the PTS reanalysis project was a closely integrated effort among three primary technical disciplines:

(1) PRA (including HRA),
(2) TH modeling, and
(3) PFM.

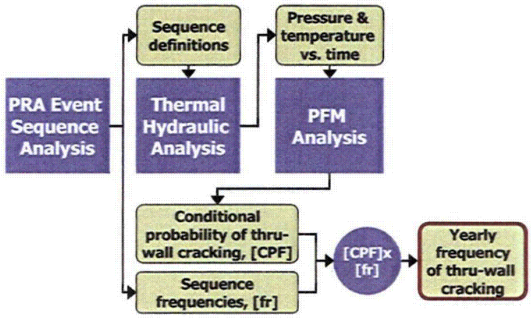


Figure 5.1. Integrated technical analyses comprising the PTS reanalysis project

As such, while this section focuses on the PRA and HRA (hereafter referred to as PRA unless specifically dealing with HRA) aspects of the reanalysis, important interfaces with the other technical disciplines are noted and cannot be completely separated from what was done in the PRA portion of the PTS reanalysis project.

A key final product of this reanalysis project is the estimation of TWCFs associated with severe overcooling scenarios. The PRA portion of the reanalysis project had three primary purposes:

(1) Define the overcooling scenarios (sequences) with the potential for being PTS challenges.

(2) Direct the TH analysis as to the specific sequences to be modeled to obtain plant TH response information to be forwarded to the PFM analysts.

(3) Estimate the frequencies, including uncertainties, for those overcooling sequences that are potentially important to the PTS results and provide that information to the PFM analysts.

In fulfilling the above purposes, the PRA analysts followed an iterative process.
The iterations were the result of (1) additional information becoming available from the other disciplines as the analyses evolved, and (2) feedback from the licensees participating in the three plant analyses (Oconee Unit 1, Beaver Valley Unit 1, and Palisades Unit 1).

For each purpose listed above, a specific product was produced. The first product, definition of the overcooling sequences, is in the form of event trees constructed by the PRA analysts for each of the three plant PTS analyses.
Event tree construction is a well-known and well-established PRA modeling tool that has been used in identifying and analyzing core damage scenarios, such as in the Individual Plant Examination (IPE) program. In this case, the same tool was used to identify and model overcooling sequences, rather than core damage sequences that could occur as a result of undercooling events. The sequences depicted by the PTS event trees represent those combinations of initiating events that disrupt normal plant operation (e.g., turbine trip), and

subsequent plant equipment and operator responses that are included in each plant model to represent overcooling sequences with the potential to be a PTS challenge.

The second product, direction by the PRA analysts to the TH analysts as to specific sequences to be modeled in their phase of the overall PTS analyses, was provided in the form of written and oral communications among the analysts. Each TH-modeled sequence was assigned a "case" number for identification purposes. For a given plant analysis, each TH "case" is a scenario that broadly represents many possible sequences on the event trees for that plant whose characteristics are similar enough that the sequences can be collectively represented by a single TH sequence (case). The TH analysts modeled each case to derive the time histories for reactor coolant pressure, reactor vessel downcomer temperature, vessel wall heat transfer characteristics, and other parameters important to defining the plant TH response during each case. This response information was subsequently provided to the PFM analysts to determine the vessel wall response (i.e., crack initiation and propagation) for the TH conditions. The modeling of multiple event tree sequences by a smaller number of "case" sequences involved a manual *binning* process that is summarized later in more detail.

The third product, sequence frequencies including uncertainties, was provided to the PFM analysts by the PRA analysts for those overcooling "case" bins that are potentially important to the PTS results. This information was provided in the form of electronic files containing a "case" bin identifier and statistical frequency information associated with that bin. These bin frequencies correspond to the "case" sequences modeled by the TH analysts and represent the combined frequencies of all event tree sequences combined into each bin. The PFM analysts then used the statistical frequency information, along with the TH information representing each bin, to estimate the TWCFs.

5.2 Methodology

A multi-step approach was followed to produce the PRA products for the PTS reanalysis project. Figure 5.2 depicts the steps followed to define the sequences of events that may lead to PTS (for input to the TH model), as well as the frequencies with which these sequences are expected to occur (for combination with the PFM results to estimate the annual frequency of through-wall cracking). Although the approach is illustrated in a serial fashion, its implementation involved multiple iterative passes through the various steps as the analyses and mathematical representations of each plant evolved. The following sections describe seven steps that together comprise the PRA analysis:

Step 1: Collect information (Section 5.2.1)

Step 2: Identify the scope and features of the PRA model (Section 5.2.2)

Step 3: Construct the PRA models (Section 5.2.3)

Step 4: Quantify and bin the PRA modeled sequences (Section 5.2.4)

Step 5: Revise PRA models and quantification (Section 5.2.5)

Step 6: Perform uncertainty analysis (Section 5.2.6)

Step 7: Incorporate uncertainty and finalize results (Section 5.2.7)

The reader should recognize that the PRA models described in this section consider *only* events internal to the operating plant (stuck-open valves, pipe breaks, etc.) as possible PTS precursors. A scoping study aimed at assessing the frequency and consequences of external initiating events (e.g., fires, floods, etc.) is detailed in a separate document [***Kolaczkowski-Ext***] and summarized in Section 9.4 of this report.

5.2.1 Step 1: Collect Information

During the initial phase of the PTS project, significant resources were expended to collect information regarding PTS in general and each plant in particular. General information-gathering activities included reviewing the basis for the current PTS Rule [10 CFR 50.61], and searching LERs for the years 1980–2000 to gain an understanding of the frequency and severity of real overcooling events [INEEL 00a]. Plant-specific information sources included the PRA analyses performed during the 1980s in support of the Integrated Pressurized Thermal Shock (IPTS) studies and the current PTS Rule [ORNL 85a, 85b, 86], as well as plant-specific design and operational information. Familiarity with all of this information provided the bases upon which the PRA analysis of each plant was conducted.

5.2.1.1 Generic Information

5.2.1.1.1 LER Review

The LER review identified a total of 128 events, demonstrating that overcooling events, or at least their precursors, do occur from time to time. These events are dominated by failure to properly control or throttle secondary side feed, a precursor that leads to relatively minor overcooling. Still, a few events have been associated with actual or potential loss of portions of secondary pressure control. These events predominantly involve equipment failures in the main feedwater, feed and steam control, and main steam systems. The results of the LER review also demonstrate that both active and passive (i.e., latent) human errors play a role, as many of the equipment failures were caused by improper maintenance or testing. Additionally, equipment in non-normal configurations can be an aggravating factor because contributing equipment faults have occurred that operators must identify, and for which they must compensate, to prevent overcooling.

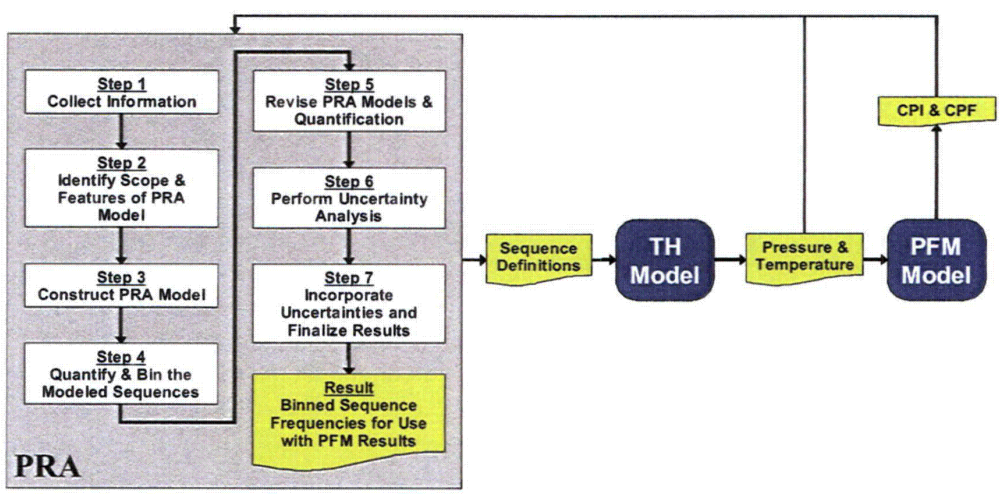


Figure 5.2. Diagrammatic representation of the PRA approach

5.2.1.1.2 Initiator Frequency and Probability Data

Initiator frequency and failure probability data are needed for initiating events, systems, and components as input to the PRA model. Since the goal of the PTS reevaluation project was to provide a PTS risk perspective for the operating fleet of PWRs, it was deemed appropriate to apply industry-wide PWR data for initiator frequencies and equipment failure probabilities in the plant-specific analyses. Hence, while the PRA model structure and the operational considerations it represented were based on plant-specific information, initiator frequencies and equipment failure probability data were generally based on industry-wide data.

Generic PWR data were obtained from two main sources. The first source, NUREG/CR-5750 [Poloski 99], summarizes industry-wide initiator experience for the years 1987–1995, along with failure probabilities for selected components. This information was updated twice. The first update was performed in an unpublished (at the time) addendum to NUREG/CR-5750 [Poloski 99], which extended the experience base through 1998. The second update dealt with loss-of-coolant initiators and was based on input intended to account for time-dependent material aging mechanisms that were not included in the experiential data [Tregoning 05]** The second source, NUREG/CR-5500 [Poloski 98], summarizes industry-wide experience for selected systems.

5.2.1.2 Specific Information

5.2.1.2.1 Previous PTS-PRA Analyses

Review of the PRA analyses performed in support of the IPTS studies and the current PTS Rule was another important input to the analyses. Of particular relevance were NUREG/CR-3770 [ORNL 86] and WCAP-15156 [Westinghouse 99] (a more recent 1999 study) since these are past analyses of two of the plants covered in this work, Oconee 1 and Beaver Valley 1, respectively. Information in NUREG/CR-4183 [ORNL 85b] concerning H.B. Robinson, and NUREG/CR-4022 [ORNL 85a] concerning Calvert Cliffs, was also considered since these documents provided additional perspectives and analytical considerations useful to this work.

5.2.1.2.2 Plant-Specific Information

At the outset of each plant-specific analysis, information was requested from the licensees pertaining to plant design, procedures, training, and other aspects of plant operation relevant to building a PRA model for analyzing PTS. Information provided in response to these requests was supplemented by information gained during plant visits and ongoing interactions (oral, written, and email exchanges) with each licensee as the analyses evolved. In total, plant-specific information was derived from the following sources:

- summaries of any recent past actual overcooling events
- current PRA model and write-ups
- final safety analysis report sections
- piping and instrument diagrams and electrical drawings
- emergency and abnormal operating procedures
- miscellaneous system design-basis information and related material
- PTS-relevant training material
- operational aspects associated with hot-shutdown conditions
- observed multiple simulator exercises at each plant involving overcooling events that were setup and run as part of a collaborative effort between each licensee and the NRC contractor PRA analysts
- periodic interactions with the licensees regarding modeling details as each analysis evolved

** Generic initiator frequency and system failure probability information (as described in Section 5.2.1.1.2) was used for Oconee 1 and Beaver Valley 1, whereas the plant-specific PRA conducted by Consumers' Energy personnel (for Palisades) incorporated plant-specific information.

- feedback from each licensee as interim results from the analyses became available

5.2.2 Step 2: Identify the Scope and Features of the PRA Model

The format, structure, and details considered in the current analyses draw considerably from the earlier PRA analyses of PTS. Aside from recognition of the results and the reasons for the results from these past analyses, limitations and conservatism associated with the past studies were identified and, to the greatest possible extent, alleviated. Other improvements were adopted with the intent of increasing both the accuracy and comprehensiveness of the PRA representations of the plants. Table 5.1 summarizes the differences between the current PRA and that used to support the current PTS Rule. These differences fall into the following three major categories:

(1) greater refinement and detail in the current PRA
(2) more realistic treatment of operator actions in the current PRA
(3) use of the latest available data on initiating event frequencies and equipment failure probabilities in the current PRA

As noted in the table, since these improvements were made with the intent of increasing both the accuracy and comprehensiveness of the PRA representations of the plants, they neither systematically increase nor reduce the estimated risk from PTS.

In addition to identifying the areas for improvement of the PRA models that are addressed in Table 5.1, review of past PRA analyses of PTS provided information in four other areas:

(1) identifying the types of sequences that needed to be included in the PRA
(2) identifying what types of initiating events should be included
(3) identifying what functions and equipment status needed to be included
(4) identifying what human actions needed to be considered

The following four subsections describe the general features of the PRA models in each area. These features were established by a team approach involving analysts skilled in both system/sequence considerations and HRA considerations. Thus, the process for building PRA models involved integrated consideration of both system/sequence and human reliability factors.

5.2.2.1 Types of Sequences

The following list details the types of sequences included in the PRA models:

- overcooling scenarios
  - at full/nominal-power operation
  - under hot-shutdown conditions
- loss of RCS pressure scenarios
- virtually sustained RCS pressure scenarios (i.e., scenarios where RCS pressure initially decreases, necessitating start of HPI to restore pressure)
- late repressurization scenarios
- scenarios that provide immediate overcooling, as well as those that begin as loss-of-cooling scenarios (i.e., undercooling) and subsequently become overcooling scenarios

Two types of scenarios commonly modeled in PRAs are not included in the current PTS analyses (as previously discussed in Section 3.3.1):

(1) ATWS scenarios
(2) ISLOCA scenarios

Sequences resulting from such scenarios were not included, based on the following considerations. First, ATWS events generally initially begin as a severe undercooling event (i.e., there is too much power for the heat removal capability) and likely involve other failures to achieve an overcooling situation. While ISLOCAs, like the LOCAs modeled in the PTS study, could involve overcooling from the start of the event, significant ISLOCAs are often assumed to fail mitigating equipment in PRAs, which ultimately causes an undercooling event and core damage. Second, with typical ATWS and sizeable (not just small leaks), ISLOCA frequency estimates in the range of 10^{-5}/yr to 10^{-6}/yr (or even lower) and with the need

Table 5.1. Comparison of PRA analyses used in this study with the PRA analyses that supported 10 CFR 50.61

Difference Between Current PRA Analyses and the PRA Analyses that Supported 10 CFR 50.61			Effect on Calculated Risk	Comments
1	Refinement of Detail Considered by the Analysis	Slight expansion of the types of sequences and initiators considered	Increase	
2		Slight expansion of support systems both as initiators and as dependencies affecting equipment response	Increase	
3		Less gross binning of TH sequences because there are more "cases" into which to bin individual TH runs	Reduce	Current work features 50–100 cases per plant whereas previous studies only considered about a dozen cases (e.g., small steamline breaks and the opening of 1–2 secondary valves were previously binned with a large guillotine steamline break, thereby treating the cooling effects of the smaller scenarios much too conservatively).
4		External initiating events considered as potential PTS precursors	Increase	See Section 9.4.
5	Treatment of Operator Actions	Credit for operator actions is based on detailed consideration of numerous contextual factors associated with the modeled sequences, on multiple simulator observations at each plant, on the latest procedures and relevant training, and on numerous discussions with operating and training staffs. Detrimental acts of commission are also considered based on these same inputs, including procedural steps that call for operator actions that can exacerbate overcooling in certain situations.	Both Increase and Reduce	
6		A greater number of discrete operator action times are considered.	Reduce	Previous studies considered success or failure of operator action generally at 1 or 2 times after the start of the event. Currently, we consider up to 3 discrete times for some operator action.
7	Use of New Data	Includes the latest industry-wide (and some plant-specific) data for initiating event frequencies, equipment failure probabilities, and common-cause considerations.	Reduce	Largest factor is the significant drop in the initiator frequencies as a result of the decrease in scram rates resulting from institutional programs executed in the 1980s and 1990s.

for other failures to occur to possibly cause a continuing and serious overcooling situation, sequences involving ATWS or ISLOCAs should not be significant contributors to PTS risk. This is because other modeled scenarios that are likely to be significant contributors to PTS risk commonly have initiator frequencies in the range of 1/yr to 10^{-3}/yr, including other LOCAs that are already modeled in the PTS study.

5.2.2.2 Initiating Events

The following internal initiating events were included in the PRA models:

- small-, medium-, and large-break LOCAs
- transients commonly modeled in PRA analyses, including:
  - reactor-turbine trip
  - loss of main feedwater
  - loss of main condenser
  - loss of offsite power (including station blackout)
  - loss of support systems, such as AC or DC buses
  - loss of instrument air
  - loss of various cooling water systems
- steam generator tube rupture (SGTR)
- small and large steam line breaks with and without subsequent isolation

5.2.2.3 Functional/Equipment Considerations

The event trees in the PRA models that depict potential overcooling sequences are based on the status and interactions of four plant functions and associated plant equipment. Figure 5.3 presents a function-level event tree depicting the four functions and resultant general types of sequences treated in the PRA models. Each plant analysis features much more detailed event trees constructed at the initiator and equipment response level that incorporate the plant-specific design and operational features. These four functions (i.e., primary integrity, secondary pressure, secondary feed, and primary flow/ pressure) are important to treat in the PTS analyses for the following reasons:

- Primary integrity: The status of this function influences the potential RCS pressure, which in turn influences the rate of cooldown (in some situations), the injection source capability, and the incoming and outgoing flow rates. All of these factors influence the vessel downcomer temperature.
- Secondary pressure: The status of this function influences the pressure and temperature in the RCS, since the RCS and the secondary side of the plant are thermal-hydraulically coupled in most scenarios. For example, a rapid drop in secondary pressure can cause rapid cooling of the RCS, affecting both the downcomer temperature and, potentially, the RCS pressure (depending on subsequent RCS injection flow and heat removal).
- Secondary feed: The status of this function influences the pressure and temperature in the RCS, since the RCS and the secondary side of the plant are thermal-hydraulically coupled in most scenarios. For example, overfeed can contribute to enhanced cooling of the RCS, affecting both the downcomer temperature and, potentially, the RCS pressure (depending on subsequent RCS injection flow and heat removal).
- Primary pressure/flow: The status of this combination of conditions influences the RCS pressure and flow conditions (forced flow or natural circulation) during the overcooling event as well as the nature of the injection that can add cooling to the vessel wall. The flow characteristics either exacerbate or mitigate flow stagnation, which can also affect the downcomer temperature.

In the plant event trees, the status of equipment relevant to each function is modeled in each PRA. This means that for each plant, the status of equipment relevant to each function is identified and included in the sequence modeling.
For illustrative purposes, the following list summarizes the equipment associated with each function in the PRA models:

- Primary integrity: Status of pipe breaks, PORVs and associated block valves,

pressurizer SRVs, and pressurizer heaters and spray considerations where appropriate.

- Secondary pressure: Status of steam line breaks, MSIVs and associated non-return valves, as well as related bypass and drain valve considerations where appropriate, turbine throttle and governor valves, steam dump/turbine bypass valves and associated isolation valves (if any), ADVs and associated isolation valves, and secondary steam relief valves (SSRVs).
- Secondary feed: Status of main feedwater (MFW), condensate, and auxiliary/emergency feedwater (AFW/EFW) systems.
- Primary pressure/flow: Status of high head safety injection, charging pumps and letdown considerations, accumulators/safety injection tanks, low head safety injection, and reactor coolant pumps (RCPs).

The status of other equipment that is relevant because of interactions with the equipment in this list is also modeled as appropriate. Such equipment includes the actuation and protection/isolation circuitry associated with the equipment in the preceding list, and support systems including cooling water, instrument air, and electric power and instrumentation. Heating and ventilation equipment was not considered in the analyses because of the slow effects of such a loss, and since the loss can often be easily identified and recovered.

5.2.2.4 Human Action Considerations

Plant records of overcooling events that have actually occurred demonstrate that operator actions and inactions can significantly influence the degree of overcooling and the RCS pressure for many types of overcooling events. Consequently, operator action directly influences, in both beneficial and detrimental ways, the potential for many types of event sequences to become serious PTS challenges. For example, early operator action to isolate the feed to a faulted (depressurizing or already depressurized) steam generator directly affects the amount of overcooling that occurs and/or how long such cooling is sustained.

Consequently, any "realistic" PTS analysis needs to consider operator actions and inactions that influence overcooling sequences. Therefore, consistent with the guiding principles of this project to adopt best-estimate models and treat uncertainties explicitly whenever practicable, a rigorous treatment of human actions is included in the PRA models. The process to identify, model, and probabilistically quantify human factors derives largely from NUREG-1624, Revision 1 [NRC 00], which uses an expert elicitation approach. In this study, the experts included both NRC contractors and licensees. These individuals considered both errors of omission and acts of commission. This process identified several general classes of human failures (see Table 5.2), which have been incorporated into the PRA models. Table 5.2 also details which of the four primary functions (identified in Section 5.2.2.3) these failures most affect.

5.2.3 Step 3: Construct the PTS-PRA Models

The well-known and well-established event tree-fault tree PRA methodology was adopted as the basis for all plant-specific analyses. However, the modeling approach varied somewhat from plant-to-plant because of the order in which the plants were analyzed (lessons learned in the Oconee analysis impacted the Beaver Valley and Palisades modeling approach, for example). Additionally, the availability of information from TH and PFM at the time PRA modeling began influenced how the PRA model evolved. A summary is provided below of the modeling approaches for Oconee, Beaver Valley, and Palisades.

Table 5.2 General classes of human failures considered in the PTS analyses

Primary Integrity Control	Secondary Pressure Control	Secondary Feed Control	Primary Pressure/ Flow Control
I. Operator fails to isolate an isolable LOCA in a timely manner (e.g., close a block valve to a stuck-open PORV) II. Operator induces a LOCA (e.g., opens a PORV) that induces/enhances a cooldown	I. Operator fails to isolate a depressurization condition in a timely manner II. Operator isolates when not needed (may create a new depressurization challenge, lose heat sink...) III. Operator isolates wrong path/SG (depressurization continues) IV. Operator creates an excess steam demand such as opening turbine bypass/atmospheric dump valves	I. Operator fails to stop/throttle or properly align feed in a timely manner (overcooling enhanced or continues) II. Operator feeds wrong (affected) SG (overcooling continues) III. Operator stops/throttles feed when inappropriate (causes underfeed, may have to go to feed and bleed possibly causing overcooling)	I. Operator does not properly control cooling and throttle/terminate injection to control RCS pressure II. Operator trips RCPs when not appropriate and/or fails to restore them when desirable III. Operator does not provide sufficient injection or fails to trip RCPs appropriately (failure to provide sufficient injection is modeled as leading to core damage; thus, such sequences are not PTS-relevant)

5.2.3.1 PRA Modeling Differences Attributable to the Organization Constructing the Model

Both the Oconee and Beaver Valley PTS analyses use the same large event tree-small fault tree modeling format adopted by the PRAs that formed the technical basis for the current PTS Rule. This approach makes best use of the earlier work in constructing updated PRA models. Since the desired outputs do not require the explicit component faults for some systems included in the model, very simple system fault trees were used with corresponding system-level failure data to represent the failure or unavailability of these systems.

In contrast, a plant-specific PRA model developed by the licensee was used to provide the starting point for the PRA model of the Palisades plant used in this project. The licensee's PRA includes more detailed component-level fault trees for all the systems included in the PTS-PRA model. However, in all three analyses, the level of resolution in the results is sufficient for the purposes of assessing the PTS risk.

5.2.3.2 PRA Modeling Differences Attributable to the Order of Plant Analysis

The PRA model of Oconee was constructed first (at a time when feedback information from the TH analysis and from the PFM analysis was not yet available). Consequently, it was not possible to screen out of the model overcooling sequences having a benign TH response or very low estimated conditional probabilities of through-wall cracking (from the PFM analysis). Hence, the Oconee PRA model contains virtually all the possible overcooling sequences with virtually no *a priori* screening out of "low significance" sequences. Subsequent feedback from both TH and PFM verified that many of the sequences included in the Oconee model could justifiably been omitted from the PRA model.

Work on the Beaver Valley PRA model was initiated after the Oconee model had been constructed, at a time when the Oconee analysis results, while still evolving, were generally well-understood. Also, as the Beaver Valley PRA model was being constructed, some advanced TH and PFM results were already available for Beaver Valley sequences (identified from "lessons learned" from the Oconee analysis). Consideration of this TH/PFM information on Beaver Valley permitted *a priori* screening of the following general categories of sequences from the Beaver Valley PRA model:

- Sequences involving certain combinations of stuck-open pressurizer PORVs or SRVs were not modeled.
- Sequences involving certain combinations of secondary valve and simultaneous pressurizer PORV/SRV stuck-open events were not modeled.
- Sequences involving only secondary valve (single or multiple) stuck-open events were not modeled.
- Sequences involving overfeed of various steam generator (SG) combinations were not modeled.
- Sources of secondary depressurization downstream of the MSIVs were not explicitly modeled.
- SGTR sequences (including those involving lack of proper feed control and even with RCPs shutdown, possibly inducing RCS loop stagnation) were not modeled.
- Other sequences were screened from modeling on a case-by-case basis if the sequence frequency could be conservatively estimated as less than $\sim 10^{-8}$/yr. This screening limit was used because, when coupled with the maximum CPTWC (i.e., failure) calculated for any type of sequence (in the 10^{-3} range) a TWCF of $<10^{-11}$/yr would be generated. Such frequencies would clearly not be important to the overall PTS results, since some other sequences were known to involve TWCFs in the 10^{-8}/yr range.

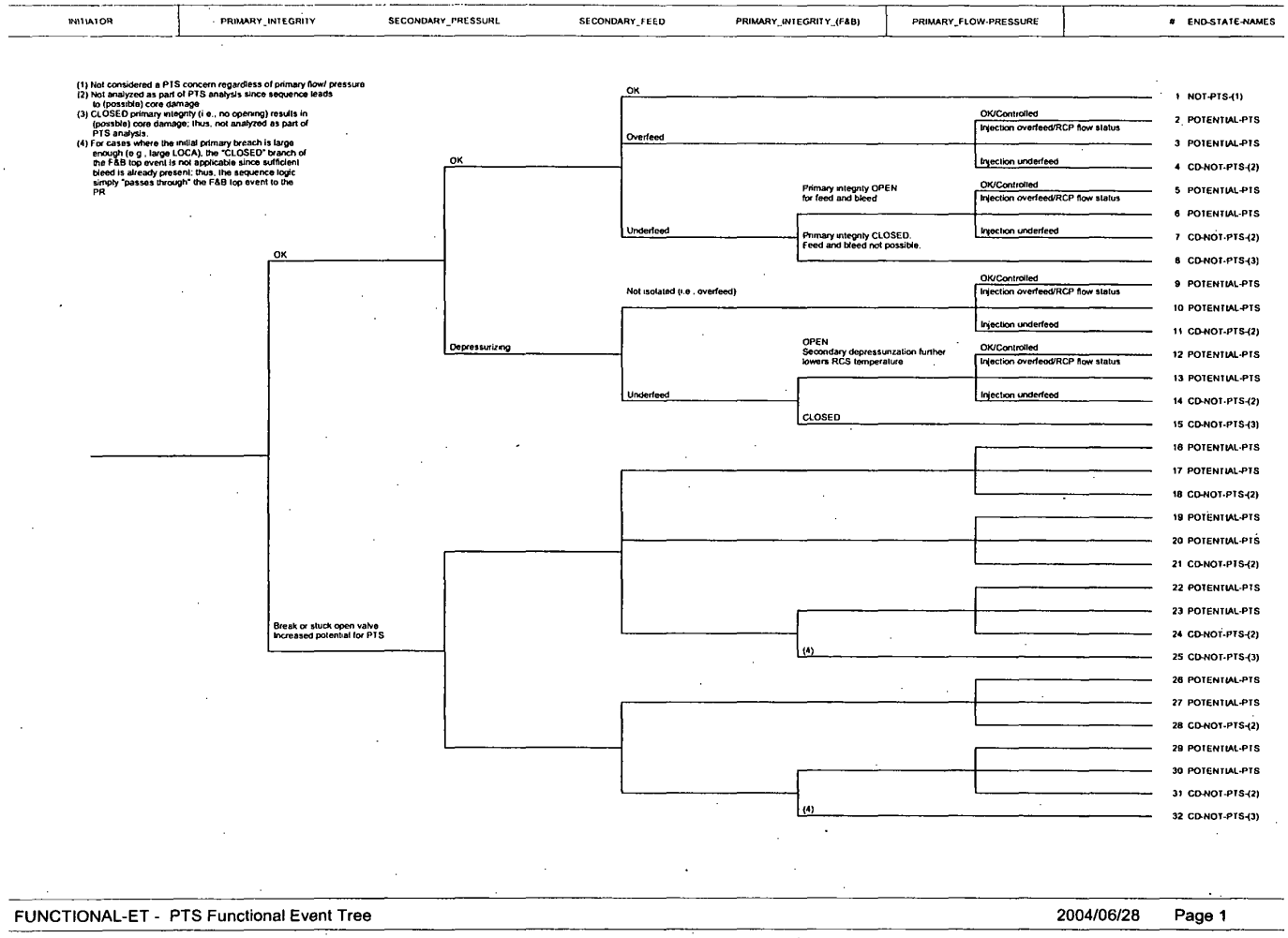


Figure 5.3. Functional event tree as the basis for PTS PRA analysis

Because the Palisades model was built starting with an already established licensee component-level PRA model with overcooling sequences, it is the most detailed model of the three. This preexisting Palisades model was augmented by the licensee, on the basis of NRC contractor review and input, to include possible scenarios and other factors not already in the preexisting model. Consequently, the "lessons learned" from the Oconee PRA also influenced the Palisades PRA model. In general, the Palisades PRA model addresses the same types of initiators and sequences, as do the Oconee and Beaver Valley models. However, the initiating event frequencies, equipment failure probability data, and human failure estimates are specific to Palisades.

5.2.4 Step 4: Quantify and Bin the PTS-PRA Modeled Sequences

For each plant, two conditions were modeled: full operating power and hot zero power (HZP). As identified in Section 5.2.3.2, little information was available to screen out potential PTS sequences for Oconee. Thus, because of a SAPHIRE code [SAPHIRE] limitation (i.e., the inability to store more than 100,000 sequences in a database); it became necessary to produce separate SAPHIRE models for full-power and HZP. Once the models (i.e., the event trees and fault trees) were constructed, the SAPHIRE code was used to generate the sequence logic for each event tree, and to solve the resulting sequences (90,629 sequences for each model) with no truncation attributable to frequency.

Given the number of potential PTS sequences for Oconee (181,258), it was necessary to group (i.e., bin) sequences with like characteristics into representative TH cases that could be analyzed with the RELAP TH code [RELAP].

Initial bins were constructed by developing event tree partitioning rules in SAPHIRE, and then applying those rules to produce the TH bins. Development of the partitioning rules required the analysts to examine the TH information available from preliminary analyses to identify the characteristics that would be important to the binning process.

Using this information, the analysts then made judgments as to whether existing TH characteristics could be used to represent new groups of sequences. If the analysts judged that existing characteristics were appropriate, either because they matched the examined sequences exactly or because the TH conditions from the new sequences were expected to be similar to, but not be worse than, the conditions from the existing analysis, the uniquely defining characteristics associated with the existing TH analyses were written in rule form for application in SAPHIRE. For those cases where the analysts were sufficiently unsure as to the appropriateness of using existing characteristics, new TH characteristics were identified. These new sets of characteristics were discussed with the TH analysts. If those discussions led to the conclusion that the expected TH conditions could be sufficiently different from prior TH analyses and the frequency of occurrence of the conditions was such that they could not be "added" to some existing TH bin without being unnecessarily conservative, a new TH calculation (and hence, TH "case") was identified. The TH characteristics associated with this new calculation were then written in rule form for subsequent application in SAPHIRE.

This iterative process continued until all accident sequence cut sets were associated with a specific TH bin. Thus, the final application of the developed rules involved the examination of each sequence cut set to determine which rule the cut set met, the subsequent "tagging" of the cut set, and the gathering of like-tagged cut sets into initial TH bins. Once all cut sets were gathered into the initial TH bins, the bins were re-quantified using a truncation limit of 10^{-10}/yr.

For Beaver Valley, essentially the same process was followed. The major difference between the Oconee and Beaver Valley analyses was in the number of sequences developed and solved (a total of 8,298 sequences for Beaver Valley for power and HZP). As discussed in the previous subsection, knowledge about what was and was not

important in the Oconee analysis was used with preliminary sequence frequency estimates and CPTWC results from early Beaver Valley TH and PFM calculations to minimize the number of sequences actually modeled in the corresponding SAPHIRE databases. Given the significantly lower number of sequences, no truncation was performed on the initial TH bins.

For Palisades, the process was somewhat different, in that the SAPHIRE model included both power and HZP sequences in the same database (only 3,425 sequences total) and the sequences were solved using truncation value of a 10^{-9}/yr. Another difference between the Palisades and Oconee or Beaver Valley analyses was how the TH bins were created. In the Palisades analysis, each sequence end state was defined to a specific TH bin and all resulting cut sets were placed in the defined bin. (Note: use of this binning process rather than the one used in the Oconee or Beaver Valley analyses did not have any significant impact on the results, which are similar across the three plant analyses. It is simply that the binning process was somewhat less refined for bins that, based on experience with Oconee and Beaver Valley, were not expected to significantly influence the estimated TWCF values.)

5.2.5 Step 5: Revise PTS-PRA Models and Quantification

With preliminary results available, reviews were conducted by both licensee and internal project staff. This allowed for formal feedback from the licensee with regard to the PTS-PRA models, inputs, assumptions, and results, and gave the analysts an opportunity for self-review of the PRA performed to date. The purposes of the reviews were to determine the following:

- whether inaccuracies existed in the models, and whether additional potential PTS sequences needed to be modeled
- whether additional TH bins should be created to reduce unnecessary conservatism based on new or updated information obtained from preliminary CPTWC calculations or needs identified by the uncertainty analysis
- which human actions were associated with the important TH bins
- which of those human actions should be reexamined to produce even more realistic (i.e., less conservative) HEPs
- what combination of the above could be accomplished within the constraints of the project

For Oconee, the reviews identified the following needs:

- to add one more type of potential PTS sequence
- to add more TH bins to address uncertainty issues and reduce conservatism (note that conservatism is reduced by not having too many sequences represented by a bin that is described by plant conditions that are too conservative for the actual conditions of the sequences)
- to reexamine some human actions to produce updated HEPs to account for more specific conditions

The Beaver Valley reviews identified the following needs:

- to add more TH bins to address uncertainty issues and reduce conservatism
- to reexamine some human actions to produce updated HEPs to account for more specific conditions

Because the Palisades analysis was performed by the utility, the results of the review described here dealt only with issues identified by the NRC review of the licensees' PTS model. The review identified the following needs:

- to add more break sizes to the LOCA class of initiating events
- to modify probabilities for a few selected basic events
- to add more TH bins to address uncertainty issues

It should be mentioned that while formal reviews were performed, such as during the second plant

visits at both Oconee and Beaver Valley, informal periodic reviews were conducted through frequent written and oral communications among the licensees and project staff. Appropriately, the models were revised and requantification was performed on the basis of these licensee inputs and as a result of self-evaluations by the project staff.

5.2.6 Step 6: Perform Uncertainty Analyses

The primary objective of the PRA portion of the PTS analyses was to produce frequencies of the set of representative plant responses to plant upsets (i.e., scenarios). These scenarios involve mitigating equipment successes and failures, as well as operator actions that result in various degrees of overcooling of the internal reactor vessel downcomer wall. The major areas of uncertainty associated with the PRA can be grouped into two broad categories:

- modeling of the representative plant scenarios
- estimation of the frequency of each modeled scenario

These areas of uncertainty and the techniques used to deal with the uncertainties are discussed in the following two subsections.

5.2.6.1 Modeling of Representative Scenarios to Characterize Aleatory Uncertainty

Each scenario in the PRA is represented by a collection of events described by the logic of the event tree and relevant fault trees for each initiating event identified in the analysis. The model initially assumed binary logic (e.g., the valve either fully recloses or sticks wide open with no intermediate states) for the events. The only explicit modeling of event timing involved the timing of operator actions (i.e., failure to take an action is modeled as failure to take that action in multiple discrete times — for example, by 10 minutes, by 20 minutes — each with a probability). Most uncertainties with regard to model structure (e.g., completeness, intermediate states) were not quantified. However, where deemed potentially important, a few aleatory uncertainties were addressed by purposely changing the model and assigning a probability to the applicability of the model change. Each of these changes became a different scenario (TH bin) with an associated frequency (e.g., area associated with a stuck-open SRV reduced 30%, timing of enclosure of a stuck-open SRV (3,000 s *vs.* 6,000 s), actual break size of small and medium LOCAs). Since it is unknown which scenario will occur following an initiating event, the complete set of scenarios, as represented by the event trees, characterize a large part of the aleatory uncertainty associated with the occurrence of a PTS challenge. The most important of these uncertainties that were explicitly handled in the analyses are addressed further in the next step, Step 7.

In addition, there is the overall uncertainty regarding the completeness of the PRA model(i.e., have all scenarios that potentially lead to PTS conditions been identified and modeled). This uncertainty issue was addressed non-quantitatively through both internal (i.e., NRC and its contractors) and external (i.e., licensee) reviews of the PRA model. As a result of this peer review process, the models are expected to produce a sufficiently complete set of potential PTS sequences and thus, any incompleteness in the models is expected to have a negligable effect on the results.

5.2.6.2 Quantification of Scenario Frequencies to Characterize Epistemic Uncertainty

Each scenario from the set of modeled scenarios is the interaction of what are treated as random events:

- initiating event (plant upset)
- series of mitigating equipment successes/failures (e.g., MFW trips, AFW starts, ADV challenges when one sticks open)
- operator actions (e.g., fails to close the ADV isolation valve by 20 minutes after the ADV sticks open)

Thus, the occurrence of each scenario is random, and the frequency of each scenario is obtained using the following equation:

Eq. 5-1

$$f_{scenario} = f_{initiating-event} \cdot Py_{equipment-response} \cdot Py_{Operator-Actions(s)}$$

where f denotes a frequency and Py denotes a probability.

Each of the variables used to obtain the scenario frequency has an epistemic uncertainty described by a distribution. The source of this information came primarily from the input data used in the analysis (i.e., the addendum to NUREG/CR-5750 [Poloski 99] for Oconee and Beaver Valley, and the plant-specific data used in the Palisades analysis). For a few specific model inputs, other data sources were also used to derive these uncertainty estimates. For the HEPs, both best-estimate values and uncertainty ranges and distributions were derived through the expert elicitation processes carried out in the human reliability analyses. Latin Hypercube sampling techniques were used to propagate these epistemic uncertainties to generate a probability distribution for each scenario frequency. Thus, the frequencies provided by the PRA analysts to the PFM analysts were described by histograms representing the resulting frequency distributions. In this way, these PRA uncertainty distributions were propagated through and combined with the PFM uncertainties to ultimately derive uncertainty distributions in the estimated TWCFs.

5.2.7 Step 7: Incorporate Uncertainty and Finalize Results

This section discusses important uncertainties (largely aleatory in nature) specifically addressed in the PRA and describes how each was handled. As described in the previous subsection, epistemic uncertainty in the frequency for each of the final TH bins was estimated using Latin Hypercube sampling techniques and is not described in this subsection.

The uncertainties discussed below were dealt with quantitatively; however, the degree of resolution associated with each specific uncertainty was limited. These uncertainties include:

- size of the LOCA within a LOCA category plus other factors (e.g., initial injection water temperature)
- size of the opening associated with single or multiple stuck-open SRV(s)
- time at which a stuck-open SRV recloses
- time at which operators take or fail to take action

These uncertainties were highlighted for specific treatment in the analysis based on (1) the scenarios found to be most important to the PTS results, and (2) a series of uncertainty analyses performed by the University of Maryland (UMD) project team members on many of the inputs and parameters potentially affecting the PTS results to see which uncertainties would most affect those results. The specific UMD analyses are discussed in [***Chang***]. The results of that work concluded that the above uncertainties are sufficiently important that they needed to be treated explicitly in the PRA model. These uncertainties and how they were addressed are discussed in the following paragraphs.

The actual break size of a LOCA for a specific LOCA class (i.e., small, medium, or large) can be any point on the spectrum of sizes defined by the two end points for that class. In addition, other factors (e.g., initial injection water temperature, break location, and injection flow rate) can contribute to the overall PTS model uncertainty, since these factors along with the specific break size affect the rate of cooling and subsequent plant response. Numerical probability results from the UMD uncertainty analysis were used to model and estimate the importance of the various modeling uncertainties examined in the UMD analysis, including different break sizes within a given class (which were assumed to be uniformly distributed). These numerical analyses provided a spectrum of different plant TH responses arising from uncertainties in these key parameters including break size. This spectrum of results was then represented by a number of discrete cases to cover the total spectrum of results (typically, five cases for small LOCAs, three for medium

LOCAs, and one for large LOCAs). Each case was assigned a probability by the UMD analysts based on how much of the total spectrum the discrete case represented. Each discrete case was assigned a new TH case number with corresponding TH curves, and the frequency of each new case was adjusted using the UMD assigned probability for that case. This was accomplished through the following steps:

- gather all cut sets from all sequences generated for a specific LOCA class into one bin
- reproduce the gathered cut sets a specified number of times corresponding to the number of discrete cases defined to represent the spectrum of results
- modify each set of reproduced cut sets to include the probability assigned to that discrete case

Thus, the new modified cut sets account for the uncertainty associated with various parameters examined in the UMD analysis, including possible variation of break sizes within a given LOCA class.

Just as with the LOCAs, the size of the opening associated with a stuck-open SRV can vary from sizes that are not PTS-significant to the valve fully stuck open. To deal with this and other relevant issues examined in the UMD analysis, the cut sets (and their associated frequencies) from stuck-open SRV sequences were modified to include a fraction that represented the uncertainty from the UMD work. In this case, it was assumed that the SRV opening size is uniformly distributed (any specific opening is equally likely) and the resulting fraction was included in the sequence frequency estimates to account for that fraction of possible SRV size openings that would be sufficient, from a cooling perspective, to be potentially important.

The time at which a stuck-open SRV recloses is unknown and can occur at any point after the valve sticks open. To approximate this, the frequencies associated with stuck-open SRV sequences with subsequent closure of the SRV were divided equally between two specific SRV reclosure times (i.e., 3,000 s and 6,000 s). These two time points were chosen after reviewing stuck-open SRV TH conditions. The 6,000 s point was chosen to coincide with the time when the change in downcomer wall temperature had "flattened out." The 3,000 s point was chosen to coincide with the time when sufficient cooling had occurred to the downcomer wall such that PTS could become an issue. Use of these two times provides a mechanism for determining some measure of the uncertainty associated with reclosure of stuck-open SRVs. Each case was assigned a 50% chance of occurring[††].

Just as the time at which a stuck-open SRV recloses is unknown, so too are the times at which operators perform actions. To address this issue, the times at which selected operator actions (i.e., those believed to be relatively important to PTS) were performed was varied. Typically, two or three different times were chosen to represent the uncertainty in when the action would be performed. Once the times were defined, typically (1) as early as could be expected, (2) as late as possible that would still affect the outcome, and (3) for some actions, some intermediate time, the probability of failing to perform the action by the specified time was developed. Use of these operator action times provides a means of estimating the uncertainty associated with when the operators actually perform their actions.

For the Oconee analysis, all issues identified above were incorporated into the analysis. For the Beaver Valley and Palisades analyses, results from the UMD analysis indicated that little uncertainty came from the sequences involving stuck-open SRVs that remained stuck open; thus, no modifications were made to those types of sequences in the Beaver Valley and Palisades analyses. However, all other

[††] Subsequent sensitivity analyses demonstrated that the 6,000 s time is nearly the worst time from a PTS challenge point of view. The worst conditional probabilities of vessel failure typically occur if the SRV is assumed to close at 7,000 s or a little beyond, but the vessel failure probabilities are within a factor of ~2 of those calculated for 6,000 s. See also the discussion in Section 8.5.3.3.2 and Comment #76 in Appendix B.

modifications were made for the analyses of Beaver Valley and Palisades.

6 Thermal-Hydraulic Analysis

6.1 Introduction and Chapter Structure

This section describes the thermal-hydraulic analysis performed on the Oconee-1, Beaver Valley-1, and Palisades nuclear power plants:

- The Oconee-1 coolant system is a lowered-loop, Babcock & Wilcox design with two steam generators, two hot legs, and four cold legs.
- The Beaver Valley-1 coolant system is a Westinghouse design with three steam generators, three hot legs, and three cold legs.
- The Palisades coolant system is a Combustion Engineering design with two steam generators, two hot legs, and four cold legs.

The discussion in this section begins in Section 6.2 with a general discussion of thermal-hydraulic issues for transients that contribute to the risk of vessel failure attributable to reactor coolant system overcooling. This section is followed by a description of the RELAP5 code and its implementation in the TH analysis in Section 6.3. The general structure of the RELAP5 code and an overview of the physical models contained in RELAP5 are included in this section.

The modeling of the plant primary and secondary systems including model initialization is discussed in Section 6.4. Section 6.5 presents an overview of the types of transients simulated, while Section 6.6 presents an overview of the results.

A summary discussion of the experimental validation of RELAP5 is presented in Section 6.7. Section 6.8 presents a discussion of sensitivity analysis and the analysis of uncertainty.

6.2 Thermal-Hydraulic Analysis of PTS Transients

The PTS analysis combines the thermal-hydraulic response of the reactor coolant system with the thermal response of the reactor vessel. These parameters, when combined with the PFM analysis, are used to estimate the probability of unstable crack propagation leading to possible vessel failure. The principal purpose of the TH analysis is to generate the time histories for key parameters for use in the FAVOR fracture mechanics analysis code, for various plant transients. The parameter responses passed to the FAVOR code are the reactor vessel downcomer fluid temperature, primary system pressure, and heat transfer coefficient on the inside of the vessel wall.

A wide variety of transients that could contribute to the risk of vessel failure were analyzed. These transients include reactor system overcooling attributable to a LOCA or a stuck-open primary side relief valve, a component failure that results in an uncontrolled release of steam from the secondary side (e.g., MSLB or stuck-open secondary side relief valve), or a control system failure that results in overfilling the steam generators. Combinations of failures are also of concern and were analyzed. The transients analyzed were defined from an event and fault tree analysis to determine possible transients (or accident sequences) and their frequencies of occurrence (see Chapter 5). Each transient and its associated frequency of occurrence are factored into the PFM analysis to estimate the risk of vessel failure.

As part of the analysis, key parameters and processes that affect the reactor vessel downcomer fluid temperature, primary system pressure and heat transfer coefficient on the inside of the vessel wall were defined. The Phenomena Identification and Ranking Table (PIRT) methodology was used to identify the most

important processes that impact reactor system thermal-hydraulic response to a transient [Shaw 88, Zuber 89].

The PIRT methodology considered number of phenomenological processes and reactor system and plant boundary condition parameters. Examples of phenomenological processes include wall-to-fluid heat transfer in the downcomer, natural circulation flow, and steam generator heat transfer. Boundary condition examples include ECCS water injection temperature, break location (in the case of a LOCA), and timing of valve reclosure (for transients involving a stuck-open relief valve).

The PIRT methodology has been applied to the Yankee Rowe and H.B. Robinson plants for PTS events. In the case of Yankee-Rowe, the PIRT is based on a 1.3-in. [3.3-cm] cold leg break. This break is approximately equivalent to a 2.8-in. [7.1-cm] break when scaled up to the larger diameter of the three current plants. The H.B. Robinson PIRT was based on a 2-in. [5.08-cm] hot leg break. A PIRT was also performed as part of the assessment of RELAP5/MOD3.2.2Gamma against data from tests performed at experimental facilities that considered the wide variation in thermal-hydraulic conditions that can occur in PTS transients. This assessment is discussed in the RELAP5 PTS Assessment Report [***Fletcher***]. Table 6.1, excerpted from that report, provides a list of the parameters and processes considered and their ranking. This list considers a broader view of the types of transients that were analyzed, rather than focusing on a single transient.

The PIRT table presented in Table 6.1 was used to focus the RELAP5/MOD3.2.2Gamma assessment on the following parameters that can be observed in the experiments:

- break flow
- primary system pressurization
- natural circulation/flow stagnation
- boiler-condensation mode and reflux condensation
- mixing in the downcomer
- condensation, mixing, and stratification in the cold leg
- integral system response

These parameters were selected because of their primary or secondary importance on downcomer conditions. The following three phenomena were deemed to be most important to downcomer conditions during PTS events:

- natural circulation/flow stagnation
- integral system response
- primary system pressurization

Natural circulation/flow stagnation is important because if loop flow continues (or restarts during a transient), warm water at the average coolant system temperature will be flushed through the reactor vessel downcomer, increasing the downcomer fluid temperature. In contrast, if the loop flow is stagnant, the cold ECCS water will not be mixed with water from other parts of the reactor system and the downcomer temperature will be colder relative to the natural circulation case. Integral system response is important because the ECCS injection behavior (flow rates, timing, and to some extent temperatures) are functions of the overall system behavior. System pressurization is itself a primary figure of merit in the PTS analysis. The other phenomena listed above were considered because of their effect on these main phenomena or because they potentially impact downcomer conditions. Fluid mixing in the downcomer is among these phenomena. These phenomena as well as the overall RELAP5/MOD3.2.2Gamma assessment are further discussed in Section 6.7.

6.3 RELAP5 Code Description

6.3.1 RELAP5 Analysis Process

The RELAP5/MOD3.2.2Gamma computer code released in June 1999 was used for transient analysis to determine downcomer fluid conditions. The RELAP5 code was developed for best-estimate transient simulation of light-water reactor coolant systems during postulated accidents and transients. The code models the coupled behavior of the reactor coolant system, core, and secondary side system for loss-of-coolant accidents and operational transients such as anticipated transients without scram, loss of

offsite power, loss of feedwater, and loss of flow. With RELAP5, a generic modeling approach is used that permits simulating a variety of thermal-hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater and steam systems.

Figure 6.1 and Figure 6.2 present top-level schematics of the RELAP5 modeling process and code structure.

The RELAP5 model input development process is portrayed on the left side of Figure 6.1. When modeling fluid systems with RELAP5, the physical systems are subdivided into networks of fluid cells that are interconnected by junctions. The RELAP5 model represents the fluid volumes, flow areas, path lengths and other characteristics of the physical system using a nodalization scheme of the fluid cells and junctions.

A RELAP5 input model is developed by assembling data that describes the thermal-hydraulic parameters of the physical system, such as pipe lengths, flow areas, volumes, and coefficients that simulate the pressure losses for flow through irregular geometry. The input model also requires the user to select various modeling options appropriate for the specific application, such as the critical flow model to be used and the locations in the model where it is to be activated.

The user must specify the initial conditions (pressures, temperatures, flow rates, etc.) for every model feature. In practice, RELAP5 plant transient event simulations begin from conditions that represent steady-state conditions. The initial condition input specifications cannot be made to an acceptable degree of accuracy using a manual approach. Instead, the user typically enters initial conditions that only approximate the desired ones and executes the plant model with RELAP5 in a steady-state mode until a smooth solution is attained with initial conditions that acceptably represent steady-state conditions. RELAP5 transient event simulations are then begun, starting from the accurate set of RELAP5-calculated steady-state initial conditions.

The user must specify the thermo-physical properties (such as thermal conductivity and heat capacity) for the materials of the model features that represent structures.

The user also defines the timing information for the calculation. This includes the problem start time, problem end time, a range of time step size and the interval between data points for the calculation printed and plotted output.

The RELAP5 code is executed using the input model described above and the code execution process is summarized in Figure 6.2. RELAP5 simultaneously solves the equations for the conservation of mass, momentum and energy for the fluid conditions and flows among the cells and junctions in the nodalization grid.

The code employs a set of steam tables to represent the steam, water and noncondensible gas physical properties (pressure, temperature, void fraction, quality, density, internal energy, etc.) in each cell as the transient calculation proceeds.

The transient calculation is advanced in time using discrete time steps, the selection of which is made to assure a stable solution. The code automatically makes this selection of time step size within the minimum and maximum time step range that is defined by the user via the input.

Table 6.1 Phenomena Identification and Ranking Table for Pressurized Thermal Shock in PWRs

Rank	Description	Comments
1	Break flow/diameter (or valve capacity)	Importance of LBLOCA has increased, pressure is less important
2	ECCS flow rate (Accumulator, HPI, LPI)	State on/off, shutdown head of pumps, accumulator initial pressure
3	Operator actions	Includes operating procedures, RCP trip, HPI throttling, feedwater isolation, etc.
4	Time of stuck valve reclosure	Pressurizer safety relief valves which reclose after sticking open
5	Plant initial state	Hot full power vs. hot zero power operation
6	Break location	Primary LOCA (hot leg, cold leg), MSLB (inside/outside containment, upstream/downstream MSIVs), SGTR
7	Unique plant features/design	Difference in steam generator design, number of loops, vent valves, etc.
8	Vessel to downcomer fluid heat transfer	Affects the rate at which heat is transferred from the vessel wall to the downcomer fluid. Affects risk of vessel failure in non-conduction limited situations.
9	ECCS injection temperatures	Seasonal/operational variations
10	Sump recirculation	ECCS temperature/flow changes after RWST drained
11	Feedwater control (or failure)	Post trip main feedwater behavior, steam generator overfeed events
12	Feedwater temperature	Oconee (using AFW instead of MFW during transient).
13	Reactor vessel wall heat conduction	In conjunction with vessel to downcomer fluid heat transfer, affects the rate at which heat is transferred from the vessel wall to the downcomer fluid. Important particularly on those situations where heat transfer from the wall is conduction limited.
14	Loop flow upstream of HPI	Scenario dependent, not as important for LBLOCAs
15	ECCS – RCS mixing in cold legs	Affects potential for formation of cold plumes in the downcomer
16	Flow distribution in downcomer	Affects mixing and potential for formation of cold plumes in the downcomer
17	Jet behavior, cold leg pipe to downcomer	
18	Loop injection upstream of safety injection	Scenario dependent, important for MSLB, not for LBLOCA
19	Steam generator energy exchange	
20	Timing of manual RCP trips	Risk of vessel failure lower if pumps remain on. Operator assumed to trip RCPs in accordance with plant procedures.
21	Interphase condensation and non-condensibles	RELAP5 overprediction of condensation
22	DC to core inlet bypass	Less important for LBLOCAs
23	Downcomer to upper plenum bypass	Less important for LBLOCAs
24	Upper head heat transfer coefficient under voided conditions	Less important for LBLOCAs

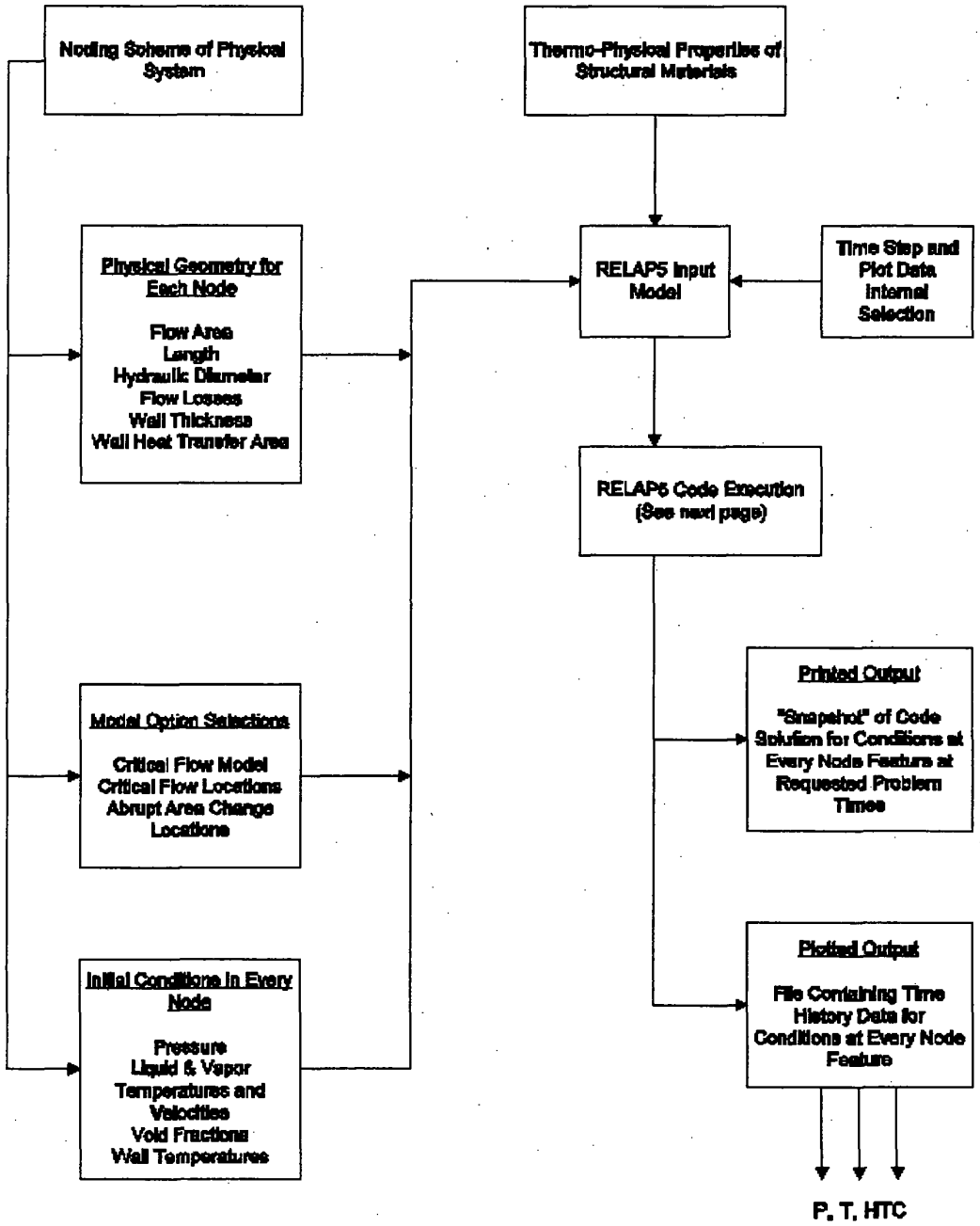


Figure 6.1. Schematic of RELAP5 Input and Output Processing

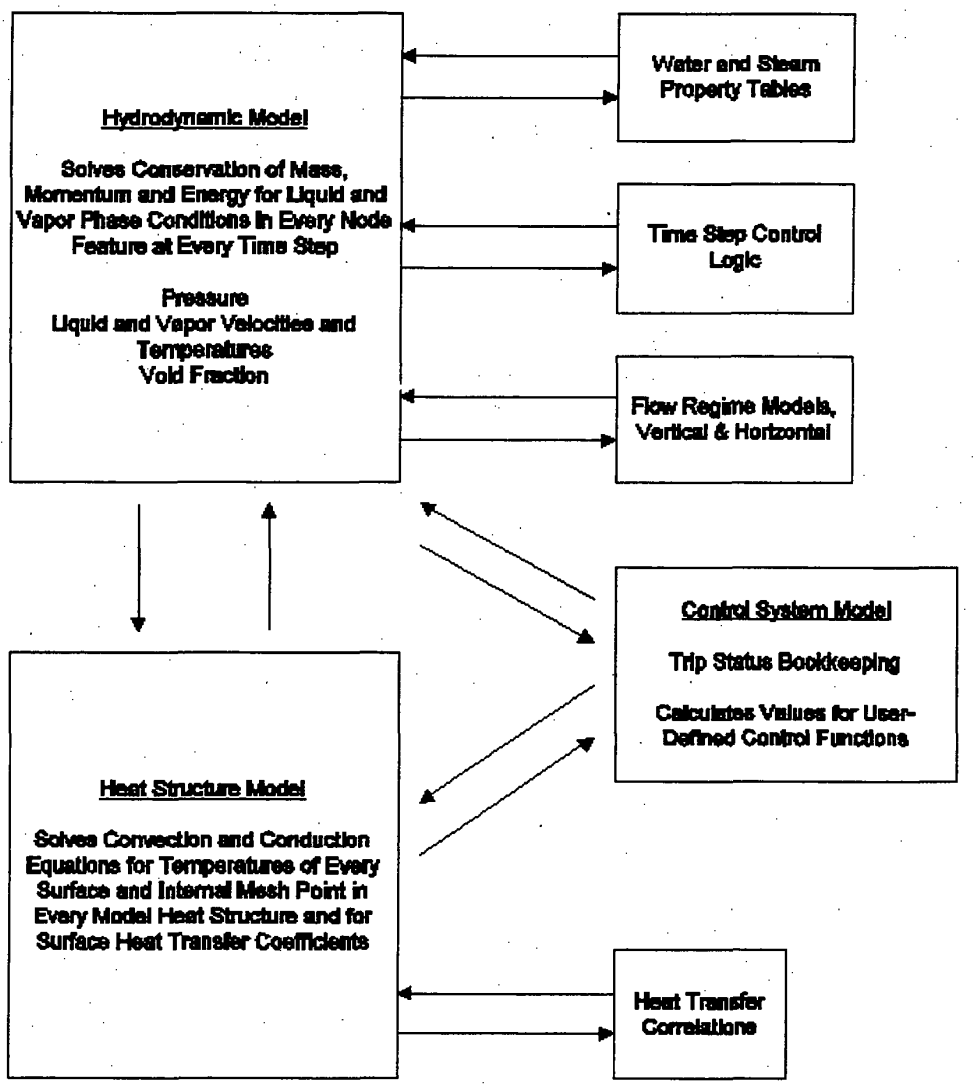


Figure 6.2. Schematic of RELAP5 Execution Processing

RELAP5 is based on a hydrodynamic model for single-phase and two-phase systems involving steam-water-noncondensible fluid mixtures in enclosed regions. The model is non-homogeneous (that is, the liquid and vapor phases at the same location may flow at different velocities) and non-equilibrium (that is, the liquid and vapor phases within the same region may exist at different temperatures).

The RELAP5 solution is based on a staggered-mesh arrangement in which the conditions representing the fluid state (pressures, temperatures, void fractions, etc.) are calculated at the center of each cell and the fluid flow behavior (liquid and vapor velocities and mass flow rates) is calculated at the junctions between the cells. The RELAP5-calculated behavior, therefore, represents flow of liquid and vapor from the center of one cell, through one-half

of the length of that cell to the interconnecting junction, and through one-half of the length of the adjacent cell to the center of the adjacent cell.

The flow through the cell regions of the flow path is subjected to the influence of losses attributable to wall friction, and the flow through the junctions may be subjected to the influence of losses attributable to the presence of irregular configurations, such as pipe bends, valves, and orifices. In addition, the model considers the effects of friction between the liquid and vapor phases.

Flow regime maps that provide characteristics for fluid behavior in vertical and horizontal cell orientations are used to determine the distribution of steam and liquid within each cell. This distribution is considered consistently throughout the RELAP5 model (for example, influencing interphase friction, liquid and steam velocities, condensation, and vaporization and fluid-to-wall heat transfer).

The RELAP5 heat structure model is used to represent the structures of the physical system, such as fuel rods, steam generator tubes, and piping walls. Heat structures may include the effects of internal heating, such as with fuel rods or electrically powered pressurizer heaters. Heat structures are connected to the fluid cells and may be "single-sided" (connecting to a fluid cell on only one side, for example when modeling a cold leg piping wall) or "two-sided" (connecting to fluid on both sides, for example, when modeling the passage of heat from the primary to secondary coolant system through the steam generator tubes).

RELAP5 calculates wall-to-fluid heat transfer on a consistent basis, with the heat transfer based on the wall surface temperature and the fluid conditions (pressure, temperatures, velocities) in the fluid cell connected to the wall. The flow of heat within the heat structure is based on the wall surface temperature and a solution of the one-dimensional conduction heat transfer equation. A wall heat transfer mode map (analogous to the flow regime map described above) is used to determine the fluid-to-wall heat transfer process based on the wall temperature and fluid conditions (pressure, steam and liquid temperatures, void fraction, steam and liquid velocities).

RELAP5 capabilities include trip and control functions that allow the system model to represent the functions of automatic and operator actions in a plant. Examples of these actions include reactor trips, feedwater termination, relief valve operation, reactor coolant pump trips, and initiation of emergency core coolant flows. The RELAP5 trip and control features are also particularly important because they provide great flexibility for linking the hydrodynamic and heat structure models together and using them for simulating transient events that realistically represent the expected behavior the prototype plant systems.

RELAP5 output, as portrayed on the right side of Figure 6.1, includes both printed and plotted output. The printed output consists of a snapshot of the RELAP5 solution for the conditions of every model feature at user-selected times during the transient calculation. The plotted output consists of a file containing the time histories of the calculated solutions for every condition in every model feature. The user specifies the data interval of the plotted output. For the PTS application, it is the RELAP5-calculated time histories for reactor vessel downcomer fluid temperature, pressure, and wall heat transfer coefficient that are passed to the fracture mechanics analysts for use as boundary conditions in their analyses.

The RELAP5 plant and code assessment calculations for the PTS project were performed consistently using the same version of the code, which is RELAP5/MOD3.2.2Gamma. Complete documentation regarding the RELAP5 code and its application is found the RELAP5 Code Manuals [RELAP, various citations].

6.3.2 RELAP5 Numerics Issues

Two potential RELAP5 problems related to unphysical flow circulations exist that are significant for PTS analysis. These problems are discussed as follows.

The first potential problem relates to circulations for plants with two cold legs per coolant loop during event sequences that result in complete stagnation of the coolant loops (LOCAs with break diameters larger than 2 inches). Potential flow networks exist for these plants, consisting of the two common cold legs and the steam generator outlet plenum on each coolant loop and the reactor vessel downcomer. Circulating flows within these networks have been observed in RELAP5 calculations during periods when cold ECC injection water is injected into the cold legs. The calculated solution initially becomes unstable, resulting in the onset of a continuous flow through the network (with forward flow through one of the cold legs and reverse flow through the other cold leg).

Recirculating cold leg flows are believed to be numerically initiated as a result of round-off error, although once initiated, physically based buoyancy forces are created that could sustain such flows. The data from certain MIST and APEX tests used in the RELAP5 assessments (discussed later) provide potential, but inconclusive, evidence of circulating flows in cold leg networks in the test facilities. If present, cold leg network flow increases the downcomer fluid temperature as a result of mixing of the ECC injection water before it enters the downcomer. Because cold leg network flow is nonconservative for PTS, and because it is not clear whether such flows are physical, large artificial reverse flow loss coefficients were added in the cold legs near the reactor coolant pumps in the Oconee and Palisades models used for the LOCA cases. These artificial flow loss coefficients prevent negative flow in either of the two cold legs, thereby preventing circulating flows within the cold network and ensuring a solution for PTS that is conservative in this respect.

The second potential problem relates to large circulating flows calculated by RELAP5 to exist within the reactor vessel downcomer region that are not physically realistic. As with cold leg network circulation (described above), downcomer circulations were noted for LOCA sequences with break diameters greater than 2-in. (5-cm). The source of the circulation was traced to the application of the RELAP5 momentum flux model within downcomer regions that are represented using two-dimensional nodalization schemes (in the axial and azimuthal directions). The root cause of this problem in the RELAP5 code has not yet been determined; however, it was found that deactivating momentum flux for the junctions within the downcomer region prevented these physically unrealistic circulating flows. As a result, momentum flux was deactivated in the downcomer regions of the plant models used for the LOCA cases.

6.4 Plant Model Development

For all three plants examined, the thermal-hydraulic analysis methodology is similar. For each plant, the best available RELAP5 input model was used as the starting point. For Oconee, the base model was that used in the code scaling, applicability, and uncertainty [CSAU] study. For Beaver Valley, the base model was the H.B. Robinson-2 model used in the original PTS study in the mid-1980s. This model was reviewed by Westinghouse and revised and updated based on the review comments to reflect the Beaver Valley plant configuration. For Palisades, the base model was obtained from CMS Energy Corporation, the operators of the Palisades plant. This model was originally developed and documented by Siemens Power Corporation to support analysis of the loss of electrical load event for Palisades. The RELAP5 models are detailed representations of the power plants and include all major components for both the primary and secondary plant systems. RELAP5 heat structures are used throughout the models to represent structures such as the fuel, vessel wall, vessel internals, and steam generator tubes. The reactor vessel nodalization includes the downcomer, lower plenum, core inlet, core, core bypass, upper plenum and upper head regions. Plant-specific design features, such as the Oconee reactor vessel vent valves, are included. To illustrate the model features and level of detail, a nodeing diagram for the Palisades plant is included in Figure 6.3, Figure 6.4, and Figure 6.5. The modeling approaches used for Oconee and Beaver Valley are similar.

The downcomer model used in each plant was revised to use a two-dimensional nodalization. This approach was used to capture the possible temperature variation in the downcomer resulting from the injection of cold ECCS water into each cold leg. Capturing this temperature variation in the downcomer is not possible with a one-dimensional downcomer nodalization. In the revised models, the downcomer is divided into six azimuthal regions for each plant. The reason for choosing six azimuthal regions is to match the geometry of the hot and cold legs around the circumference of the reactor vessel and so that water from each of the cold legs would flow into a separate downcomer node.

The safety injection systems modeled for the Oconee, Palisades, and Beaver Valley plants include high-pressure injection (HPI), low-pressure injection (LPI), other ECCS components (e.g., accumulators, core flood tanks (CFTs), and/or safety injection tanks (SITs), depending on the plant designation), and makeup/letdown as appropriate. The secondary coolant system models include steam generators, main and auxiliary/emergency feedwater, steam lines, safety valves, main steam isolation valves (as appropriate) and turbine bypass and stop valves. Each of the models was updated to reflect the current plant configuration, including updating system setpoints (to best estimate values) and modifying control logic to reflect current operating procedures. Other model changes include adding control blocks to calculate parameters for convenience or information only (e.g., items such as minimum downcomer temperature).

Detailed information regarding the specific individual RELAP5 input models for the Oconee, Beaver Valley and Palisades plants can be found in [***Arcieri-Base***].

The RELAP5 model does not include an explicit containment model. A volume held at constant atmospheric pressure is used to represent the containment. This approach was used for the simulation of adverse containment conditions during a main steam line break in the containment. In this situation, the reactor coolant pumps are tripped because of high containment pressure.

The RELAP5 analysis considers the increase in injection water temperature resulting from switchover of the ECCS suction from the refueling water storage tank (RWST) to the containment sump. This switchover occurs when the water inventory in the RWST is depleted as a result of the combined pumping of the ECCS and containment spray pumps. After switchover, the ECCS and containment sprays operate in a recirculation mode, taking suction from the containment sump. At the point of suction switchover, ECCS injection water temperature will increase from a typical range of 283 to 305 K (50 to 90°F) to 325 to 335 K (120 to 140°F) or higher. Increase in ECCS injection temperature resulting from switchover to the containment sump is modeled to reflect the change in ECCS injection temperatures.

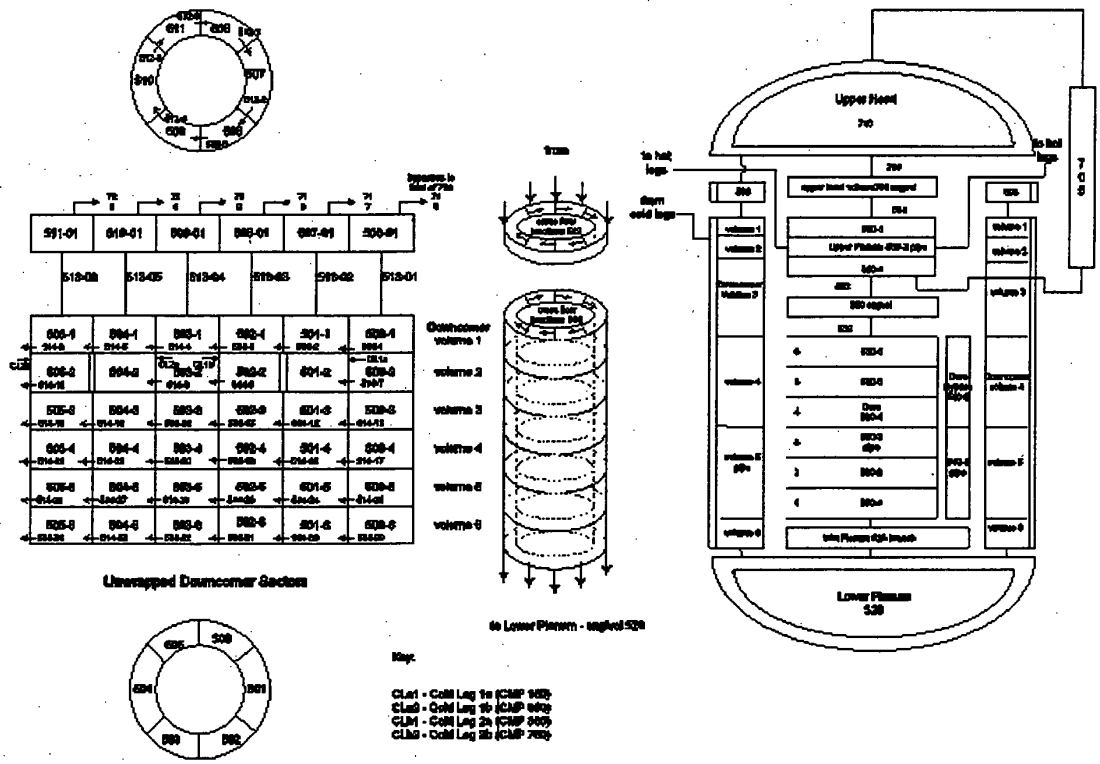


Figure 6.3. Palisades Reactor Vessel Nodalization

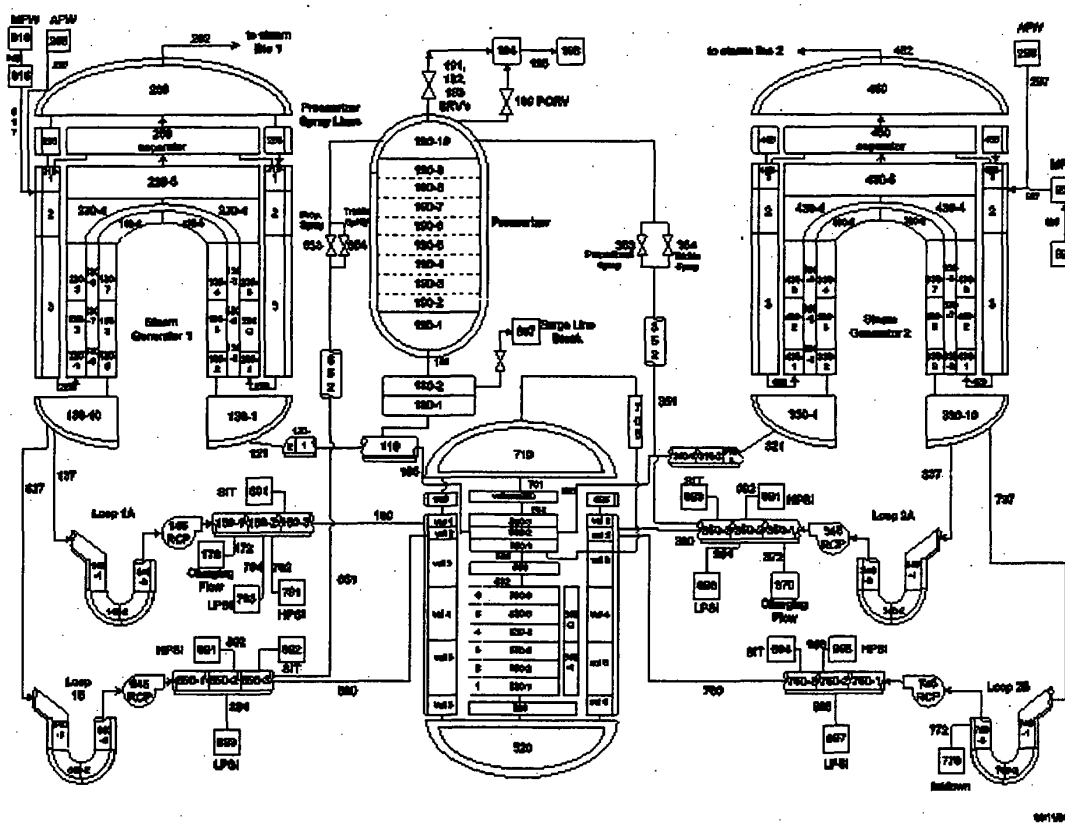


Figure 6.4. Palisades Coolant Loop Nodalization

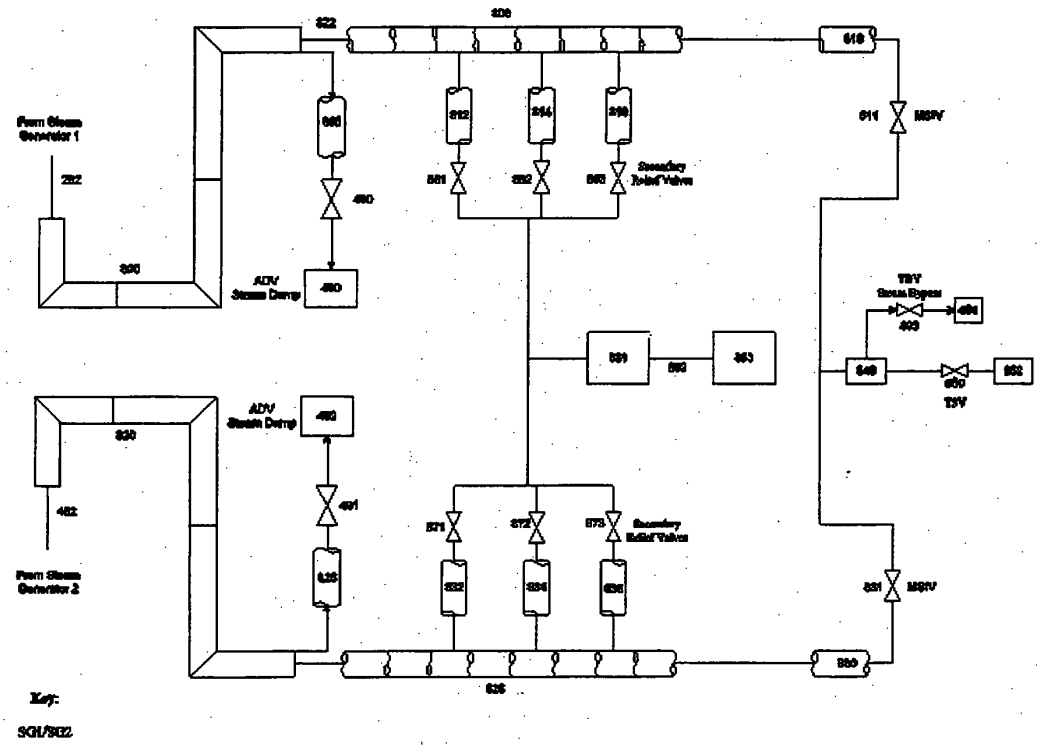


Figure 6.5. Palisades Main Steam System Nodalization

6.5 Transient Event Simulations

Transient events were selected for evaluation based on probabilistic risk assessment (PRA) analysis. Since each plant possesses unique thermal-hydraulic, hardware failure, and operational characteristics, there necessarily was variation in the transients events analyzed for the three plants. Examples of plant-to-plant differences important for PTS that affect transient selection include variations in shutoff heads for HPI pumps; initial pressure and temperature conditions in the accumulator (safety injection tank, core flood tanks); initial ECCS fluid temperature and allowed range; initial steam generator (SG) water masses; sizes and configurations of various valves and automatic controllers; and plant-specific operating procedures.

The development of the transient case list for each plant was an evolutionary process defined by the transient or sequence definition analysis. Generally, transients were selected based on the rate of primary system cooldown after transient initiation. Most transient event cases simulated generally fell into the categories of LOCAs and reactor/turbine trips with various complicating hardware and operator failures. Scenarios that consider stuck-open relief valves that either remain open or subsequently reclose later in the transient, system failures that cause steam generator overfeed, main steam line breaks, and others were analyzed. Evaluations were also performed for other types of events, such as steam generator tube rupture, recovery from a loss-of-all-feedwater event, and feed-and-bleed recovery from a LOCA with HPI failure.

The transient event simulations were run as RELAP5 restart calculations beginning from steady plant operating conditions. Total simulation time is 15,000 seconds for Palisades and Beaver Valley. For Oconee, the total simulation time is 10,000 seconds. .

6.5.1 Loss of Coolant Accidents

The smallest LOCA break size evaluated was 1.0-in. (2.54-cm) in diameter. Larger break diameters were also evaluated where the break flow area was progressively doubled, up to 22.63-in. (57.47-cm) in diameter. Break diameters considered in the analysis, therefore, range over the full break spectrum. The breaks for most LOCA cases are assumed to be on the hot side of the reactor coolant system (in the pressurizer surge line for smaller breaks and in the hot leg for larger breaks). The hot leg break location was selected for most evaluations because it results in the greatest reactor coolant system cooldown rate, an intentionally conservative treatment. The ECCS injection rates are also maximized in this situation. Evaluation of cold leg break LOCAs was also performed.

For all LOCA cases, the discharge and flow loss coefficients used for break junctions are assumed to be equivalent to those used in AP600 work. While these coefficients may not be appropriate for a specific break, the wide spectrum of break diameters accounts for any uncertainties in loss coefficients.

6.5.2 Reactor/Turbine Trips

The majority of cases analyzed are initiated by a reactor/turbine trip followed by various primary or secondary side failures. These failures include relief valve failures, steam generator level control failures, and others. In the RELAP5 model for all cases, a reactor trip is considered the same as a turbine trip. In reality, if a reactor trip signal is generated, there is a small delay before a turbine trip is generated. Since the long-term downcomer temperature and pressure are of interest, this delay is considered negligible. There are numerous cases where stuck-open valves (pressurizer or steam generator PORVs, safety relief valves, etc.) are modeled as failures following a reactor/turbine trip. In these cases, the valve is assumed to spuriously open at transient initiation. Primary side stuck valves (pressurizer SRVs or PORVs) are similar to LOCAs where the "break" is located at the top of the pressurizer, rather than

in the surge line, hot leg, or cold leg. In most cases, the RELAP5 models use a single valve component to model several valves in parallel. For example, in Beaver Valley, three pressurizer PORVs are modeled with a single RELAP5 valve component. In order to have a single PORV fail by sticking open, the RELAP5 valve component is opened to one-third of the full flow area.

In a number of cases, the valve that stuck open was assumed to reclose at some later time. The time of reclosure was defined as either 3,000 seconds or 6,000 seconds depending on the transient definition from the PRA analysis. (Occasionally, a different time was chosen.) Various times were chosen since it would not be known when the valve would reclose (if it were to reclose). The 6,000-second reclosure time was selected as a point far enough out in time where the primary pressure and temperature reached a minimum.

Another set of failures is overfeeding of the steam generators. As with other cases, the initiating event is the reactor/turbine trip. These cases will result in an overcooling event. The failure could be anything from equipment/component failure to control failure or operator error. Cases have been run where a single steam generator is filled to the top, and the water level is maintained at that level. There are cases where multiple steam generators are filled to the top. Cases were run where the steam generator was filled to the top, then feedwater was stopped and the steam generator was allowed to boil dry.

6.5.3 Main Steam Line Break

Main steam line break cases were selected because they cause rapid depressurization of the steam generator. This rapid depressurization is one of the most limiting overcooling transients from a single failure on the secondary side. Large breaks considered were modeled as double-ended guillotine breaks. These breaks were assumed to occur at the connection of the steam line to the steam generator (upstream of the main steam isolation valves). Smaller steam line breaks were simulated with stuck secondary side valves (SRVs, ADVs, etc.) Turbine bypass valves were also assumed to stick open. In plants with main steam isolation valves, some of these stuck valves (breaks) were isolated by the MSIVs.

6.5.4 Operator Actions

Various operator actions are considered in the RELAP5 analyses based on the transient definition from the PRA analysis. For cases involving a primary system LOCA, the operator is assumed to take no action since automatic systems are presumed to operate and provide the core and primary system cooling. In these situations, the primary operator function is to monitor system conditions. For various transients involving reactor/turbine trips combined with component failures that lead to primary system overcooling, operator actions are a major factor and were modeled. Generally, the two categories of operator actions considered are (1) the operator correctly diagnoses the plant situation and performs the correct actions based on the emergency operating procedures, and (2) the operator fails to correctly diagnose the situation or takes an incorrect action.

A significant operator action for the plants analyzed is HPI control/throttling. Depending on the transient scenario, continued HPI injection can cause the system to refill and repressurize to the HPI pump shutoff pressure and/or the pressurizer PORV opening setpoint pressure. A good example of a transient where system repressurization can occur is a stuck-open primary safety valve that recloses after the system has depressurized. Continued HPI will cause the primary system to repressurize in this case unless the operator recognizes that the faulted valve has reclosed and takes action to control HPI injection.

Different plants have different HPI control methods. In Oconee, the operator can throttle HPI flow to obtain a desired flow rate and maintain a certain pressurizer water level. In Beaver Valley, however, the operator can either have a pump running or not. There is no "throttling"; rather, pumps are turned off if conditions are met. In Palisades, the operator

can throttle HPI if auxiliary feedwater is operating with the steam generator wide-range level greater than -84% and the reactor coolant system subcooling greater than 13.9 K (25°F). In this case, HPI is throttled to maintain pressurizer level between 40 and 60%.
HPI control is a crucial component in the overall PTS risk. An event where there is no HPI control can produce a much greater challenge to vessel integrity because of primary system repressurization than would the same event with HPI control because system repressurization does not occur. One significant variable in the HPI control is operator timing. Since the time that the operator will take control of the HPI is variable depending on the transient situation, several times are analyzed based on PRA input to determine the variation in overall system (downcomer) conditions. As an example, for Beaver Valley, cases were run where the operator does not control HPI, controls HPI 1 minute after the criteria for control are met, and controls HPI 10 minutes after the criteria are met.

Another example of an operator action is control of the reactor coolant pumps. The different plants use different criteria for tripping the RCP. At Oconee, the operator is assumed to trip the RCPs on low subcooling. At Beaver Valley, the RCP trip criterion is based on the difference between steam generator and pressurizer pressures. At Palisades, RCP trip criteria are based on primary system pressure and subcooling margin. In some events, the RCPs were not predicted to trip; however, various operating procedures could have caused the operators to trip the pumps. Therefore, in some cases, the RCPs were set to trip as an operator action. An additional note about RCPs is that they will be tripped if there are adverse containment conditions (i.e., main steam line break). Since the RELAP5 models used do not include the containment, the pumps were tripped manually if it was deemed necessary.

Failure of the operator to correctly diagnose the situation and take the correct action was also considered in the transient analysis. Failure to isolate the auxiliary/emergency feedwater to a faulted steam generator during a steam line break is an example of an operator failure considered in this analysis. This failure will result in an overcooling event where the faulted generator continues to remove heat, thus lowering the primary temperature. Timing of operator action was also analyzed. As an example, analyses were performed assuming that the operator stops AFW/EFW to the faulted generator (at 30 minutes for Beaver Valley). Time of operator action was determined by PRA analysis.

6.6 RELAP5 Analysis Results

The parameters that are used in the probabilistic fracture mechanics analysis are the reactor vessel downcomer fluid temperature, primary system pressure and reactor vessel wall heat transfer coefficient as a function of transient time. Post-processing of the RELAP5 results is performed to generate files that are transmitted to ORNL for analysis. Averaged values for the downcomer fluid temperature, system pressure, and downcomer fluid to vessel wall heat transfer coefficient were provided.

A large number of cases were analyzed for the Oconee, Beaver Valley, and Palisades plants to meet the requirements of the PRA analysis. A total of 177 cases were run for Oconee, 67 cases for Palisades, and 130 cases for Beaver Valley. These cases were needed to support the PRA model, particularly to support the development of transient bins needed to categorize the large number of transients that must be considered in developing a nuclear plant risk model. Because of the large number of cases, the results that are used in the probabilistic fracture mechanics analysis are separately presented in [***Arcieri-Base***].

6.7 RELAP5 Assessment Against Experimental Data

Assessments are performed to establish the suitability of the RELAP5/Mod 3.2.2Gamma code for analyzing plant transients that are significant risk contributors for PTS. The RELAP5 code version used for the assessment calculations is the same that is used for the PTS plant calculations. Assessment principally