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Class 3

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H.B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK RISK STUDY

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H. B. Robinson Unit 2

Pressurized Thermal Shock

Risk Study

WCAP-10318

EXECUTIVE SUMMARY

Methodology

A Pressurized Thermal Shock (PTS) Risk Study was performed for H. B. Robinson Unit 2 (HBR). This study made extensive use of the methodology developed by Westinghouse for the Westinghouse Owner's Group (WOG) for a generic evaluation of PTS [1, 2, 4] that was previously accepted by the NRC. The methodology and results of the WOG generic program were applied in combination with considerations applicable to the H. B. Robinson vessel in this plant specific study. The major elements of the study are:

1. Determination of the likelihood of transients of interest for PTS.

2. Characterization of the severity of these events with regard to the pressure and degree of cooling.

3. Determination of the stresses, vessel toughness and the resultant probability of a flaw growing to a significant depth in the vessel wall.

4. Combination of the above results to obtain the frequency of significant flaw extension.

Scope of Work

To accomplish these tasks, generic event trees and generic thermal hydraulic transient analyses for three main contributors to pressurized thermal shock were used. Plant specific analyses of fluid mixing and probabilistic fracture mechanics were completed to identify the main contributors to risk. Further deterministic fracture mechanics evaluations better quantified the risk resulting from the main contributors.

This approach minimized the number of plant specific analyses carried out in Items 1, 2, and 3 described above in order to quickly assess the PTS Risk for HBR. The steps identified above result in an HBR PTS Risk curve that can be compared to previous NRC generic evaluations. The parameter used for this purpose is a measure of the reactor vessel toughness called RT_{NDT} (for reference nil-ductility transition temperature).

The study was conducted so that risk results could be directly compared to those associated with the NRC proposed screening criterion: 270°F upper bound surface RT_{NDT} for axial welds, and national safety goals: one large scale core melt event per 10,000 reactor years of operation (10^{-4}/R-Yr). (No correlation in terms of risk can be made to the current NRC RT_{NDT} screening value of 300°F set for circumferential welds, which is applicable to the HBR weld of interest, since the NRC did not perform a risk evaluation to substantiate this value).

It is anticipated that the uncertainties inherent in the H. B Robinson PTS risk study are similar to those contained in the generic NRC study (SECY-82-465[3]) since both studies were done on a consistent basis. Therefore, any limitations inherent in the HBR risk analysis are comparable to those associated with the generic NRC results.

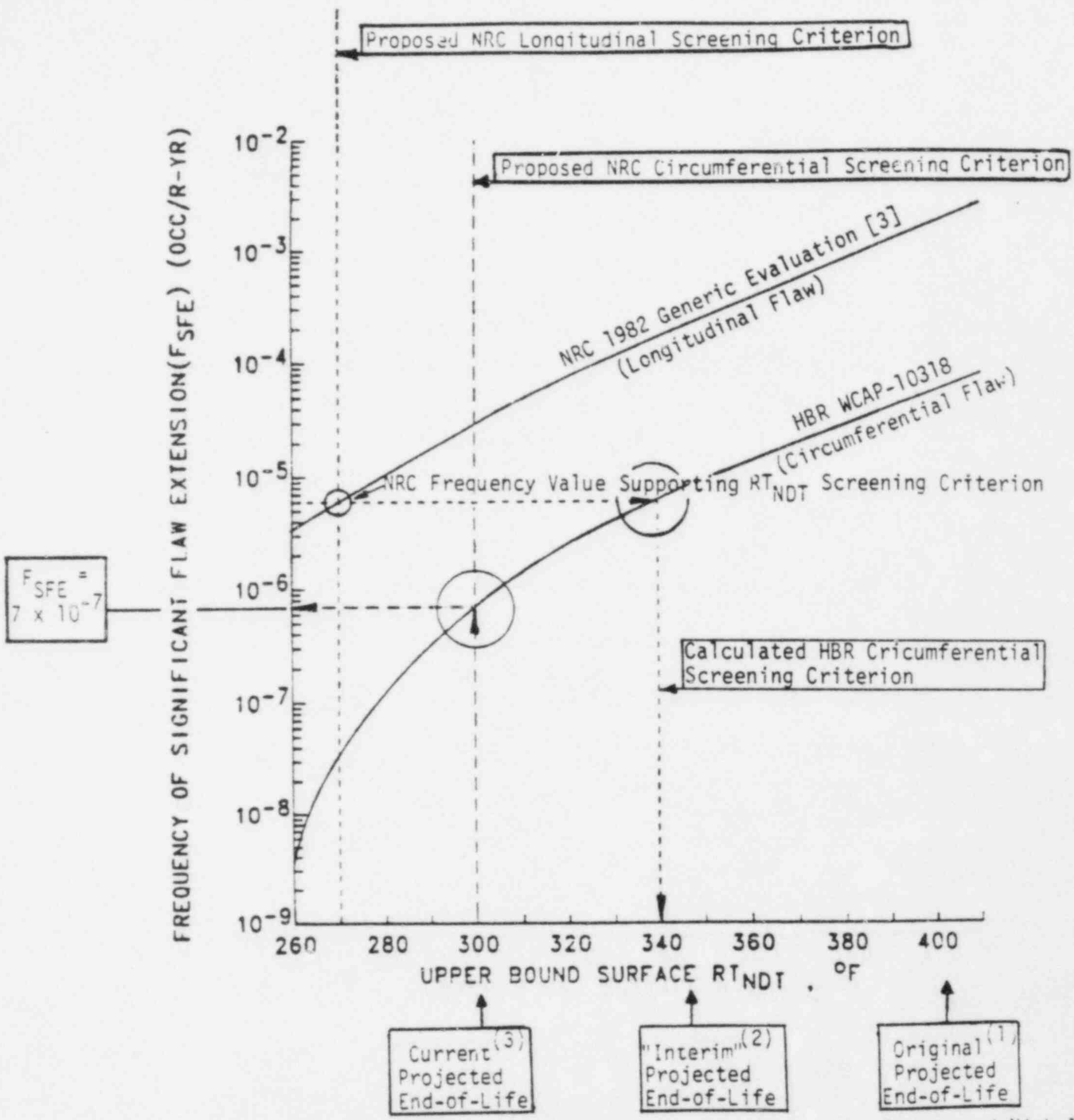
Results and Conclusions

These comparisons have resulted in the following conclusions about H. B. Robinson PTS risk:

- o The screening criterion calculated for H. B. Robinson that compares to the NRC generically determined screening criterion was calculated to be 340°F as shown in Figure 1.

- o The single major contribution to this improvement is the circumferential versus axial orientation for the welds of interest. (Note that only one circumferential weld at HBR was predicted to exceed the NRC's screening criteria prior to the present flux reduction program.)

FIGURE 1 DETERMINATION OF TOTAL RISK OF FLAW EXTENSION FROM PRESSURIZED THERMAL SHOCK OF THE H.B. ROBINSON UNIT 2 REACTOR VESSEL



(1) Based on original core configuration with "upper bound" Copper and Nickel.

(2) Based on currently installed "Low Leakage Core" with "upper bound" Copper and Nickel, not including additional planned reduction.

(3) Based on currently installed "Low Leakage Core" and planned flux reduction using "Part Length Shielding" with assumed "upper bound" Copper and Nickel.

- This result demonstrates the considerable conservatism in the NRC proposed screening criterion of 300°F RT_{NDT} for circumferential welds since the NRC proposed value of 300°F was based on limited calculational results and the HBR value is based on more extensive results.

- The frequency of significant flaw extension beyond 75% of the reactor vessel wall for H. B. Robinson is determined to be 7×10^{-7} occurrences per reactor year at the end of plant life with current flux reductions and planned part length fuel assembly shielding. To determine the HBR PTS risk for comparison with the national safety guideline for large scale core melt, other factors must be considered. These include the probability that the flaw will continue to extend entirely through the vessel wall, the probability that the resultant breach of the vessel wall will prevent adequate core cooling, and the probability that core melt will result. Assuming a value of one for these other factors (i.e., saying that they are 100 percent probable), the HBR PTS risk is still more than a factor of 100 lower than the large scale core melt safety guideline [6], i.e., 7×10^{-7}/R-Yr versus 10^{-4}/R-Yr.

<u>CP&L Actions to Reduce the Severity of PTS to HBR</u>

The plant specific study includes consideration of the following actions Carolina Power & Light Company has taken to significantly reduce the PTS risk:

- use of part length fuel assembly shielding to reduce neutron fluence on the critical weld. This weld should not exceed the screening criteria based on the current estimating technique [3]

- extensive operator training on PTS type transients and revision of plant operating procedures. These procedures, which are being revised, are based on the WOG developed emergency response guidelines

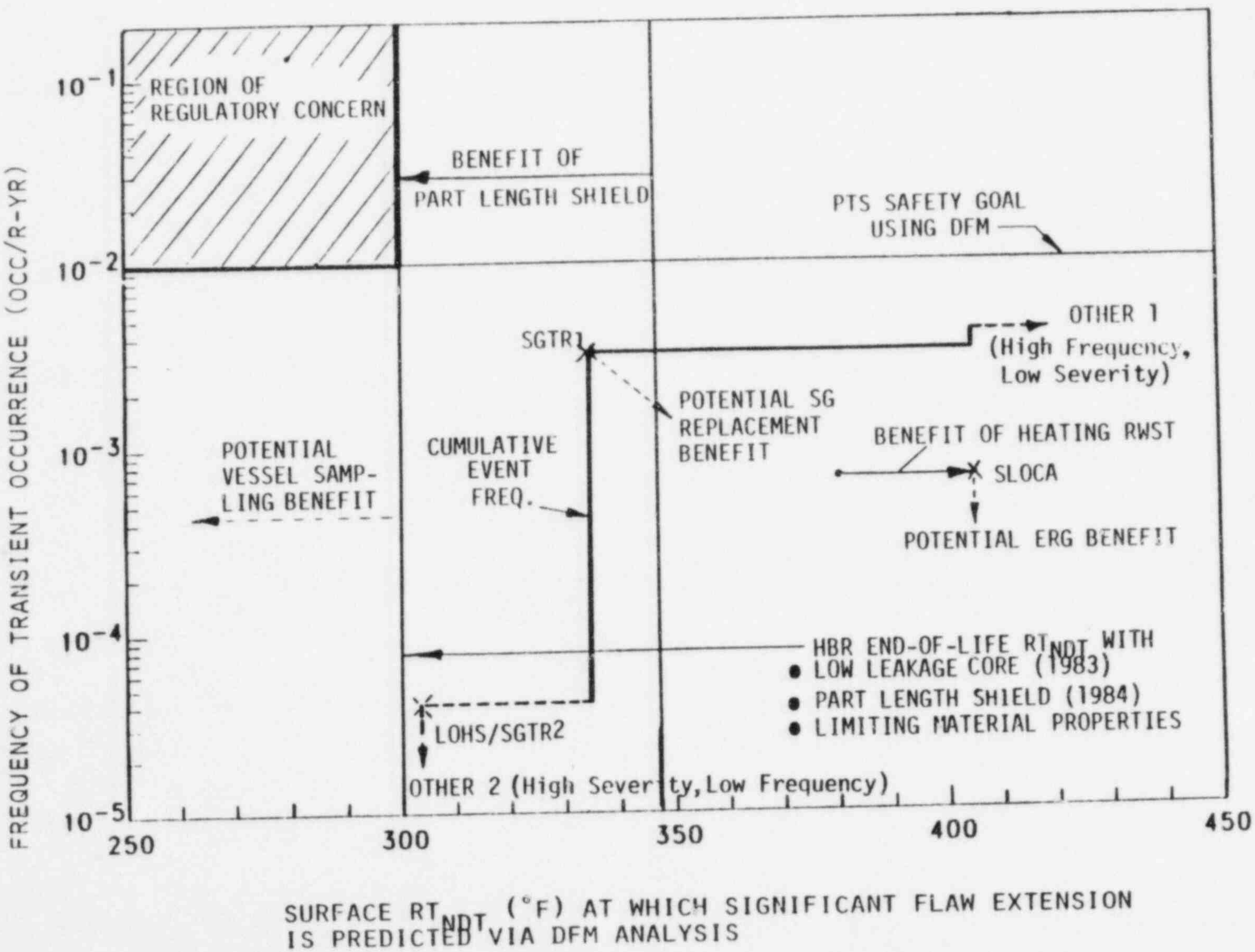
- RWST tank is being heated to increase the temperature of injected water to further reduce the effect of PTS transients

- reactor vessel sampling program to determine the actual copper and nickel content of the critical reactor vessel weld. The actual chemistry could reduce the estimated end-of-life RT_{NDT} to less than 300°F.

In order to provide additional insight relative to those PTS events that are most important, the potential benefit of these actions, as well as the steam generator replacement program, is shown on Figure 2. Figure 2 portrays the results of deterministic analyses against a frequency of event occurrence goal. This goal is different from that identified in Figure 1 because Figure 1 portrays the results of probabilistic analyses against a frequency of significant flaw extension goal. Both approaches, along with the respective goals, were used by the NRC in the development of the RT_{NDT} screening criteria for PTS.

These and other programs carried out by Carolina Power & Light Company have increased and will further increase the margins of safety for H.B. Robinson. In addition, the analyses demonstrate that even without such initiatives the H. B. Robinson PTS risk at end-of-life RT_{NDT} is equivalent to the NRC probabilistic safety criteria used to substantiate the proposed screening criteria.

FIGURE 2 H.B. ROBINSON- UNIT 2 FREQUENCY OF TRANSIENT OCCURRENCE LEADING TO SIGNIFICANT FLAW EXTENSION USING DETERMINISTIC FRACTURE MECHANICS ANALYSIS WITH CP&L INITIATIVES



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GLOSSARY OF ABBREVIATIONS AND TERMS

ABBREVIATIONS

AFW	-	Auxiliary Feedwater
ATWS	-	Anticipated Transient Without Trip
β	-	Stylized Transient Cooldown Time Constant (MIN^{-1})
BFREQ	-	Frequency of the BIN (OCC/R-YR)
CP&L	-	Carolina Power & Light Company
dPa	-	Displacements Per Atom
DFM	-	Deterministic Fracture Mechanics
ERG	-	Emergency Response Guidelines
ETA	-	Event Tree Analysis
EXFW	-	Excessive Feedwater
F_{SFE}	-	Frequency (or Risk) of Significant Flaw Extension
HBR	-	H.B. Robinson Unit 2
LEFM	-	Linear Elastic Fracture Mechanics
LOCA	-	Loss of Coolant Accident
LOHS	-	Loss of Heat Sink
MSIV	-	Main Steam Isolation Valve
P	-	Characteristic Pressure (PSIA)
PFM	-	Probabilistic Fracture Mechanics
PORV	-	Power Operates Relief Valve
PRA	-	Probabilistic Risk Assessment
PTS	-	Pressurized Thermal Shock
PWR	-	Pressurized Water Reactor
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RWST	-	Refueling Water Storage Tank
RT_{NDT}	-	Reference Nil-Ductility Transition Temperature
SD	-	Secondary Depressurization
SG	-	Steam Generator
SI	-	Safety Injection
SLOCA	-	Small LOCA
STGR	-	Steam Generator Tube Rupture
TAVG	-	RCS Hot and Cold Leg Average Temperature
T_f	-	Stylized Transient Final Temperature (°F)
T&H	-	Thermal-Hydraulic
W	-	Westinghouse
WOG	-	Westinghouse Owners Group
WCAP	-	Westinghouse Technical Publication Serial Number

GLOSSARY OF ABBREVIATIONS AND TERMS

TERMS

PTS Risk	-	PTS risk is defined to be the frequency of occurrence of significant crack extension through the vessel wall. The risk may be specified to be associated with a specific PTS Bin or Category, or it may be a total risk due to all PTS transients. PTS risk is expressed in terms of occurrences per reactor year (OCC/R-YR)..
PTS Category	-	A group of cooldown scenarios which are caused by a common mechanism such as a rupture of the Primary System Boundary. LOCA, SGTR, and SD are examples of PTS categories. A frequency can be associated with each category.
PTS Scenario	-	A cooldown transient which is described by a specific discrete set of parameters (e.g., A small steamline break caused by a .11 Ft^2 rupture which occurs at a decay heat of .05 percent of rated power,..., etc.,).
PTS BIN	-	A bin describes a subset of PTS scenarios within a single PTS category. The bin is defined by ranges of parameters that are particularly significant to PTS risk. The arithmetic sum of the bin Frequencies of a certain category equals the category frequency.
Stylized Transient Characteristics	-	A cooldown event can be approximated by an exponential cooldown. The final temperature (FTEMP) and time constant (BETA) of the exponential represent two of the three stylized transient characteristics. The system pressure during the period of crack extension (CPRESS) represents the third.
RT_{NDT}	-	Reference nil-ductility transition temperature is a measure of the temperature at which the vessel material undergoes a transition from ductile to non-ductile behavior. This parameter varies as a function of vessel life, fluence, and material properties, and therefore can be a measure of vessel age for the PTS issue.

SECTION I

INTRODUCTION

I.1 PURPOSE OF THE PLANT SPECIFIC PTS RISK STUDY

During 1982, Westinghouse, supported by the Westinghouse Owners Group (WOG), developed a methodology for the probabilistic assessment of risk from Pressurized Thermal Shock (PTS) [1]* and [2]. The methodology was used to quantify PTS risk on a typical Westinghouse pressurized water reactor. Much of the information that was generated in the WOG effort was used by the U.S. Nuclear Regulatory Commission in developing a position on the PTS issue [3].

The purpose of the H. B. Robinson plant specific Risk Study is 1) to use the above methodology, along with the results from the WOG programs, in combination with plant specific refinements applicable to the H.B. Robinson Unit 2 reactor vessel and plant system in a plant specific study of PTS risk, 2) to determine if the plant specific risk is less than the NRC risk determined on a generic basis for PTS [3], and 3) provide support for a higher acceptable RT_{NDT} value for H. B. Robinson than that which has been established on a generic basis. Another objective is to show that the margin of safety in terms of risk is clearly adequate for demonstrating continued safe plant operation as a result of the combined effect from all Carolina Power & Light Company programs related to PTS.

I.2 OVERVIEW OF THE REPORT

This report has been written to facilitate two types of assessment: executive and technical. As used here, "executive assessment" is an evaluation of the method and results without the burden of technical detail and is provided for in the main body of the report. "technical assessment" is an in depth

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* Numbers in square brackets refer to references listed in Section VIII of this document.

evaluation of the technical details to verify the approach and the appropriateness of the conclusions. It is provided for in the Appendices.

This report is organized as follows:

- The remaining paragraphs in Section I provide background and an explanation of the use of probabilistic methods in the assessment of reactor vessel PTS.

- Section II defines the scope of this PTS risk study.

- Section III outlines the general PTS risk assessment methodology and Appendix A provides a more detailed description of the methodology.

- Section IV provides a general description of the application of the approach to assessing the integrity of the H.B. Robinson Unit 2 reactor vessel. Appendix B provides the detailed description of the application.

- Section V presents a summary of the results from the risk assessment as well as a discussion of the margins and uncertainties inherent in the analysis.

- Section VI presents the conclusions and recommendations.

- Appendix C presents a sensitivity study on the effect of circumferential variation in neutron flux on flaw shape change during flaw growth. It demonstrates the conservatism inherent in the use of a semi-infinite, continuous flaw for crack arrest as used in this risk study.

- Appendix D gives the results of a deterministic fracture mechanics evaluation of a steam generator tube rupture which is one of the events that dominates risk resulting from PTS. The evaluation given in Appendix D was used in the detailed narrow scope risk analysis given in Section IV and Section 8 of Appendix B. The detailed narrow

scope risk analysis given in Section IV and Section 8 of Appendix B also includes an integration of the benefits from Carolina Power & Light Company programs related to PTS. These results quantify the margin of safety inherent in the continued future operation of the plant.

I.3 DISCUSSION OF THE USE OF STATISTICAL METHODS IN RISK ASSESSMENT FOR PTS

Risk assessment of Pressurized Thermal Shock (PTS) is a newly developed application of probabilistic risk assessment (PRA) methodologies. Risk assessment techniques in general, such as PRA, have been applied in a variety of fields and are increasing rapidly in importance and diversity of application. The technology however, is also subject to misunderstanding of how it is applied to aid decision making. For these reasons the following sections are provided to describe the technique and to help clarify the application.

I.3.1 BASIS FOR UTILIZING STATISTICAL METHODS

The approach to evaluating pressurized thermal shock for nuclear reactor systems has evolved to include technologies based on mathematical methods that utilize probabilistic and statistical theories. Prior to 1982, such methods were primarily "deterministic" and based on standard design procedures which consider only the severity of events. The word "deterministic" here is used to distinguish it from a "statistical" approach that considers event frequencies as well as resulting severities. This section addresses the importance of this distinction and the reasons why incorporation of statistical methods is an advancement in technology.

Standard design procedures use a deterministic approach whereby maximum loads are defined and used in the design calculation without consideration of the likelihood of the event. However, literally thousands of events and sequences of events could occur. Some of these events may lead to a condition of concern but most do not. Some are more likely to occur than others. A deterministic approach cannot realistically account for all of these variables. For this reason a means of evaluating the importance of events in

terms of the likelihood of occurrence as well as the resulting severity should be used. The benefit of using the statistical or probabilistic method is that the important events can be identified through consideration of the likelihood of occurrence in combination with severity. The most important events can then be addressed minimizing the effort expended on events of much less relative importance. This capability is important in light of the type of event sequences being considered for the PTS issue. That is, these sequences include simultaneous consideration of a wide range of control system operation, operator action time, status of plant operation and many other variables.

I.3.2 DISCUSSION OF THE USE OF STATISTICAL METHODS

This section describes the importance of statistical methods, why they are more suitable than a solely deterministic approach and why they were required to help address an issue that could not be satisfactorily resolved by standard methods.

These methods are not called upon to solve the problem in an absolute sense but were found to be required to put all the elements into perspective and to prioritize the thousands of pertinent quantities and the associated variabilities: to identify the most important parameters and to find what parameter variations would cause the most significant effects.

This allows the industry to address the important variables rather than ones that have little affect. The method minimizes the number of parameters and variations to consider only the most important and does it with great efficiency. This not only makes the best use of resources but saves years of evaluation effort that would be required to look at each concern one by one and to determine what to do for each. This allows the whole PTS issue to be addressed quickly and avoids years of delay in improving or verifying the safety of the operating plants. It allows appropriate and effective decisions to be made by the NRC, the utilities and the industry by identifying the highest priority concerns and the proposed modifications. By incorporating these probabilistic methods into the analysis procedure, the proposed modifications can be readily evaluated for their effectiveness and overall

impact. This again improves safety by enabling the utilities to take the most effective action when required and by evaluating the effect of the actions on the entire system.

Example of Evaluation of Parametric Variation

The reason that these probabilistic or statistical methods are so effective for example, in the area of fracture mechanics analysis, is that they take into account the variation of parameters in a way that indicates how sensitive the results are to the variations. This can be visualized in terms of three levels of sensitivities where 1) slight variations causes a large change in the results, 2) a very large variation does not affect the results and 3) some moderate level of variation causes some moderate change in the results. Situation described by the first and second cases are "black and white" and are usually readily identifiable. However, the third case is by far the more important case where "shades of gray" need to be evaluated and the "relative grayness" can be prioritized. Once the priorities are set, that is, once the importance or lack of importance of the variation in a parameter is identified, then the proper action to take is much more obvious and it can be taken with great effectiveness.

Example of Evaluation of Event Sequence

Another benefit that can be obtained through use of this method is the determination of event frequencies. When a known result (called an "end state") is identified, the sequence of events required to reach it from some starting condition must be evaluated. This evaluation determines 1) what sequence of events is required, 2) if the end state can actually be reached, and 3) what the frequency of occurrence is, knowing all the possible alternative sequences. The likelihood of a sequence of events to occur over the lifetime of the plant can be determined, even when no end state of the type being considered has actually occurred.

Overall, the capability to consider thousands of variables, to determine the sensitivity of variations in the variables, to prioritize the importance of variables, to evaluate the effectiveness of modifications, and to determine

the frequency of sequences of events makes this method a powerful tool for decision making that few other analytical methods can provide. When the probabilistic methods are combined with deterministic evaluations, reliable and precise conclusions can be made and solutions can be provided where needed.

SECTION II

SCOPE AND INTENT OF THE PLANT SPECIFIC PTS RISK STUDY

II.1 GENERAL REVIEW

As was noted previously the purpose of the H. B. Robinson PTS risk study is to quantify the margin of safety for the H. B. Robinson plant. In addition, this study is to provide a realistic assessment of the components of the PTS risk envelope and to provide additional insights into the benefits of the implemented and planned CP&L programs aimed at eliminating pressurized thermal shock as a safety issue for the H. B. Robinson plant. In order to accomplish this it was recognized that the plant specific study draws on previous and recent WOG PTS studies [1,2,4]. These studies are the most complete studies to date, both with respect to methodology and results. This study is sufficiently realistic and includes appropriate levels of conservatism to adequately reflect the plant specific application. Figure II.1-1 shows the relation of the HBR plant specific study to the current generic risk analyses.

II.2 DEVELOPMENT OF THE SCOPE OF STUDY

The objectives of this study are:

- o Provide a conservative H. B. Robinson plant specific evaluation of PTS risk
- o Utilize a methodology consistent with previous WOG and NRC efforts.

In order to accomplish these objectives in a timely and cost effective manner the scope of the study was reviewed to identify the appropriate level of detail required. In addition, consistent with previous WOG and NRC efforts and the NRC Policy Issue SECY-82-465 [3], the study contains plant specific considerations and utilizes probabilistical methods. The study also treats fracture mechanics considerations in a way that is consistent with the development of event (end state) frequencies. The use of probabilistic fracture mechanics (PFM) analysis provides this consistency and is a natural and realistic means of assessing sensitivity to parameter variations.

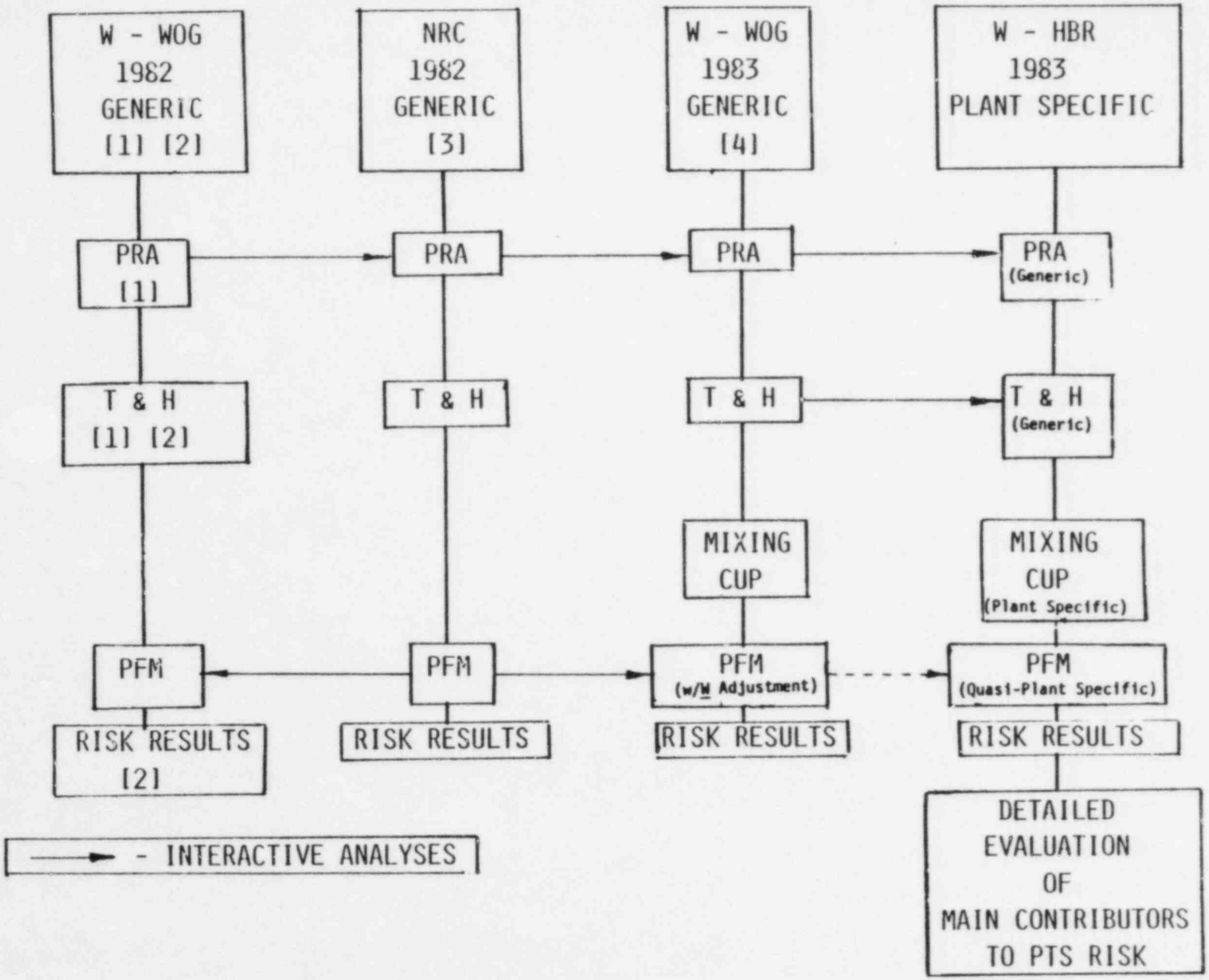


FIGURE II.1-1: RELATIONSHIP OF HB ROBINSON RISK STUDY TO CURRENT GENERIC PTS EVALUATIONS.

Since the general methodology has been developed by the NRC and WOG and since PTS risk results were available from a generic plant of the same vendor, two considerations were evaluated to determine the level of plant specific detail required in this study:

- o To what degree is the plant type, detailed safety and control systems, emergency procedures and other important aspects similar to or different from those used in the generic PTS risk study?

- o To what extent are the proposed NRC screening values in terms of RT_{NDT} met during the service life of the plant and thus how extensive are the analyses used to justify margins of safety based on the probabilistic results?

Given the above, a plant that has significant differences in safety systems would require a very detailed plant specific analysis. Such an analysis would probably make more minimal use of the generic study results. In addition, a plant attempting to justify continued plant operation at an RT_{NDT} significantly above the NRC screening criteria may also need to justify a very detailed plant specific analysis.

However, as is the case for H. B. Robinson, a plant that utilizes a PTS risk study to demonstrate existing margins of safety, and which is similar to the generic plant system and operation, should be able to draw much more heavily on the generic study results. H. B. Robinson meets this condition as follows:

1. The H. B. Robinson plant is a Westinghouse three loop design with a configuration, safety and control system configurations and operating procedures that are similar to the generic plant model used in the generic PTS risk study. The original study was carried out utilizing a majority of the information and analysis for a three loop plant.

2. The core modifications, including partial length shielding and modified low leakage core loading, will provide significantly reduced vessel material embrittlement. Specifically, it is antipicated that the end of life RT_{NDT} will be near the NRC current screening value of 300°F for circumferential flaws.

3. CP&L has carried out other programs (beyond the flux rate minimization program identified in the above) that are discussed in detail in Section IV and Appendix B, Section 8. These range from heating the RWST for providing preheated safety injection fluid, typically in excess of 90°F, to vessel sampling and search programs to identify conservatisms in the assumed material properties. Such programs provide additional justification of conservatisms and margins present in the PTS risk study of the H. B. Robinson plant.

4. The existing NRC analyses supporting the screening criterion (and previous WOG PTS risk studies) have concentrated on longitudinal weld orientations and the NRC has used conservative assessments to arrive at a 300°F RT_{NDT} value for circumferential welds. The H. B. Robinson weld of interest is circumferential and consequently the margin of safety (if comparison is made to the NRC screening criterion and the bases supporting them) is larger than previously documented.

Based on the above considerations, the generic PTS risk analysis carried out for the Westinghouse Owners Group can be used extensively to demonstrate the safety margins and the effectiveness of actions that further improve the safety margins for H. B. Robinson. The plant specific evaluation that is performed realistically assesses the benefits of the actions taken to date (and potential future considerations) as well as provides a study that is on the same bases and that demonstrates compliance with the risk goals used to formulate the NRC screening criteria.

Given these considerations and conclusions the major elements of the H. B. Robinson plant specific PTS study were identified and were compared to the generic PTS program. This process served to identify those areas where additional development was necessary or desirable and those areas where generic program results were to be used with minor modifications. The results of this process (see Section IV.2 for additional detail and justification) are summarized as follows:

Event Sequence Analysis - Use of WOG generic study results are justified with minor modifications.

Thermal Hydraulic (T/H) - Use of WOG generic study results are justified, with plant-specific responses incorporated where appropriate. Plant specific analysis of the mixing phenomena was required.

Probabilistic Fracture Mechanics Analysis - WOG analyses used NRC PFM results directly. Re-analysis of the conditional probability was required using Westinghouse PFM analysis for circumferential weld orientation and HBR considerations using NRC assumptions as a starting point. Evaluation of a limited number of cases is justified.
Importance sampling methodology for cost considerations was applied.

Post-Probabilistic Deterministic Fracture Mechanics Analysis - Not previously part of WOG studies and provides additional insight. It is particularly useful since little previous in-depth analysis for circumferential welds has been performed.

II.3 ANTICIPATED IMPACT OF EXTENDED ANALYSES ON HBR RISK EVALUATION

The more extensive analyses and the possible benefits are briefly described below. The reasons why they were not performed for this analysis are given.

In general, the benefits of these possible extensive analyses are that the results will be more plant specific and that the conservatisms and uncertainties associated with the use of the generic evaluations would be reduced. However, for the purpose of this analysis, more detailed evaluations

were assessed to not be required since the plant specific parameters are not expected to effect the results to the extent that it would change the conclusions of this report.

II.3.1 Event Sequence Analysis

If a complete event sequence analysis was performed the analysis would include the following:

- o Complete event sequence analysis of the H. B. Robinson system.
- o Consideration of plant operating history.
- o Evaluation for each of the transient scenarios evaluated.

The benefit of this analysis is that it would reduce the possibility of overlooking plant specific parameters that affect the event frequencies, both in a positive and negative sense. The effect of this analysis is to limit the uncertainty in the event sequence analysis by evaluation of plant specific parameters.

This complete event sequence analysis was not performed, however, since it has been shown in the past that the use of conservative criteria in the selection of transient categories provides an accurate measurement of risk, especially when the plant is similar to the generic plant, as is H. B. Robinson.

II.3.2 T&H Analysis

If a complete event sequence analysis was performed the analysis could be extended to include the following:

- o Plant specific thermal hydraulic analysis of the transient scenarios that are shown to contribute significantly to risk.
- o Inclusion of other transients for evaluation.

- o Development of the mixing cup model to determine if a larger mixing cup volume is applicable.

The above improvements would reduce the range of uncertainty in the results and may improve the analysis results in that the predicted transients would be less severe. Evaluation of a larger number of transients would reduce the uncertainty that sufficient cases have been considered and that the general conclusions from generic analysis apply.

The effect of these improvements are expected to be within the uncertainty band of the present results and are not expected to change the conclusions of this report.

II.3.3 PFM Analysis

If a PFM analysis was performed on a quasi-plant specific basis, the analysis could be extended to consider the following:

- o a larger matrix of transient cases and RT_{NDT} values to minimize interpolation and extrapolation of the conditional probabilities of significant flaw extension

- o various heat transfer coefficients to reflect cases where stagnation does and does not exist

- o an improved initial flaw size distribution and appropriate distributions reflecting the plant specific material properties and fluence

- o consideration of various vessel failure criteria.

Although more extensive PFM analysis results would be useful, the current results provide sufficient information to perform an appropriate PTS risk assessment.

SECTION III
GENERAL METHODOLOGY FOR PRESSURIZED THERMAL SHOCK RISK ASSESSMENT

III.1 INTRODUCTION

This section of the report discusses the methodology in general terms. The technical description of the methodology is contained in Appendix A.

The risk assessment methodology that was applied to the Pressurized Thermal Shock issue on a generic basis evolved through interaction between the Westinghouse Owners Group and the NRC in 1982. The basic methodology has been utilized and accepted by the NRC.

The following section describes and discusses the importance of the steps used in the methodology. A succinct description of each step is provided at the beginning of each discussion; the importance is stated in each concluding paragraph.

III.2 DESCRIPTION OF THE RISK ASSESSMENT METHODOLOGY FOR PTS

The basic steps taken in evaluating the potential risk of significant flaw extension in the reactor vessel wall resulting from pressurized thermal shock are as follows:

STAGE 1 - BROAD SCOPE DETERMINATION OF RISK FROM PTS

1. Determine a) what sequences of events can lead to reactor vessel pressurized thermal shock and b) the "Frequency of Occurrence"* of each sequence.

   This step assesses of the sequence of events and the Frequency of Occurrence for the sequences giving a "broad perspective look" at literally thousands of cooldown transients. The purpose of this step is

* Certain terms are capitalized since they are used repeatedly in this discussion.

to identify which of the thousands of possible scenarios will result in conditions that could cause "Pressurized Thermal Shock". This step in the analysis uses a technique based on "Event Trees". An event tree starts at the initiating event entry point and branches when the system functions can take multiple paths. The branch point is represented by a node in the event tree where the up branch is "success" and the down branch is "failure" for that system or component. This branching occurs many times in the event tree and represents all the many alternative occurrences possible relative to the PTS concern including errors by the operator and malfunction in equipment. This means that a very large number of possibilities are evaluated. At every point where two directions are possible (i.e., "node points"), the possibility of leading to a PTS concern is considered. This process of branching continues until all of the many end states are reached. Of the multiple end states possible, only those that may be "PTS Contributors" are selected for closer scrutiny. In addition, for each point at which the tree branches, the likelihood of a potential PTS concern is determined. These likelihoods are then combined with the initiating event frequencies to determine the frequency of occurrence of the end state. In this way, all the many possible ways to reach a PTS end state are considered and the likelihood of occurrence is determined. The end states are then grouped into PTS Categories.

The importance of this step is: 1) it defines the sequences that can result in a PTS contribution, and 2) it defines the "Frequency of Occurrence" for each sequence.

2. Determine the rate of cooldown, the final temperature and the representative (characteristic) pressure resulting from each sequence. These values characterize the severity of the sequences for pressurized thermal shock.

   This step associates thermal and hydraulic characteristics with the events defined in the first step. There can be literally thousands of end states but only a small number prove to be of practical concern as contributors to PTS risk. Therefore, it is prudent to use simple yet conservative approximations to represent the key variables. There are three key variables for each sequence of events being evaluated:

1) the rate at which the cooldown occurs
2) the final equilibrium temperature of the downcomer fluid
3) the representative (characteristic) pressure for the event.

Since the pressure may drop then again rise in many cases, a limiting characteristic pressure is chosen for these transients. In general, where the transient pressure and temperature vary, all approximations are conservatively applied.

The word "transient" refers to the fluid temperature and pressure variations that take place in the reactor vessel resulting from the sequence of events being considered. In the thermal hydraulic evaluation, a reliable and detailed computer model uses the sequence of events and calculates the temperature, rate of cooldown and pressure.

The importance of this step is that the temperature, rate of cooldown and pressure are determined. These three parameters are important since they are the factors that produce the stresses on the vessel that are used in the next step.

3. Determine the "Conditional Likelihood of Significant Flaw Extension" for a set of cooldown rates, final temperatures and pressures representative of the results expected from Step 2. Take into account variation of important parameters related to the fracture mechanics analyses of the reactor vessel.

   The third step is to quantify the Conditional Likelihood of Significant Flaw Extension in the reactor vessel if the cooldown transients of Step 2 occur. This likelihood is termed "Conditional" since it is "conditioned" on the assumption that the transient occurs. However, it must be combined with the Frequency of Occurrence of the event (from Step #1) before it can be related to the actual risk or likelihood of significant flaw extension in the vessel (this will be done in Step #3). For each case of cooldown rate, final temperature and characteristic pressure, a large number (thousands) of deterministic fracture mechanics analysis trials are simulated by varying the analysis parameters about a mean value. The term "trials" refers to calculations that vary several input parameters over a range of values. For instance, parameters that are varied include such

values as the copper content that effects the ability of the reactor vessel material to resist flaw growth and the probability that a flaw is present. This step addresses such questions as "What if the initial flaw depth was really la.ger or smaller than the mean value?" or "What if the copper content was really higher or lower than the mean value?". The probabilistic approach answers these questions, not just individually but in all possible combinations of the variables in many trial cases. This means that the likelihood of overlooking some important effect is reduced and that the sensitivity of the results to the variation in each of the variables becomes obvious. This information could NOT readily be learned if a purely deterministic approach was used. Knowing the variables to which the analysis is sensitive and the degree of sensitivity allows focusing on the dominant parameters. (For instance, a set of "base case transients" are generated from which the Conditional Likelihoods are found).

The importance of this step is that the method, 1) takes into account many possible variations, 2) determines the Likelihood of Significant Flaw Extension for the transients identified in Step 2, and 3) considers all the possible combinations that could occur.

4. Combine the Conditional Likelihood of Significant Flaw Extension, (from Step #3) with the Frequency of Occurrence (from Step #1) to find the Frequency of Significant Flaw Extension, for the sequences of events considered within a given category at a given vessel age:*

Frequency of Significant Flaw Extension	=	Frequency of Transient Occurrence	x	Conditional Likelihood of Significant Flaw Extension

---

* Age as used here is in terms of the effect of radiation on the resistance to flaw extension in the vessel. The aging process depends on a number of factors, including the fluence due to radiation and the material properties, and not solely on the passage of time. RT_{NDT} is a func.ion of the fluence and material properties and has been adopted as the measure of age for the PTS issue. For convenience, the term "vessel age" is used to describe various times in vessel life, given the core configuration(s). Thus, it is equivalent to fluence, RT_{NDT} or other related parameters.

5. Repeat the process in Step 4 for each category as a function of age of the reactor vessel, see Figure IV.2-4.

   These steps combine the Conditional Likelihood of Significant Flaw Extension with the Frequency of Occurrence to determine the total likelihood of flaw extension considering all potential events that could occur for each category. The question being answered is "We know the Likelihood of Flaw Extension if a particular transient occurs AND we know the Frequency of Occurrence of the particular transient, but what is the Total Frequency of Significant Flaw Extension to be expected during operation of the plant?" This frequency, which can turn out to be one in a thousand (for one category), or million (for another category), or billion (for yet another category) years of reactor operation, is found by multiplying the Frequency of Occurrence times the Conditional Likelihood of Significant Flaw Extension for each of the possible end states reflecting the various conditions considered in each PTS category.

   The importance of these steps is that they identify the frequency of significant flaw extension for the vessel in terms of occurence per reactor year for each of the PTS transient categories. Once the frequencies are known the transient categories can be compared to identify which are the most important. The calculation is done as a function of age through the end of vessel life.

6. In order to assess the relative importance of each transient, plot the Frequency of Significant Flaw Extension as a function of vessel age for all of the sequences (see Figure IV.2-4 as an example).

   In this step the Frequency of Flaw Extension for each transient found in Step 5 are plotted together to show the relation between each transient. The transients with the greater Frequency of Significant Flaw Extension are of more concern for PTS than the others.

   The importance of this step is that it avoids concentrating effort inappropriately on transients that are not the most important. Without looking at the whole picture in this way, this vital information could not be known. It also shows the relationship between the transients over the life of the vessel. For instance, some transients may become more of a concern as the vessel ages.

7. Determine the Total Frequency of Significant Flaw Extension by summing the Likelihoods from all of the scenarios considered for the vessel as a function of vessel age (for example, see Figure IV.2-4).

   This step sums all the contributors to PTS Risk and gives the total risk value as a function of age of the reactor vessel. This result is called the PTS Risk Envelope. This Risk Envelope shows the rate and amount of increase in risk with vessel age. The Risk Envelope also gives the Total PTS Risk for the reactor vessel over its lifetime so that it can be compared to a Region of Regulatory Concern defined in the next step.

   The importance of this step is that the total risk is defined. That is, the risk likelihood from each transient category is added together giving the maximum total risk due to all PTS Contributors. By knowing the total risk from all contributors, a comparison can be made to an allowable maximum risk to assess if there is a concern. In addition, if there is a concern, the relative severity of the concern is readily observable.

8. Select the "Main Contributors", which are defined to be those transients that have the highest likelihood of a PTS concern, for further detailed assessments.

   The transients with the greatest frequency of flaw extension are called the Main Contributors. Those transients that exceed or that are near the frequency of regulatory concern at the vessel age of interest can be assessed in more detail. Transients that are below the frequency of regulatory concern but within a band accounting for uncertainty in the Stage 1, Broad Scope Evaluation are also selected. Transients with very low likelihood of a PTS concern are eliminated from further consideration. A more in-depth look at the Main Contributors can now be performed to evaluate their significance and, if necessary, to determine how their significance can be reduced.

   The importance of this step is that 1) the Main Contributors can be identified, 2) it defines what to look at in Stage 2 where a more detailed assessment is made for the Main Contributors, and 3) work can be concentrated to reduce the associated risk and to directly improve the safety for the reactor vessel.

STAGE 2 - DETAILED ASSESSMENT OF TRANSIENT SCENARIOS THAT DOMINATE TOTAL RISK

9. Taking a closer look at the assumptions and characterizations associated with each of the Main Contributors identified from Step 8.

   This step is performed so that the identified Main Contributors are appropriately considered in a more detailed fashion. This more detailed assessment is accomplished by looking at the assumptions and variables of the Main Contributors. The simplified representations defined in Step 2 are reassessed to determine a more appropriate characterization for the transient.

   The importance of this step is that the detailed characterizations are defined for the Main Contributors.

10. For each Main Contributor, determine what the vessel age would have to be when the transient would actually cause significant flaw extension.

    This step determines the vessel age at which Significant Flaw Extension would be predicted if the transients, which are the main contributors to risk, actually occur. In general, deterministic calculations evaluate a limited number of transients that adequately represent the Main Contributors.

    Once the transient characterization and other factors are evaluated, it is appropriate to perform deterministic evaluations of the Main Contributors. The deterministic calculations evaluate the flaw growth behavior (using Fracture Mechanics) in the reactor vessel wall when subjected to the transients associated with the Main Contributors to PTS Risk. The analyses use conservative values for the parameters of concern in the fracture mechanics calculations.

    The importance of this step is that it determines the effect of the Main Contributor transients on the vessel. This step could have been performed without the previous steps, but it would not have been known which transients were the important ones to evaluate.

11. To aid visualization, plot the results from Step #9 as a function of the Frequency of Transient Occurrence. This plot is a stair step line that defines the frequency for the dominating transient scenarios as a function of the age of the vessel when significant flaw extension is predicted (for example, see Figure IV.3-1).

    This step plots the results of the deterministic fracture mechanics analyses of the Main Contributors as a function of vessel age. By adding up the Frequency of Significant Flaw Extension for all of the Main Contributors, a stair-step line is formed. This line defines the total or cumulative Frequency of flaw extension over the vessel life.

    The importance of this plot is 1) that the total frequency of flaw extension can be obtained and 2) that it is a visualization of the relative importance of the transients throughout the vessel lifetime.

12. Define the Region of Regulatory Concern on the plot developed in Step 11. Again, the Frequency of Transient Occurence leading to Significant Flaw Extension should not intersect the Region of Regulatory Concern.

    This step defines a Region of Regulatory Concern on the plot of Step 11 that takes into account many factors and is based on an acceptable risk goal. The goal in this figure, however, is different from that of Step 8. The goal in Step 8 considers both the Likelihood of Occurrence of the transients and the Conditional Likelihood of Significant Flaw Extension, whereas the goal in this step considers only the Frequency of Occurrence at the vessel age for which flaw extension is conservatively calculated to occur from deterministic analyses.

    The importance of this step is that the relative importance of the Main Contributors, the effect of vessel age and the important parameters for improvement all can be readily identified.

13. If the Main Contributor does intersect the Region of Regulatory Concern, determine what action(s) can be taken to move the Main Contributor away from the Region of Regulatory Concern. These actions are the means of improving the margin of safety (i.e., decreasing risk by reducing severity or frequency or both or by changing other relevant factors).

This step evaluates what action can be taken to assure that the cumulative frequency of flaw extension does not intersect the the Region of Regulatory Concern. Various modifications can be chosen for future analysis to determine the actual benefit. The relative costs and benefits from various actions can be compared against each other.

The importance of this step is that the most effective actions can be identified to improve the safety of the vessel.

14. Evaluate what actions are most efficient to best resolve the concern and implement (for example, see Figure IV.3-2).

    This step evaluates what remedies can best resolve the concern and then implements the most cost efficient actions. The evaluation takes into account a number of factors that consider the effectiveness of the action and determination of detrimental effects that could occur. In this way, the alternatives can be weighed to find the solution that is balanced for overall benefit to safety and plant operation.

    The importance of this step is that the procedure used assures that the actions taken 1) accomplish the goal of improved safety, 2) are actually warranted, 3) will continue to be effective throughout the life of the plant, and 4) the method provides a basis for limiting the potential actions to only those that are of real benefit.

SECTION IV
APPLICATION OF THE METHODOLOGY TO H. B. ROBINSON

IV.1 INTRODUCTION

This section of the report discusses, in general terms, the application of the methodology described in Section III to H. B. Robinson. The technical description of the application of the methodology to H. B. Robinson is presented in Appendix B.

IV.2 DESCRIPTION OF STAGE 1, "BROAD SCOPE" EVALUATION FOR H. B. ROBINSON

In general, this analysis combines generic and plant specific analyses and data to evaluate the PTS risk for H. B. Robinson. These and other evaluations have aided decision making and demonstrated the significance of actions taken to reduce the PTS risk. A discussion of the steps, described in Section III of this report, as applied to the H. B. Robinson are given below. The first two steps draw extensively upon the analyses of WCAP-10319[4], which is the most complete study of PTS events to date including the evaluation of transient scenarios potentially leading to flow stagnation in the reactor coolant system.

Step 1 (Sequence and Frequency Analysis)

This step is primarily taken from generic studies that have been developed by major efforts funded by the Westinghouse Owners Group to address the pressurized thermal shock issue[1,4]. The event sequences and frequencies were judged (based on evaluation of the differences and similarities between the H. B. Robinson and the generic plant descriptions) to be applicable to H. B. Robinson.

The following significant differences exist between the H.B. Robinson plant design and the generic plant that was used in the WOG PTS risk analyses [1, 2, 4]:

(1) The design shutoff head of the HBR SI pumps, 1,450 PSIG, is lower than that of the generic plant, 2,500 PSIG.

(2) The frequency of non-isolatable small secondary depressurizations is somewhat increased because the steam dump control system drives the secondary power operated relief valves which are upstream of the main steam isolation valves (MSIV's).

(3) The RWST temperature, 90°F, is higher than that of the generic plant.

The only significant design difference from the point of view of the probabilistic plant model is the use of the secondary power operated relief valves for steam dump. This difference was not evaluated since it was apparent that it would not affect the total PTS risk nor change the significant contributors to PTS risk. This is true since the Secondary Depressurization category was found to have minimal effect on risk, as can be seen in Figure IV.2-4.

The only significant departure from the generic analysis is the estimate of the initiating event frequency for a large secondary depressurization upstream of the MSIVs. The value for this event (1.7×10^{-4}) was taken from SECY-82-465 [3] and was analyzed for potential PTS risk. This estimate is consistent with the generic frequency of large pipe breaks in the primary coolant loop and is judged to be conservative.

The key event sequence and frequency results from this evaluation are provided in Appendix B.

Step 2 (Cooldown Rate, Final Temperature and Representative Pressure)

This step is a combination of generic and plant specific work. The temperature and cooldown rate is determined in two parts: 1) detailed thermal or hydraulic computer analysis of the transient, and 2) computer analysis of the mixing of the cooler safety injection water with the hot water already in the reactor coolant system. The generic results for the first part of the analysis were judged to be appropriate and were used in this analysis. H. B. Robinson specific analyses were performed in the second part for small LOCA steam generator tube rupture and secondary depressurizations, which were previously identified as contributing transients [1,2]. The results from this analysis are given in Appendix B.

Step 3 (Conditional Likelihood of Significant Flaw Extension)

This analysis is predominantly plant specific. The analysis was first compared to NRC results from generic analyses [3]. Before the actual analyses were performed on the HBR model, the effect of plant specific considerations were compared in a parametric evaluation. These plant specific considerations were evaluated one by one so that the relative effect (positive and negative) on the magnitude of the result could be seen. Table IV.2-1 defines the analytical cases that were established to evaluate the H. B. Robinson plant specific considerations starting with the NRC parameters as the benchmark case. The results from this sensitivity study are given in Figure IV.2-1. Case G was taken to be representative of the H. B. Robinson considerations for PTS with the net effect of all of the plant specific considerations being a benefit of about a 2 order of magnitude reduction in the conditional likelihood of significant flaw extension below the NRC results. The significant contributor to this improvement was the consideration of a circumferential weld rather than a longitudinal weld. The analysis is performed for the circumferential weld since it is the limiting location for the H. B. Robinson vessel as determined considering the fluence and assumed weld chemistry. The longitudinal welds in the H. B. Robinson vessel, which have sufficiently high toughness, would make an insignificant contribution to the PTS risk, and are not considered in the evaluation.

The H. B. Robinson considerations in Case G of Table IV.2-1 (with the exception that a constant heat transfer coefficient similar to the NRC value was used) were applied to a set of four "base case transients" that represent the characteristics of the dominant H. B. Robinson transient scenarios. The results of this analysis are given in the curves of Figure IV.2-2. This figure also includes the corresponding NRC data [3]. The conditional likelihoods found from these curves are coupled with the results from Steps 1 and 2 to obtain the frequencies of significant flaw extension. An additional analysis was performed to conservatively adjust the conditional likelihood for transients with characteristic pressures greater than 1,000 psig.

TABLE IV.2-1: H.B. ROBINSON CONSIDERATIONS BEYOND NRC RESULTS [3]

TRANSIENT: Final Temperature (T_f) = 150°F, Cooldown Rate (β) = 0.15/min., Pressure (P) = 1,000 PSI

CASE	VESSEL MODEL	HEAT TRANSFER COEFFICIENT	FLAW ORIENTATION	FAILURE CRITERION	FLUENCE ATTENUATION THROUGH WALL	FLAW SHAPE
A [3]	NRC	$\bar{h}$ = 320 Btu/hr-ft^2-°F	Longitudinal	Arrest $\geq$1.0 a/w	NRC Equation for Neutron Energies > 1.0 MeV	Continuous-Initiation and Arrest
B		W Free Convection [5]				
C	HBR					
D			Circumferential			
E1				Arrest $\geq$ 0.5 a/w		
E2				Arrest $\geq$ 0.75 a/w		
F					HBR Equation for displacements per atom model	
G						6:1 Finite-Initiation/ Continuous-Arrest

NOTES:

- SIMULATION PROCESS: STRAIGHT MONTE CARLO
- a = flaw depth
  w = wall thickness

FIGURE IV.2-1

PFM CONDITIONAL PROBABILITIES OF SIGNIFICANT FLAW EXTENSION FOR HBR CONSIDERATIONS (T_F = 150°F, β = 0.15/MIN., P ≈ 1000 PSI)

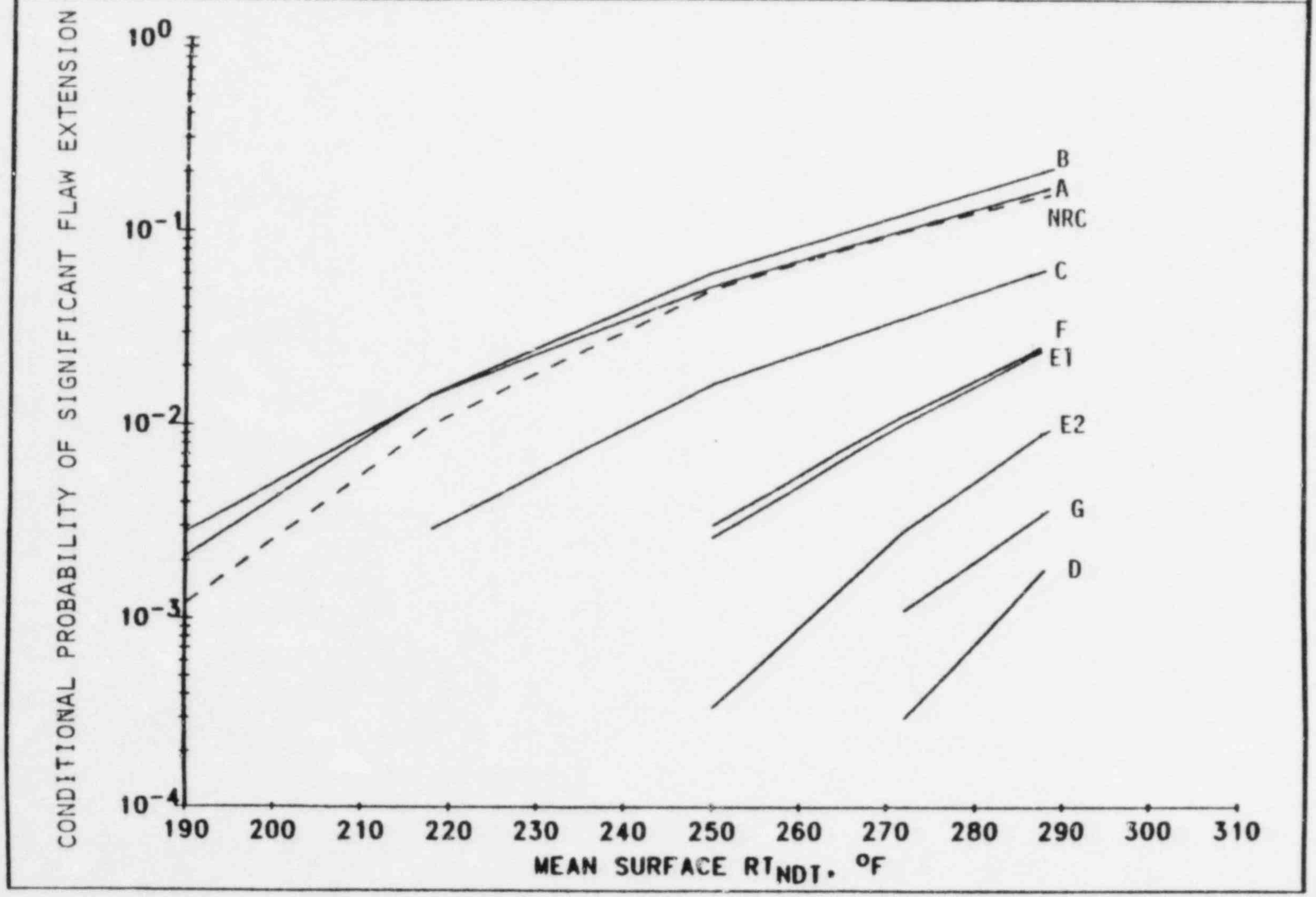
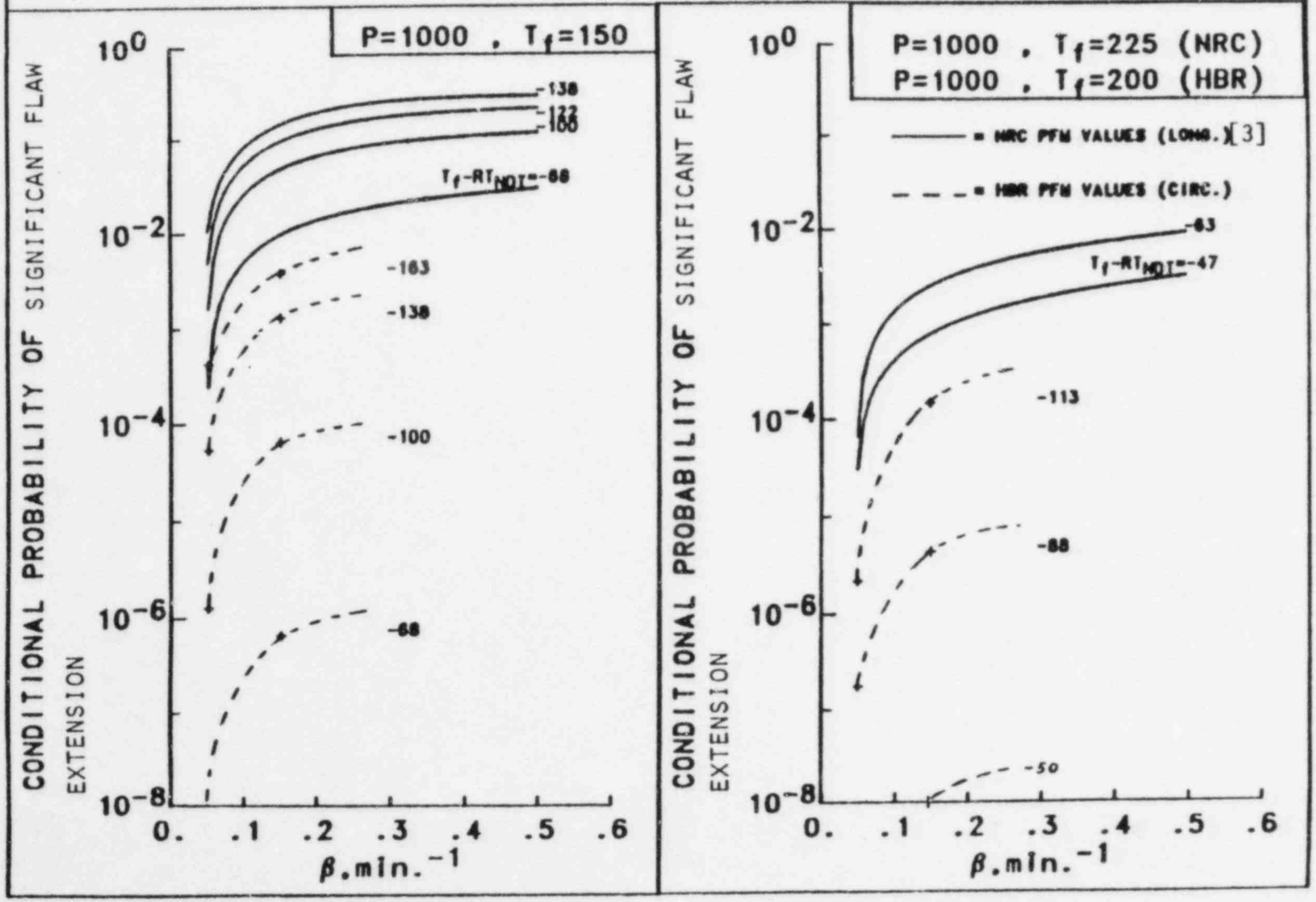


FIGURE IV.2-2

CONDITIONAL PROBABILITIES OF SIGNIFICANT FLAW EXTENSION IN THE REACTOR VESSEL FOR A SINGLE

SIX FOOT LONG BELTLINE WELD-P=1000 psig



It should be mentioned at this point that two specific conservatisms exist in the above analysis. First, any possible benefits from the effect of warm prestressing, which inhibits flaw extension, have not been considered in order to maintain consistency in the "broad scope" evaluation of all the transient scenarios. Second, a continuous flaw has been assumed for flaw arrest. A flaw shape change study given in Appendix C lends support to the use of other than continuous flaws for analysis of arrest since the circumferential neutron flux variation affects the flaw growth of circumferential flaws.

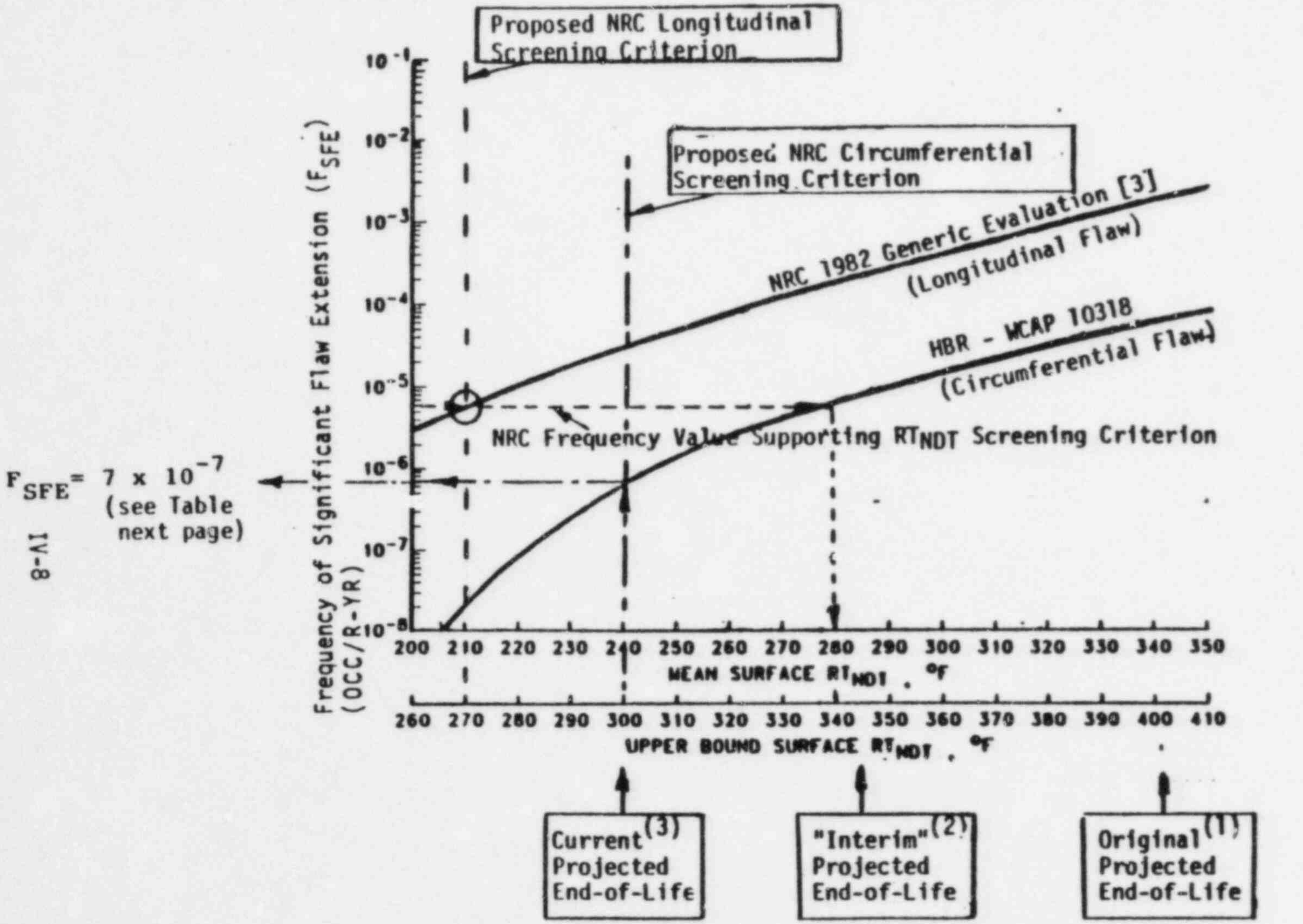
Steps 4, 5, 6, 7 and 8 (Determination of the Combined Effects of the Various Transients)

These are specific evaluations of the H. B. Robinson vessel. They are important to formulate an overall perspective of the PTS risk. The perspective they provide is not only useful for identifying the relative importance of the various transients, but also because it provides a means for comparing this evaluation to other plant specific or generic evaluations that are made on a similar basis.

The risk curves of significant flaw extension from PTS, which are the object of this analysis, are determined from a combination of the analysis results from steps 1, 2 and 3 of the "broad scope" evaluation. Figure IV.2-3 is the total risk envelope of significant flaw extension for H. B. Robinson from this analysis (WCAP-10318) along with the corresponding curve from the NRC 1982 generic evaluation for PTS [3], which was used to support the selection of the RT_{NDT} screening criterion of 270°F for longitudinal flaws.

At each corresponding RT_{NDT} value, the H. B. Robinson frequency is about a factor of 30 lower than the NRC generic results at the RT_{NDT} values of interest. The major sources of this improvement in risk arise from specific differences in the two analyses that include use of the H. B. Robinson geometry, the circumferential flaw orientation and the finite flaw for first initiation. Note that these factors reduce the risk below the NRC results even though factors are also included that tend to raise the calculated risk

FIGURE IV.2-3 DETERMINATION OF TOTAL RISK OF FLAW EXTENSION FROM PTS FOR HBR



(1) Based on original core configuration with "upper bound" Copper and Nickel.

(2) Based on currently installed "Low Leakage Core" with "upper bound" Copper and Nickel, not including additional planned reduction.

(3) Based on currently installed "Low Leakage Core" and planned flux reduction using "Part Length Shielding" with assumed "upper bound" Copper and Nickel.

level such as, use of "displacements per atom" (dPa) in the determination of neutron fluence (rather than neutron damage for greater than 1 mev neutron energy only), and taking no credit for the benefit of warm prestressing for the small LOCA transient. The NRC evaluation did not consider the effect of dPa and did include the benefit of warm prestressing for small LOCA. These differences are all associated with the analyses to generate the conditional likelihood of significant flaw extension.

The benefit of the reduced risk can be translated to an increased allowable RT_{NDT} value for H. B. Robinson and is interpretable relative to the screening criteria. Figure IV.2-3 shows that at a "mean surface RT_{NDT}" of 210°F, the risk level on the NRC curve is about 6×10^{-6}. At this same level of risk, the results from this study show that the acceptable mean RT_{NDT} for H. B. Robinson would be about 280°F (i.e., the circumferential weld in the H. B. Robinson vessel also has acceptably low risk through a 280°F mean RT_{NDT}).

The NRC screening criteria use the "upper bound" RT_{NDT} defined by the Guthrie Correlation [B.3] which is 60°F above the "mean" RT_{NDT} used in the NRC and H.B. Robinson risk analyses (and which is consequently used in many of the figures). In terms of this "upper bound" RT_{NDT}, the longitudinal NRC screening criterion is 270°F, which is 60°F above the 210°F "mean" surface RT_{NDT} described in the above paragraph. Likewise, the corresponding criterion for the H.B. Robinson circumferential weld is 340°F RT_{NDT} (which is 60°F above the mean value of 280°F). The relation between the "mean" and "upper bound" RT_{NDT} is explicitly shown on the horizontal axis of Figure IV.2-3. The "mean" value of RT_{NDT} is used in the PFM analysis because it is one of the quantities that is analyzed for variation about its mean. The "upper bound" is used in specifying the material property of the reactor vessel (and consequently for the screening crteria) because it conservatively bounds the possible value of RT_{NDT} for that material.

The NRC specified RT_{NDT} circumferential screening criterion is 300°F. However, this value was not developed in the same manner as the longitudinal screening criterion. The NRC RT_{NDT} criterion for circumferential flaws, 300°F, was extrapolated from the longitudinal case using deterministic

fracture mechanics analysis for flaw initiation. However, no risk analysis was performed for circumferential flaws. For this reason, no correlation in terms of risk can be made between the NRC circumferential screening criterion and the H. B. Robinson value. However, since the H. B. Robinson value of 340°F is above the NRC value of 300°F, the NRC screening criterion is viewed as a conservative value for application to H. B. Robinson.

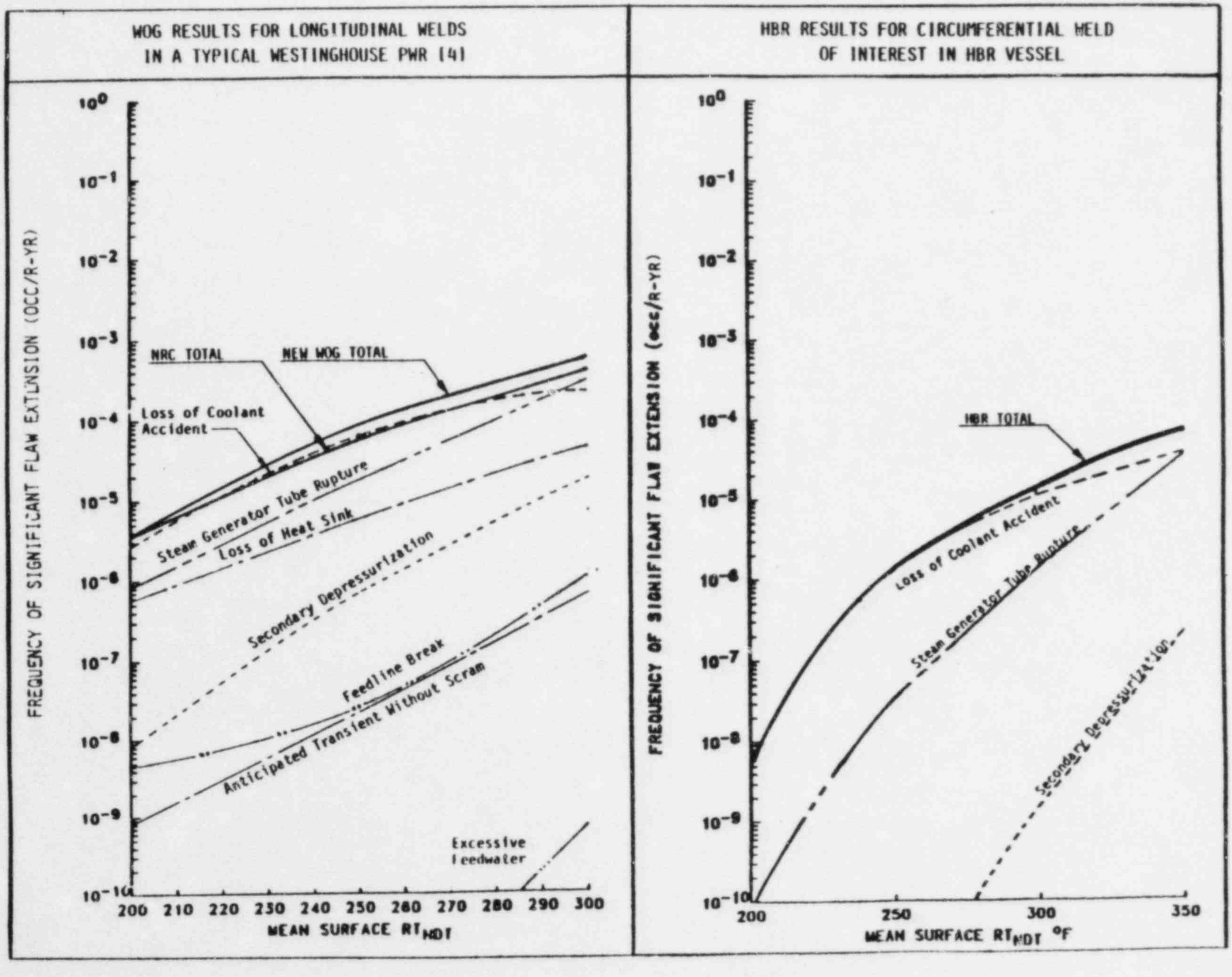
Considering the flux reduction currently being implemented by CP&L for HBR, the end-of-life upper bound surface RT_{NDT} using assumed high copper and nickel properties is projected to be near the 300°F NRC screening value for circumferential flaws. This value translates into a frequency of significant flaw extension (F_{SFE}) of 7×10^{-7}/reactor-year from the HBR total risk curve. This value, which does not take into account the conditional probability of core melt given that significant flaw extension occurs, is already more than 2 orders of magnitude below the National Safety Guideline for large scale core melt of 10^{-4}/reactor-year from all accidents [6].

Even though the total risk envelope from the "broad scope" evaluation, at the RT_{NDT} considering implementation of part length shielding, is below the NRC frequency value supporting the screening criterion, a detailed narrow scope evaluation is performed as described in the following section. This evaluation provides additional insight from the detailed narrow scope approach to show margins and to show how CP&L initiatives are clearly eliminating the PTS issue from further consideration as a safety issue for H. B. Robinson. Before selecting the transients for the detailed narrow scope (Stage 2) analysis, a discussion is given relative to HBR results versus the results recently obtained from the WOG generic PTS risk assessment [4].

Figure IV.2-4 provides a summary of both the WOG and HBR risk (or equency) of significant flaw extension versus RT_{NDT} curves. The plots are given respectively for longitudinal welds in a typical Westinghouse PWR and for the circumferential weld of interest in the HBR vessel. These plots also show the total risk envelope that was found by summing the results from the transient categories that were evaluated in each study. As discussed previously, the WOG evaluation [4] is a complete study of PTS events, including scenarios that potentially lead to flow stagnation in the reactor coolant system. Since the WOG results are directly comparable to those of the HBR study and because the relation of the contributors to each other remain the same between the two risk studies, the HBR study can also be considered to effectively address the major PTS events.

In both studies the loss of coolant accident and steam generator tube rupture categories are the dominant contributors to the total risk. Both studies demonstrate that secondary depressurizations no longer require focused attention as was previously suggested [1,2] since they make an insignificant contribution to the overall risk. Therefore, the LOCA and SGTR categories are investigated in depth in the detailed narrow scope (Stage 2) evaluation in the next section. The loss of heat sink (LOHS) category from the WOG assessment is also included in Stage 2 to assess the effect of LOHS for H. B. Robinson since its was the third largest contributor to risk. Proper treatment and inclusion of this category in the HBR "broad scope" (Stage 1) evaluation would not be expected to change the total HBR risk results. This is true since the inclusion of these transients would not increase the risk significantly; that is, the risk envelope of Figure IV.2-4 would remain the same.

FIGURE IV.2-4 WOG/HBR PTS RISK STUDIES - TOTAL FREQUENCY OF SIGNIFICANT FLAW EXTENSION



IV.3 DESCRIPTION OF STAGE 2 - DETAILED NARROW SCOPE ASSESSMENT OF TRANSIENT SCENARIOS THAT DOMINATE TOTAL RISK FOR H. B. ROBINSON

In general, the Stage 2 assessment is a plant specific analysis of the dominating transient scenarios, which were identified from the H. B. Robinson "broad scope" risk assessment, using thermal-hydraulic and deterministic fracture mechanics (DFM) analyses. The approach is similar to that used by the NRC to formulate the RT_{NDT} screening criteria for PTS. The results are presented in a cumulative frequency of significant flaw extension distribution plot to provide additional insight relative to the inherent margin of safety of H. B. Robinson for PTS. CP&L initiatives are then evaluated to show the benefit derived from each initiative and to indicate the resulting increased margins of safety for HBR.

Steps 9, 10, 11, 12 (Deterministic Evaluation of Risk)

These steps are an evaluation of transients representative of the dominating scenarios in the loss of coolant accident and steam generator tube rupture categories, which were previously identified to dominate the PTS risk for HBR. However, before discussing the results of this evaluation, it is important at this point to show that the method used in summarizing the results for HBR is consistent with the approach employed by the U.S. NRC to select the RT_{NDT} screening values for PTS.

Events which occurred during the first 350 total PWR reactor years of operation in the United States, were used by the U.S. NRC as the basis for selecting the RT_{NDT} screening criteria for pressurized thermal shock (see Section 4 of SECY-82-465 [3]). The eight significant events were each characterized in terms of final cooldown temperature (T_f) and in terms of critical RT_{NDT} (RT_c), the limiting vessel material condition establishing the onset of vessel failure (i.e. significant flaw extension). To select the RT_{NDT} screening criterion, a cumulative frequency distribution was plotted as a function of the T_f values for the eight events. Similarly, the RT_c results for the eight events were used to develop a plot of the cumulative frequency of events versus the RT_c for which the deterministic fracture

mechanics calculations predict unacceptable crack extension will occur. The NRC [3] also states that

> The RT_c evaluation is, in many ways, the better way to characterize an event than using T_f alone. Calculating RT_c includes the actual time variation of temperature and pressure and is preferable to the stylized constant pressure and simple experimental temperature behavior approximation inherent in the T_f evaluation (of the eight events). Moreover, characterization of events by T_f alone requires neglect of the effect of different pressure and different time decay constants on PTS severity.

The RT_{NDT} screening criterion of 270°F, selected for longitudinal flaws, was based on earlier plots of T_f and RT_c that yielded values of approximately 260°F and 280°F, respectively, for a nominal event frequency of 10^{-2} per reactor-year. "The justification for choosing 10^{-2} was only that this is comfortably lower than the range of 'anticipated operating occurrences'" [3]. Nevertheless, the 10^{-2} frequency was and still is a reasonable place to start for evaluating safety margins for results obtained from deterministic fracture mechanics analyses, as long as probabilistic analyses, which are consistent with those performed by the NRC to support the RT_{NDT} screening criterion, are also performed to demonstrate the degree of conservatism. Conservative assumptions are generally used in deterministic fracture mechanics analyses thereby supporting a goal higher than that used in probabilistic evaluations, which use "mean" or best estimate analyses. In theory, both goals should represent an identical status of reactor vessel integrity.

Consistent with the intent of selecting the RT_{NDT} screening value of 270°F, a plot of cumulative frequency of events versus the RT_c obtained for all PTS event sequences, which dominate the total risk of significant flaw extension as determined from a probabilistic risk assessment, can be used to assess the margin of safety for a given vessel.

Applying the above approach, a plant specific small LOCA mixing cup, stress and DFM evaluation was previously performed in WCAP-10309 [7] for a range of Refueling Water Storage Tank (RWST) temperatures. The transient that was analyzed is representative of the small LOCA scenarios at the various decay heat levels that dominated the risk of significant flaw extension for the loss of coolant accident category in the "broad scope" risk evaluation for HBR. Using a criterion that significant flaw extension is defined as a flaw being unable to arrest within 75% of the vessel wall thickness, the RT_c for the small LOCA without the benefit of heating the injection water (i.e. RWST temperature = 40°F) is 380°F. The benefit of the warm prestressing phenomenon in prohibiting further flaw extension was applied in this deterministic evaluation consistent with previous NRC and WOG analyses for this transient [3,8]. Details of this plant specific transient, thermal, stress, and DFM analysis are given in WCAP-10309. A frequency of occurrence equal to 6.3×10^{-4} occ/r-yr is conservatively associated with this transient to represent the sum total of the event frequencies for the small LOCA scenarios dominating the PTS risk for HBR.

Similarly, a plant specific evaluation of a steam generator tube rupture transient, which is representative of the scenarios that dominate the risk of significant flaw extension for the SGTR category, was performed and is presented in Appendix D. This detailed analysis, which used the same methods as those applied in WCAP-10309 [7] with the exception that the warm prestressing benefit was not applied, yielded a critical RT_{NDT} of 335°F. A frequency of occurrence equal to 3.2×10^{-3} occ/r-yr is conservatively associated with this transient to represent the total event frequencies for the dominating SGTR scenarios (SGTR1).

Other transient scenarios worthy of consideration, but not via detailed thermal-hydraulic and DFM analyses, are also discussed and presented relative to the impact of these events on the total evaluation.

After the above SLOCA and SGTR representations, the next highest contributors to the risk from PTS for HBR are two of the remaining SGTR bins (SGTR2) that were not included in the above evaluatiuon. Based upon the WOG PTS risk assessment [4], the loss of heat sink (LOHS) transient category would also be expected to yield the same PTS risk as the two mentioned SGTR bins. A

transient that would be representative of these SGTR2 and LOHS scenarios would have transient characteristics more severe than those evaluated for the dominating SLOCA and SGTR1 events. Therefore, the critical RT_{NDT} for this transient would be expected to be less than 335°F, the value obtained for the representative dominating SGTR scenarios. The total event frequency for a representative SGTR2/LOHS transient would however be a low value of 4.1 x 10^{-5} occ/r-yr, the sum total of the two SGTR bins and LOHS category of interest.

All other transient scenarios are not significant contributors to PTS risk because they fall into one of the following two categories:

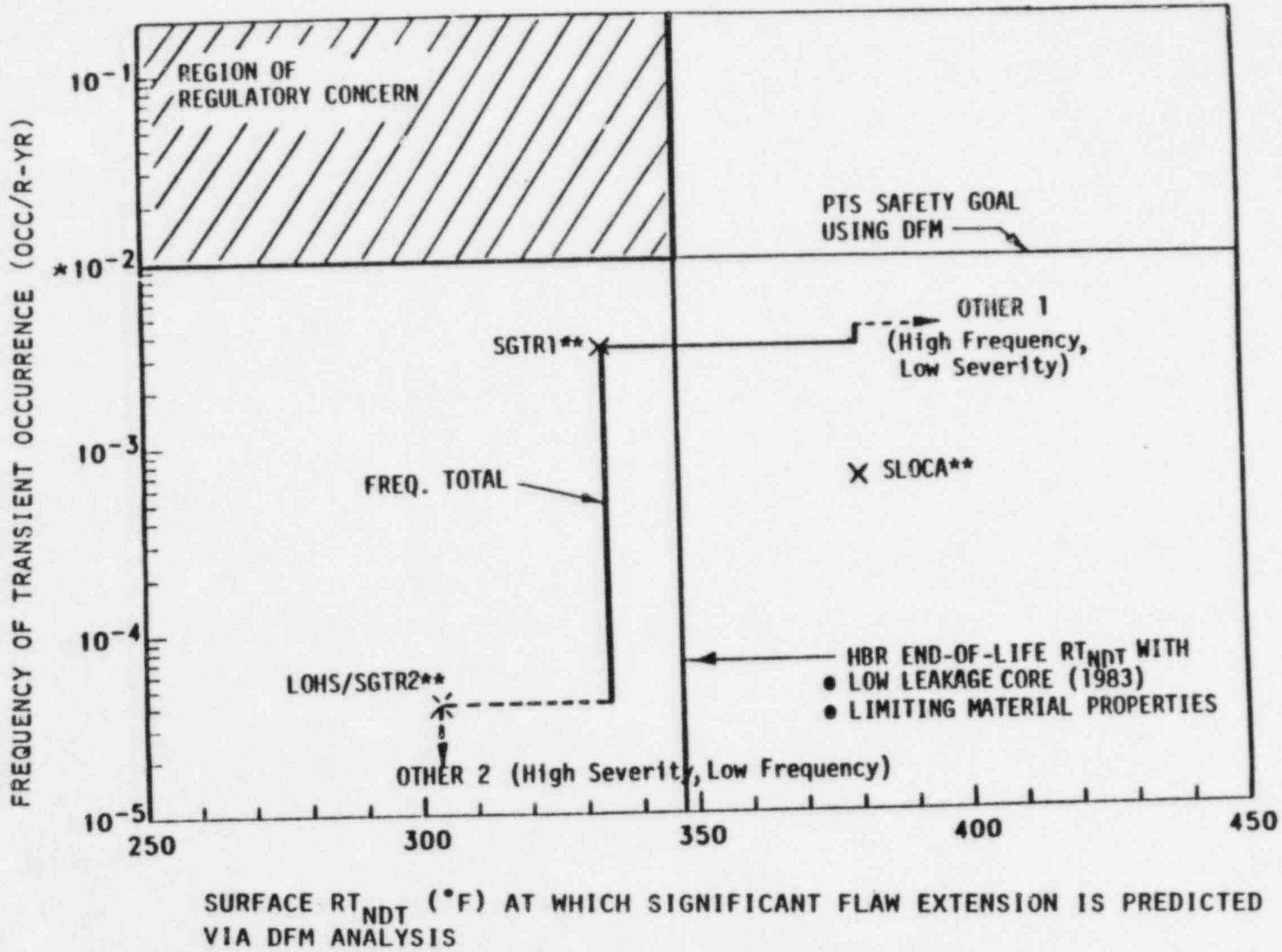
OTHER 1) The severity of the transient is not significant resulting in low conditional probabilities of significant flaw extension. Even if the scenario has a transient frequency of occurrence equal to one, the frequency of significant flaw extension is very low.

OTHER 2) The transient severity is significant, but the frequency of transient occurrence is extremely low yielding an insignificant frequency of significant flaw extension.

Most of the other transient scenarios fall into the first category (other 1), while a few transients belong in the second category (other 2), e.g., a large secondary depressurization between the main steam isolation valves and the steam generator with operator error in never terminating auxiliary feedwater to the faulted steam generator.

Figure IV.3-1 presents a plot of HBR frequency of transient occurrence leading to significant flaw extension as a function of vessel age using the above DFM results. Any consideration of CP&L initiatives that are planned and some that have already been implemented are not taken into account. Starting with the same goal of 10^{-2} occ/r-yr as used by the NRC in the formulation of the RT_{NDT} screening criteria for PTS, the analyses demonstrate that the HBR margin of safety at end-of-life, even without such initiatives, is equivalent

FIGURE IV.3-1 H.B. ROBINSON-UNIT 2 FREQUENCY OF TRANSIENT OCCURRENCE LEADING TO SIGNIFICANT FLAW EXTENSION USING DETERMINISTIC FRACTURE MECHANICS (DFM) ANALYSIS



*NRC GOAL USED TO ESTABLISH RT_{NDT} SCREENING CRITERIA

**DOMINATING TRANSIENTS IDENTIFIED FROM HBR AND WOG PTS RISK STUDIES AND DEFINED IN SECTION IV.3

to that inherent in the NRC screening criteria. Although the region of regulatory concern is not intersected, the next steps evaluate the CP&L initiatives to demonstrate the further significant increase in the margin of safety for HBR that is resulting from these actions.

Steps 13, 14 (Evaluation of CP&L Initiatives)

In addition to the PTS risk study given in this report, Carolina Power and Light Company has recently sponsored several other programs to address concerns related to pressurized thermal shock of the H.B. Robinson Unit 2 reactor pressure vessel. These programs have been undertaken to further improve the safety of the vessel for PTS and to show that modifications already in place or planned will demonstrate a sufficient margin of safety for continued plant operation. A short description of each of these programs, including the individual results expected from their implementation, is provided as follows:

- o Study of Benefit in Heating RWST for SLOCA (WCAP-10309) [7] - The impact of heating the Refueling Water Storage Tank for a range of temperatures above those encountered in past operating conditions on the integrity of the H.B. Robinson Unit 2 vessel during a postulated small break LOCA, which is a dominating transient scenario identified from the PTS risk study, was examined. Plant specific transient, thermal, stress, and deterministic fracture mechanics analyses were performed to obtain the results. The water in the RWST will be heated such that its temperature should not fall below 90°F. Using this minimum temperature, the deterministic results in WCAP-10309 can be readily interpolated to yield a critical RT_{NDT} of 405°F, a 25°F increase over the result without the benefit of heating the RWST water.

- o Flux Rate Minimization Program [9] - This flux reduction program involves moving thrice burned fuel to the exterior of the core. This initiative was implemented in fuel cycle 9 and results in the

projected end-of-life RT_{NDT} of 347°F assuming high copper and nickel properties. Further flux reductions will be achieved by using part length shielding to selectively reduce the neutron flux rate to the lower circumferential weld, the weld of concern for the HBR vessel. The projected end-of-life RT_{NDT} with the limiting material properties will be reduced to about 300°F as a result of this action scheduled to be implemented in fuel cycle 10 in 1984. 300°F also happens to be the NRC screening value for circumferential welds for PTS.

- Vessel Sampling/Search Program - A material property search was conducted to find the chemical properties of the HBR vessel weld material. Since the search did not yield chemical properties, the study showed that certain welds in the upper head corresponded to the weld wire and flux of the specific vessel welds of interest. These head welds will be sampled in the near future to determine the copper and nickel content. The benefit expected from this CP&L initiative is to further reduce the projected end-of-life RT_{NDT}. It should be noted that all studies and evaluations performed to date on the HBR vessel have assumed limiting high copper and nickel properties.

- Operator Action and Procedures - Carolina Power and Light Company has implemented Revision 1 of the Emergency Resonse Guidelines [10] on HBR that were developed by the Westinghouse Owners Group to mitigate the effects of reactor vessel pressurized thermal shock. These revised guidelines provide a significant benefit by reducing the frequency of occurrence of PTS events and, to some extent, lessen the associated transient severity via improved operator guidance. These benefits have only been taken into account for the SGTR category in both the HBR and WOG [4] risk analyses. Extension of these results particularly with respect to pump trip criteria at low decay heat levels, to other transient categories would provide additional benefit to the PTS risk results.

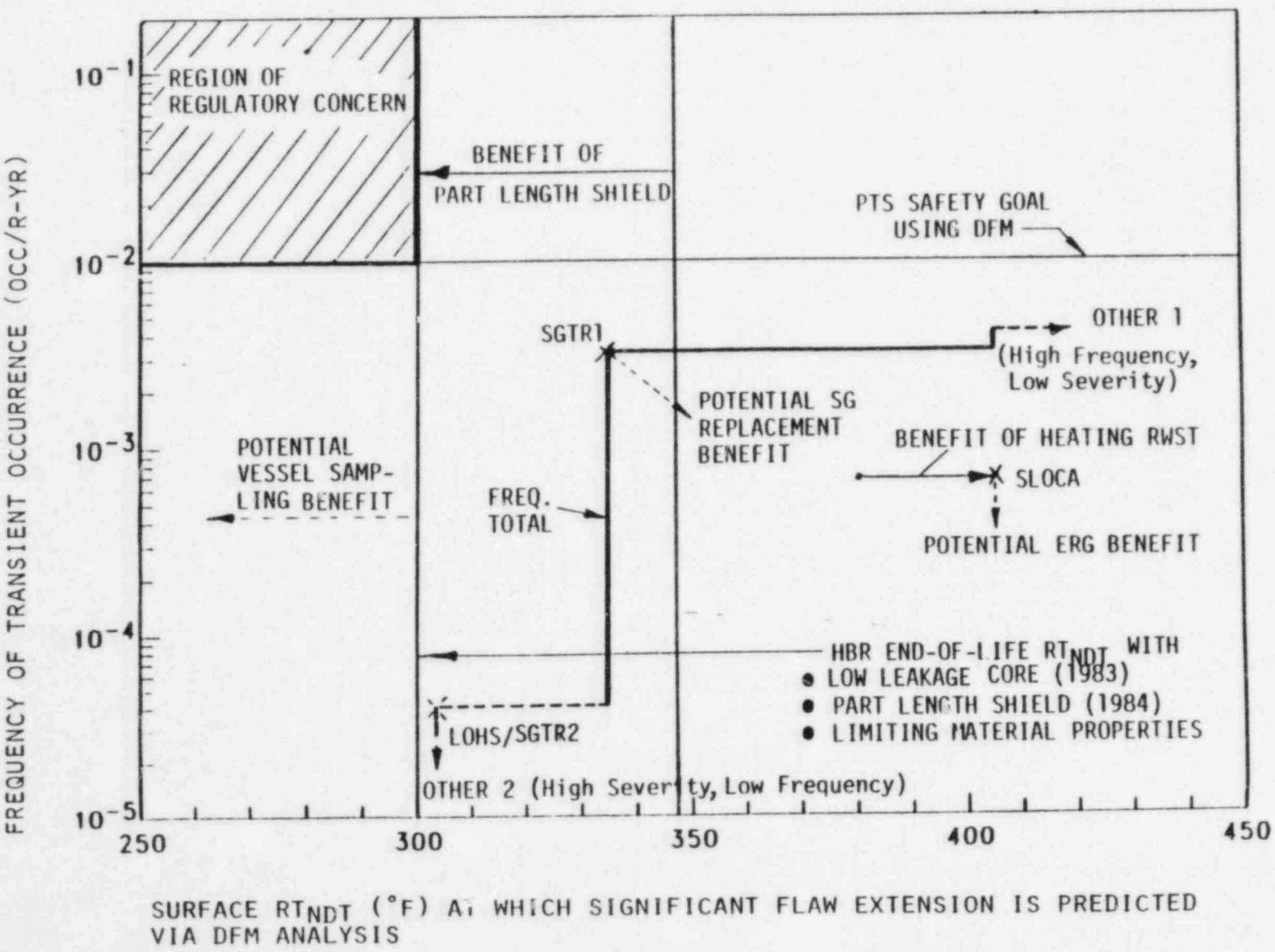
- o Steam Generator Replacement Program - Although this program is not being implemented because of PTS considerations, the new steam generators planned to be installed in 1984 should have a possible benefit on SGTR initiating frequencies because of stainless steel support plate material, improved heat treatment, and favorable water chemistry. Changes being implemented or planned that will improve the water chemistry should have a positive benefit by reducing corrosion concerns for the steam generators. Some improvement in the secondary depressurization transient severity because of the flow limiter in the outlet nozzle could also be expected.

The objective of the following evaluation is to integrate the results from the above programs to demonstrate that a sufficient margin of safety exists for continued safe operation of the H.B. Robinson Unit 2 reactor vessel through end-of-life. The approach for integrating the results, which again is consistent with the method used by the U.S. NRC to select the RT_{NDT} screening values for PTS is exactly the same as that used in the evaluation carried out for steps 9 through 12.

Figure IV.3-2 presents a plot that provides a summary of the impact of all CP&L initiatives on the HBR frequency of significant flaw extension versus vessel age. This figure is synonymous to Figure IV.3-1, which provided results without consideration of most CP&L initiatives for PTS. From comparison of the two figures, the margin of safety from PTS events will have a significant further increase through end-of-life for HBR as a result of the above CP&L actions.

The largest contributor and most cost-effective measure for this increase is the planned implementation of part length shielding to maximize the reduction in neutron flux to the circumferential weld of interest. A similar increase in margin may result from the vessel sampling program to define less limiting properties for the circumferential weld. Less limiting material properties will result in a lower radiation embrittlement rate and consequently any flaws

FIGURE IV.3-2 H.B. ROBINSON-UNIT 2 FREQUENCY OF TRANSIENT OCCURRENCE LEADING TO SIGNIFICANT FLAW EXTENSION USING DETERMINISTIC FRACTURE MECHANICS ANALYSIS WITH CP&L INITIATIVES



in the weld will be less susceptable to flaw extension. Because the steam generator tube rupture transient bins are closest to the region of regulatory concern and, thus dominate the risk in the DFM evaluation, heating of RWST water has a minimal impact on the overall margin of safety. Similarly, accounting for benefits from the revised ERG's will not impact the safety margin because the benefits were already considered in the SGTR evaluations. However, the CP&L steam generator replacement program may help to reduce the SGTR frequencies. The RWST heat-up and revised ERG initiatives may then play a more significant role in increasing the margin of safety for PTS since other transients that depend on these factors could then be relatively more important.

In summary, the detailed narrow scope assessment demonstrates that the HBR margin of safety for PTS events at end-of-life is equivalent to that inherent in the NRC screening criteria even without the benefit of all future CP&L initiatives. Taking all the initiatives into account, the margin of safety is further increased and demonstrates continued safe operation through the remainder of the service life of the vessel relative to PTS considerations.

SECTION V

RESULTS, MARGINS AND UNCERTAINTIES

V.1 INTRODUCTION

This section presents the final results of both the Stage 1 and Stage 2 analysis. The results are compared with other major analyses to give a perspective on this work. Section V.2 presents the final results and comparisons from the Stage 1 analysis; Section V.3 presents the results and comparisons from the Stage 2 analysis. Section V.4 discusses the margins and uncertainties inherent in the analyses.

V.2 STAGE 1 RESULTS

The principal results from the Stage 1 analyses are a) the risk curves for each transient category and b) the risk envelope, based on the summation of the individual risk curves. The risk curves for the individual categories are given perspective by comparison to results of the most recent generic analyses performed for the Westinghouse Owners Group [4]. The risk envelope is given perspective by comparison to the NRC risk envelope [3] that was used to support the RT_{NDT} screening criteria.

Figure IV.2-4 presents the Stage 1 risk curves and the 1982 NRC and 1983 WOG results for the individual transients. The risk envelope for each analysis is also shown. In general, the results for individual transient categories and the risk envelop for H.B. Robinson are lower than those found in the generic analysis. This improvement is because of plant specific considerations for H.B. Robinson, the most notable of which is the evaluation of the circumferential flaw rather than the longitudinal flaw evaluated in the generic case. Note that the NRC and WOG risk envelopes are essentially the same. Further discussion of the difference in the analysis considerations are in Section IV.2 and Appendix B. The conclusions drawn from this analysis are that the main contributors to risk are the loss of coolant accident (primarily

the Small LOCA bin) and Steam Generator Tube Rupture transient categories, and that the secondary depressurization category contributes minimally. The relative position of these transients as risk contributors is the same for the generic analysis. This implies that the relative contribution of other transient categories not included in the H.B. Robinson study is the same as for the generic results.

Figure IV.2-3 presents the Stage 1 risk envelope in comparison to the NRC risk envelope. The curves are shown in terms of "Frequency of Occurrence of Significant Flaw Extension per Reactor Year" versus RT_{NDT}. Two significant values can be taken from this curve presenting the results. First, the value that compares to the NRC generically determined screening criterion is 340°F RT_{NDT}. Second, the risk value that corresponds to the currently predicted end-of-life RT_{NDT} for H.B. Robinson based on the proposed vessel shielding is 7×10^{-7} occurrences per reactor year. These two values are discussed in the following paragraphs.

The NRC screening criterion for longitudinal flaws, 270°F RT_{NDT}, was formulated from deterministic fracture mechanics analyses and was supported by a risk analysis [3]. The RT_{NDT} criterion for circumferential flaws, 300°F, was extrapolated from the longitudinal case using deterministic fracture mechanics analyses for flaw initiation. However, no risk analysis was performed for circumferential flaws. The value of risk at the 270°F longitudinal screening criterion is 6×10^{-6} occurrences per reactor year. As shown in Figure IV.2-3, comparing the HBR and NRC results shows that the RT_{NDT} for HBR at this same risk is about 340°F. The significance of this value is that it gives one measure of the RT_{NDT} that is allowable for the H.B. Robinson lower circumferential weld and the measure is on the same basis as the one used by the NRC to support the 270°F screening value. The risk value of 6×10^{-6} is said to "support" the NRC screening criterion since the NRC considered other determinations as well.

The value of RT_{NDT} that corresponds to the currently predicted end-of-life value for the H.B. Robinson lower circumferential weld is about 300°F. The RT_{NDT} for H.B. Robinson was predicted to be about 347°F based on the fluence expected for the current core configuration. However, a modification developed by CP&L resulted in a means to further limit the fluence so that the currently predicted end-of-life RT_{NDT} is at 300°F. At this value of RT_{NDT}, the predicted risk level of 7×10^{-7} is much lower than the value used by the NRC as discussed above. The significance of this result is that the risk level associated with H.B. Robinson is lower than that associated with the generic plant at the same RT_{NDT}.

In addition, the material properties used for this evaluation are "upper bound". This means that residual elements that contribute most significantly to vessel embrittlement (that is, copper and nickel) are assumed to be the highest possible values. Further programs by CP&L to determine the actual weld chemistry values are underway. (If the weld chemistry has a smaller percent of residual elements that affect the radiation embrittlement rate of the weld then the RT_{NDT} and risk level are lower than currently predicted. This effect has potential for having significant benefit for PTS).

Stage 2 evaluations were performed for the dominating transient categories that contributed most to the risk level. These determinations put the transient categories in perspective using deterministic analyses.

V.3 STAGE 2 RESULTS

Stage 2 is a detailed narrow scope evaluation of risk. The scope is "narrower" than Stage 1 since it evaluates the main contributors identified in Stage 1. It is more "detailed" since these main contributors are evaluated using thermal hydraulic, stress and deterministic fracture mechanics analyses, (they were identified using stylized transient characteristics and probabilistic fracture mechanics analyses).

The principal result from the Stage 2 analysis is the risk plot showing the relation of the main contributors to the Region of Regulatory Concern. Various aspects of the development of these results are given in Section IV.3 and Section 8 of Appendix B. Figure IV.3-1 shows the relation of the stair step risk plot to the Region of Regulatory Concern derived from the deterministic analyses without CP&L initiatives to reduce risks. Figure IV.3-2 shows the relation of the risk curve to the Region of Regulatory Concern after the initiatives are taken into account. The primary conclusion is that the reduced risk is sufficiently removed from the Region of Regulatory Concern. This then supports the conclusion that PTS is not a safety concern for the H. B. Robinson plant.

The margin between the stair step risk curve and the Region of Regulatory Concern is increased by other considerations as well. One factor that adds to this margin is the results of the analysis of Appendix C. This analysis evaluates the change of shape of the flaw as it grows while subjected to a stylized severe transient (with a high ratio of cooling, a low final temperature, and a higher pressure than expected for H. B. Robinson). This analysis takes into account the reduction in radiation damage to the vessel wall away from the peak fluence region. The results of the analysis lend support to the fact that something other than a continuous flaw (i.e., one that extends around the vessel) may be applicable as an assumption in the fracture analysis. Use of the "continuous flaw criteria" in the fracture analysis, because of the above considerations is a conservatism in the analysis. This adds additional, although undefined, margin to the results. Further considerations of this type are discussed in the following section.

V.4 MARGINS AND UNCERTAINTIES

Calculated risk levels in Stage 1 are a relative measure of how a reactor vessel is affected by PTS transients. They are not an absolute measure of the actual or specific risk related to a vessel. Rather the best uses are: 1) comparisons with other measurements on the same and other vessels or with generic evaluations and 2) comparison of the relative risk of transients.

The probabilistic analyses of the Stage 1 risk of PTS is based on a number of assumptions and a collection of a broad range of considerations each with its range of uncertainties. The accumulation of uncertainties in Event Tree Analysis, Transient Analysis and Probabilistic Fracture Mechanics analyses precludes direct application of the results in an absolute sense. However, even though the magnitude of risk is uncertain and cannot be applied on an absolute basis, as opposed to the use of deterministic analyses, the relative measures between analyses have a considerably reduced degree of variability and uncertainty between them since they are made on the same basis, with the same type of assumptions, with readily quantifiable differences. For these reasons, they are directly comparable. The results should be utilized with the knowledge that there is some range in variability and uncertainty in the risk measurement.

The sections that follow provide a list of conservatisms and uncertainties applicable to the analysis. The elimination of any conservatisms would tend to improve the results, while the uncertainties could increase or decrease the absolute magnitude of the risk results. Neither effect would change the basic conclusions or change the effectiveness of the relative measure of risk. The deterministic analyses of Stage 2 are subject to a much smaller range of variation since the variation in certainty is primarily a result of the estimate of transient frequency of occurrence and depends little on other variations since conservative assumptions are used in the T&H and detailed stress and DFM analyses as well as in the material properties.

V.4.1 Event Tree Analysis

1. One major conservatism in the event tree analysis is the treatment of decay heat. The separation of the transient frequencies into decay heat initial conditions or bins is significantly skewed toward the lowest decay heat bin. The currently available information to separate transients is EPRI-2230 [11] which only contains one bin for transients at 25% power or less. Transients that initiate at 15%, 10%, or even 5% should still have

relatively high values of decay heat present, especially if operation is at power immediately preceding the incident. However, in the absence of such data, it was conservatively assumed in the analysis that the transients initiating below 25% power are split between the three decay heat bins including the lowest decay heat bin at the calculated percentage.

2. The principal uncertainty associated with this analysis is that an in depth plant specific evaluation was not performed. However, the results from PRA evaluations are based in a large part on operating plant history and not solely on plant specific concerns. In addition, the resulting uncertainty in this area is within the bounds of the overall uncertainties in the analysis.

V.4.2 T&H Analysis

1. One of the remaining uncertainties associated with the T&H Analysis (at the time of writing of this report) is the still not fully defined mixing phenomenon in the cold leg and downcomer region. While the mixing cup volume used in this study was obtained conservatively by comparing with the 1/5 scale CREARE tests [12] and with some 3-D analytical predictions [13], the results are highly dependent on the appropriateness of the scaling laws and scaling parameters used. More experimental data, preferably from a larger scale prototypic, geometric configuration, will help to further qualify the extent of the mixing phenomenon. In particular, the degree of fluid mixing in the reactor lower plenum, the loop seal, the pump internal, and the SG plenum will need to be better understood. Also the effects of the accumulator lines and the SI lines on mixing should be quantified. The 1/2 scale CREARE tests and the Purdue Mixing tests currently being conducted are expected to provide the additional qualification. Based on currently available data, the downcomer temperatures calculated by the present mixing cup model are believed to be slightly conservative (i.e., lower temperatures).

2. The effect of a possible buoyant plume (warm boundary layer) next to the vessel wall can conceivably shield the vessel wall from direct contact with cold SI fluid. The extent of this plume effect has not been fully investigated, but is expected either to reduce or to not affect the risk of significant flaw extension.

3. In the present mixing cup model, loop to loop fluid mixing in the downcomer annulus region is not considered. For asymmetric stagnant loop transients (e.g., SD, SGTR), this nonmixing assumption will result in a more conservative estimate of fluid temperatures (i.e., lower temperatures) in the affected loop. If a 3-D mixing model is used, it is anticipated that some mixing in the downcomer annulus region between the affected and non-affected loops will be present. The size of the mixing cup model volume used in this analysis is conservatively small. Based on the current mixing volume selection methodology developed in Reference [4], the volume used for the mixing cup calculation could be increased to include part of the lower plenum and the horizontal section of the loop seal.

4. Because of the limited number of PTS scenarios that were analyzed, the stylized transients selected to represent the PTS bins generally result in a higher predicted frequency of significant flaw extension, for a particular PTS category than if a finer classification of the PTS bins had been applied.

5. The assumptions used in the T&H analyses for the different transients are generally biased to give a conservative low prediction of the downcomer fluid temperatures (e.g. max SI, max AFW, and early RCP trip). Incorporating best estimate assumptions can provide some positive benefit in the estimated risk of vessel failure.

6. The value of the RWST temperature used was conservatively low for the transient T&H analyses (70°F vs. 90°F for H. B. Robinson) and for the mixing cup calculations (80°F vs. 90°F).

7. The transient results input to the mixing cup and the fracture mechanics analyses are generally obtained from the T&H analyses for generic Westinghouse plants. While HBR T&H characteristics are similar to typical Westinghouse plant, there are some uncertainties as to the degree of accuracy in applying the generic results to HBR. However, the degree of uncertainty, in all cases, is relatively minor and should not affect the conclusion of this report.

V.4.3 <u>Probabilistic Fracture Mechanics Analysis</u>

1. A heat transfer coefficient ~ 300 Btu/hr-ft^2-°F, which was judged to be appropriate for use with stylized transients that are representative of stagnant loop conditions (i.e., free convection heat transfer), has been used in the PFM evaluation of the set of PTS transient characterizations. This assumption will be nonconservative for some transient scenarios where the reactor coolant pumps continue to run and forced convection is the prevailing heat transfer mechanism. However, these scenarios are not the dominating events in the risk assessment (primarily because the T_f is relatively high and β is relatively small).

2. The effect of the stainless steel cladding has not been considered in the PFM evaluation. The clad on the inner surface of the reactor vessel can produce both positive and negative effects on the prediction of flaw initiation. The clad may provide an effective increase in flaw initiation fracture toughness because of its relatively high fracture toughness causing the clad to constrain flaws in the vessel material. The residual stresses that result from the application of the clad may adversely affect the stress intensity or driving force at the crack tip.

3. The number of transient cases evaluated using PFM was limited to 4 representative cases at 1000 psi and 1 case, at a higher pressure reflecting the HBR SI pump shutoff head. This limits the number of T_f values considered as well as the number of cooldown rates. Although the time duration and time step intervals for the transient cases that were evaluated are consistent with those used by the NRC,

some uncertainty exists relative to the applicability of these values for the circumferential flaw orientation. This uncertainty exists because flaw arrest is more important than flaw initiation in the generation of the conditional probability results for HBR.

4. The use of an infinitely long, continuous flaw for both crack initiation and arrest in the PFM analysis is a conservative approach. This is because other than infinitely long arresting cracks, e.g., those with an aspect ratio up to 20:1, finite flaw for arrest may be justified for use in some situations yielding a further benefit in the risk of significant flaw extension. This is fully discussed in Appendix C.

5. The OCTAVIA flaw distribution that has been used in the PFM analyses is more conservative than the Marshall distribution, which is currently being recommended for use. Incorporating flaw non-detection probabilities to account for "non-destructive examination reliability" could result in a neutral effect or reduction in the risk of significant flaw extension.

6. The largest uncertainty in the PFM analysis is in the distributions used to represent the parameters affecting RT_{NDT} and to define the material properties K_{Ic} and K_{Ia}. Until further statistical work has been completed, it is unknown if the impact will be positive or negative.

7. Use of LEFM in the PFM analysis is generally conservative when compared to results generated by elastic-plastic fracture mechanics However, the level of conservatism is uncertain at this time particularly with respect to what consitutes "vessel failure".

8. Benefits of warm prestressing effects, which have not been considered in this PFM analysis, will apply to some transient scenarios. The benefit could be a reduction of several orders of magnitude in risk for some transients. Consequently, it will have a positive benefit on the total PTS risk of significant flaw extension.

9. In the evaluation of flaws in welds, weld residual stresses that may exist could have a positive or negative impact on the risk of vessel failure although the actual effect is unknown at this time.

10. The HBR analysis employed the "Monte Carlo" simulation process modified by the "importance sampling technique". Results generated with this modification checked reasonably well with the values obtained from an analysis using an unmodified Monte Carlo analysis for two cases where some comparison could be made. However, this does not insure that biases do not exist when using the importance sampling technique in all situations.

SECTION VI
CONCLUSIONS AND RECOMMENDATIONS

VI.1 CONCLUSION

The primary conclusion of this analysis is that there is sufficiently strong justification to eliminate Pressurized Thermal Shock as a Safety Concern for H. B. Robinson. On the basis of plant specific analysis and CP&L plant specific programs to reduce the risk from Pressurized Thermal Shock, continued operation of H. B. Robinson through end of life will have an acceptably low level of risk. In addition, this evaluation shows that H. B. Robinson compares favorably with other plants in that the risk is lower than that accepted by the NRC for generic acceptance criterion. This conclusion is supported by the results discussed in Section V.

As previously stated in Section V, this study shows that:

1. For H. B. Robinson the applicable screening criterion should be an RT_{NDT} of about 340°F.

2. Conclusion 1 results primarily from consideration of a circumferentially oriented flaw using analysis techniques consistent with those used by the NRC to support the 270°F criterion for axially oriented flaws.

3. At an RT_{NDT} of 300°F, the risk of H. B. Robinson vessel failure from PTS is about an order of magnitude lower than the acceptable risk level used to develop the NRC screening criteria.

4. The end-of-plant life estimated frequency of significant flaw extension for H. B. Robinson, 7×10^{-7} occurrences per reactor year, is lower than the proposed core melt melt safety goal by at least two orders of magnitude. Thus, the risk from PTS at H. B. Robinson is approximately 1% of the proposed acceptable core melt risk even without considering the conditional probability of core melt given significant flaw extension.

5. Based on the above conclusions, Pressurized Thermal Shock is not a safety concern for the H. B. Robinson Plant.

VI.2 RECOMMENDATIONS

It is recommended that this study be used to:

1. Provide part of the basis for continued operation of the H. B. Robinson Plant.

2. Assess results of other PTS studies currently underway such as the USI A-49 effort.

3. Provide information and training material for plant operators as to what transients will result in higher PTS risk and why.

4. Provide an information base to generate effective procedures for dealing with high PTS risk transients.

5. Assess relative benefit of proposed modifications to mitigate PTS.

SECTION VII

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SECTION VIII
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APPENDIX A

GENERAL PTS RISK ASSESSMENT METHODOLOGY

A.1 INTRODUCTION

The general PTS risk assessment methodology employed by Westinghouse is described in this appendix. In Stage 1, as described in the main body of this report, the first step in the approach is the event tree analysis (ETA) which is used to determine the frequency of occurrence for each event. Thermal-hydraulic analysis (T&H) is then performed for these events. Given the thermal-hydraulic characteristics, the conditional probabilities of significant flaw extension in the reactor vessel are determined using probabilistic fracture mechanics (PFM). Finally the PTS risk associated with each event is determined by synthesizing the results from the ETA, T&H and PFM.

The PTS risk assessment methodology that is described in this report for Stage 1 results in the identification of a mean PTS Risk as a function of mean surface RT_{NDT} for the H.B. Robinson Unit 2 pressurized water reactor. In addition, specific cooldown scenarios that contribute significantly to the total risk are identified. Further evaluation of these transients are performed in Stage 2 to determine the contribution of each to the overall risk.

Because literally thousands of postulated cooldown scenarios must be considered, it is essential that various approximation techniques be employed when performing the PTS risk assessment. However, it is appropriate to employ detailed models to further analyze the relatively small number of cooldown scenarios identified from the PTS risk assessment as being significant. This consideration is addressed in Appendix B. The Stage 2 detailed analytical models yield a similar total PTS risk to that which the Stage 1 assessment methodology described herein identifies. Section V.2 of the main report provides a discussion of the margins and uncertainties in this PTS Risk Assessment.

A.2 EVENT TREE ANALYSIS (ETA) (STEP 1)

The first step in the PTS risk assessment is to identify, using event tree analysis, the broad categories of events that could potentially result in a pressurized thermal shock of the reactor vessel. This categorization technique facilitates accumulation of the results and allows orderly presentation. A PTS category is a group of cooldown events caused by a common mechanism such as a primary system boundary rupture. The criteria that are used to identify the PTS transient categories documented in this report are:

(1) One of the following conditions must exist in a PTS scenario within the category:

- o Sustained loss of primary coolant
- o Sustained loss of secondary coolant
- o Sustained injection of SI into a stagnant RCS loop
- o Sustained excessive main or auxilary feedwater flow to one or more steam generators.

(2) The PTS risk associated with the category (summed for individual events) is greater than 10^{-6} OCC/R-YR, using conservative assumptions discussed in Appendix B.

The three primary transient categories, which have been chosen using these criteria and which are considered in this analysis, are listed in Table A.2-1. These categories have been selected using experience based upon results from earlier analyses [A.1, A.2, A.3]. However, a further detailed assessment of other possible contributions to PTS has been recently completed for the WOG in Reference [A.4].

In addition to the more dominating PTS transient scenarios given above (i.e., SD, LOCA, and SGTR), that were also considered in the NRC PTS evaluation [A.5], the NRC identified certain events where primary coolant loop flow is lost, including single or multiple failure situations, that may warrant consideration in a detailed PTS evaluation. This concern arises because once

TABLE A.2-1
PTS Transient Categories Addressed*

Category (Abbreviations)	Description	Examples
1. Secondary Depressurization (SD)	Sustained Excessive Steaming Of One Or More Steam Generators (Except Feedline Break)	Steamline Breaks Stuck Open Secondary Valves Reactor Trip Without Turbine Trip Steam Dump Control System Failures
2. Loss Of Coolant Accident (LOCA)	Sustained Loss Of Primary Coolant (Except SGTR and LOHS)	Primary Piping Breaks Stuck Open Primary Valves Control System Failures RCP Seal Failures
3. Steam Generator Tube Rupture (STGR)	Rupture Of Steam Generator Tube	SGTR

*Addressed by this study based on the 1982 generic evaluation [A.1, A.2] using a conservative criteria for category selection.

natural circulation loop flow is lost, the Reactor Cooling System (RCS) cooldown will be accelerated as a result of the smaller mixing volume available to the safety injection flow. In the case of an event where all reactor coolant pumps are tripped, natural circulation flow, which will normally develop in the RCS in at least one steam generator, is available to transfer decay heat out of the core. Natural circulation flow through each loop is generated by the thermal head resulting from the density difference between the hot and cold sides of the RCS. Mechanisms, which significantly alter the necessary temperature distribution or pressure drops through a loop, have the potential to impede flow in one or more loops. Study of these mechanisms, which lead to stagnant loop conditions, has led to the identification of four additional transient categories that were evaluated in Reference A.4 but have not been explicitly considered in this HBR risk study. Results from the detailed assessment in Reference A.4 are discussed in Appendix B in relation to this H. B. Robinson analysis.

PTS categories are further subdivided by such parameters as break size, decay heat levels, location of breaks and various operator action times. Each of the subdivisions are defined as a PTS "bin". Binning offers greater resolution and accuracy to the risk assessment by grouping similar scenarios within a PTS category. Each bin is described by a combination of ranges in certain parameters that are influential to the severity of the PTS category. A frequency of occurrence is associated with each bin. The sum of the bin frequencies is equal to the category frequency. Tables A.2-2 through A.2-4 show the parameters which are used to segregate the categories into bins.

A great deal of insight into the sensitivity of the analysis to Event Tree, Thermal and Hydraulic, and Probabilistic Fracture Mechanics parameters is required when binning a PTS category. Some of the considerations are listed below:

(1) Minimizing the number of bins simplifies the analysis but in general increases uncertainty and conservatism. A large number of bins is most appropriate for the dominating PTS risk categories.

TABLE A.2-2

SECONDARY DEPRESSURIZATION (SD)
BIN PARAMETERS

	1	2	3	4	5
(DH) Decay Heat Level (Percent of Rated Power)	DH > 1	.5 < DH < 1	DH < .5		
(S) Size of Equivalent Break Area (ft^2)	S < .11	S > .11			
(L) Location of Equivalent Break	Downstream of MSIVs	Between MSIVs and SGs			
(OA) Time to Terminate Auxiliary Feedwater to the Affected Steam Generator (min)	OA=5	OA=10	OA=20	OA=60	OA=Indef.

TABLE A.2-3

LOSS OF COOLANT ACCIDENT (LOCA)
BIN PARAMETERS

	1	2	3
(DH) Decay Heat Level (Percent of Rated Power)	DH > 1	.5 < DH < 1	DH < .5
(S) Size of Pipe Diameter with Equivalent Break Area (ft^2)	S < 1.5	1.5 < S < 6.0	S > 6.0
(L) Equivalent Break Location	Hot Leg	Cold Leg	Elsewhere
(OP) System Pressure During Period of Flaw Extension (PSIA)	OP < 1000	1000 < OP < 2000	OP > 2000

TABLE A.2-4
STEAM GENERATOR TUBE RUPTURE (SGTR)
BIN PARAMETERS

	1	2	3
(DH) Decay Heat Level (Percent of Rated Power)	DH > 1	0.5 < DH < 1	DH < o.5
(S) Size Equivalent Number Of Tubes	S ≤ 1	S > 1	
(OSI) Time To Terminate SI After Criteria Are Met (MIN)	OSI < 3	OSI > 3	
(OR) RCP Status	Always Running	Not Running	

(2) The ris. of significant flaw extension associated with a bin frequency is found by multiplying the bin frequency by the conditional probability of significant flaw extension associated with the stylized transient characteristics of the scenario (using the mean parameters of the bin). For example, the stylized transient characteristics that are determined for small secondary depressurization bins are

- o Decay Heat Level as a percent of rated power.
- o Size of break (eg .11).
- o Operator action time to terminate AFW to the faulted SG.
- o Operator action to keep pressure below a given pressure during the period of potential flaw extension.

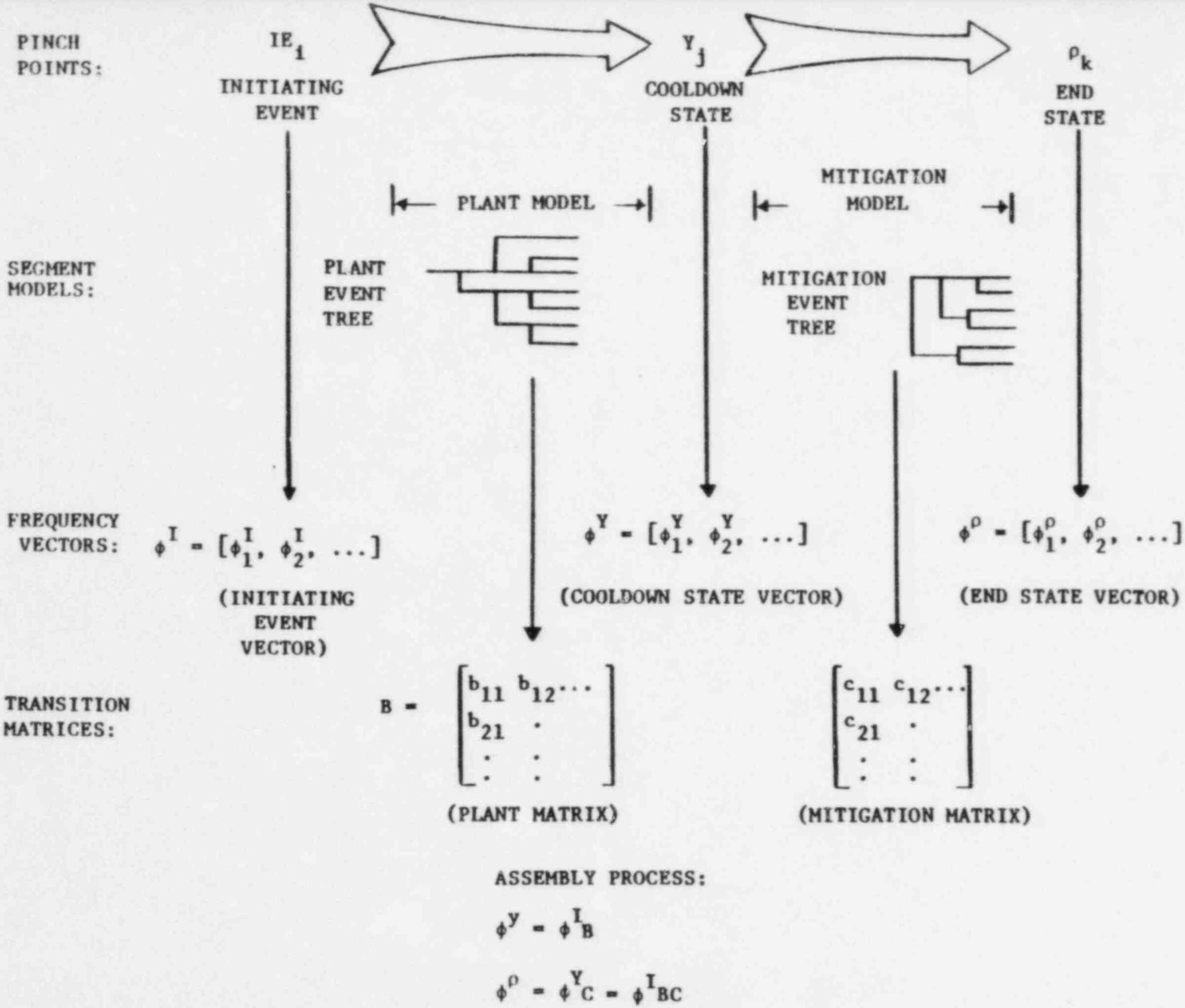
Thus, the bins must be selected so that the calculated risk for the bin is meaningful i.e., not overly conservative.

(3) Only the parameters, which have a significant influence on the severity of a PTS category, are selected for binning in order to simplify the analysis.

(4) For the remaining parameters appropriate values are assumed. The values assumed are based on judgment since bounding values are difficult to define. This is because the choice of values may affect the results in different ways in different parts of the analysis.

Many insights were gained from the extensive generic studies performed for the Westinghouse Owners group during 1982 [A.1, A.2] in which this methodology evolved. Further insights have also recently been gained from the WOG PTS study in Reference [A.4]. These methods have been generally accepted by the industry and the NRC [A.5] and are considered applicable to the H.B. Robinson specific study.

The event tree approach displayed in Figure A.2-1 is used to identify the transient frequencies for each of the bins within the categories. The frequencies depend upon both independent occurrences of these transients as



OVERVIEW OF THE ASSEMBLY PROCESS SHOWING RELATIONSHIP OF PINCH POINTS, FREQUENCY VECTORS, EVENT TREES, AND TRANSITION MATRICES

Figure A.2-1

initiating events and consequential occurrences from other events that are not by themselves cooldown transients. These "other transients" may be transformed into PTS transients as a result of plant system malfunctions or operator error.

Event Frequencies

Significant initiating events and the associated frequencies, which are representative of U.S. PWR operating experience, are shown in Table A.2-5. This set of initiating events is applied to the plant response event tree with four possible outcomes for each initiating event: 1) a non-PTS initiator remains a non-PTS event, 2) a non-PTS initiator is transformed into a consequential PTS transient because of system malfunctions, 3) a PTS initiator remains as is, and 4) a PTS initiator is transformed into a more severe PTS transient because of system malfunctions.

By using the plant event tree, the total frequency of occurrence of these four possible outcomes is determined. There are two basic steps in this process. The first step is to determine the probability of plant systems availability and malfunction for each possible sequence by using the event tree for each initiating event. The output of the model is a set of cooldown states of varying degrees of severity and frequencies resulting from the possible sequences of plant response to all expected initiating events. In the second step, these plant cooldown sequences are further broken down by levels of decay heat expected during the initiating event, again based on US PWR operating experience. Decay heat level is singled out since it has a significant effect on the resulting cooldown transient: high decay heat levels inhibit most cooldown transients from reaching low enough temperatures to produce a PTS challenge.

After the frequency of occurrence is defined, the events are further evaluated to consider recovery possibilities. The end result from the above cooldown evaluation (the cooldown state vector) is further mitigated by the effect of automatic or operator actions to terminate or lessen the effect of cooldown sequences. The overall result of the event tree approach is a set of end states of the potentially unmitigated PTS scenarios, grouped by common end state categories and decay heat levels, and the associated frequencies of occurrence.

TABLE A.2-5

Initiating Event Frequencies [A.1]

	Event	Frequency (/R-Year)
1.	Loss of Main Feedwater	3.41
2.	Closure of one Main Steam Isolation Valve	6.00×10^{-1}
3.	Loss of Primary Flow	3.21×10^{-1}
4.	Core Power Increase	4.77×10^{-2}
5.	Turbine Trip	4.00
6.	Spurious Safety Injection Activation	1.59×10^{-1}
7.	Reactor Trip	4.11
8.	Turbine Trip due to Losses of Offsite Power	1.01×10^{-3}
9.	Steam Generator Tube Rupture	3.92×10^{-2}
10.	Small LOCA, < 1.5 in. diameter	9.07×10^{-3}
11.	Small LOCA, > 1.5 in. and { 6 in diameter	6.11×10^{-4}
12.	Large LOCA, > 6 in. diameter	3.88×10^{-4}
13.	Excessive Main Feedwater	2.50×10^{-1}
14.	Steamline Rupture Inside Containment	3.88×10^{-4}
15.	Steamline Rupture Outside Containment	3.87×10^{-2}

A.3 THERMAL-HYDRAULIC ANALYSIS (T&H) (Step 2)

The next step in the assessment is to associate thermal-hydraulic characteristics with each PTS bin (i.e., the end states). Because there can be literally thousands of end states and since only a small number prove to be of any practical concern, it is prudent to use simple approximations at this point. Using sensitivity studies and judgement based on experience, each end state is fit with a simple temperature versus time exponential curve, and a pressure is selected that is representative of the system pressure during the period of potential flaw extension. These approximations allow any cooldown transient to be characterized by three quantities; a final temperature (T_f) reflecting the extent of the cooldown, a time constant (β) reflecting the rate of the cooldown, and a characteristic pressure (p). In cases where the actual transient has fluctuations, the approximations are conservatively applied.

A.4 PROBABILISTIC FRACTURE MECHANICS ANALYSIS (PFM) (Step 3)

The third step is to quantify the conditional probability of significant flaw extension given that an exponential cooldown occurs. Data are generated from probabilistic fracture mechanics (PFM) analyses using the Monte Carlo technique. A matrix of cases for given transient characterization quantities and vessel lifetimes are evaluated to obtain the data. For each case, a large number of deterministic fracture mechanics analysis trials are simulated using random values that are selected by a random generator from distributions defined for the pertinent input properties. The input properties, which are treated as random variables, include: initial crack depth, initial RT_{NDT}, copper content, fluence, and the critical stress intensity values for flaw initiation and arrest. The conditional probability of significant flaw extension for each case is determined by dividing the number of failures by the number of trials. Conditional probability curves are plotted from the matrix of results in terms of the transient characterization quantities and as a function of vessel lifetimes.

A.5 SYNTHESIS OF ETA, T&H, AND PFM INTO PTS RISK (Steps 4 through 8)

The final step in the PTS Risk Assessment is to associate a frequency of significant flaw extension (F_{SFE}) with each end state by multiplying the frequency of the end state itself by the conditional probability of significant flaw extension given that the scenario has occured. If the probabilities of significant flaw extension that are associated with the end states are summed together, the total risk of significant flaw extension from PTS can be ascertained as a function of RT_{NDT}, and dominating transient scenarios can be identified for further evaluation if the risk is near the frequency of regulatory concern at the vessel age of interest. The total PTS risk and the risk associated with each cooldown scenario evaluated for the HBR study are provided in Appendix B.7, as a function of plant RT_{NDT} values.

A.6 DETAILED ASSESSMENT OF TRANSIENT SCENARIOS THAT DOMINATE TOTAL RISK (Steps 9 through 12)

Detailed thermal-hydraulic and deterministic fracture mechanics analyses are then performed on the small group of selected transients. The results are plotted on a graph where the abscissa is the lowest age at which significant flaw extension is predicted to occur for a specific transient. The ordinate is the frequency of occurrence of the transient scenario. The total PTS risk plot is constructed by simply adding together the frequencies associated with each scenario that causes significant flaw extension at a given vessel age. If the total PTS risk plot extends into a region of regulatory concern, some modifications can be examined on a cost benefit basis to determine if an acceptable result can be achieved.

The PTS safety goal, which is used in the rigorous transient specific analysis, will be different in magnitude from that used in the broad PTS risk assessment. Although in theory, both PTS safety goals represent an identical status of reactor vessel integrity.

Evaluation of Modifications (If Necessary) (Steps 13, 14)

There are three general ways of affecting the PTS risk associated with a specific scenario. The frequency of the scenario can be reduced through the use of system or man-machine interface modifications. For example, installation of a block valve upstream of the secondary power operated relief valves might reduce the frequency of non-isolable small secondary depressurizations. The severity of the transient, if it occurs, can be reduced through the use of system modifications or man-machine interface improvements. Heating of emergency core cooling water is an example of such a modification. Finally, methods such as flux reductions may be employed to decrease the reduction in fracture resistance as the vessel ages through plant life (the vertical boundary of the region of regulatory concern shifts to the left).

A cost-benefit analysis can be performed on individual modifications or combinations thereof, that are found to affect PTS risk. In this way, the most cost effective method of achieving PTS goals can be selected.

A.7 REFERENCES

A.1 "Summary of Evaluations Related to Reactor Vessel Integrity", report performed for the Westinghouse Owner's Group, Westinghouse Electric Corporation, Nuclear Technology Division, May, 1982.

A.2 Letter from O. D. Kingsley, WOG to Dr. Denton, NRC, "Westinghouse Owner's Group Activities Related to Pressurized Thermal Shock", OG-73, July 15, 1982.

A.3 Meyer, T. A., "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants", WCAP-10019, December 1981.

A.4 Ackerson, D. S. et al., "A Generic Assessment of Significant Flaw Extension, Including Stagnant Loop Conditions, From Pressurized Thermal Shock of Reactor Vessels on Westinghouse Nuclear Power Plants", WCAP-10319 Draft.

A.5 NRC Policy Issue, Enclosure A, "NRC Staff Evaluation of Pressurized Thermal Shock", SECY-82-465, November 23, 1982.

APPENDIX B

TECHNICAL DESCRIPTION OF THE APPLICATION OF THE METHODOLOGY TO H. B. ROBINSON

B.1 INTRODUCTION

This appendix provides a detailed description of the considerations, technology and direct applications in the H. B. Robinson evaluation. In this introduction, a perspective of the contents of the appendix and the extent of work performed for this study are given.

The H.B. Robinson analysis implements results from generic studies that are deemed applicable. Figure B.1-1 shows the relationship of the H. B. Robinson analysis to these current generic PTS evaluations.

The generic analyses performed for the Westinghouse Owners Group utilized a typical three loop Westinghouse PWR system model. The H.B. Robinson three loop system and its operation, including use of the Westinghouse Owners Group sponsored Emergency Response Guidelines [B.5], is similar to the generic model. Section 2 of this appendix specifies the three major differences between the H.B. Robinson and the generic plant systems.

From the three sections of the PTS risk assessment: 1) PRA - establish the transient scenarios and frequencies; 2) T&H - determine the transient characteristics of final temperature, cooldown rate and pressure; 3) PFM - determine the conditional probabilities of significant flaw extension, generic results were used where applicable. The following three paragraphs describe where generic and plant specific analyses were used in each case.

First, the transient scenarios and frequencies are based on the generic evaluation [B.1, B.4]. They are assessed to be applicable to H.B. Robinson as described in Section 3 of this appendix: Event Sequence Analysis with the HER Plant Model.

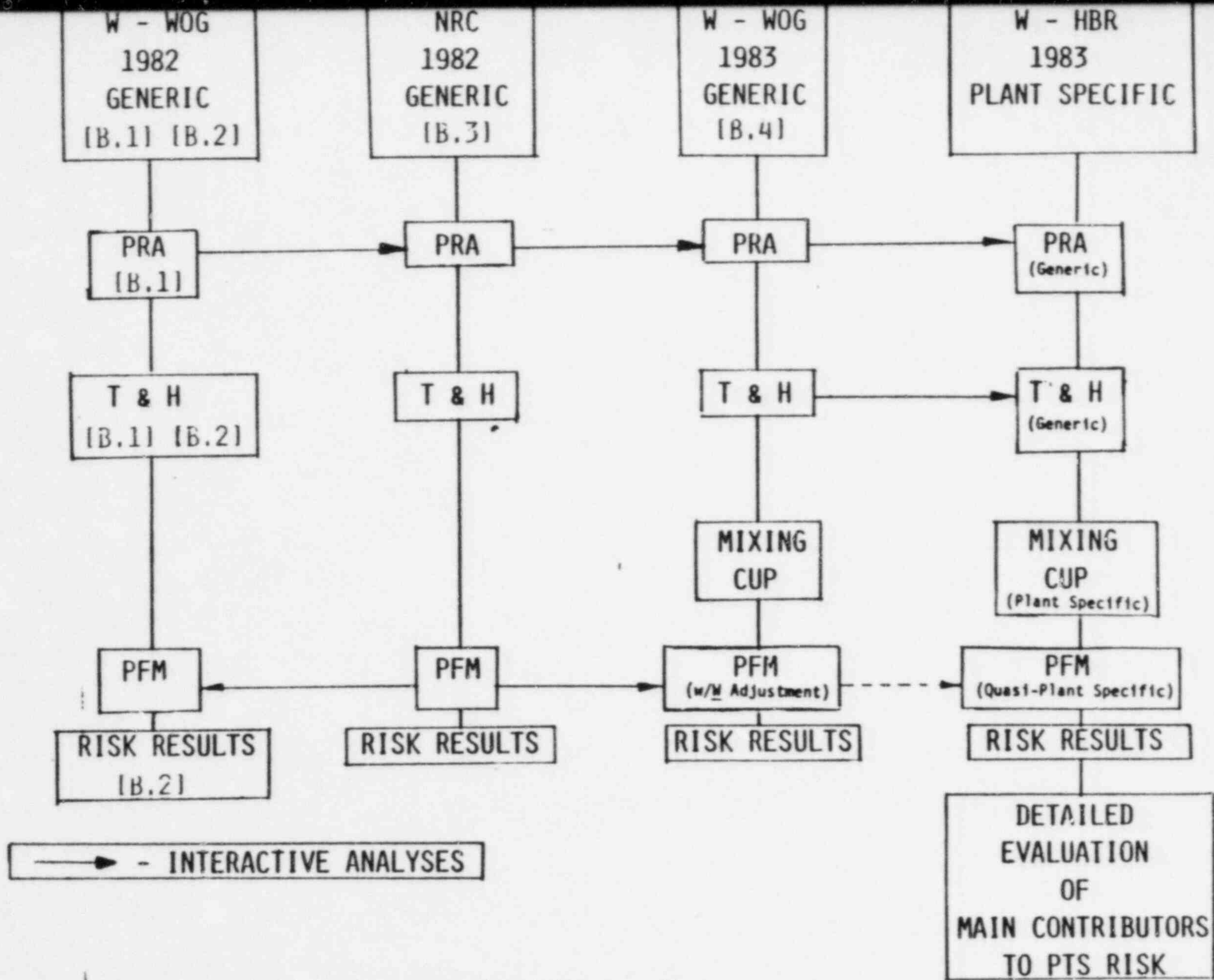


FIGURE B.1-1: RELATIONSHIP OF HB ROBINSON RISK STUDY TO CURRENT GENERIC PTS EVALUATIONS.

Second, the transient characteristics (temperature, cooldown rate and pressures) were evaluated partially on a generic basis and partially on an HBR specific basis since the evaluation is essentially a two part process. The first part utilized generic results and the second part is based on H. B. Robinson specific characteristics. The two parts are 1) perform a system thermal and hydraulic computer analysis to determine the transient characteristic, and 2) use these transient characteristics in a mixing evaluation to determine the temperature at the midplane of the downcomer of the reactor vessel. The part 2 mixing calculation is required since, for some of the limiting PTS transients, one or more coolant loops are expected to be stagnant (that is, to have no coolant flow). This additional downcomer cooling comes from the relatively cool SI water being injected into the loop during "no flow" conditions, which are postulated to exist in these accident transient scenarios. Tests (Creare [B.6]) have shown that this cool water mixes with the hot water already in the vessel downcomer and piping. Thus, part 2 is performed using the amount of hot water that mixes with the cooler SI water to obtain the temperature of the water in the vessel as a function of time during the transient. For the H.B. Robinson evaluation, the generic results [B.4] of part 1 were used and a plant specific evaluation of part 2 was performed for each of the transient categories analyzed: secondary depressurization, LOCA and steam generator tube rupture. Section 4 of this appendix discusses the generic evaluations of part 1 for the three transient categories evaluated. Section 5 discusses part 2 for the H. B. Robinson specific mixing cup model.

The probabilistic fracture mechanics analysis is then performed for the H.B. Robinson vessel. The PFM analyses are performed for two reasons. First, the generic results for longitudinal welds are not directly applicable to the evaluation for the circumferential weld of concern for H.B. Robinson and second, several other factors were taken into account that were not considered in the generic evaluation. Some of these factors "improve" the results, while others may have a negative impact. Details of these considerations are in Section 6 of this appendix. The H.B. Robinson specific PFM analysis was performed using the W PFM computer code. Before the above mentioned factors and circumferential orientation were taken into account, the W PFM and the NRC PFM computer codes were benchmarked and the results compared well.

The combination of the three PTS risk assessment components (PRA, T&H and PFM) and the overall results that define the PTS risk are discussed in Section 7 of this appendix.

Specific deterministic evaluations were then performed on the major PTS contributors for H. B. Robinson. The analyses, as well as the significance of these analyses as an aid to decision making, are described in Section 8 of this appendix.

B.2 HBR SYSTEM DIFFERENCES

The following significant differences exist between the H.B. Robinson Plant design and the generic plant that was used in the WOG PTS risk analysis [B.1, B.4].

(1) The shutoff head of the SI pumps, 1,450 PSIG, is lower than that of the generic plant, 2,500 PSIG.

(2) The frequency of non-isolable small secondary depressurization is somewhat increased because the steam dump control system drives secondary power operated relief valves which are upstream of the main steam isolation valves.

(3) The RWST temperature, 90°F, is higher than that of the generic plant.

B.3 EVENT SEQUENCE ANALYSIS WITH HBR PLANT MODEL (Step 1)

The derivation of the event sequence frequencies was based in general upon the models and results reported in the Westinghouse Owners Group reports [B.1, B.4]. The basis for this decision is that the generic plant analyzed in these references (a generic three loop plant) is similar to the H.B. Robinson plant design. The only significant design difference from the point of view of the plant model is the use of the secondary power operated relief values for steam dump as noted in Section B.2. This difference was not evaluated since it effects the secondary depressurization and it became apparent that it would neither affect the total PTS risk nor change the significant contributors to PTS risk.

The generic event sequence analyses were judged to be applicable to the H.B. Robinson plant. The generic initiating event frequencies were based on U.S. PWR operating history, of which H.B. Robinson is part. H.B. Robinson does not have differences significant enough from the generic model to change the major conclusions of this report. The generic consequential failure models (event trees) and component failure rates are unchanged in the H.B. Robinson evaluation. These unchanged models include the consequential small LOCA's attributable to failed pressurizer PORVs; excessive feedwater from failed main feedwater control system failures; secondary depressurization from failed steam dump valves down stream of the MSIVs and the single steam generator tube rupture event. The generic breakdown of event categories by decay heat support states, or bins, was also judged to be applicable to H. B. Robinson.

The only significant departure from the generic analysis is the estimate of the initiating event frequency for a large secondary depressurization upstream of the MSIVs. The value for this event (1.7×10^{-4}) was taken from reference 3 and was analyzed for potential PTS risk as discussed in the following sections. This estimate is consistent with the generic frequency of large pipe breaks in the primary coolant loop and is judged to be conservative.

B.4 TRANSIENT CHARACTERIZATIONS WITH HBR PLANT MODEL (Step 2)

B.4.1 Secondary Depressurization (SD)

B.4.1.1 Transient Description

The SD category of cooldown transients includes steam dump malfunctions, steamline ruptures of all sizes, reactor trip without turbine trip, and control system failures, or operator errors that could result in any of these malfunctions. The transient is characterized by a rapid cooldown of the reactor coolant system (RCS) with a rapid primary depressurization until safety injection is actuated and a rapid primary repressurization occurs. The pressurizer empties in most cases but refills shortly after safety injection is actuated. Natural circulation and therefore good mixing conditions are maintained in the faulted loop in this transient in excess of 30 minutes unless low (< 0.5 percent) decay heat levels exist.

The specific SD scenarios, that are judged to be contributors to PTS risk, are analyzed. This judgement is based upon the results of references [B.1], [B.2] and [B.8], which show these scenarios to be dominant on a generic basis. These scenarios are defined using combinations of the following critical parameters:

1. Decay Heat Level
2. Operator Action Time To Terminate Auxiliary Feedwater To The Faulted Steam Generator
3. Equivalent Break Size
4. Operator Action To Control RCS Pressure

The transient characteristics of the SD scenarios were extrapolated from six computer simulations of a plant similar to H.B. Robinson. These six simulations were performed using the LOFTRAN Code. Using H. B. Robinson plant specific input, the Mixing Cup model computer code called MXGCUP was used to model stagnant loop conditions to predict the downcomer core midplane fluid temperature.

B.4.1.2 Initial Conditions And Assumptions for Secondary Depressurization

The transient temperature histories for the six LOFTRAN/MXGCUP simulations are shown in Figure B.4.1-1. A graph that is used for determining the quasi-equilibrium, which the RCS reaches after safety injection and auxiliary feedwater are terminated, is shown in Figure B.4.1-2. Listed below are the assumptions and initial conditions which were made on a generic basis and that were used for the six LOFTRAN simulations.

1) A constant decay heat level of 0.25, 0.75 or 1.50 percent of rated power was assumed for the three decay heat bins.

2) Equivalent break area equals 0.11 ft^2 (bounds the flow capacities of stuck-open relief, safety or steam dump valve) or 0.60 ft^2 (representative of a medium size break) for each size bin.

Figure B.4.1-1 TYPICAL COOLDOWN ASSOCIATED WITH A .11 AND .6 SQUARE FOOT EQUIVALENT STEAM BREAK ON A 3-LOOP PLANT. NO OPERATOR ACTION TO THROTTLE AUXILIARY FEED TO ANY STEAM GENERATOR

T_{COLD} AFFECTED LOOP (°F)

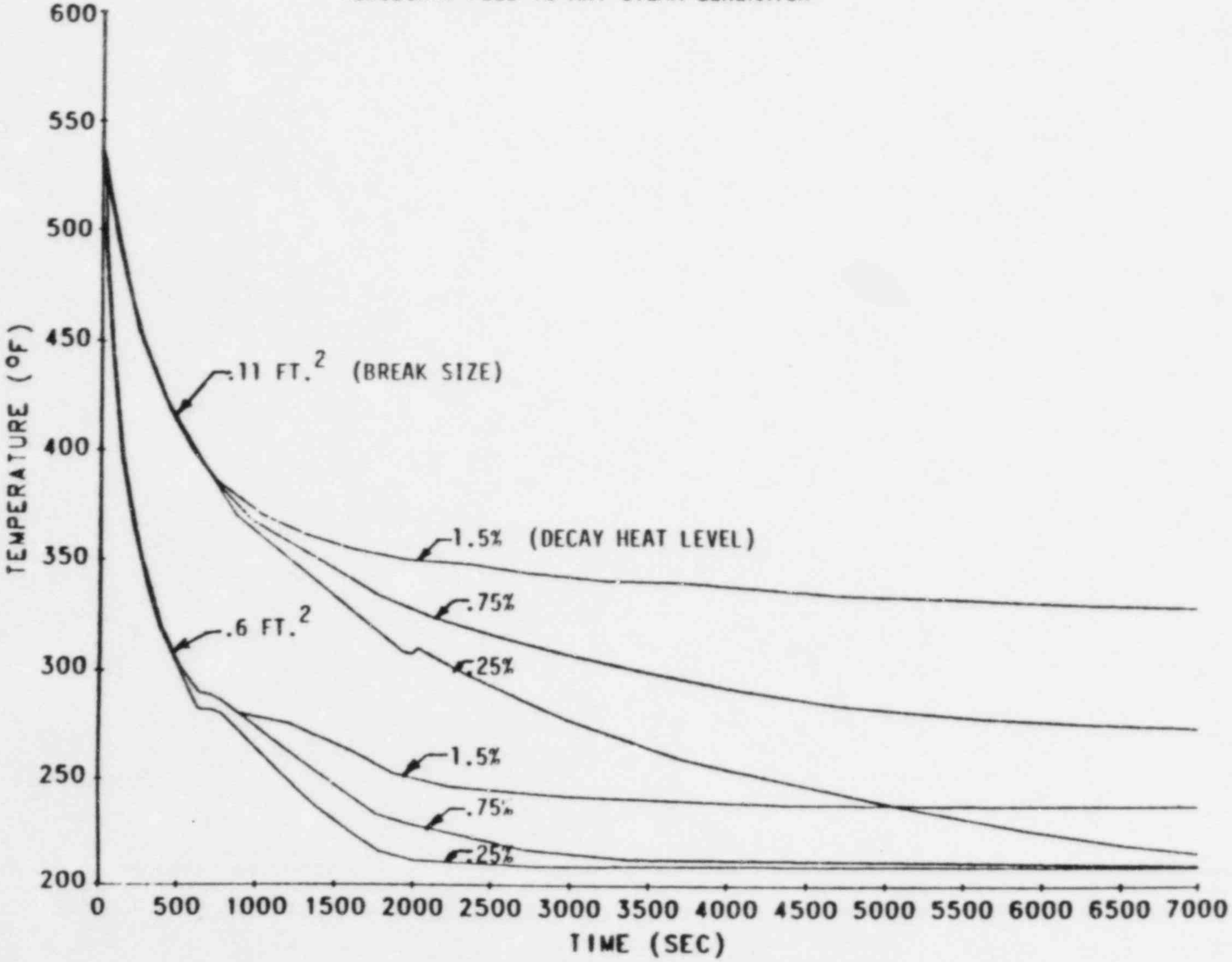
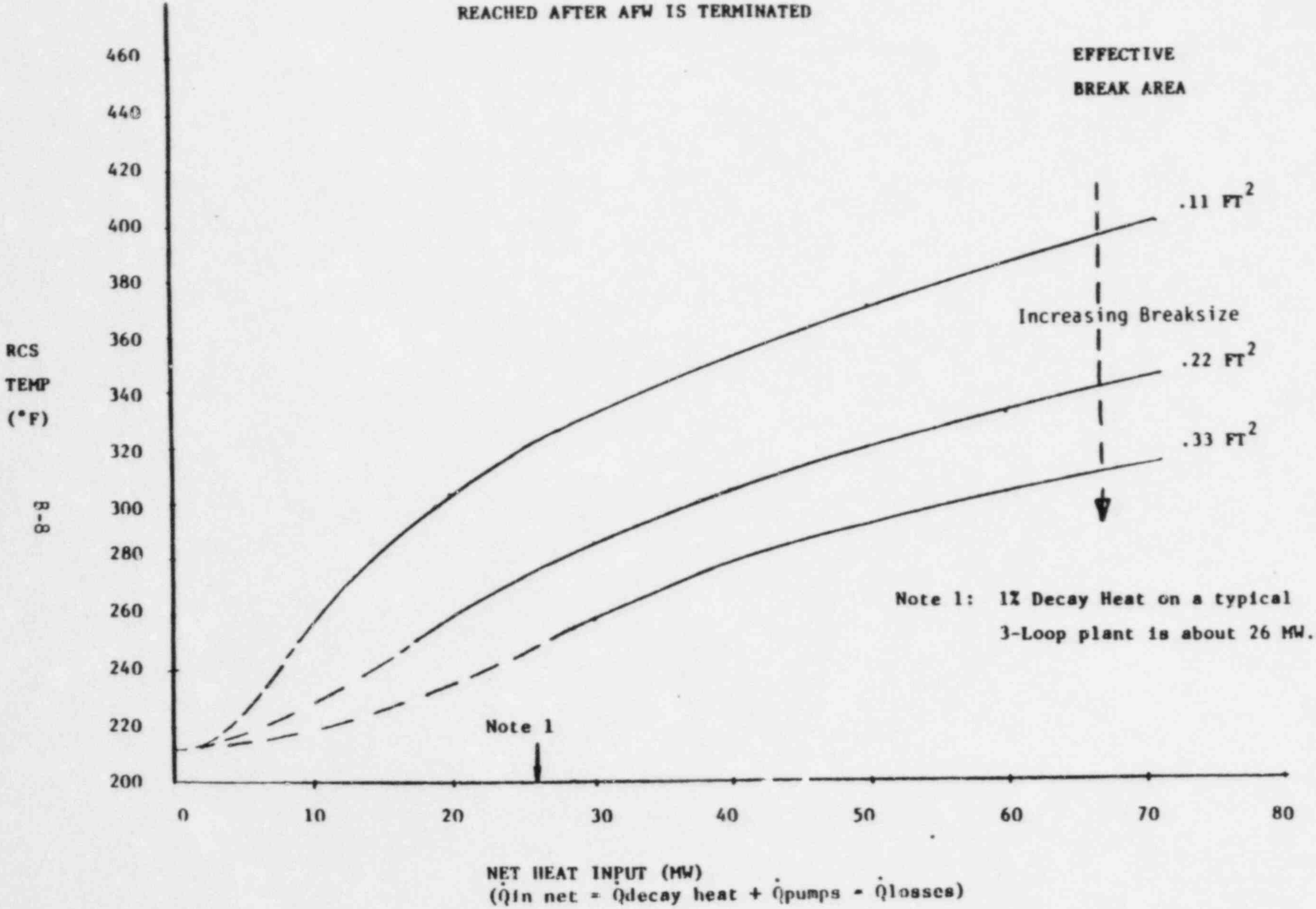


FIGURE B.4.1-2

QUASI - EQUILIBRIUM RCS TEMPERATURE

REACHED AFTER AFW IS TERMINATED



3) Initial condition of the plant is hot zero power with a zero power steam generator inventory in the two lower decay heat bins, and a full power inventory for the high decay heat bin.

4) All reactor coolant pumps trip immediately at start of transient.

5) Heat transfer from reactor plant thick metal sections to the reactor coolant is considered.

6) Steam generator reverse heat transfer from the secondary to the primary side is accounted for.

7) Steamline isolation occurs immediately at the start of the transient.

8) Safety injection initiates when low pressurizer pressure setpoints are reached. Full flow is assumed 10 seconds after the pumps start.

9) Auxiliary feedwater pumps start immediately when the SD is initiated. Both steam and motor driven pumps are available.

10) Auxiliary feedwater flow split between the steam generators is modeled based on the pump head-flow curves, system line resistances, and flow paths between the pumps and the steam generators.

11) Safety injection and auxiliary feedwater temperatures are assumed to be at 70°F.

12) Auxiliary feedwater to the affected steam generator is assumed to be terminated manually either at 5, 10, 20, 60 or an indefinite number of minutes for the AFW operator action bins.

13) Auxiliary feedwater to the non-affected steam generators are assumed to be terminated manually at 10 minutes, or upon achieving a steam generator level of 50 percent of narrow-range span, whichever is longer.

14) Safety injection is terminated when the pressurizer level reaches approximately 50 percent of span and all other SI termination criteria have been satisfied.

15) One charging pump is running at the start of the transient. The operator manually starts all remaining charging pumps after 10 minutes to restore pressurizer level to about 50 percent of span if required.

16) No return to power is permitted as a result of reactivity addition during a small secondary depressurization.

17) Letdown is isolated immediately at the start of the transient.

18) The LOFTRAN prediction of affected loop cold leg outlet temperature is indicative of the bulk fluid temperature in the vicinity of the reactor vessel downcomer at the core midplane except during the period when SI is on and the non-affected loop(s) is (are) stagnant.

19) During this period of uncertain mixing, the mixing cup model is applied to one of the stagnant loops to predict the bulk fluid temperature.

B.4.2 Loss of Coolant Accident (LOCA)

B.4.2.1 Transient Description

The LOCA category for cooldown transients for the primary system includes hot and cold leg breaks of various sizes. Within these different categories are RCP Seals, primary PORV or safety valve leakage or failure as well as actual primary piping leaks. The transient is characterized by a rapid cooldown and depressurization of the Reactor Coolant System. The main characteristic that distinguishes the smaller break hot leg LOCAs (less than and up to 1.5 inches in diameter) from larger size breaks is the maintenance of continuous flow in the loops and through the downcomer. The mechanism that drives this flow is the density difference due to the delivery of cold SI water to the cold leg and downcomer and heat addition in the core. As long as the SI flow matches

the break flow and keeps the steam generator tubes full of fluid, this flow will continue and help to distribute energy from the core and steam generator secondary side throughout the system. The result is a higher temperature in the downcomer than for a case where flow is interrupted and the downcomer is more effectively isolated from the core and steam generator energy.

The factors that affect the downcomer pressure and the rate of cooldown following a LOCA are

- o Break location
- o Break size
- o SI flow and temperature
- o Decay Heat
- o Secondary pressure (steam dump and feedwater)
- o Mixing of SI and RCS fluid

The following describes the rationale for break location , critical break size, and other model selection.

B.4.2.2 Model Description

The objective of the small LOCA transient analyses is to generate system transients that, for a given break size, minimize fluid temperature in the reactor vessel downcomer and maximize pressure in the Reactor Coolant System (RCS). The maximum temperature cooldown rate and minimum depressurization rate are desired. The input assumptions used in developing these transients are selected to insure maximum heat removal from the RCS. All systems that provide cooling to the RCS are assumed to operate at maximum capability, e.g., all trains of safety injection, all auxiliary feedwater pumps, etc. All water sources are at conservatively low temperatures.

The equilibrium version of the NOTRUMP code is used in this study to model the small LOCA. However, the downcomer fluid temperatures are modified using the mixing cup post-processor, MXGCUP, for the periods when the flow stagnates. The NOTRUMP evaluation was performed using a generic 3 loop plant with high head SI. The mixing cup is H.B. Robinson specific.

Modeling considerations include the following:

A) A reactor coolant pump (RCP) trip is assumed to occur coincident with reactor trip signal since this will maximize downcomer fluid temperature cooldown.

B) The system cooldown is dependent on the rates of energy and mass removal or addition. A hot leg break location is chosen because: 1) it insures that all SI flow is delivered to the downcomer, 2) it reduces primary loop flow required for break energy removal, 3) the cold leg remains filled with subcooled liquid, 4) it leads to the maximum cooldown rate for a given break size.

C) In Reference [B.8] a generic analysis of a spectrum of small loss of coolant hot leg break sizes (1 to 4 inches diameter) was performed to establish the critical break size for the study of vessel integrity for a generic Westinghouse PWR plant. Based on the results from this study the critical break size where loop flow first begin to stagnate for a 3 loop plant is between 1.4" to 2.0". For hot leg breaks smaller than 1.4" (including a stuck open PORV) the RCS maintains natural circulation. For the latter case a significant amount of energy is transfered to the downcomer fluid, consequently the fluid temperature is higher. Therefore the PTS risk from small LOCA is dominated by the break sizes larger than 1.4".

D) For the LOCA analyses, automatic no-load TAVG steam dump control and maximum auxiliary feedwater flowrates at a minimum Condensate Storage Tank (CST) temperature are assumed. The operator is assumed to control auxiliary feedwater to maintain steam generator water level in the narrow range span.

E) The role of the steam generator in maintaining continuous loop flow is minimal. The heat transfer from secondary to primary system is negligible compared to the core heat flow. The effect on the loop flow rate of operator action to throttle auxiliary feedwater is also negligible since

the steam generator does not provide enough thermal driving head to maintain the flow. This action does slow the secondary cooldown and increase heat flow into the primary. The operator action is expected to occur as outlined in the Emergency Response Guidelines (ERGs) [B.5] in order to prevent filling of the steam generator. Operator actions that reduce the secondary pressure would increase the cooldown rate but would have a favorable impact on natural circulation.

F) Similar to the steamline break transients, three ranges of decay heat level are identified (i.e., 0 - .5 percent; .5 to 1 percent and extended full power). After the cessation of natural circulation, the downcomer fluid temperatures are determined by the mixing cup model described in Section B.5. Heat transfer from the thick metal to the reactor coolant in the cold leg and downcomer section is considered. In addition, heat transfer from the core is considered for the high decay heat bin (a trip from extended full power operation) For the lower decay heat cases, the temperature differentials between the core and the downcomer fluid are much smaller, thus the small benefit derived is not included in the calculations.

A subclass of the small LOCA that warrants discussion is the isolatable LOCA scenario. The isolatable LOCA was shown in WCAP-10019 [B.8], on a deterministic basis, not to be a PTS concern and subsequent work in WCAP-10319 [B.4] has demonstrated this conclusion to still be valid.

For break sizes greater than 6 inches (large break), results from Reference [B.8] demonstrated that no significant contribution to PTS risk is involved because of the low pressure expected throughout a large LOCA accident and that the benefit of warm prestressing would be applicable for this event.

Based on these considerations the only break size range to be coupled with PFM analyses is between 1.4" and 6".

B.4.2.3 Assumptions and Initial Conditions

1. The significant generic plant parameters assumed include:

Plant Design	-	3 Loop W PWR	
Core Power	-	2298	MWT
Reactor Trip Setpoint	-	1860	PSIA
Main Feedwater Isolation	-	1860	PSIA
Max Auxiliary Feedwater	-	353	GPM/SG
RWST Temperature	-	40	°F
Condensate Store Tank	-	40	°F
No-Load TAVG	-	547	°F
SI Flow Rates	-	Figure	B.4.2-1*
Reactor Pump Trip Setpoint	-	1860	PSIA
SI Pump Trip Setpoint	-	1780	PSIA

2. The initial conditions assumed are Hot Zero Power for the two low decay heat bins and full power for the high decay heat bin. A constant decay heat level of .25 percent and .75 percent of rated power during the transients is assumed for the two low decay heat (0 to .5 percent, and .5 to 1 percent) bins. For the high decay heat bin, the ANS 5.1 Decay Heat Standard of 1979 [B.9] is used for best estimate of the decay heat release after a trip from extended full power operation.

3. The break is assumed to occur at time zero. The LOCA break area analyzed is equal to .049 ft^2 (~ 3" dia). This break size insures that a stagnant loop condition would occur.

4. All reactor coolant pumps are tripped after the pump trip setpoint has been reached.

5. Safety injection is initiated when associated setpoint is reached. No delay in the injection flow is assumed.

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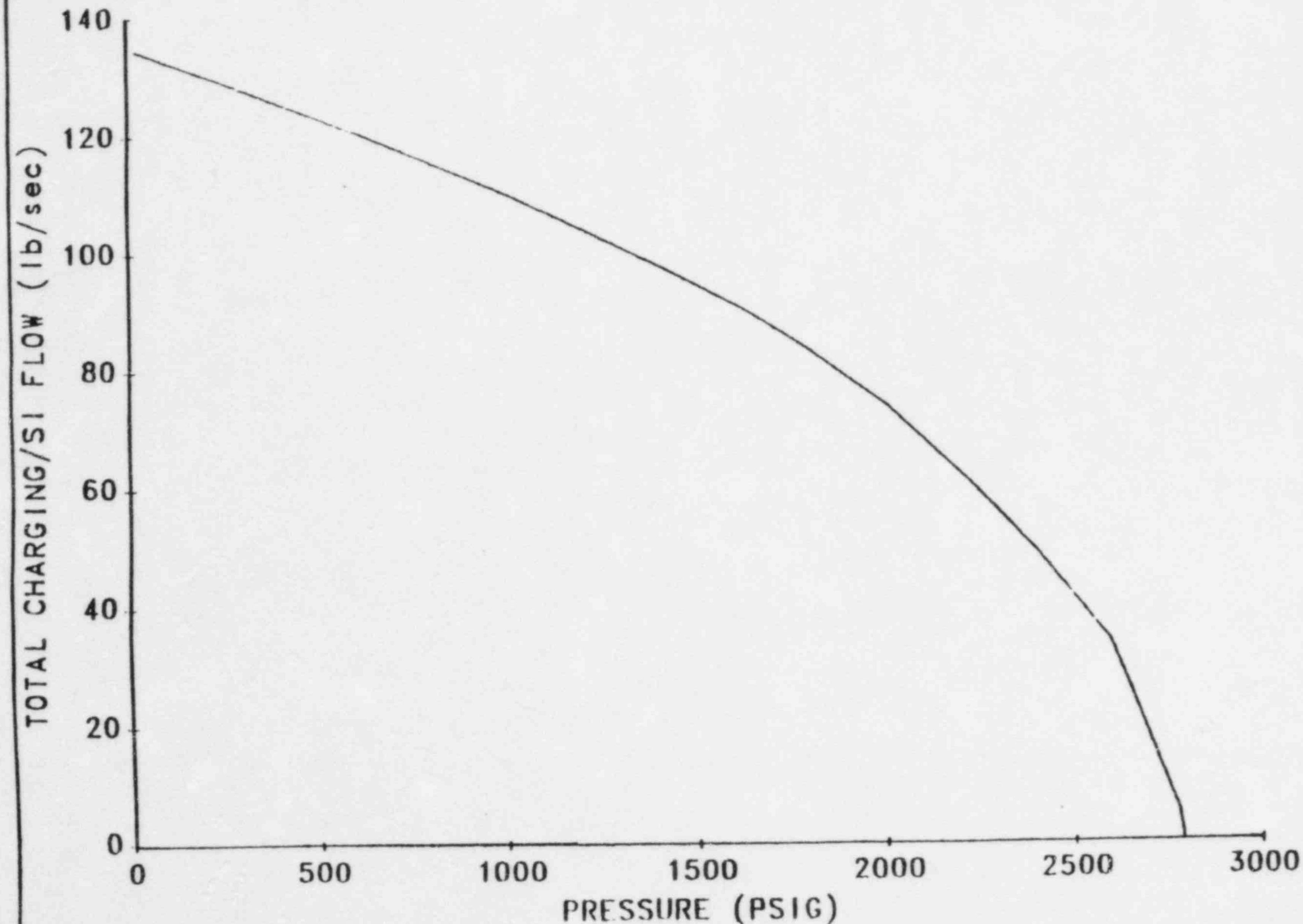
* (As used in NOTRUMP generic simulation, mixing cup used HBR flow rates).

FIGURE B.4.2-1

MAXIMUM SI/CHARGING FLOW VERSUS RCS PRESSURE

FOR S LOCA -- GENERIC 3 LOOP HIGH HEAD PLANT



6. Main feedwater flow will coast down in 11 seconds after the main feed isolation setpoint has been reached.

7. Auxiliary feedwater flow starts 13 seconds after the main feed isolation setpoint. The maximum auxiliary feedwater flow will last for 5 minutes then control will be switched to limit steam generator level.

B.4.3 Steam Generator Tube Rupture

B.4.3.1 Transient Description

The response of the reactor coolant system to a variety of steam generator tube failure events, up to the complete severance of a single tube, has been analyzed for a generic W PWR using LOFTRAN. These analyses simulate automatic protection systems, such as reactor trip and emergency core cooling systems, as well as operator actions. Unlike other design basis events, a steam generator tube failure by itself will not result in a rapid cooldown of the primary system or in an excessively high Reactor Coolant System pressure. Furthermore, natural circulation will develop in all primary loops and mix with incoming safety injection flow to preclude local temperature depressions if the RCPs are stopped. The subsequent operator actions to terminate primary-to-secondary system leakage, however, may rapidly cool the Reactor Coolant System for short periods and may stagnate the faulted loop. In that case, local temperature depressions resulting from continued safety injection flow may occur. In addition, consequential failures may result in a more severe transient.

Table B.4.3-1 presents the sequence of automatic actions (A) and simulated operator action (O) modeled in the WOG Stagnant loop PTS SGTR analysis [B.4]. The exact times and responses are plant and event specific.

The Reactor Coolant System (RCS) pressure initially decreases toward the reactor trip setpoint as flow through the tube rupture, in excess of the normal charging pump capacity, depletes the primary coolant inventory. The reactor trip for the generic plant analyzed occurs on over temperature delta-T. But for other plants, depending on the abnormal actual trip set point, reactor trip can occur on low pressure first.

TABLE B.4.3-1

Sequence of Events for SGTR

1. Reactor Trip - pumps lost if offsite power is not available (A).

2. Turbine Trip (A).

3. Loss of Offsite Power (A).

4. Steam Generator Safety Valves Open (A).

5. Safety Injection Actuated (A).

6. Auxiliary Feedwater Actuated (A).

7. Main Steamline Isolation (O).

8. Steam Dump From Non-Ruptured SGs - reduce RCS temperature to 50°F below no-load (O).

9. One Pressurizer PORV Opened - reduce RCS pressure to the ruptured steam generator pressure (O).

10. Terminate SI (O).

11. Reestablish Charging and Letdown - maintain pressurizer level at 20 percent span (O).

12. Pressurizer Spray Initiated - terminate break flow (O).

(A) Automatic
(O) Operator

Following reactor trip, the primary pressure decreases rapidly as the energy transfer to the secondary side rapidly cools the RCS and tube rupture flow continues to deplete primary inventory. This decrease in RCS pressure results in a low pressure Safety Injection (SI) signal soon after reactor trip. For smaller tube ruptures the operator may manually initiate SI prior to automatic actuation. Normal feedwater flow is automatically terminated on reactor trip and the auxiliary feedwater system is actuated to deliver flow to all steam generators. Eventually, operator action is required to adjust auxiliary feedwater flow to maintain the steam generator water level on the narrow range span.

Secondary pressure will increase rapidly as automatic isolation of the turbine following reactor trip momentarily stops steam flow from the steam generators trapping the steam in the steamlines. Steam dump control is expected to establish and maintain programmed no-load RCS temperature after reactor trip. If a reactor coolant pump continues running, only a small core delta-T will exist. Consequently, the core inlet and core outlet temperatures will tend to stabilize at the no-load temperature until manual cooldown of the RCS is initiated. For the WOG stagnant loop SGTR evaluation [B.4] both cases are considered (i.e. with and without RCP running).

For the case with no RCP running, a simultaneous loss of offsite power is conservatively assumed, and hence, all reactor coolant pumps are tripped at the time of reactor trip. Steam dump to condenser is assumed unavailable as a result of the loss of offsite power. Manual control of steam dump to the atmosphere is assumed. SI flow and auxiliary feedwater flow eventually reduce the RCS average temperature to near no-load temperature until auxiliary feedwater is manually controlled to maintain steam generator level in the narrow range.

After reactor trip, the primary coolant shrinkage associated with the rapid post-trip cooldown and the continued loss of RCS inventory through the tube rupture result in a rapid decrease in pressurizer level until SI flow occurs. Pressurizer level may be offscale on the low side prior to SI actuation. The minimum level is dependent upon tube rupture size, SI setpoint and capacity, and steam dump control. Maximum SI flow capacities were assumed for the stagnant loop SGTR analyses [B.4].

When the post-trip RCS cooldown subsides, SI flow is expected to begin refilling the pressurizer and increase RCS pressure until SI flow equals break flow, thus maintaining a constant primary system inventory. Pressurizer level may not return to span during repressurization of the RCS. For multiple tube ruptures or reduced SI capacity, RCS pressure may continue to decrease toward the ruptured steam generator pressure until equilibrium is established. Manual actions may reduce RCS pressure and SI flow prior to reaching equilibrium.

The analysis assumes that the operator identifies the ruptured steam generator by a high steam generator water level and subsequent operator actions in accordance with the WOG Emergency Response Guidelines [B.5], are modelled to mitigate the steam generator tube rupture accident.

Once the ruptured steam generator has been identified, it is manually isolated to maintain the ruptured steam generator pressure above the non-ruptured SG pressure and to minimize activity releases. Auxiliary feedwater flow is terminated to the ruptured steam generator to reduce the chance of filling the steamlines with water. As the cooldown of the intact steam generator continues, flow through the faulted loop decreases significantly and may eventually stagnate. Steam dump from the non-ruptured steam generators to the condenser, via the steam dump system or through the atmospheric relief valves, is initiated to reduce the RCS temperature to 50°F below the saturation temperature at the ruptured steam generator pressure. This assures adequate subcooling of the RCS after depressurizaton to the ruptured steam generator pressure and creates a rapid cooldown of the RCS.

After the primary temperature is reduced, the RCS is depressurized to the ruptured steam generator pressure to terminate break flow. In the generic analysis, the primary pressure is reduced by opening one pressurizer PORV. The pressurizer water level increases during the depressurization as SI flow, in excess of break flow, replaces vented steam with water.

When the RCS pressure is equal to the ruptured steam generator pressure, the PORV is closed. Pressurizer level and, consequently, RCS pressure continue to

increase until safety injection is terminated. Once SI is terminated, the primary RCS cooldown will stop and extrapolation can be used to determine the downcomer fluid temperatures.

B.4.3.2 Model Description

The LOFTRAN computer code [B.7] is used for the generic PTS SGTR analysis. An HBR plant specific mixing cup model was then used to define the downcomer temperatures for the SGTR scenarios analyzed. The plant response to the automatic protection system and operator recovery actions as outlined in the previous section are incorporated into the configurated version of LOFTRAN. Modeling improvements incorporated for the generic evaluation [B.4] include:

a) A thick metal heat transfer calculation is added to the vessel downcomer.

b) SI flow is no longer automatically forced into the downcomer. The code will determine the SI flow split between the vessel downcomer and the cold leg loop seal based on the path resistance.

c) Enhanced modeling flexibility to allow simulation of operator actions to cycle the PORV, turn on pressurizer spray, throttle auxiliary feedwater flow, and termination of SI has been incorporated.

d) The HBR mixing cup post-processor model is used to calculate the downcomer fluid temperature when the faulted loop losses natural circulation and stagnates.

Multiple tube ruptures were not considered separately but the probability of multiple tube ruptures is included in the tube rupture frequencies in Table A.2-5. This approach is acceptable because LOFTRAN predicts that the faulted loop will stagnate even for a single tube failure. The RCS cooldown transients for both single and multiple tube failures, after the primary and secondary pressure have equilibrated and natural circulation has stopped, should be similar and can be combined together for risk assessment.

SGTR together with an uncontrolled secondary steam release may lead to a further decrease in the average RCS temperature below the no-load TAVG temperature. However, no significant additional contribution to PTS risk is expected from this combined failure for the following reasons:

a) The major causes for a possible uncontrolled steam release during a SGTR are a stuck open secondary safety or relief valve, and steamline fracture. For a main steamline break downstream of the MSIV, closing the MSIV and bypass valves will terminate the steam release. The RCS cooldown will be the same as that of a SGTR accident.

b) For a main steamline break upstream of the MSIV, or a stuck open safety valve, the secondary pressure will continue to decrease below the no-load pressure until steam release through the fault in the secondary system matches steam generated via energy transfer from the primary system. Because of the continual maintenance of heat transfer between the primary and the secondary systems, natural circulation will be maintained in the faulted loop. The resulting RCS cooldown transient is therefore bounded by the results of the large secondary depressurization.

c) Finally the frequency of a SGTR coincidental with an uncontrolled secondary steam release is significantly lower than the frequency of either accident happening alone [B.4].

B.4.3.3 <u>Assumptions and Initial Conditions Used in the Generic Study</u>

1. The initial conditions assumed for the two low decay heat bins and the full power bin are hot zero power and full power operation respectively.

2. A constant decay heat level of .25 percent and .75 percent of rated power during the cooldown is assumed for the two low decay heat (0 to .5 percent, and .5 percent to 1 percent) bins. For the high decay heat bin the best estimate decay heat release (based on ANS 5.1 Decay Heat Standard of 1979) [B.9] after a trip from extended full power operation is used.

3. A break flow area equivalent to a double ended rupture of a single tube at the SG tube sheet on the cold side is assumed.

4. For cases without RCP operating SGTR with simultaneous loss of offsite power is assumed. All RCPs are therefore tripped at the start of the transient.

5. Heat transfer from the vessel thick metal to the reactor coolant is considered.

6. Steamline isolation occurs at normal protection set points.

7. Maximum AFW was assumed.

8. Auxilary feedwater to the non-affected steam generators is controlled to maintain a SG level of 15 percent of the narrow-range span. Auxiliary feedwater to the affected steam generator is terminated manually at 10 minutes.

9. Safety injection termination criterion is based on an indication of increasing RCS pressure after the PORV is closed.

10. Two operator action times for the successful termination of SI, after the termination criterion is met, are considered. The first bin assumes manual termination within 3 minutes of meeting the criterion. The second bin assumes manual termination of SI between 3 to 30 minutes after the criterion is met.

B.5 MIXING CUP MODEL

B.5.1 Introduction

The fluid conditions in the reactor vessel downcomer near the core midplane are of primary concern in the PTS risk analysis because the vessel RT_{NDT} is generally highest there. PWR system T-H codes such as LOFTRAN predict downcomer temperatures accurately under a wide range of loop flow conditions

(roughly speaking greater than 0.1 percent of rated flow).* Under low flow or stagnant conditions, a model has been developed that uses boundary conditions which are provided by the system codes to accurately predict downcomer conditions. The model is called the mixing cup, and has been implemented by way of the MXGCUP code for the WOG stagnant loop code evaluation program [B.4].

In the following sections, the MXGCUP code itself is described, its use in the PTS risk assessment is discussed, and a discussion on the Plume Effect is provided.

B.5.2 Mixing Cup Description

A computational model that predicts fluid temperatures as a function of time at the reactor vessel beltline or midplane has been developed for use in PTS structural evaluations. This model accounts for two phenomena:

- o the bulk mixing of cooler SI water with warmer loop water, and,

- o the bulk energy exchange between the fluid in a region and the metal structures bounding that region.

A Schematic diagram of the Mixing Cup model appears in Figure B.5.1-1.

The fluid mixing model is based on The First Law Of Thermodynamics. The model uses constant volume fluid nodes. Perfect or thermodynamic equilibrium mixing is assumed for all fluid within the fluid nodes. ASME steam table subroutines are entered at every time step to determine fluid conditions. Initial conditions and boundary conditions for the model are transferred automatically via tape or via manually generated data files.

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* The point where stagnation is assumed.

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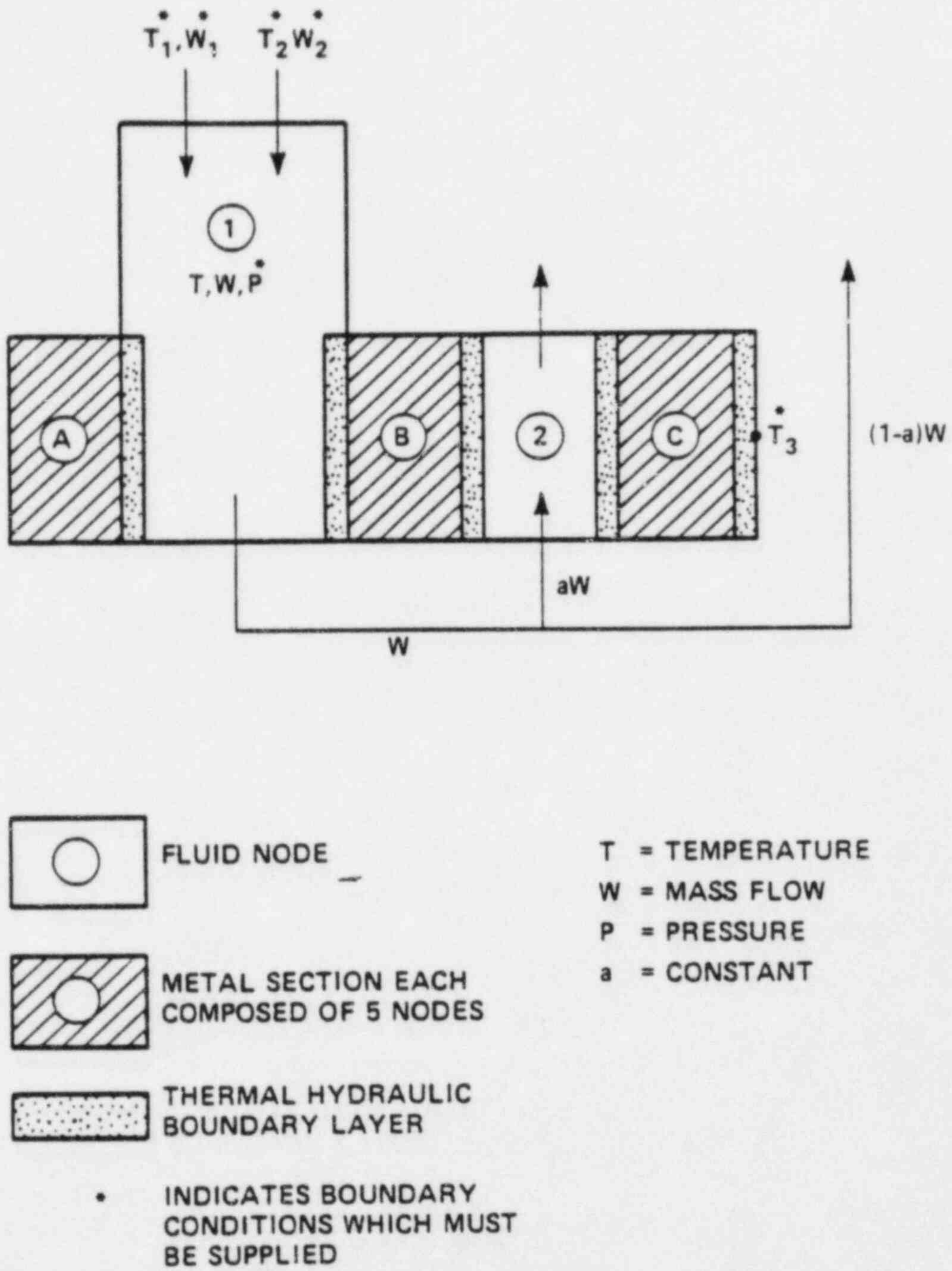


Figure B.5.1-1 Schematic Diagram of the Mixing Cup Model in MXGCUP

Heat conduction through the three 5-node slabs of metal in MXGCUP is based upon standard explicit difference equations [B.10]. Both a forced and free convection heat transfer coefficient are calculated at each time step for each boundary layer. The larger of the two coefficients is used during that step in the conduction equation for a particular slab. The heat transfer correlations implemented in the model are as follows:

- Forced convection [B.11]

  $$Nu = 0.023\ (Re)^{0.8}\ (Pr)^{0.4}$$

- Free convection [B.12]

  $$\text{For } GrPr > 1 \times 10^9: \quad Nu = .021(GrPr)^{2/5}$$
  $$\text{For } GrPr < 1 \times 10^9: \quad Nu = .555(GrPr)^{1/4}$$

B.5.3 Use Of The MXGCUP Code

During periods of loop stagnation in all loops, the cold leg and downcomer are decoupled from the rest of the RCS as far as mixing of SI with primary coolant is concerned. Experimental evidence to date supports the use of a thermodynamic equilibrium mixing model with a volume as shown in Figure B.5.3-1 for downcomer core midplane temperature prediction.

The input to the MXGCUP code which was used in the PTS risk assessment is summarized below (refer to Figure B.5.1-1 for the Noding Scheme):

<u>Input Quantity</u> (Remarks Below)

Vessel Thickness	-	.78 FT
Core Barrel Thickness	-	.17 FT
Average Baffle Thickness	-	.59 FT
Core Wrapper Thickness	-	.10 FT
Vessel Surface Area Per Loop	-	271 FT^2
Core Barrel, Wrapper, and Baffle Surface Area Per Loop	-	152 FT^2

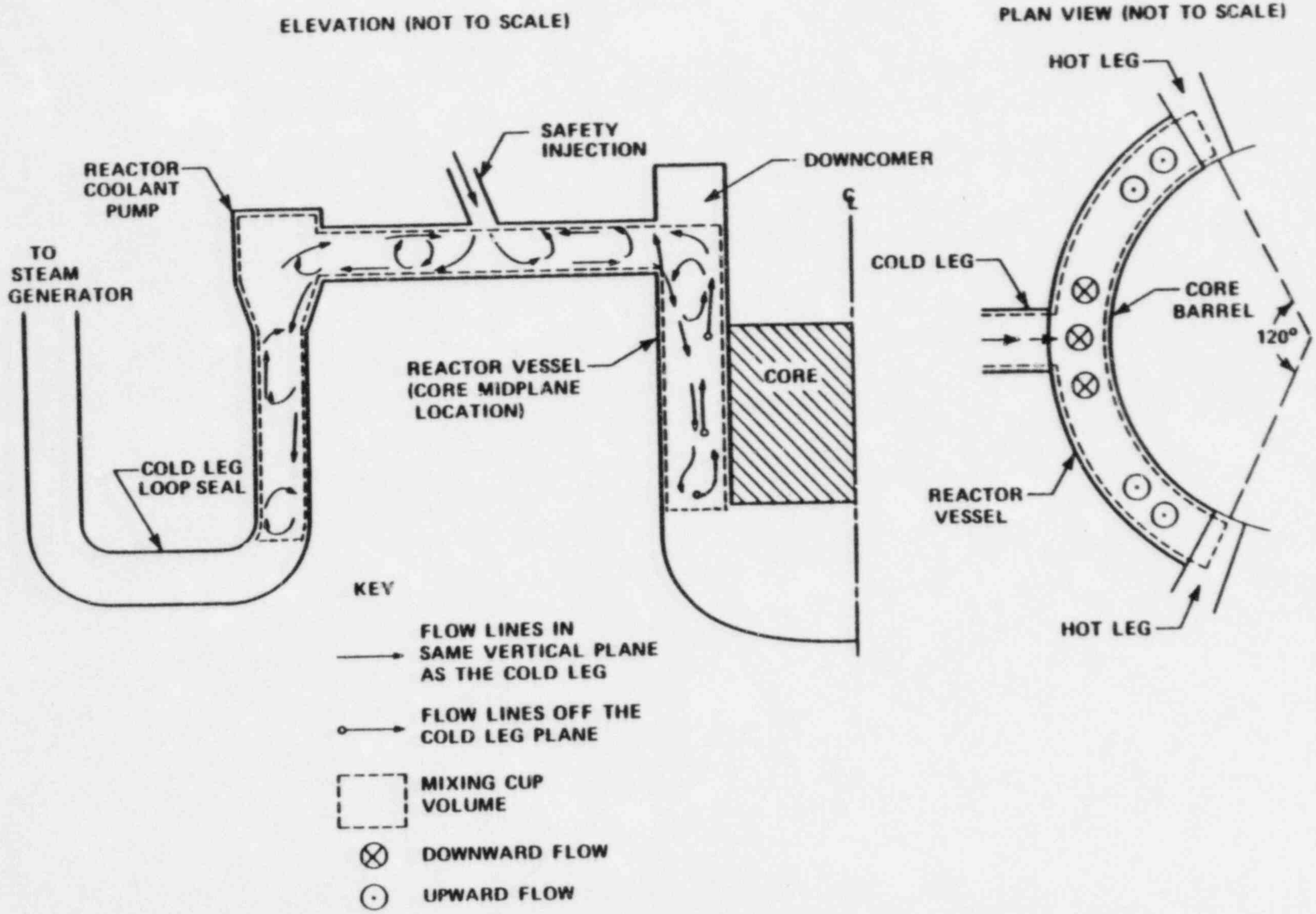


Figure B.5.3-1 Mixing Cup Volume and Typical Gross Flow Paths Under Completely Stagnant Conditions

5498-2

Input Quality (Continued)

Flow below which the mixing cup model is actuated. (1)	-	15 lbm/SEC
Fraction of downcomer flow that goes through the baffle (2)	-	.05
Characteristic length of the downcomer (used in free convection correlation)	-	20 FT
Characteristic length of the core barrel, downcomer side, and core wrapper, core side	-	12 FT
Characteristic length of the baffle, both sides	-	1.5 FT
Cross sectional area of the downcomer and baffle associated with one loop (used in forced convection correlation)	-	6.5 FT
Cross sectional area of the core	-	13.9 FT^2
Hydraulic diameter of the Downcomer associated with one loop (used in forced convection correlation)	-	.5 FT

Input Quality (Continued)

Hydraulic diameter of the baffle associated with one loop	-	1.0 FT	
Hydraulic diameter of the core	-	.042 FT	
Volume of the mixing cup	-	400 FT^3	(3)

(1) The actual number which is used here could range from 10 to 100 lbm/sec without significantly affecting the stylized characteristics of a severe cooldown transient because the onset of stagnation is very abrupt. The 15 lbm/sec value is chosen because, below it, backflow is expected to occur on a typical PWR per the Creare Test [B.6].

(2) The 0.05 value is conservatively high. The higher the bypass flow fraction, the less energy is conducted from the core to the downcomer.

(3) This mixing cup volume is conservatively small (~ 30 percent) relative to the final WOG methodology since it was used before the acceptable size was finalized. The core heat transfer into the mixing cup is accounted for only in the LOCA analysis because stagnant fluid in the baffle is assured. For assymetric cooldown transients, such as a small secondary depressurization that involve loop stagnation, the mixing cup is conservatively used for the stagnant loops. The comparison of the mixing cup results with experimental results is fully discussed in Reference [B.4].

B.5.4 Plume Effect

Scoping work in the area of two-dimensional fracture mechanics shows that actual temperature deviations from the mixed mean temperature of the kind found in experimentation to date, have a small effect on peak stresses in the

reactor vessel. These findings further support the use of a mixed mean temperature in a PTS risk assessment. A discussion of the Plume Effect follows.

Under completely stagnant conditions, a layer of relatively colder fluid can be expected to exist at the bottom of the cold leg and in the downcomer directly under the inlet nozzle. In the past, analyses have been performed either assuming the entire downcomer was subjected to a step change in temperature or an exponential decay to the RWST temperature. In this brief study, recent two-dimensional theoretical results [B.13] are summarized and possible benefits are explored.

In order to understand the effects of thermal plumes on thermal stresses in the vessel, it is necessary to consider at least a 2D model with radial and azimuthal temperature variation. It has been shown in [B.13] that if the width of the colder region (plume) is restricted to less than approximately 120°, then the peak stresses will be smaller than the peak stress in a cylinder subjected to the same uniform cold temperature. Figures B.5.4-1 and B.5.4-2 which are taken from [B.13], respectively show various cooling spans in a cylinder and display stress profiles for various widths of plumes. It is immediately apparent that the peak stress occurs in the center of the cold region and the magnitude of the peak stress is significantly lower as compared to the peak stress resulting from uniform cooldown.

Mixed mean temperatures can be calculated fairly well with mixing cup models (provided that the mixing volume is well known) but it is difficult to predict the coldest or plume temperature. The available Creare experiments and 3-D numerical results [B.1-] show a difference of less than 50°F below the mixed mean temperature. The cases used for comparison included a single plume with a square-well temperature distribution and two neighboring plumes with square-well temperature distributions. Figure B.5.4-3 shows the hoop stresses as a function of azimuthal location at 500 seconds for the two cases. The peak values are very similar. It is apparent that this difference between the 1D analysis based on the mixed mean temperature and a 2D analysis based on a ΔT of 50°F between the coldest and mixed mean temperatures is expected to be

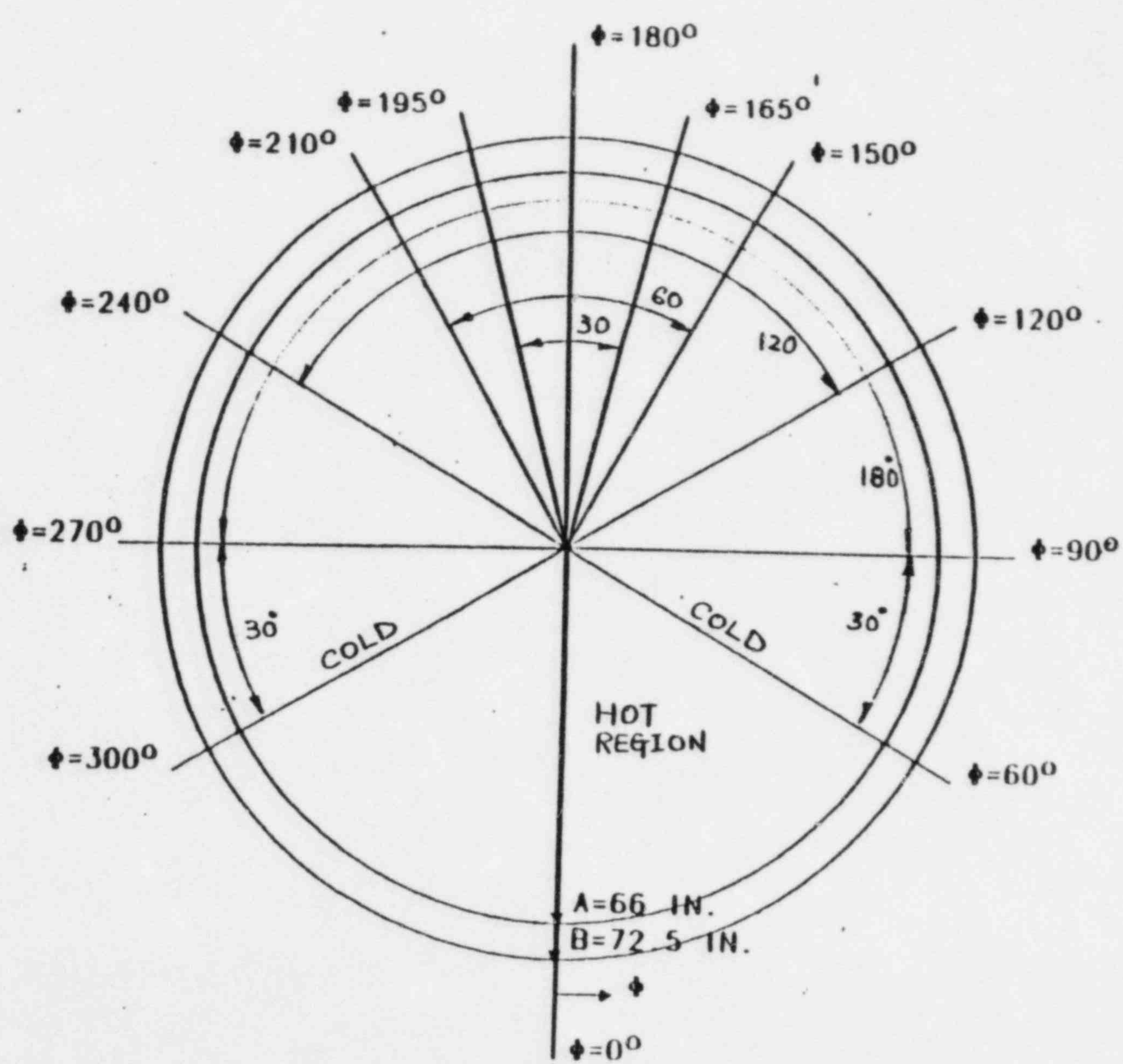


FIGURE B.5.4-1 GEOMETRY OF CYLINDER SHOWING VARIOUS COOLING SPANS

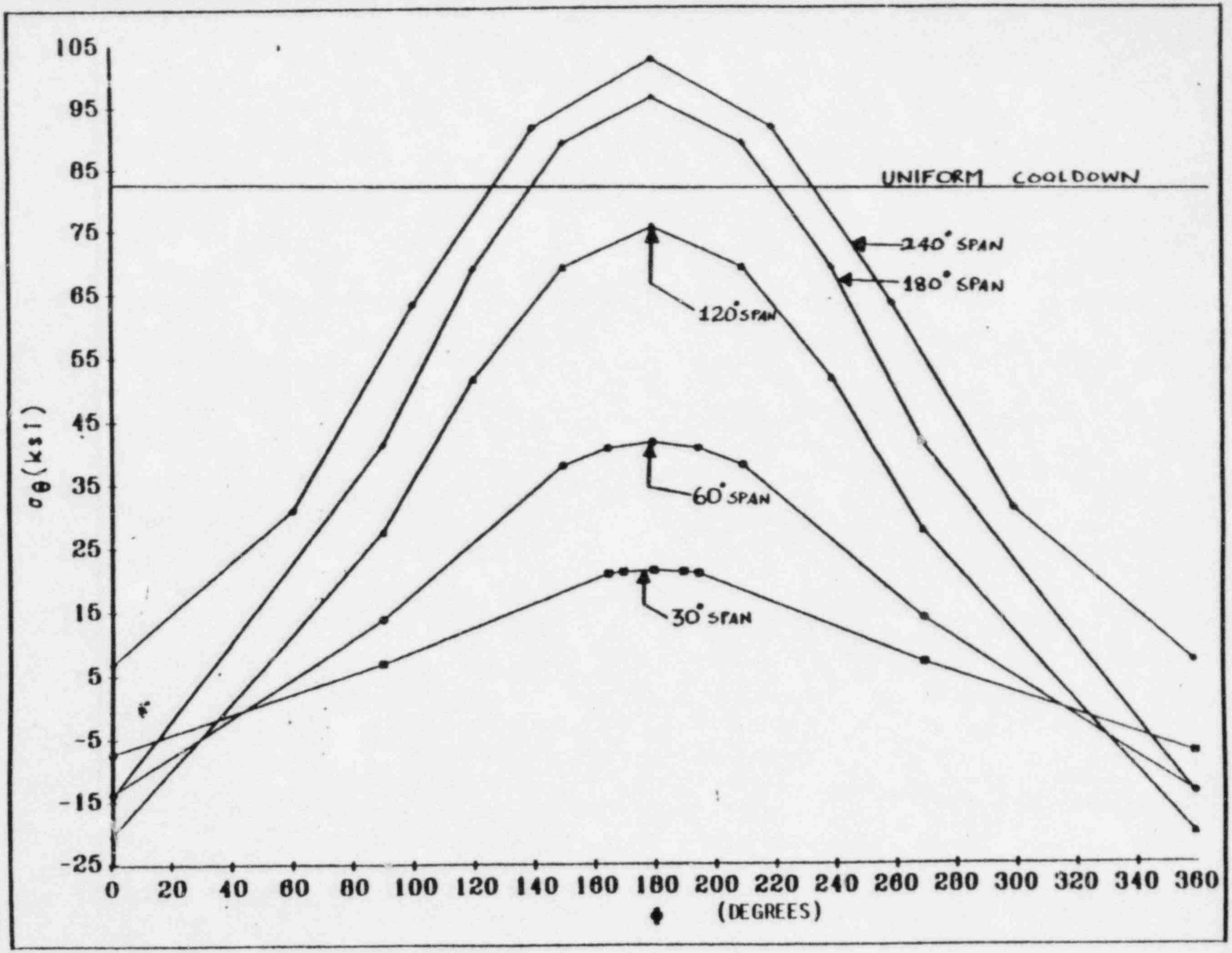


FIGURE B.5.4-2 VARIATION OF HOOP STRESS WITH ANGLE Ø AT TIME = 500 SECONDS FOR DIFFERENT SPANS OF COLD REGION

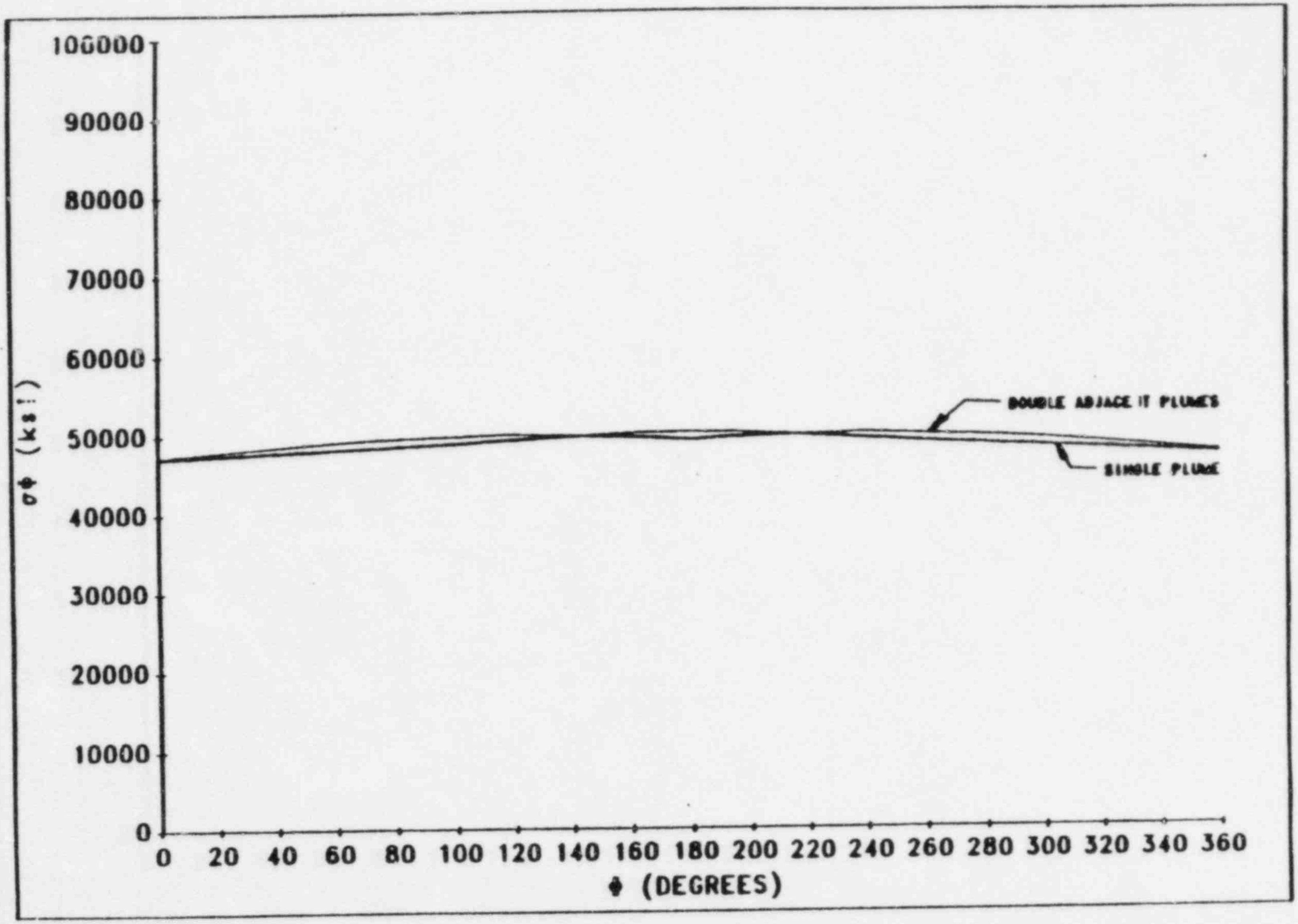


FIGURE B.5.4-3 HOOP STRESSES AS A FUNCTION OF AZIMUTHAL LOCATION AT 500 SECONDS

a Gaussian profile. Thus, the actual peak stresses are expected to be less than the peak stresses obtained from the assumed square well temperature distribution. It, therefore, follows that as long as there is evidence to show that there could be no larger temperature difference than a ΔT of 50°F between the coldest plume temperature and the calculated mixed mean temperature, one-dimensional analyses (i.e. uniform cooldown) based on mixed mean temperatures from a mixing cup model will be representative of the actual scenario.

It appears that the information given here, when evaluated together with existing experimental and analytical mixing data, provide sufficient evidence to support the use of the simpler 1-D model for thermal and stress analyses even for transients that result in flow stagnation.

B.6 PROBABILISTIC FRACTURE MECHANICS (PFM) (Step 3)

As discussed previously, probabilistic fracture mechanisms analyses are performed as part of the risk assessment to quantify the conditional probabilities of significant flaw extension for given PTS transient characterizations. PFM analysis results were generated and reported by the U.S. NRC in Reference [B.3] to lend support to the selection of the RT_{NDT} screening criteria for PTS. These results, while providing sufficient information to perform a reasonable generic PTS risk assessment, were generated for a longitudinal flaw on a typical PWR vessel and included other assumptions that should be modified in a plant specific assessment. Therefore, significant attention is first given to the assumptions and methods used in the U.S. NRC PFM analysis since it is the starting point for the HBR evaluation. Then, a model is developed that is essentially identical to the U.S. NRC model for studying the effects of various refinements applicable to the HBR vessel. Using the results from this study, which is done on a consistent basis with the NRC analysis, several of the refinements are chosen to generate the HBR conditional probabilities of significant flaw extension from a small set of PTS transient thermal-hydraulic characteristics.

B.6.1 NRC PFM Analysis

Appendix H of Reference [B.3] gives a detailed explanation of the PFM analysis methods and assumptions used by the U.S. NRC in generating failure probabilities for a set of hypothesized PTS transients for the probabilistic assessment done to support the RT_{NDT} screening criteria. A summary of the key assumptions is given as follows:

NRC Reactor Vessel Model

- o Wall thickness = 9"
- o Mean radius = 90"
- o Stainless steel cladding is not modeled

NRC Thermal and Stress Analysis and Stress Intensity, K_I, Determination

- o Transient characterization - represented by

$$T(t) = T_f + (550 - T_f)e^{-\beta t}, \quad P(t) = \text{constant}$$

where $T(t)$ = transient temperature, °F
T_f = final temperature, °F
β = cooldown rate constant, min^{-1}
t = time, min
P = pressure, psig

- o Heat transfer coefficient = 320 Btu/hr-ft^2-°F at the vessel wall inside surface taking into account the fluid heat transfer coefficient and the thermal resistance of the stainless steel cladding.

- o Thermal diffusivity = 0.98 in^2/min
- o Parameter $\frac{E\alpha}{1-\nu}$ = 0.322 ksi/°F where

E = Young's Modulus
α = Coefficient of Thermal Expansion
ν = Poisson's ratio

- o Stresses resulting from the differences in thermal expansion between the cladding and the base metal of the reactor vessel have not been included.

- o Flaw orientation - longitudinal

- o Flaw shape - infinitely long (i.e. continuous) for both crack initiation and arrest.

- o K_I is determined using linear elastic fracture mechanics (LEFM) methods

NRC Probabilistic Model

- o Flaw distribution for crack depth - OCTAVIA (computer code) in discrete form. The weld volume associated with the OCTAVIA flaw distribution was defined as the volume of longitudinal weld material in the beltline region of a PWR. To obtain the flaw distribution for a single beltline weld (~ 72" long in the NRC study), the OCTAVIA flaw distribution was adjusted assuming that the flaws were equally distributed among six longitudinal beltline welds to represent the probability of having a flaw in one weld. Flaw non-detection probabilities to account for non-destructive examination reliability were not considered in the PFM analysis of the hypothesized PTS transients.

- o Parameters affecting reference nil-ductility transition temperature (RT_{NDT}).
  - Inital RT_{NDT} ~ Normal distribution with a mean (μ) value of 20°F and standard deviation (σ) of 15°F
  - Copper content ~ Normal distribution - μ = 0.30 percent, σ = 0.025 percent
  - Nickel Content ~ Constant value of 0.75 percent
  - Fluence ~ Normal distribution with five mean values considered ranging from 0.5 x 10^{19} neutrons/cm^2 to

4.0×10^{19} neutrons/cm^2 with a standard deviation = 30 percent of respective mean value at vessel inside surface. (This yields a range of RT_{NDT} values, thereby providing a means to evaluate the effect of vessel life on the conditional probabilities of significant flaw extension.) The fluence attenuation through the vessel wall is represented by the equation

$$F(a) = F_{ID}e^{-.33a}$$

where,

a = depth into the wall, in

$F(a)$ = fluence at any depth in wall, neutrons/cm^2

F_{ID} = fluence at inside wall, neutrons/cm^2

This equation is representative of damage for E > 1.0 MeV.

- $\Delta RT_{NDT} = [-4.83 + 476\ Cu + 267\ CuNi][F/10^{19}]^{0.218}$, which is the original HEDL mean irradiation damage trend curve (For equivalent ΔRT_{NDT} values at the vessel inside surface, the given curve provides a conservative assessment of significant flaw extension when compared to the revised HEDL trend curve prescribed in Reference [B.3], because the slope of the original trend curve is less steep).

o Fracture toughness curves

- Crack initiation toughness, K_{Ic} ~ Normal distribution represented by the following mean equations:

$$\bar{K}_{Ic} = 36.2 + 49.4 \exp (0.0104 (T - RT_{NDT})) \text{ for}$$

$$T - RT_{NDT} < -50°F$$

$$\bar{K}_{Ic} = 55.1 + 28.0 \exp (0.0214 (T - RT_{NDT})) \text{ for}$$

$$T - RT_{NDT} > -50°F$$

and the standard deviation is defined as $\sigma = 0.10\ \overline{K}_{Ic}$

- Crack arrest toughness, K_{Ia} ~ Normal distribution represented by the following mean equations:

$$\overline{K}_{Ia} = 19.9 + 43.9 \exp(0.00993\,(T - RT_{NDT})) \text{ for}$$

$$T - RT_{NDT} < 50°F$$

$$\overline{K}_{Ia} = 70.1 + 6.5 \exp(0.0196\,(T - RT_{NDT})) \text{ for}$$

$$T - RT_{NDT} > 50°F$$

and the standard deviation is defined as $\sigma = 0.10\ \overline{K}_{Ia}$

- Both the mean crack initiation and crack arrest toughnesses were truncated at a mean upper shelf value of 200 ksi-in$^{1/2}$ with a standard deviation equal to 10 percent of the 200 ksi-in$^{1/2}$ value.

NRC Constitution of Failure

- o If crack initiation is predicted (i.e., $K_I > K_{Ic}$), failure occurs when
  - $K_I > K_{Ia}$ anywhere within the vessel wall, or
  - $\sigma > \sigma_{flow}$ in the remaining vessel wall ligament after crack arrest. (σ = pressure stress and σ_{flow} = material flow stress).

- o Multiple initiation and arrests are considered. If crack arrest is not predicted before K_I reaches the upper shelf toughness value, vessel failure is predicted.

- o Any possible benefits in the prohibiting of crack extension from the effect of warm prestressing were not considered since the warm prestressing benefit should not be arbitrarily applied to a family of stylized transients to determine the conditional probabilities of failure for the PTS risk assessment. However, the NRC has previously used the benefit of warm prestressing in the evaluation of a small LOCA transient [B.3].

NRC Simulation Process

- o Straight Monte Carlo approach up to 10^6 trials/case

- o Frequency of failure =

$$\frac{\text{Number of failures}}{\text{Number of trials}}$$

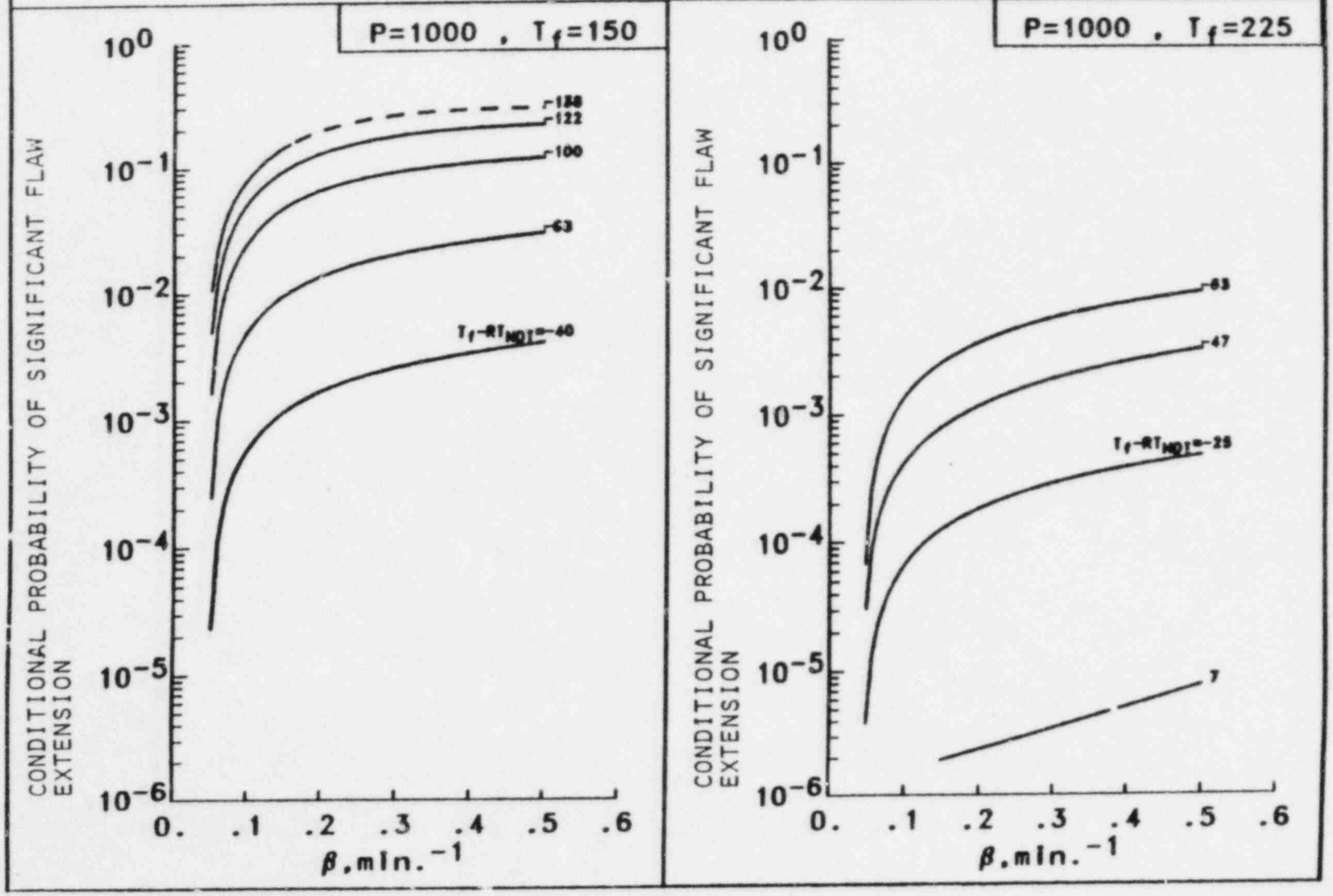
Using the above assumptions and methods, conditional probabilities were generated for a set of 45 hypothesized PTS transients at five levels of fluence (or RT_{NDT} or effective full power years of operation). Referring to the transient characterization presented above, three values of T_f = 150°F, 225°F, and 300°F; three values of β = 0.05 min^{-1}, 0.15 min^{-1}, and 0.5 min^{-1}; and five constant pressure levels of 0 psig, 500 psig, 1000 psig, 1500 psig, and 2000 psig were evaluated to constitute the 45 transients, which represent a broad range of possible PTS scenarios. All of the conditional probability results are given in Table H-2 of Reference [B.3]. The results have been converted into plots of conditional probability versus β using the normalization of T_f - RT_{NDT} for the NRC PFM data of interest as shown in Figure B.6.1-1 for the 1,000 psig case.

B.6.2 HBR PFM Analysis

To address the applicability of the NRC PFM conditional probability results to the H.B. Robinson vessel, additional studies were performed using the Westinghouse PFM analysis code. There are significant differences between the H.B. Robinson plant and the "generic model" studied by the NRC; in particular the H.B. Robinson circumferential weld is of interest whereas the NRC generic

FIGURE B.6.1-1

NRC CONDITIONAL PROBABILITY OF SIGNIFICANT FLAW EXTENSION
FOR A SINGLE LONGITUDINAL BELTLINE WELD - P = 1000 psig [B.3]



study considered a longitudinal flaw. (The Westinghouse PFM Code has been recently benchmarked against the U.S. NRC PFM analysis code "VISA" under a Westinghouse Divisional General Research Program). A sensitivity study was first performed for one base case transient using refinements to the NRC PFM analysis approach that are applicable to the HBR vessel. Using the results from this study, several of the refinements were chosen to generate the HBR conditional probabilities from the evaluation of four base case transients. These values were then used in a PTS risk assessment of the HBR vessel.

B.6.2.1 Sensitivity Study of HBR Refinements

Using the NRC transient characterization discussed in Section B.6.1, a representative PTS base case transient of T_f = 150°F, β = 0.15/min, and P = 1,000 psi was chosen to evaluate various refinements beyond the NRC PFM analysis approach that are directly applicable to the H.B. Robinson vessel. The various considerations that have been studied using the Westinghouse PFM code are given as follows:

- H.B. Robinson Vessel Model
  - Wall thickness = 9.3125"
  - Vessel inside radius at clad = 77.75"
  - Clad thickness = 0.156" (for thermal analysis only)
- Heat Transfer Coefficient - The free convection correlation given in Reference [B.8] was employed.
- Circumferential Flaw Orientation - The lower circumferential beltline weld is the limiting HBR vessel location for PTS considerations.
- Flaw Shape - 6:1 semi-elliptical finite flaw for first initiation, continuous flaw for arrest and subsequent reinitiations. (This assumption is conservative considering the azimuthal temperature and neutron fluence variation for the HBR vessel as shown in Appendix C.)

- Through Wall Fluence Attentuation - Represented by an equation using displacements per atom (dPa) as the damage function.

- Failure Criterion - Crack propagation or arrest beyond 50 percent, 75 percent, and 100 percent of the vessel wall thickness without consideration of plastic instability (i.e., $\sigma > \sigma_{flow}$ in the remaining vessel wall ligament after crack arrest was not checked).

Table B.6.2-1 defines the analytical cases that were established to evaluate the above refinements starting with the NRC parameters as the benchmark case. All other input parameters and assumptions remained essentially the same as in the U.S. NRC PFM analysis with one exception and a straight Monte Carlo simulation process using up to 50,000 trials/case was employed.

The exception to the U.S. NRC PFM analysis assumptions for the sensitivity study is in the characterization of the upper shelf toughness for crack initiation and crack arrest. The upper shelf is characterized with a value of 200 ksi-in$^{1/2}$, but a deviation about this value is not considered. For HBR, the 200 ksi-in$^{1/2}$ value is assumed to represent a lower limit on toughness rather than a mean value of toughness.

The conditional probability of significant flaw extension results from the sensitivity study of HBR refinements are given in Figure B.6.2-1. The NRC results, shown with a dashed line, are taken directly from Appendix H of Reference [B.3] for the base case transient under study and are considered as the baseline analysis. Case A, which is essentially a benchmark of the Westinghouse PFM Code against the NRC "VISA" code, yielded values that are very similar for a probabilistic analysis and can also be considered as baseline results. The effect of the various HBR refinements on the baseline results is discussed in the following:

- Use of the free convection heat transfer coefficient from Reference [B.5] essentially had no effect on the conditional probability of vessel failure as shown in Case B.

TABLE B.6.2-1 H.B. ROBINSON CONSIDERATIONS BEYOND NRC RESULTS [B.3]

TRANSIENT: Final Temperature (T_f) = 150°F, Cooldown Rate (β) = 0.15/min., Pressure (P) = 1,000 PSI

CASE	VESSEL MODEL	HEAT TRANSFER COEFFICIENT	FLAW ORIENTATION	FAILURE CRITERION	FLUENCE ATTENUATION THROUGH WALL	FLAW SHAPE
A [B.3]	NRC	$\bar{h}$ = 320 Btu/hr-ft^2-°F	Longitudinal	Arrest $\geq$1.0 a/w	NRC Equation for Neutron Energies > 1.0 MeV	Continuous-Initiation and Arrest
B	↓	W Free Convection [B.8]	↓	↓	↓	↓
C	HBR	↓	↓	↓	↓	↓
D	↓	↓	Circumferential	↓	↓	↓
E1	↓	↓	↓	Arrest $\geq$0.5 a/w	↓	↓
E2	↓	↓	↓	Arrest $\geq$0.75 a/w	↓	↓
F	↓	↓	↓	↓	HBR Equation for Displacements per atom model	↓
G	↓	↓	↓	↓	↓	6:1 Finite-Initiation/Continuous Arrest

NOTES:

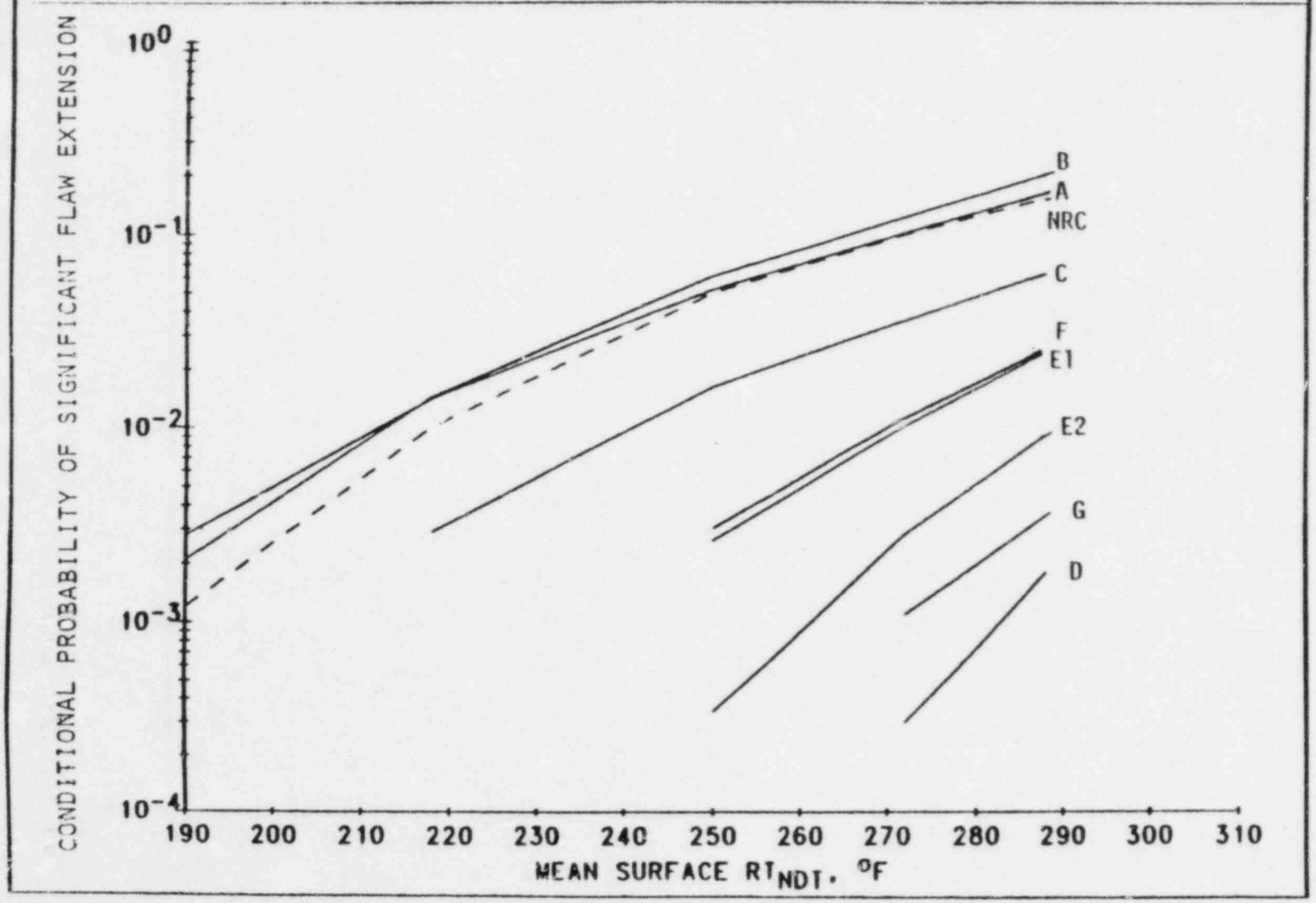
o SIMULATION PROCESS: STRAIGHT MONTE CARLO

o a = flaw depth

w = wall thickness

FIGURE B.6.2-1

PFM CONDITIONAL PROBABILITIES OF SIGNIFICANT FLAW EXTENSION FOR HBR CONSIDERATIONS (T_F = 150°F, β = 0.15/MIN., P = 1000 PSI)



- The HBR vessel model, because of its smaller radius/thickness ratio yielding lower pressure stresses, provided greater than half an order of magnitude in benefit to the vessel failure probability as demonstrated in Case C.

- A significant benefit of about 2 orders in magnitude is derived from the use of the circumferential flaw orientation when the results of Case D are compared to those of Case C. This benefit in risk occurs because: 1) the pressure stresses in the remaining ligament are lower in the circumferential case, and 2) the bending stiffness in the remaining ligament is higher for the circumferential orientation providing greater constraint of flaw growth.

- The benefit derived in Case D for the circumferential flaw orientation is reduced when other failure criteria are considered as evidenced by the results from Cases E1 and E2. This effect would not occur for a longitudinal flaw orientation assuming a continuous flaw for arrest since the probability of failure equals the probability of initiation in Cases A, B and C. That is, once a flaw initiates, crack arrest does not occur. (This fact was also verified in the Westinghouse/ U.S. NRC benchmarking studies. For high pressure transients, an NRC re-evaluation showed that their results in Reference [B.3] are unaltered even if a check of the plastic instability in the remaining ligament is not considered). However, for the circumferential flaw case, the failure criterion is important regardless if it is defined by exceeding a prescribed crack arrest depth, by failing a plastic instability check, or by not meeting some other mechanism. Further study to define an appropriate criterion representative of "vessel failure" can reduce the uncertainty associated with the prediction of "vessel failure".

- Fluence attenuation represented by the dPa damage function also reduces the benefit in risk derived from the circumferential flaw orientation as demonstrated by comparing Case F with Case E2. Once again, the longitudinal flaw situation results are not affected

because flaw initiation prevails as the failure mechanism and the fluence represented by either dPa or E > 1.0 MeV is nearly equivalent near the vessel wall inside surface. However, the fluence level is significantly higher for the dPa model versus the E > 1.0 MeV model near the vessel outer wall where crack arrest would be predicted to occur for circumferential flaws.

o Employing a 6:1 semi-elliptical finite flaw for first initiation and continuous flaw for arrest provides about an order of magnitude in benefit to the conditional probability of vessel failure as evidenced by comparing Case G with Case F. (This same benefit also occurs for the longitudinal flaw orientation) [B.4]. Further benefit in risk may also be derived by use of other than infinitely long cracks for arrest as supported by the shape change study of postulated circumferential flaws given in Appendix C.

The difference in the characterization of the upper shelf toughness between the HBR and NRC model would probably not have affected any of the results. The longitudinal results would be the same for either characterization because once the flaw initiates, crack arrest does not occur. The circumferential cases might have been slightly affected in a negative manner by the NRC characterization. However, the maximum K_I values for the transient of interest only reach 160 ksi-in$^{1/2}$, which is about 2 standard deviations below the mean NRC upper shelf value of 200 ksi-in$^{1/2}$.

In summary, several of HBR considerations reduce the conditional probability of significant flaw extension while others partially negate the benefit that is gained. However, if Case G is taken to be representative of the HBR vessel situation for PTS, the net effect of all of the refinements is a benefit of about 2 orders of magnitude reduction in risk of significant flaw extension below the NRC baseline results. The following section discusses the refinements used to determine the HBR failure probabilities from four base case transients as part of the PTS risk scoping study. The Westinghouse PFM analysis is also discussed and the HBR results are presented against the comparable NRC values.

B.6.2.2 HBR Conditional Probabilities of Significant Flaw Extension

In order to perform the Westinghouse PTS risk analysis as part of this HBR scoping study, four base case transients have been selected to determine the HBR conditional probabilities of significant flaw extension based upon expected transient characterizations of HBR dominant transient scenarios. These base case transients are presented in Table B.6.2-2. Since the four base case transients are defined with a pressure of 1,000 psig, a sub-case (Case 1A) with a pressure of 1750 psig is also evaluated. This analysis provides a basis for adjusting the conditional probabilities of vessel failure in the HBR risk analysis for transients characterized with a pressure greater than 1,000 psig.

TABLE B.6.2-2

H.B. Robinson Base Case Transients* for
Westinghouse Probabilistic Fracture Mechanics Analysis

Parameter**	Case 1 (Case 1A)	Case 2	Case 3	Case 4
β	.15 min^{-1}	.15 min^{-1}	.05 min^{-1}	.05 min^{-1}
T_f	150°F	200°F	150°F	200°F
T_i	550°F	550°F	550°F	550°F
P	1000 psig (1750 psig)	1000 psig	1000 psig	1000 psig

* Identified from dominating transient scenarios
** Transient characterization is represented by

$$T(t) = T_f + (T_i - T_f)e^{-\beta t}, \quad P(t) = P$$

The time duration and time step interval values used in the evaluation of the above transients are consistent with those used by the NRC [B.3]. These values are judged to be appropriate for this risk assessment of the HBR circumferential weld of interest based upon limited verification.

Based upon the results of the previous section, the following HBR considerations are used in generating the conditional probabilities of significant flaw extension:

- o H.B. Robinson Vessel Model

- o Constant Heat Transfer Coefficient = 300 Btu/hr-ft^2-°F. This value is judged to be appropriate for use with stylized transients, which are representative of stagnant loop conditions.

- o Circumferential Flaw Orientation

- o Flaw Shape - 6:1 semi-elliptical finite flaw for first initiation, continuous flaw for arrest and subsequent reinitiations.

- o Through Wall Fluence Attentuation - dPa damage function

- o Failure Criterion - Crack propagation or arrest beyond 75 percent of the vessel wall thickness. The 50 percent arrest value was not chosen since it is too conservative for probabilistic analyses and the 100 percent arrest value was not used since it exceeds the capability (or accuracy) of the linear elastic fracture mechanics method. The 75 percent arrest value is within the capability of LEFM and is conservative for probabilistic analyses. A check of plastic instability in the remaining ligament without consideration of the crack arrest depth (i.e., NRC approach) was not used.

With the exception of the assumed heat transfer coefficient, which does not effect the conditional probability of significant flaw extension results of interest, the HBR considerations that are used are comparable to Case G in the previous section. All other input parameters and assumptions are essentially the same as used in the U.S. NRC PFM analysis and the HBR sensitivity study with the exception that a mean surface fluence of 6.0 x 10^{19} neutrons/cm^2 was also considered to obtain a higher mean RT_{NDT} value (> 300°F) for

evaluation. Also, the HBR characterization of upper shelf toughness for flaw initiation and arrest, as used in the sensitivity study, is applied. In addition, the Monte Carlo simulation process was modified by applying importance sampling to lower computer costs associated with generating expected low conditional probabilities (see the results of Case G, which is representative of the most limiting of the four base case transients under evaluation, in the previous section).

Importance sampling is a modification of the Monte Carlo simulation in which the simulation is biased for greater efficiency. When determining conditional probabilities that are quite small (say on the order of 10^{-4} or less), a large number of trials (> 50,000) is required in order to have any failures occur. In importance sampling, the sampling is done only in the tail of the distribution in order to insure that simulated failures occur. The lower the probability of failure, the farther out in the tail the sampling must be done. Because the sampling carried out this way is biased, a correction factor is used in determining the probability of failure. Shifting the copper, K_{Ic}, K_{Ia}, and initial RT_{NDT} distributions appropriately one standard deviation and using 5,000 trials/case, the importance sampling results for transient Case 1 reasonably matched the Case G sensitivity study values, which were generated using a straight Monte Carlo simulation, for two fluence levels where comparable results were available. To gain more confidence in the use of the importance sampling technique, some further checks against straight Monte Carlo results may be necessary. The same importance sampling technique was used in the evaluation of the other transient cases and is considered to be adequate for the risk assessment.

The results of the Westinghouse probabilistic fracture mechanics analysis for HBR are given in Figures B.6.2-2 and B.6.2-3. A direct comparison of HBR conditional probability of significant flaw extension results against NRC comparable values [B.3] using the normalization of T_f - RT_{NDT} is shown in Figure B.6.2-2.

Consistent with the HBR sensitivity study, the reduction in the conditional probabilities for HBR is significant for similar T_f - RT_{NDT} values. Because of the T_f - RT_{NDT} normalization, the respective results should not

FIGURE B.6.2-2

CONDITIONAL PROBABILITIES OF SIGNIFICANT FLAW EXTENSION IN THE REACTOR VESSEL FOR A SINGLE SIX FOOT LONG BELTLINE WELD-P=1000 psig

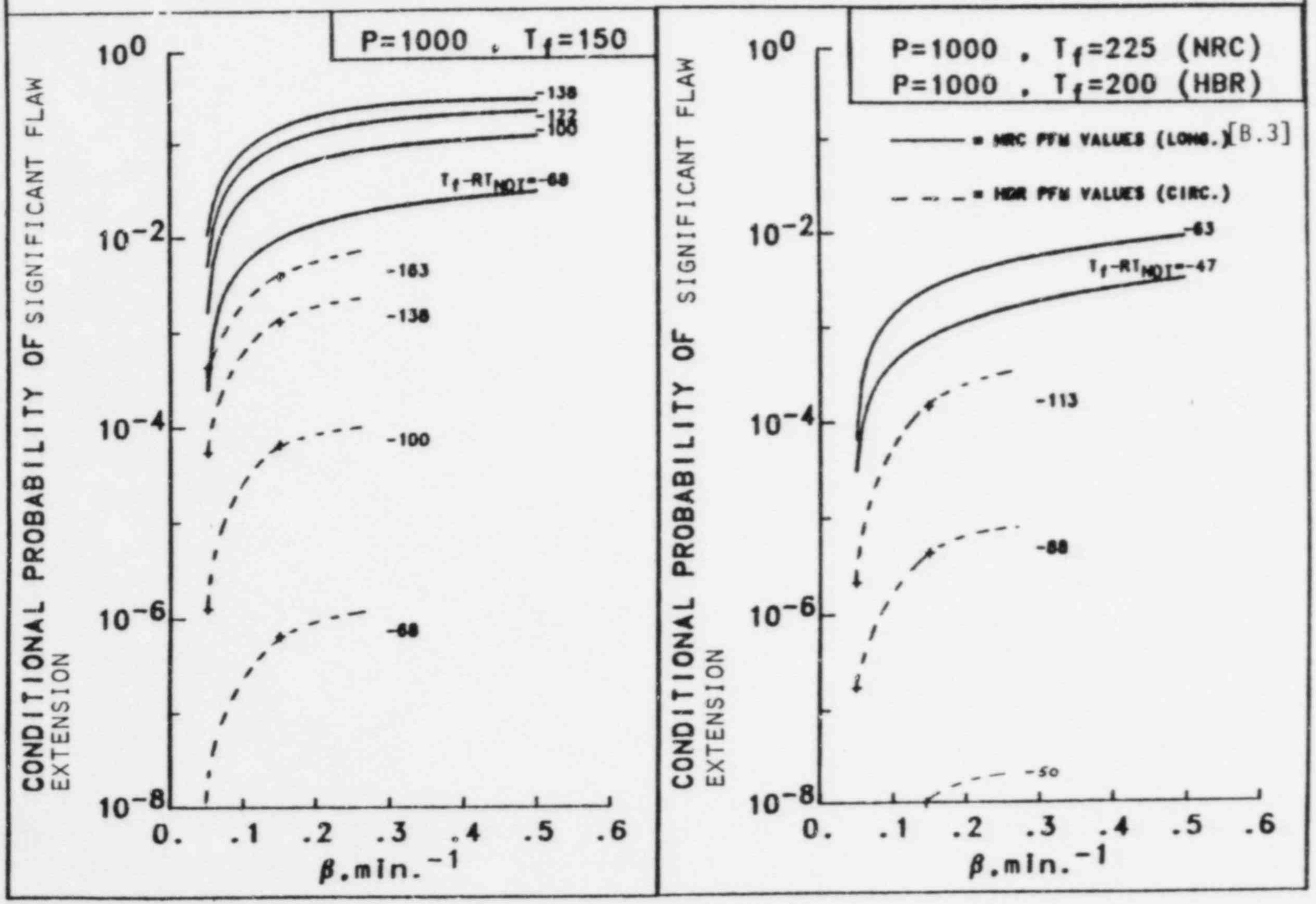
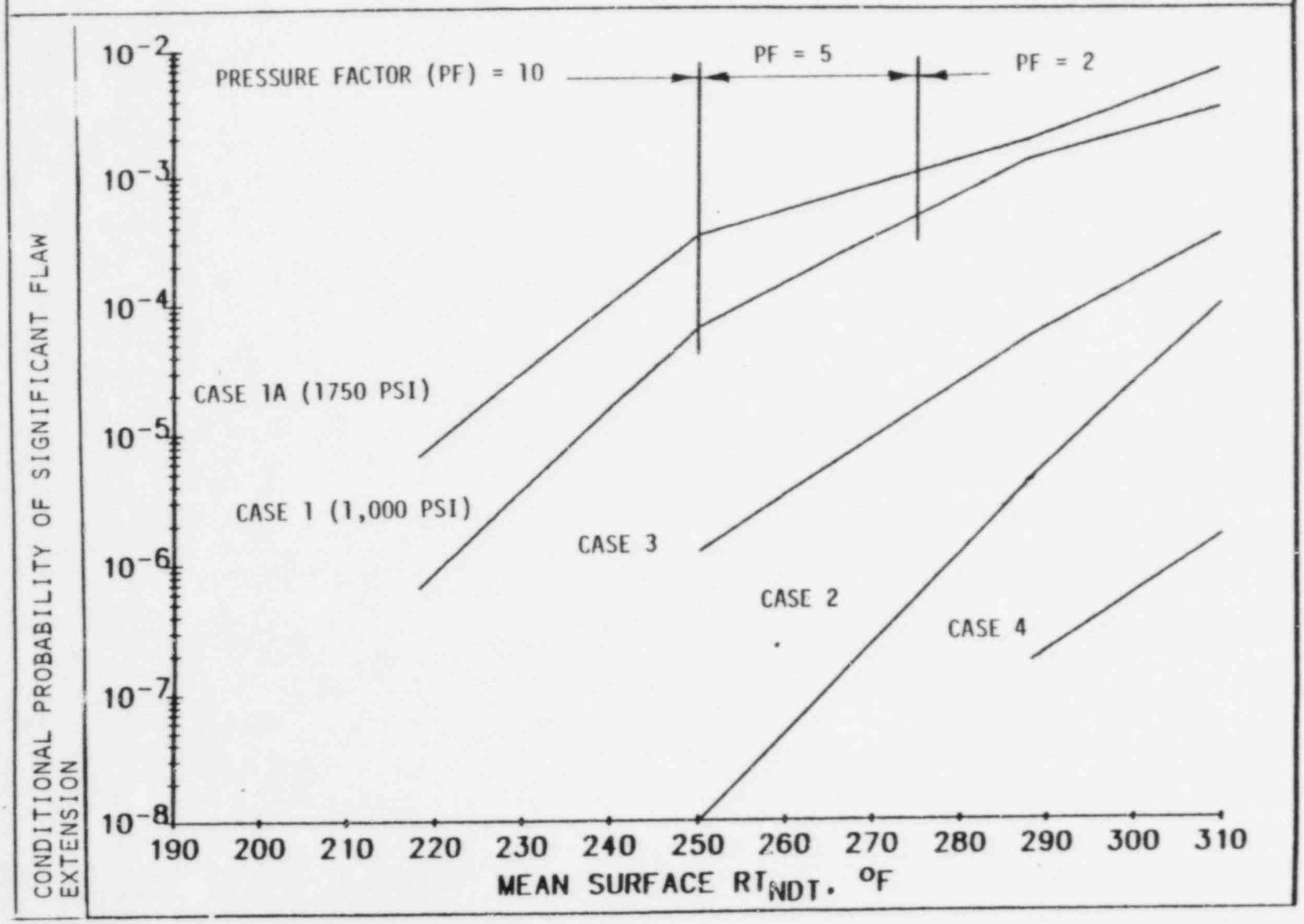


FIGURE B.6.2-3

PFM CONDITIONAL PROBABILITIES OF SIGNIFICANT FLAW EXTENSION FOR HBR BASE CASE TRANSIENTS



be used for transients that are characterized by T_f values that go well below the T_f value used in the PFM analysis. For transients characterized by T_f values above those evaluated in the PFM analysis, the conditional probability results are conservative becau.. the thermal shock to the vessel is less severe.

Figure B.6.2-3 presents the HBR results for the four base case transients as a function of mean surface RT_{NDT} including those for Case 1A to obtain the effect of pressures above 1,000 psig. Based upon the simple results from Case 1A, the conditional probabilities taken from Figure B.6.2-2 should be multiplied by the appropriate pressure factors shown in Figure B.6.2-3, depending upon the mean RT_{NDT} of interest, for transients with characteristic pressures above 1,000 psig.

Finally, to obtain the total conditional probability of significant flaw extension for use in the HBR risk analysis, the probabilities given in Figure B.6.2-2 for a 6 foot long weld should be multipled by a factor of 7. The length of the HBR circumferential weld of interest for PTS is about 43 feet in length. This weld volume factor is synonymous to the NRC's factor of 6 to determine the total probability of significant flaw extension for a vessel limited by longitudinal welds [B.3].

The PFM analysis results that have been generated and presented are most appropriately used in a relative sense for identifying plant specific variables and considerations having a significant effect on the risk of significant flaw extension for HBR. Because of the uncertainties and limitations in the PFM analysis methods and assumptions, identified above and collectively in Section V of the main report, use of the results in an absolute sense is still inappropriate at this time. Although there may be a tendency to view the results in an absolute sense when performing the probabilistic risk assessment, the restrictions in the use of the PFM analysis must be kept in mind.

B.7 SYNTHESIS OF ETA, T&H, AND PFM INTO RISK (STEPS 4 THROUGH 8)

B.7.1 Synthesis Process For Each Transient Category

The frequency curves of significant flaw extension in the HBR vessel, which are the object of this analysis, are found from a combination of the result of the several analyses described in the preceding sections. They are developed using (1) the Transient Probabilities, BFREQ, generated in the Event Tree Analysis; (2) the cooldown rates, β, and Final Temperatures, T_f, developed from the Transient Analyses and (3) the Conditional Probabilities of Significant Flaw Extension, developed from the Probabilistic Fracture Mechanics (PFM) analysis. The results are sorted into appropriate "bins" for various conditions considered for each transient. As an example, each transient is evaluated for three reactor core decay heat levels.

The risk curves are the Frequency of Significant Flaw Extension (F_{SFE}) expressed in terms of "occurrences per reactor year" and are displayed as a function of mean surface RT_{NDT}. This allows the risk for the vessel to be seen as a function of vessel age. The risk of significant flaw extension curves, which are plots of F_{SFE} vs RT_{NDT} are the combination of the transient probability (BFREQ) with the conditional probabilities of significant flaw extension if the transient does occur. Each transient was evaluated in specific bins defined within the respective transient category.

The conditional probability of significant flaw extension used for each transient is selected using the curves from the PFM analysis based on the respective stylized pressure, β, T_f and relative $T_f - RT_{NDT}$. Conditional probabilities for transients with final temperatures greater than 200°F were selected from the curves shown on the right of Figure B.6.2-2 generated at T_f = 200°F for HBR. Probabilities for transients with final temperatures less than 200°F were selected from the curves shown on the left of Figure B.6.2-2, which is based on a lower T_f = 150° for HBR.

Pressure effects the magnitude of the Conditional Probabilities. For transients with characteristic pressures above 1000 psi, conditional probabilities were determined from Figure B.6.2-2 for a pressure of 1,000 psi, then were adjusted appropriately using the pressure factors given in Figure B.6.2-3 depending upon the RT_{NDT} of interest.

Once the conditional probabilities were found, they were combined with the BFREQ values to generate the F_{SFE} values for each bin as described below. The F_{SFE} values for each bin are then combined to provide the the risk of significant flaw extension for each transient category. As described in Section B.6, these probabilities were then multiplied by a factor of 7 to account for the relative amount of weld volume being considered. The process is repeated for each RT_{NDT} of interest.

B.7.1.1 Loss of Coolant Accident (LOCA)

Table B.7.1-1 presents the results for the SLOCA transient category showing Transient Probabilities, BFREQ, cooldown rates, β, final temperatures, T_F, and Significant Flaw Extension, F_{SFE}. Only SLOCA was analyzed because all other LOCA events result in insignificant F_{SFE} values and do not contribute to PTS risk as demonstrated in WCAP-10319[B.4]. The results are shown for three core decay heat bins where the BFREQ is partitioned based on methods described in Reference [B.1, B.4]. The T_f and β values, were determined from the transient analysis described in Section B.4. The benefit of warm prestressing was not taken into account to maintain consistency in the risk assessment results. The NRC analysis in Reference [B.3] did use the benefit of warm prestressing for SLOCA.

The transient probabilities decrease for lower decay heat levels, which leads to lower risk levels, but the final temperature also reduces by about 6°F (~ 5 percent). The cooldown rate stays approximately the same at β = .088/min. However, the F_{SFE} values for each bin are within an order of magnitude of one another.

TABLE B.7.1-1

FREQUENCY (PER R-YR) OF SIGNIFICANT FLAW EXTENSION VERSUS RT_{NDT} FOR LOSS OF COOLANT ACCIDENT

PTS Bins*	Bin Frequency	T_f	β	RT_{NDT} 200°F	250°F	300°F	350°F
LOCA-DH1-S2-L1-OP1	4.7×10^{-4}	121	.088	4.7×10^{-10}	1.2×10^{-7}	1.2×10^{-6}	4.2×10^{-6}
LOCA-DH2-S2-L1-OP1	9.5×10^{-5}	116	.088	1.71×10^{-10}	4.8×10^{-8}	2.9×10^{-7}	9.5×10^{-7}
LOCA-DH3-S2-L1-OP1	6.3×10^{-5}	115	.088	1.26×10^{-10}	3.2×10^{-8}	1.9×10^{-7}	6.3×10^{-7}
Total Category Frequency	6.28×10^{-4}						

* Large break (> 6") and small break (< 1.5" non-isolable and including all isolable small LOCAs) LOCAs are shown not to be of PTS concern. See Reference [B.4].

	200°F	250°F	300°F	350°F
Risk per 6' weld length	7.7×10^{-10}	2.0×10^{-7}	1.7×10^{-6}	5.8×10^{-6}
Risk for total vessel	5.4×10^{-9}	1.4×10^{-6}	1.2×10^{-5}	4.1×10^{-5}

B.7.1.2 Steam Generator Tube Rupture (SGTR)

Table B.7.1-2 shows the transient probabilities and the frequencies of significant flaw extension for the SGTR. Again, the BFREQ, ß and T_F values are provided for three decay heat states evaluated. The lower decay heat bins have lower frequencies than the higher decay heat bins as was true for the SLOCA.

The binning for the SGTR not only includes results for three decay heat cases, it is further separated into two bins representing "prompt" and "delayed" operator action to terminate SI. Prompt termination is defined in this case as within three minutes, delayed action is between 3 and 30 minutes. The transient probability for each decay heat bin then is also split between delayed and prompt action. Bins are also provided for pumps running and pumps tripped based upon R.C. Pump trip criteria given in Revision 1 to the WOG Emergency Response Guidelines, which are being implemented for HBR.

B.7.1.3 Secondary Depressurization (SD)

Tables B.7.1-3 shows the BFREQ, T_f, ß, and F_{SFE} values for SD. In these cases, the binning is for three levels of decay heat as above and, in addition, is refined to consider 5 different times of operator action to control auxiliary feedwater, for each decay heat level.

For the large SD, which represents breaks greater than the area equivalent to one valve, the BFREQ and T_f are less for longer delays in operator action. The cooldown rate, which is rather rapid, remains the same for all bins at ß = 0.12 min^{-1}.

For the small secondary depressurizations, the event frequency represents all those applicable breaks equal to or smaller than the area equivalent to one valve. The BFREQ and T_f values again are less for longer delays in operator action. However, the cooldown rate is slower for lower decay heat bins where ß drops to .05 min^{-1}.

TABLE B.7.1-2

FREQUENCY (PER R-YR) OF SIGNIFICANT FLAW EXTENSION VERSUS RT_{NDT} FOR STEAM GENERATOR TUBE RUPTURE

PTS Bins	Bin Frequency	T_f	ß	RT_{NDT} 200°F	250°F	300°F	350°F
SGTR-DH1-S1-OSI1-OR1	1.99×10^{-2}	490	0.8	N	N	N	N
SGTR-DH1-S1-OSI2-OR1	2.01×10^{-4}	490	0.8	N	N	N	N
SGTR-DH2-S1-OSI1-OR1	3.98×10^{-3}	490	0.8	N	N	N	N
SGTR-DH2-S1-OSI2-OR1	4.02×10^{-5}	490	0.8	N	N	N	N
SGTR-DH3-S1-OSI1-OR1	2.05×10^{-3}	490	0.8	N	N	N	N
SGTR-DH3-S1-OSI2-OR1	2.68×10^{-3}	450	0.8	N	N	N	N
SGTR-DH1-S1-OSI1-OR2	9.20×10^{-3}	314	.14	N	N	N	1.8×10^{-10}
SGTR-DH1-S1-OSI2-OR2	9.29×10^{-5}	179	.09	4.6×10^{-13}	2.3×10^{-10}	2.2×10^{-8}	3.7×10^{-7}
SGTR-DH2-S1-OSI1-OR2	1.89×10^{-3}	230	.12	N	9.4×10^{-12}	9.4×10^{-10}	7.6×10^{-7}
SGTR-DH2-S1-OSI2-OR2	1.86×10^{-5}	150	.08	9.3×10^{-13}	1.8×10^{-9}	3.4×10^{-8}	1.1×10^{-7}
SGTR-DH3-S1-OSI1-OR2	1.23×10^{-3}	194	.11	N	3.1×10^{-10}	1.2×10^{-7}	4.2×10^{-6}
SGTR-DH3-S1-OSI2-OR2	1.24×10^{-5}	138	.08	9.3×10^{-12}	3.1×10^{-9}	3.7×10^{-8}	9.9×10^{-8}
Total Category Frequency	4.13×10^{-2}						
Risk per 6' weld length				1.1×10^{-11}	5.4×10^{-9}	2.1×10^{-7}	5.5×10^{-6}
Risk for total vessel				7.7×10^{-11}	3.8×10^{-8}	1.5×10^{-6}	3.8×10^{-5}

TABLE B.7.1-3

FREQUENCY (PER R-YR) OF SIGNIFICANT FLAW EXTENSION VERSUS RT_{NDT} FOR SECONDARY DEPRESSURIZATION

PTS Bins*	Bin Frequency	T_f	β	RT_{NDT} 200°F	250°F	300°F	350°F
SD-DH1-S1-L2-OA	2.2×10^{-3}	340	.1	N	N	N	N
SD-DH1-S1-L2-OA2	9.3×10^{-4}	340	.1	N	N	N	N
SD-DH1-S1-L2-OA3	8.8×10^{-5}	340	.1	N	N	N	N
SD-DH1-S1-L2-OA4	4.4×10^{-6}	340	.1	N	N	N	N
SD-DH1-S1-L2-OA5	2.3×10^{-7}	330	.1	N	N	N	N
SD-DH2-S1-L2-OA1	1.2×10^{-4}	285	.06	N	N	N	9.6×10^{-12}
SD-DH2-S1-L2-OA2	5.0×10^{-5}	285	.06	N	N	N	4.0×10^{-12}
SD-DH2-S1-L2-OA3	4.7×10^{-6}	235	.06	N	N	N	3.8×10^{-13}
SD-DH2-S1-L2-OA4	2.3×10^{-7}	285	.06	N	N	N	1.8×10^{-14}
SD-DH2-S1-L2-OA5	1.2×10^{-8}	275	.06	N	N	N	2.2×10^{-15}
SD-DH3-S1-L2-OA1	5.1×10^{-5}	220	.05	N	N	1.0×10^{-11}	1.0×10^{-9}
SD-DH3-S1-L2-OA2	2.2×10^{-5}	220	.05	N	N	4.4×10^{-12}	4.4×10^{-10}
SD-DH3-S1-L2-OA3	2.1×10^{-6}	220	.05	N	N	4.2×10^{-13}	4.2×10^{-11}
SD-DH3-S1-L2-OA4	1.0×10^{-7}	220	.05	N	N	2.0×10^{-14}	2.0×10^{-12}
SD-DH3-S1-L2-OA5	5.4×10^{-9}	212	.05	N	N	2.2×10^{-15}	2.2×10^{-13}
SD-DH1-S2-L2-OA1	1.1×10^{-4}	245	.2	N	N	4.4×10^{-11}	1.1×10^{-8}
SD-DH1-S2-L2-OA2	4.6×10^{-5}	245	.2	N	N	1.8×10^{-11}	4.6×10^{-9}
SD-DH1-S2-L2-OA3	4.3×10^{-6}	245	.2	N	N	1.7×10^{-12}	4.3×10^{-10}
SD-DH1-S2-L2-OA4	2.2×10^{-7}	245	.2	N	N	8.8×10^{-14}	2.2×10^{-11}
SD-DH1-S2-L2-OA5	1.1×10^{-8}	212	.2	N	2.8×10^{-15}	1.3×10^{-13}	1.3×10^{-11}
SD-DH2-S2-L2-OA1	5.9×10^{-6}	215	.2	N	1.5×10^{-12}	7.1×10^{-11}	7.1×10^{-9}
SD-DH2-S2-L2-OA2	2.5×10^{-6}	215	.2	N	6.3×10^{-13}	3.0×10^{-11}	3.0×10^{-9}
SD-DH2-S2-L2-OA3	2.3×10^{-7}	215	.2	N	5.8×10^{-14}	2.8×10^{-12}	2.8×10^{-10}
SD-DH2-S2-L2-OA4	1.1×10^{-8}	215	.2	N	2.8×10^{-15}	1.3×10^{-13}	1.3×10^{-11}

TABLE B.7.1-3 (Con't)

FREQUENCY (PER R-YR) OF SIGNIFICANT FLAW EXTENSION VERSUS RT_{NDT} FOR SECONDARY DEPRESSURIZATION

PTS Bins*	Bin Frequency	T_f	β	RT_{NDT} 200°F	250°F	300°F	350°F
SD-DH2-S2-L2-OA5	5.9×10^{-10}	212	.2	N	1.5×10^{-16}	7.1×10^{-15}	7.1×10^{-13}
SD-DH3-S2-L2-OA1	2.5×10^{-6}	212	.2	N	6.3×10^{-13}	3.0×10^{-11}	3.0×10^{-9}
SD-DH3-S2-L2-OA2	1.1×10^{-6}	212	.2	N	2.8×10^{-13}	1.3×10^{-11}	1.3×10^{-9}
SD-DH3-S2-L2-OA3	1.0×10^{-7}	212	.2	N	2.5×10^{-14}	1.2×10^{-12}	1.2×10^{-10}
SD-DH3-S2-L2-OA4	4.9×10^{-9}	212	.2	N	1.2×10^{-15}	5.9×10^{-14}	5.9×10^{-12}
SD-DH3-S2-L2-OA5	2.7×10^{-10}	200	.2	N	1.5×10^{-16}	2.2×10^{-14}	5.4×10^{-13}
Total Category Frequency	3.64×10^{-3}						

* Secondary depressurizations caused by equivalent breaks downstream of the MSIVs do not significantly affect the total PTS risk from SD's, see Reference [B.4]. Therefore, T&H and PFM analyses have not been performed on this bin.

	200°F	250°F	300°F	350°F
Risk per 6' weld length	N	3.1×10^{-12}	2.3×10^{-10}	3.2×10^{-8}
Risk for total vessel	N	2.2×10^{-11}	1.6×10^{-9}	2.2×10^{-7}

This type of breakdown in bins within the transient category allows improved estimate of the frequencies by minimizing the conservatisms inadvertently incorporated without such resolution. For instance, rather than considering all small secondary depressurizations in one bin, the small SD's at low decay heats, which have lower final temperatures (a worse condition), have a reduced impact because of slower cooling rates (a better condition). With this type of improvement in mind, further resolution and reduction in F_{SFE} can be obtained by binning the secondary depressurization relative to operator action to control pressure, where prompt operator action would minimize the potential repressurization.

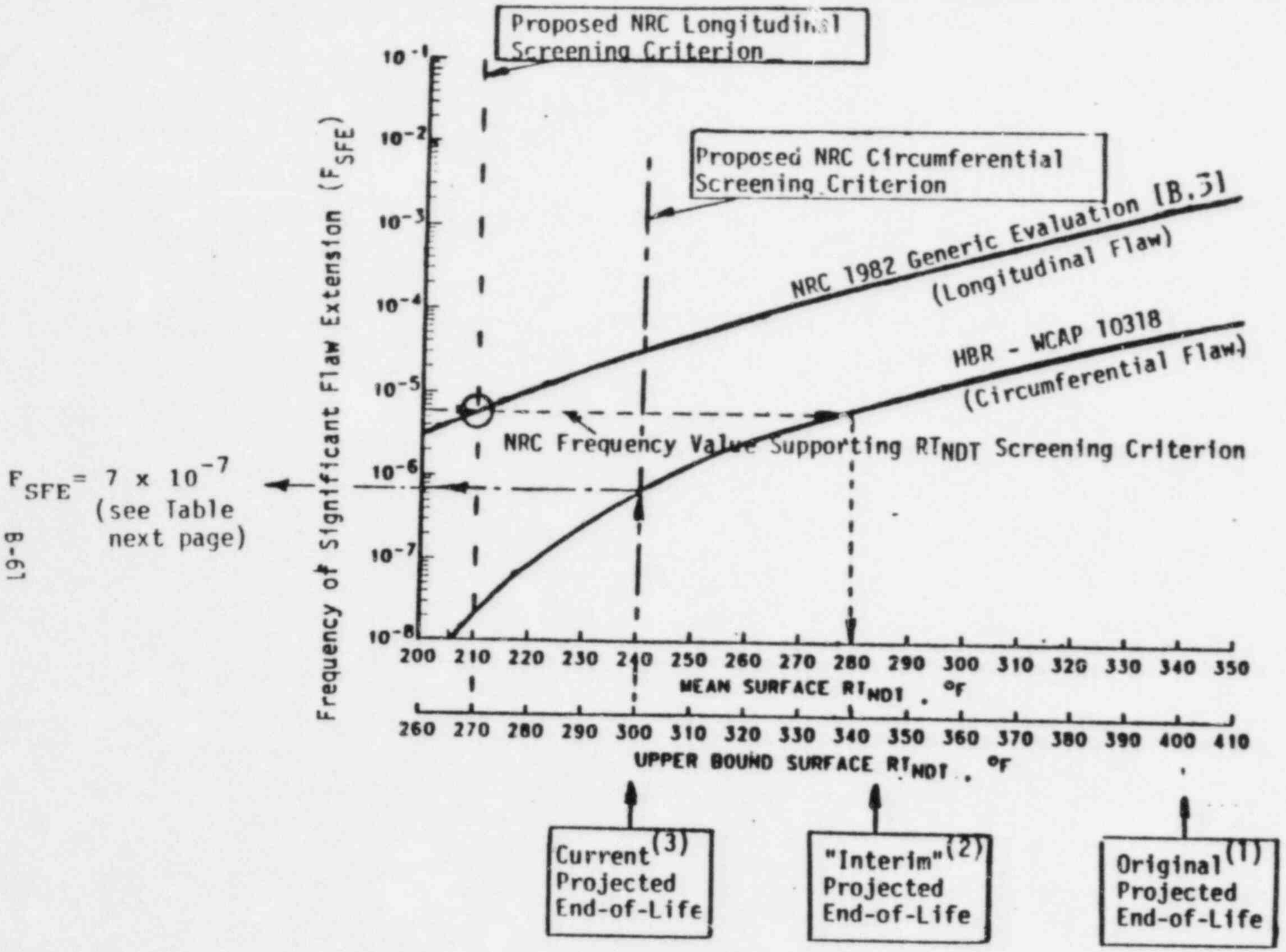
B.7.2 Risk Curves

B.7.2.1 Comparison of HBR and NRC Total Risk Curves

The risk curves of significant flaw extension from PTS, which are the object of this analysis, are determined from a combination of the analysis results from the above section. Figure B.7.2-1 provides the total risk curve of significant flaw extension for H. B. Robinson from this analysis (WCAP-10318) along with the corresponding curve from the NRC 1982 generic evaluation for PTS [B.3], which was used to support the selection of the RT_{NDT} screening criterion of 270°F for longitudinal flaws.

At each corresponding RT_{NDT} value, the H. B. Robinson frequencies are about a factor of 30 lower than the NRC generic results at the RT_{NDT} values of interest. The major sources of this improvement in risk arise from specific differences in the two analyses that include use of the H. B. Robinson geometry, the circumferential flaw orientation and the finite flaw for first initiation. Note that these factors reduce the risk below the NRC results even though factors are also included that tend to raise the calculated risk level such as, use of "displacements per atom" (dPa) in the determination of neutron fluence (rather than neutron damage for greater than 1 mev neutron energy only), and taking no credit for the benefit of warm prestressing for the small LOCA transient. The NRC evaluation did not consider the effect of dPa and did include the benefit of warm prestressing for small LOCA. These differences are all associated with the analyses to generate the conditional probability of significant flaw extension described in Appendix B.6.

FIGURE B.7.2-1 DETERMINATION OF TOTAL RISK OF FLAW EXTENSION FROM PTS FOR HBR



(1) Based on original core configuration with "upper bound" Copper and Nickel.

(2) Based on currently installed "Low Leakage Core" with "upper bound" Copper and Nickel, not including additional planned reduction.

(3) Based on currently installed "Low Leakage Core" and planned flux reduction using "Part Length Shielding" with assumed "upper bound" Copper and Nickel.

FIGURE B.7.2-1 (CONT'D)

DETERMINATION OF TOTAL RISK OF CORE MELT FROM PRESSURIZED THERMAL SHOCK OF THE H.B. ROBINSON UNIT 2 REACTOR VESSEL (CONT'D)

RISK OF CORE MELT FROM PTS (R_{CM}) = F_{SFE} X CP_{WP} X CP_{LOC} X CP_{CM}

(From Previous Page)

o $F_{SFE} = 7 \times 10^{-7}$/R-YR	Frequency of Significant Flaw Extension in the Reactor Vessel wall determined by Frequency of all PTS events x Conditional Probability of Significant Flaw Extension.
o X $CP_{WP} < 1$	Conditional Probability of (Vessel) wall penetration given that Significant Flaw Extension has occurred.
o X $CP_{LOC} < 1$	Conditional Probability of Significant Loss of Coolant given that wall penetration has occurred.
o X $CP_{CM} < 1$	Conditional Probability of Core Melt given that sufficient core cooling is available to keep up with Significant Loss of Coolant.

$= < 7 \times 10^{-7}$/R-YR, which is significantly less than the National Safety Guideline for Large Scale Core Melt of 10^{-4}/R-YR [B.15].

- Considering the Conditional Probability of Containment release at the site given that core melt occurs, the risk to public health from PTS should be significantly less than 7×10^{-7}/Reactor Year.

The benefit of the reduced risk can be translated to an increased allowable RT_{NDT} value for H. B. Robinson and is interpretable relative to the screening criteria. Figure B.7.2-1 shows that at a mean surface RT_{NDT} of 210°F, the risk level on the NRC curve is about 6×10^{-6}. At this same level of risk, the results from this study show that the acceptable mean RT_{NDT} for H. B. Robinson would be about 280°F (i.e., the circumferential weld in the H. B. Robinson vessel also has acceptably low risk through a 280°F mean RT_{NDT}).

The NRC screening criteria use the "upper bound" RT_{NDT} defined by the Guthrie Correlation [B.3] which is 60°F above the "mean" RT_{NDT} used in the NRC and H.B. Robinson risk analyses (and which is consequently used in many of the figures). In terms of this "upper bound" RT_{NDT}, the longitudinal NRC screening criterion is 270°F, which is 60°F above the 210°F "mean" surface RT_{NDT} described in the above paragraph. Likewise, the corresponding criterion for the H.B. Robinson circumferential weld is 340°F RT_{NDT} (which is 60°F above the mean value of 280°F). The relation between the "mean" and "upper bound" RT_{NDT} is explicitly shown on the horizontal axis of Figure B.7.2-1. The "mean" value of RT_{NDT} is used in the PFM analysis because it is one of the quantities that is analyzed for variation about its mean. The "upper bound" is used in specifying the material property of the reactor vessel (and consequently for the screening crteria) because it conservatively bounds the possible value of RT_{NDT} for that material.

The NRC specified RT_{NDT} circumferential screening criterion is 300°F. However, this value was not developed in the same manner as the longitudinal screening criterion. The NRC RT_{NDT} criterion for circumferential flaws, 300°F, was extrapolated from the longitudinal case using deterministic fracture mechanics analyses for flaw initiation. However, no risk analysis was performed for circumferential flaws. For this reason, no correlation in terms of risk can be made between the NRC circumferential screening criterion and the H. B. Robinson value. However, since the H. B. Robinson value of 340°F is above the NRC value of 300°F, the NRC screening criterion is viewed as a conservative value for application to H. B. Robinson.

Considering the flux reduction currently being implemented by CP&L for HBR, the end-of-life upper bound surface RT_{NDT} using assumed high copper and nickel properties is projected to be near the 300°F NRC screening value for circumferential flaws. This value translates into a frequency of significant flaw extension (F_{SFE}) of 7×10^{-7}/reactor-year from the HBR total risk curve. This value, which still does not take into account the conditional probability of core melt given that significant flaw extension occurs as described in the "risk" notes in Figures B.7.2-1, is already more than 2 orders of magnitude below the National Safety Guideline for large scale core melt of 10^{-4}/reactor-year from all accidents [B.15]. Considering the conditional probability of containment release at the site given that large scale core melt occurs, the risk to public health from PTS should be significantly less than 7×10^{-7} occurrences per reactor year.

Even though the total risk envelope from the "broad scope" evaluation, at the RT_{NDT} considering implementation of part length shielding, is below the NRC frequency value supporting the screening criterion, a detailed narrow scope evaluation is performed as described in the following section. This evaluation provides additional insight from the detailed narrow scope approach to show margins and to show how CP&L initiatives are clearly eliminating the PTS issue from further consideration. Before selecting the transients for the detailed narrow scope analysis, a discussion is given relative to HBR results versus the results recently obtained from the WOG generic PTS risk assessment [B.4]. Table B.7.2-1 summarizes the key parametric differences between the WOG, NRC, and HBR risk studies for informational purposes.

B.7.2.2 Relative Importance of the Various Transients

Figure B.7.2-2 provides a summary of both the WOG and HBR frequency of significant flaw extension versus RT_{NDT} curves for longitudinal welds in a typical Westinghouse PWR and for the circumferential weld of interest in the HBR vessel, respectively. These plots show the total risk curves along with the results from the various transient categories that were evaluated in each study to generate the total risk curves. As discussed previously, the WOG evaluation [B.4] is a complete study of PTS events, including scenarios that potentially lead to flow stagnation in the reactor coolant system. Since the

TABLE B.7.2-1

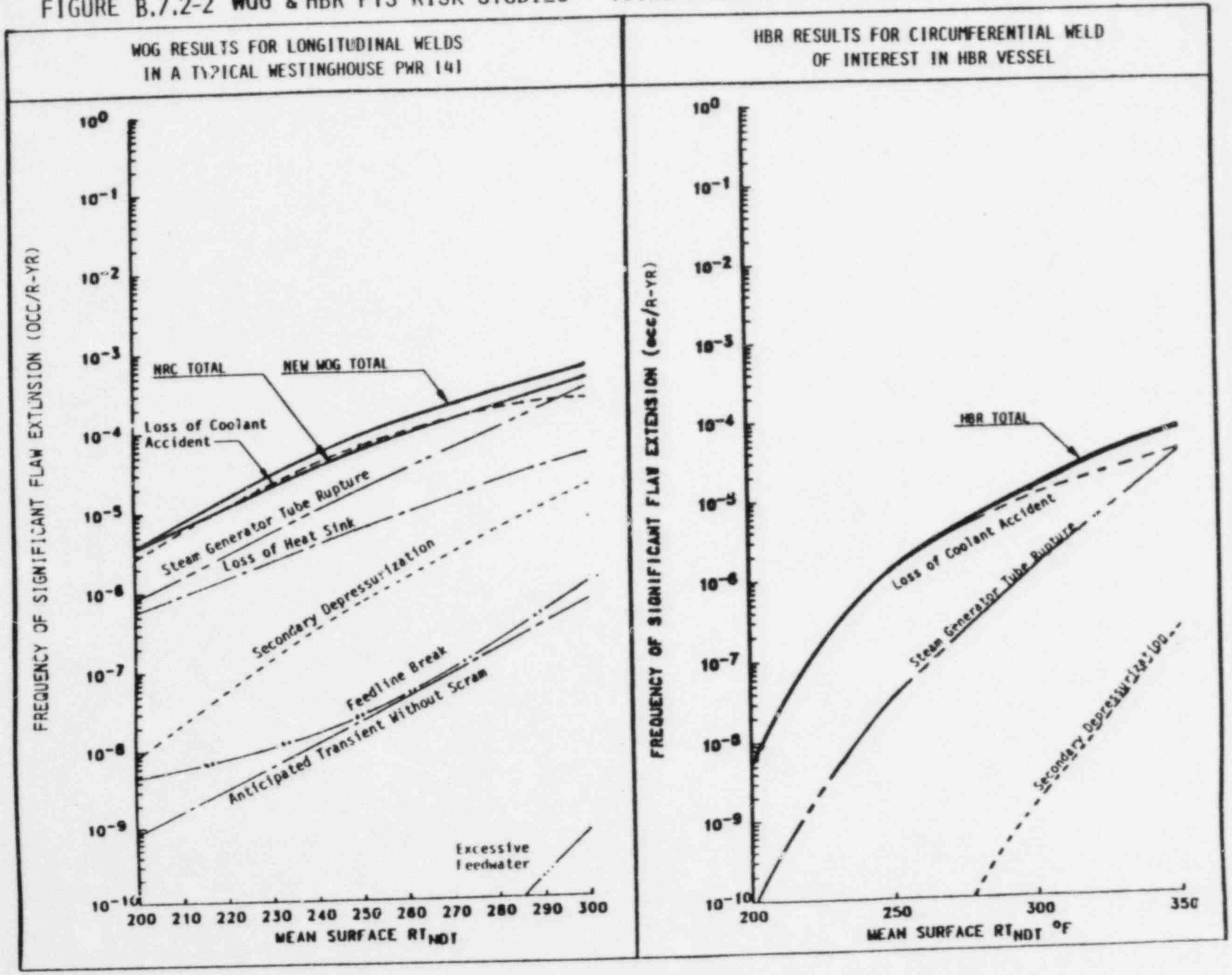
PARAMETRIC DIFFERENCES BETWEEN PTS RISK ANALYSES

Condition (Parameter)	1982 WOG [B.1,B.2]	1982 NRC[B.3)	1983 WOG STAG LOOP[B.4]	1983 HBR
o PFM Model	NRC-generic	NRC-generic	NRC-generic w/Finite Flaw for Initiation Adjustment	HBR-specific*
o Flaw Orientation	Longitudinal	Longitudinal	Longitudinal	Circumferential
o Small LOCA(1.5"-6")				
- Was it considered?	No	Yes	Yes	Yes
- w/Warm pre-stressing	-	Yes	No	No
o Transient categories other than LOCA, SGTR, SD	No	No	Yes	No
o Weld volume factor to adjust PFM probabilities	1.0	6.0	6.0	7.0
o dPa Model for fluence attenuation	No**	No**	No**	Yes

* Uses HBR model with circumferential rather than longitudinal weld, finite flaw initiation/continuous flaw arrest with Westinghouse PFM code.

** Although the dPa model was not applied, the results would not change because of the longitudinal flaw orientation and flaw shape assumptions.

FIGURE B.7.2-2 WOG & HBR PTS RISK STUDIES - TOTAL FREQUENCY OF SIGNIFICANT FLAW EXTENSION



WOG results are directly comparable to those of the HBR study and because the relation of the contributors to each other remain the same between the two risk studies, the HBR study can also be considered to effectively address the major PTS events.

In both studies the loss of coolant accident and steam generator tube rupture categories are the dominant contributors to the total risk. The other categories, including secondary depressurization, are insignificant contributors. Both studies demonstrate that secondary depressurizations no longer require focused attention as was previously suggested [B.1, B.2, B.8]. Therefore, the LOCA and SGTR categories are investigated in depth in the detailed narrow scope evaluation in the next section. The loss of heat sink (LOHS) category from the WOG assessment is also included in this analysis to assess the effect of LOHS for H. B. Robinson since it was the third largest contributor to risk. Proper treatment and inclusion of this category in the HBR PTS risk assessment would not have changed the total HBR risk results.

B.8 DETAILED RISK EVALUATION INCORPORATING ALL CP&L INITIATIVES

B.8.1 Introduction

In general, this evaluation is a plant specific analysis of the dominating transient scenarios, which were identified from the H. B. Robinson "broad scope" risk assessment, using generic thermal-hydraulic and H. B. Robinson plant specific deterministic fracture mechanics (DFM) analyses. Using an approach similar to that used by the NRC to formulate the RT_{NDT} screening criteria for PTS, the results are presented in a cumulative frequency of significant flaw extension distribution plot to provide additional insight relative to the inherent margin of safety for H. B. Robinson for PTS. CP&L initiatives are then evaluated to show the benefit derived from each initiative and to indicate the increased margins of safety for HBR.

B.8.2 Deterministic Evaluation of Risk (Steps 9, 10, 11, 12)

These steps are an evaluation of transients representative of the dominating scenarios in the loss of coolant accident and steam generator tube rupture categories, which were previously identified to dominate the PTS risk for HBR. However, before discussing the results of this evaluation, it is important at this point to show that the method used in summarizing the results for HBR is consistent with the approach employed by the U.S. NRC to select the RT_{NDT} screening values for PTS.

The experienced events, which occurred during the first 350 total PWR reactor years of operation in the United States, were used by the U.S. NRC as the basis for selecting the RT_{NDT} screening values for pressurized thermal shock (see Section 4 of SECY-82-465 [B.3]). The eight significant events were each characterized in terms of final cooldown temperature (T_f) and in terms of critical RT_{NDT} (RT_c), the limiting vessel material condition establishing the onset of vessel failure (i.e. significant flaw extension). In order to select the RT_{NDT} screening value, a cumulative frequency distribution was plotted as a function of the T_f values for the eight events. Similarly,

the RT_c results for the eight events were used to develop a plot of the cumulative frequency of events versus the RT_c for which the deterministic fracture mechanics calculations predict unacceptable crack extension will occur. The NRC [B.3] also states that

> The RT_c evaluations is, in many ways, the better way to characterize an event than using T_f alone. Calculating RT_c includes the actual time variation of temperature and pressure and is preferable to the stylized constant pressure and simple experimental temperature behavior approximation inherent in the T_f evaluation (of the eight events). Moreover, characterization of events by T_f alone requires neglect of the effect of d.ıferent pressure and different time decay constants on PTS severity.

The RT_{NDT} screening value of 270°F, selected for longitudinal flaws, was based on earlier plots of T_f and RT_c that yielded values of approximately 260°F and 280°F, respectively, for a nominal event frequency of 10^{-2} per reactor-year. "The justification for choosing 10^{-2} was only that this is comfortably lower than the range of 'anticipated operating occurrences'" [B.3]. Nevertheless, the 10^{-2} frequency was and still is a reasonable place to start for evaluating safety margins for results obtained from deterministic fracture mechanics analyses, as long as probabilistic analyses, which are consistent with those performed by the NRC to support the RT_{NDT} screening criteria, are also performed to demonstrate the degree of conservatism. Conservative assumptions are generally used in deterministic fracture mechanics analyses thereby supporting a goal higher than that used in probabilistic evaluations, which take into account mean or best estimate analyses. In theory, both goals should represent an identical status of reactor vessel integrity.

Consistent with the intent of selecting the RT_{NDT} screening value of 270°F, a plot of cumulative frequency of events versus the RT_c obtained for all PTS event sequences, which dominate the total risk of significant flaw extension as determined from a probabilistic risk assessment, can be used to assess the margin of safety for a given vessel.

Applying the above approach, an evaluation was performed and reported in WCAP-10309 [B.16] for a range of Refueling Water Storage Tank (RWST) temperatures using a generic small LOCA transient and plant specific mixing cup, stress and DFM analyses. The transient that was analyzed is representative of the small LOCA scenarios at various decay heat levels that dominated the risk of significant flaw extension for the loss of coolant accident category in the "broad scope" risk calculations for H. B. Robinson as shown in Table B.8-1. Using a criterion that significant flaw extension is defined as a flaw being unable to arrest within 75% of the vessel wall thickness, the RT_c for the small LOCA without the benefit of heating the injection water (i.e., RWST temperature = 40°F) is 380°F. The benefit of the warm prestressing phenomenon in prohibiting further flaw extension was applied in this deterministic evaluation consistent with previous NRC and WOG analyses for this transient [B.4, B.17]. Details of this plant specific transient, thermal, stress, and DFM analysis are given in WCAP-10309. A frequency of occurrence equal to 6.3×10^{-4} occ/r-yr is conservatively associated with this transient to represent the sum total of the event frequencies for the small LOCA scenarios dominating the PTS risk for H. B. Robinson.

Similarly, an evaluation of a steam generator tube rupture transient, which is representative of the scenarios that dominate the risk of significant flaw extension for the SGTR category (see Table B.8-1), was performed and is presented in Appendix D. This detailed analysis, which used the same methods as those applied in WCAP-10309 [B.16] with the exception that the warm prestressing benefit was not applied, yielded a critical RT_{NDT} of 335°F. A frequency of occurrence equal to 3.2×10^{-3} occ/r-yr is conservatively associated with this transient to represent the total event frequencies for the dominating SGTR scenarios (designated SGTR1).

TABLE B.8-1

SELECTION OF REPRESENTATIVE TRANSIENTS FOR DOMINATING TRANSIENT SCENARIOS TO PTS RISK

	Transient*	F_{SFE}* at 300-350°F RT_{NDT}	T_f*	B*	Bin* Frequency	Combined Frequency
SLOCA	LOCA-DH1-S2-L1-OP1	$10^{-7} \rightarrow 10^{-6}$	121	0.088	4.7×10^{-4}	6.3×10^{-4}
	LOCA-DH2-S2-L1-OP1	$10^{-7} \rightarrow 10^{-6}$	116	0.088	9.5×10^{-5}	
	LOCA-DH3-S2-L1-OP1	$10^{-7} \rightarrow 10^{-6}$	115	0.088	6.3×10^{-5}	
SGTR 1	SGTR-DH1-S1-OSI2-OR2	$10^{-7} \rightarrow 10^{-6}$	179	0.09	9.3×10^{-5}	3.2×10^{-3}
	SGTR-DH2-S1-OSI1-OR2	$10^{-7} \rightarrow 10^{-6}$	230	0.12	1.9×10^{-3}	
	SGTR-DH3-S1-OSI1-OR2	$10^{-7} \rightarrow 10^{-6}$	194	0.11	1.2×10^{-3}	
SGTR 2	SGTR-DH2-S1-OSI2-OR2	$10^{-8} \rightarrow 10^{-7}$	150	0.08	1.9×10^{-5}	4.1×10^{-5}
	SGTR-DH3-S1-OSI2-OR2	($10^{-6} \rightarrow 10^{-5}$ WOG)**	138	0.08	1.2×10^{-5}	
	LOHS	($10^{-6} \rightarrow 10^{-5}$ WOG)**	117	0.08	1.0×10^{-5}	

* See Table B.7.1-1 through B.7.1-3.

** Based upon WOG results from Reference [B.4].

Other transient scenarios worthy of consideration, but not via detailed thermal-hydraulic and DFM analyses, are also discussed and presented relative to the effect of these events on the total evaluation.

After the above SLOCA and SGTR representations, the next highest contributors to the risk from PTS for H. B. Robinson are two of the remaining SGTR bins (SGTR2) that were not included in the above evaluation as shown in Table B.8-1. Based upon the WOG PTS risk assessment [B.4], the loss of heat sink (LOHS) transient category would also be expected to yield the same PTS risk as the two above mentioned SGTRs bins as also shown in Table B.8-1. A transient that would be representative of these SGTR and LOHS scenarios would have transient characteristics more severe than those evaluated for the dominating SLOCA and SGTR1 events. Therefore, the critical RT_{NDT} for this transient would be expected to be less than 335°F, the value obtained for the representative dominating SGTR scenarios. The total event frequency for a representative SGTR2/LOHS transient would however be a low value of 4.1×10^{-5} occ/r-yr, the sum total of the two SGTR bins and LOHS category of interest.

All other transient scenarios are not significant contributors to PTS risk because they fall into one of the following two categories:

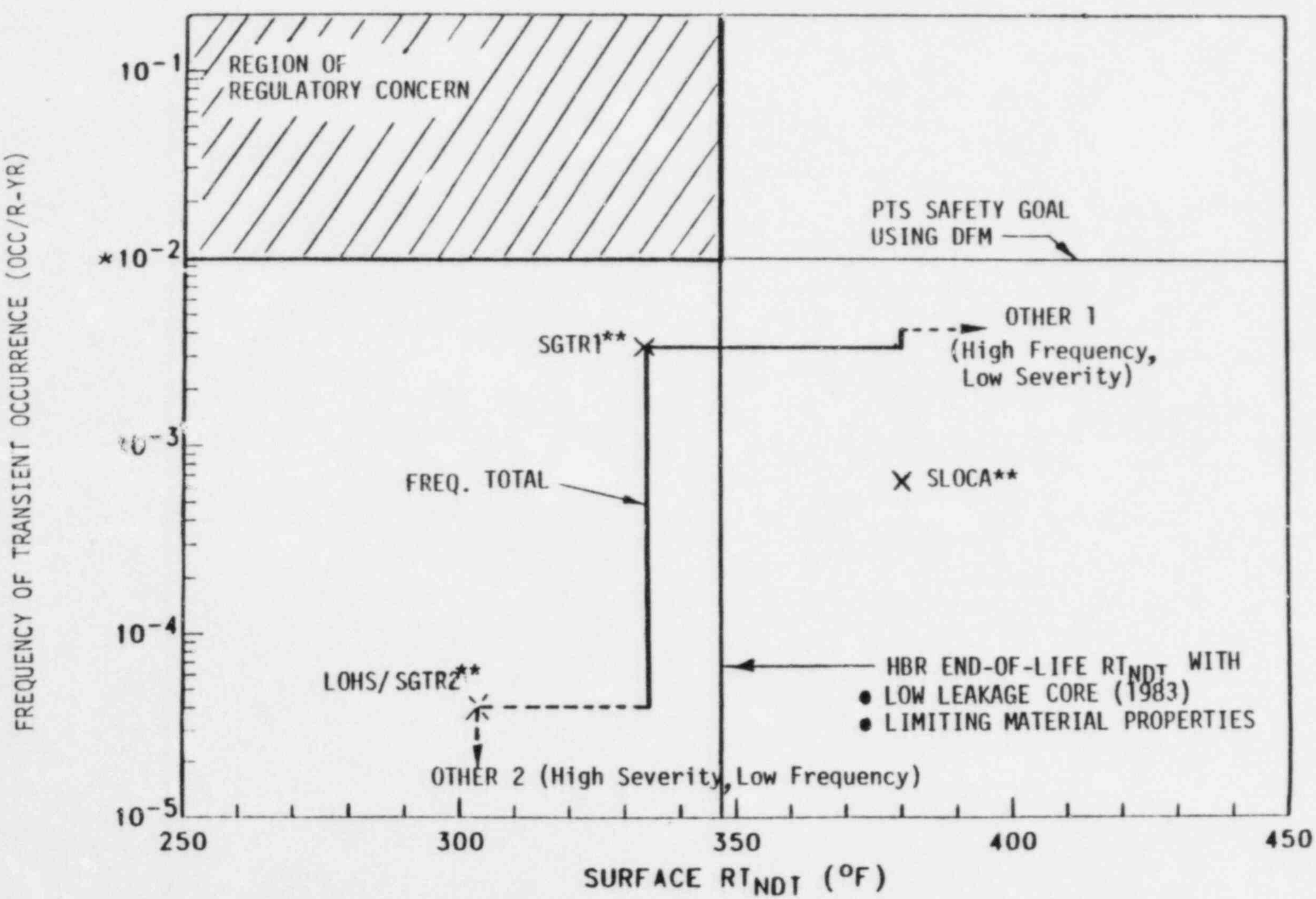
OTHER 1) The severity of the transient is not significant resulting in low conditional probabilities of significant flaw extension. Even if the scenario has a transient frequency of occurrence equal to one, the frequency of significant flaw extension is very low.

OTHER 2) The transient severity is significant, but the frequency of transient occurrence is extremely low yielding an insignificant frequency of significant flaw extension.

Most of the other transient scenarios fall into the first category (OTHER 1), while a few transients belong in the second category (OTHER 2).

Figure B.8-1 presents a plot of H. B. Robinson frequency of transient occurrence leading to significant flaw extension as a function of vessel age using the above DFM reuslts. Any consideration of CP&L initiatives that are

FIGURE B.8-1 H.B. ROBINSON-UNIT 2 FREQUENCY OF TRANSIENT OCCURRENCE LEADING TO SIGNIFICANT FLAW EXTENSION USING DETERMINISTIC FRACTURE MECHANICS (DFM) ANALYSIS



* NRC GOAL USED TO ESTABLISH RT_{NDT} SCREENING CRITERIA

** DOMINATING TRANSIENTS IDENTIFIED FROM HBR AND WOG PTS RISK STUDIES

planned are not taken into account. Starting with the same goal of 10^{-2} occ/r-yr as used by the NRC in the formulation of the RT_{NDT} screening criteria for PTS, the analyses demonstrated that the H. B. Robinson margin of safety at end-of-life, even without such initiatives, is equivalent to that inherent in the NRC screening criteria. Although the region of regulatory concern is not intersected, the next section evaluates the CP&L initiatives to demonstrate the further significant increase in the margin of safety for H. B. Robinson that is resulting from these actions.

B.8.3 Evaluation of CP&L Initiatives (Steps 13, 14)

In addition to the PTS risk study given in this report, Carolina Power and Light Company has recently sponsored several other programs to address concerns related to pressurized thermal shock of the H. B. Robinson Unit 2 reactor pressure vessel. These programs have been undertaken to further improve the safety of the vessel for PTS and to show that modifications already in place or planned will demonstrate a sufficient margin of safety for continued plant operation. A short description of each of these programs, including the individual results expected from their implementation, is provided as follows:

- Study of Benefit in Heating RWST for SBLOCA (WCAP-10309 [B.16]) - The impact of heating the Refueling Water Storage Tank for a range of temperatures above those encountered in past operating conditions on the integrity of the H. B. Robinson Unit 2 vessel during a postulated small break LOCA, which is a dominating transient scenario identified from the PTS risk study, was examined. Transient, thermal, stress, and deterministic fracture mechanics analyses were performed to obtain the results.

  The water in the RWST will be heated such that its temperature should not fall below 90°F during any weather conditions at the HBR site. Using this minimum temperature, the deterministic results in WCAP-10309 can be readily interpolated to yield a critical RT_{NDT} of 405°F, a 25°F increase over the result without the benefit of heating the RWST water.

- Flux Rate Minimization Program - This flux reduction program involves moving thrice burned fuel to the exterior of the core. This initiative was implemented in fuel cycle 9 and results in the projected end-of-life RT_{NDT} of 347°F assuming high copper/nickel properties. Further flux reductions will be achieved by using part length shielding to selectively reduce the neutron flux rate to the lower circumferential weld, the weld of concern for the HBR vessel. The projected end-of-life RT_{NDT} with the limiting material properties will be reduced to about 300°F as a result of this action scheduled to be implemented in fuel cycle 10 in 1984 [B.18]. 300°F also happens to be the NRC screening value for circumferential welds for PTS.

- Vessel Sampling/Search Program - A material property search was conducted to find the chemical properties of the HBR vessel weld material. Since the search did not yield chemical properties, the study showed that certain welds in the upper head correspond to the weld wire and flux of the specific vessel welds of interest. These head welds will be sampled in the near future to determine the copper and nickel content. The benefit expected from this CP&L initiative is to further reduce the projected end-of-life RT_{NDT}. It should be noted that all studies and evaluations performed to date on the HBR vessel have assumed limiting high copper and nickel properties.

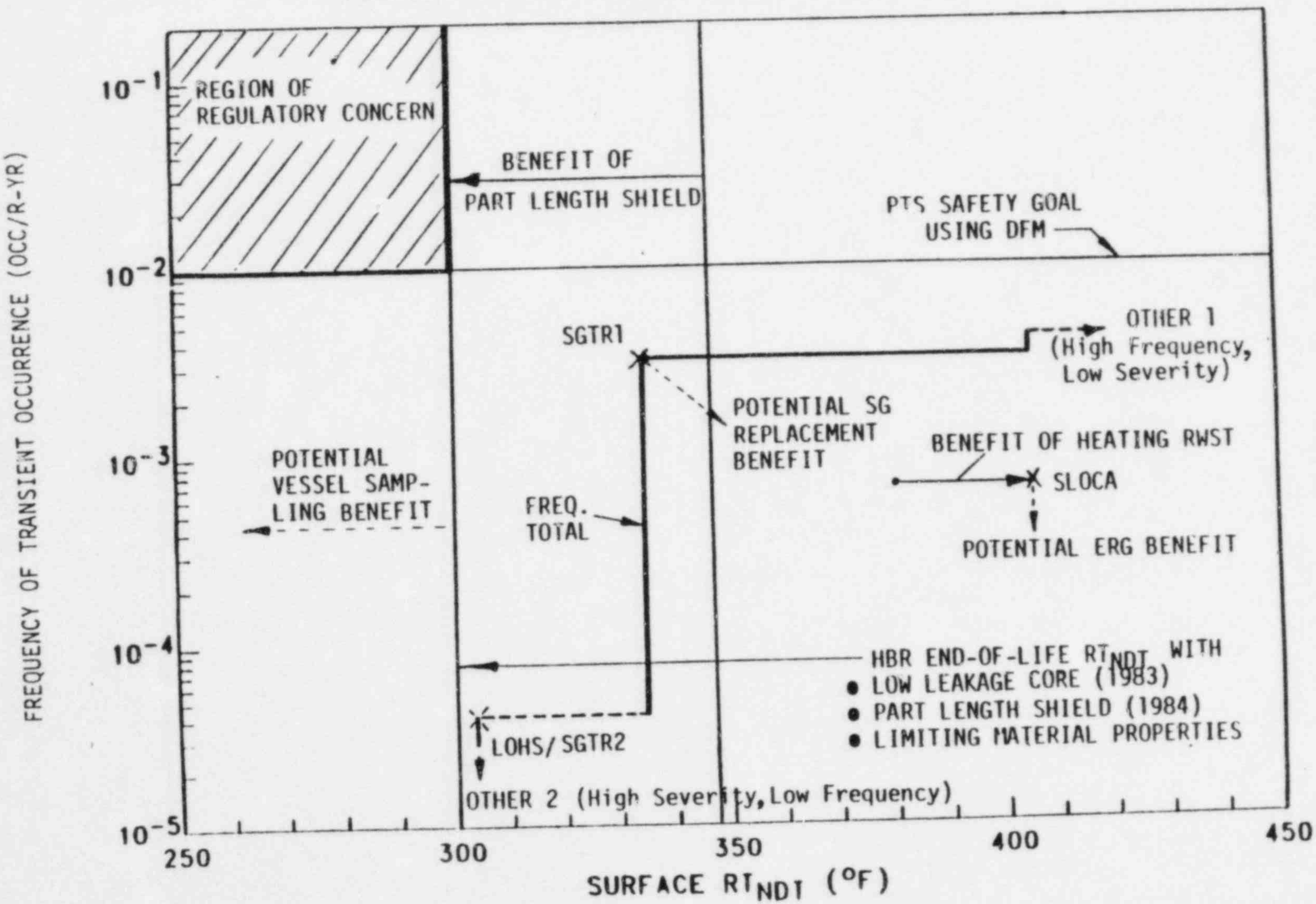
- Operator Action and Procedures - Carolina Power and Light Company has implemented Revision 1 of the Emergency Response Guidelines [B.5] on HBR that were developed by the Westinghouse Owners Group to mitigate the effects of reactor vessel pressurized thermal shock. These revised guidelines provide a significant benefit by reducing the frequency of occurrence of PTS events and, to some extent, lessen the associated transient severity via improved operator guidance. These benefits have only been taken into account for the SGTR category in both the HBR and WOG risk analyses. Extension of these results, particularly with respect to pump trip criteria at low decay heat levels, to other transient categories would provide additional benefit to the PTS risk results.

- o <u>Steam Generator Replacement Program</u> - Although this program is not being implemented because of PTS considerations, the new steam generators planned to be installed in 1984 should have a possible benefit on SGTR initiating frequencies because of stainless steel support plate material, improved heat treatment, and favorable water chemistry. Changes being implemented or planned that will improve the water chemistry should have a positive benefit by reducing corrosion concerns for the steam generators. Some improvement in the secondary depressurization transient severity due to the flow limiter in the outlet nozzle could also be expected.

Figure B.8-2 presents a plot that provides a summary of the impact of CP&L initiatives on the HBR frequency of significant flaw extension versus vessel age. This figure is synonymous to Figure B.8-1, which provided results without consideration of CP&L initiatives for PTS. From comparison of the two figures, the margin of safety from PTS events will be significantly further increased through end-of-life for HBR as a result of the above CP&L actions.

The largest and most cost-effective contributor for this increase is the planned implementation of part length shielding to maximize the reduction in neutron flux to the circumferential weld of interest. A similar increase in the margin may result from the vessel sampling program to define less limiting properties for the circumferential weld and further add to the vessel toughness. Because the steam generator tube rupture transient bins are closest to the region of regulatory concern and thus dominate the risk in the DFM evaluation, heating of RWST water has a minimal impact on the overall margin of safety. Similarly, accounting for benefits from the revised ERG's will not impact the safety margin because the benefits were already considered in the SGTR evaluations. However, the CP&L steam generator replacement program may help to reduce the SGTR frequencies and associated transient severity causing the RWST heat-up and revised ERG initiatives to play a more significant role in increasing the margin of safety for PTS.

FIGURE B.8-2 H.B. ROBINSON-UNIT 2 FREQUENCY OF TRANSIENT OCCURRENCE LEADING TO SIGNIFICANT FLAW EXTENSION USING DETERMINISTIC FRACTURE MECHANICS ANALYSIS WITH CP&L INITIATIVES



In summary, the detailed narrow scope assessment demonstrates that the HBR margin of safety for PTS events at end-of-life is equivalent to that inherent in the NRC screening criteria without the benefit of CP&L initiatives planned or implemented. Taking these initiatives into account, the margin of safety is further increased and demonstrates continued safe operation through the remainder of the service life of the vessel relative to PTS considerations.

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B.15 Federal Register, Vol. 48, No. 50, March 14, 1983, "Nuclear Regulatory Commission, Safety Goal Development Program."

B.16 Turner, R. L., R. P. Ofstun, S. L. Anderson, "Fracture Mechanics Evaluation of Heating the Refueling Water Storage Tank for the H. B. Robinson Reactor Vessel Beltline Following a Postulated Small LOCA Transient," WCAP 10309, April, 1983.

B.17 Letter OG-78, Alabama Power Co. (O. Kingsley) to U.S. NRC (H. Denton), "Westinghouse Owners Group Activities Related to Pressurized Thermal Shock", September 2, 1982.

B.18 Letter to Dr. Harold Denton of the U.S. NRC from CPL, "H. B. Robinson Steam Electric Plant Unit #2, Docket 50-261, License CPR-23 Pressurized Thermal Shock Flux Reduction Program", September 30, 1983.

APPENDIX C
CIRCUMFERENTIAL NEUTRON FLUX VARIATION EFFECT ON FLAW SHAPE CHANGE DURING FLAW GROWTH

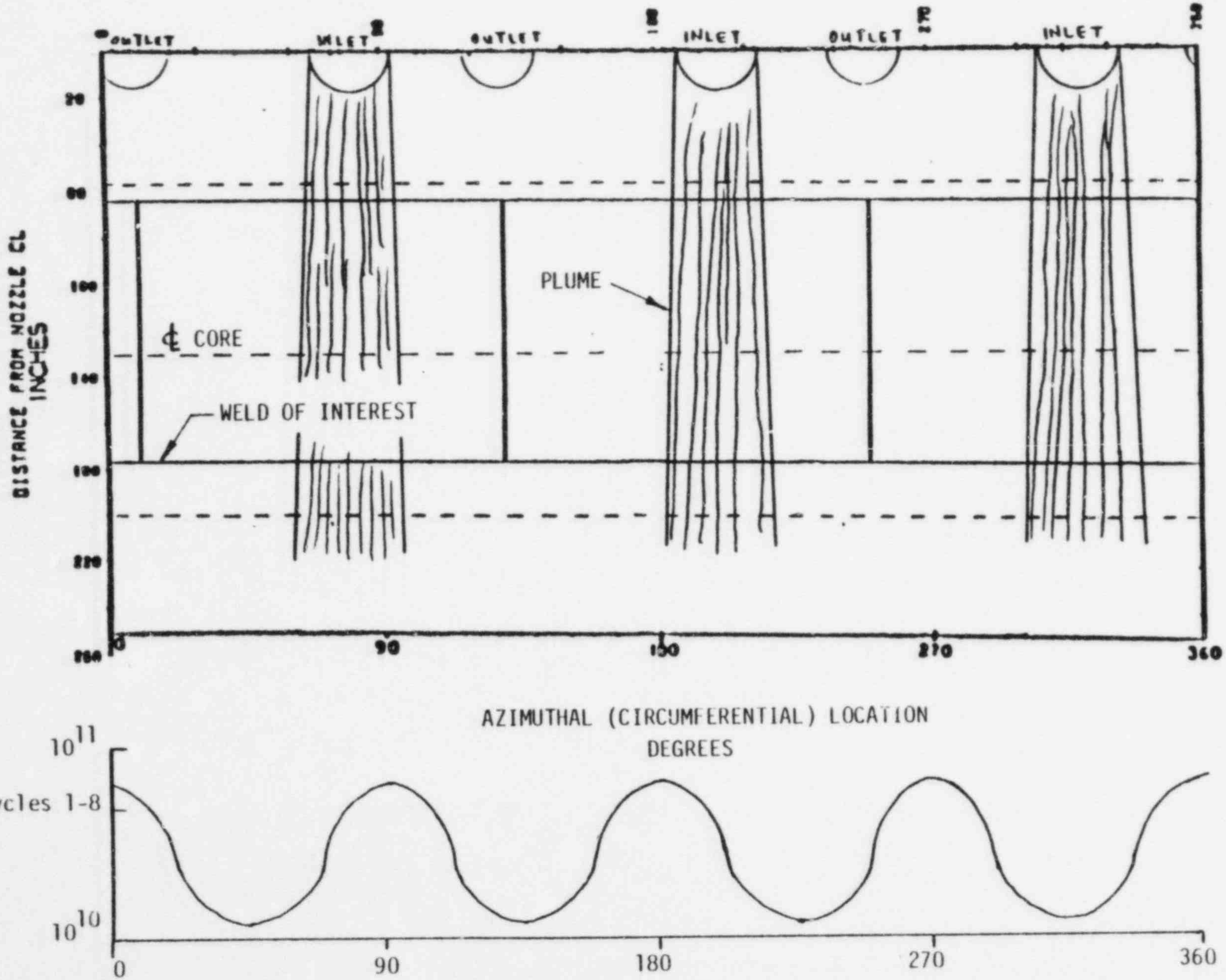
This appendix deals with new analyses using a linear elastic fracture mechanics computer program that accounts for flaw shape change effects as well as radial (through wall) and azimuthal (circumferential) variation in neutron flux. An analysis was performed for a postulated circumferential flaw, taking into account both radial and azimuthal neutron flux variation for HBR (see Figure C-1), and compared with a case where only the radial variation of neutron flux was considered. These shall henceforth be referred to as the 2D flux and 1D flux results, respectively.

The stylized transient chosen for this analysis was one with an exponential decay ($\beta = 0.15\ min^{-1}$) to 250°F from an initial temperature of 550°F. The heat transfer coefficient was maintained at 300 Btu/hr-ft^2-°F and the pressure was maintained at 2,000 psi throughout. These characteristics are actually too severe for HBR, particularly with respect to the pressure value that has been used.

A standard one-dimensional thermal and stress analysis was performed and the results were used in the multi- dimensional fracture mechanics model at a conservative peak surface RT_{NDT} of 350°F, the most limiting value of RT_{NDT} projected for HBR. The cases considered were those of a 1D flux and a 2D flux. In the ID flux case, only radial variation of flux through the wall was considered whereas in the 2D case, both radial and circumferential variations were considered. It must be pointed out that the results presented here are approximate because the K_I expressions that have been used are those for a semi-elliptical surface flaw in a flat plate. Therefore, no account is taken for curvature effects as the flaw elongates. However, relative comparisons can be made without specific regard to the accuracy of the results.

The analysis of the 1D flux showed that the initial 6:1 semi-elliptical finite flaw would extend to a very high aspect ratio (i.e., ~ infinite) and grow through the wall thickness for the severe transient being analyzed. However, when circumferential variation of fluence was considered, the flaw arrested at

FIGURE C-1 H.B. ROBINSION UNIT 2 REACTOR VESSEL WELD MAP WITH AZIMUTHAL FLUENCE PROFILE AND THERMAL PLUMES



75 percent of the wall thickness and remained "finite". The other important difference between the two results is that the aspect ratios were considerably lower in the 2D flux case. This is highly desirable and indicates the effect of circumferential variation of neutron flux on crack elongation and lends support to the use of other than infinitely long cracks for arrest. Infinitely long cracks were used for arrest in all of the probabilistic and deterministic fracture mechanics analyses presented in this report. Summary plots of the results are shown in Figures C-2 and C-3, where 2c/a is the aspect ratio, 2c the crack length and "a" the crack depth.

The benefit of considering a 2D flux is obvious but there could be an even greater benefit if this analysis were coupled with a multi-dimensional thermal and stress analysis accounting for the plume effects discussed in Appendix B of this report. The plume effect occurs under the vessel inlet nozzles, which are also shown in Figure C-1.

FIGURE C-2 SHAPE CHANGE EFFECT FOR CIRCUMFERENTIAL FLAW USING 1-D NEUTRON FLUX VARIATION

1 - 0 secs , 2C/A= 6.00 , A/T=.097
2 - 200 secs , 2C/A=28.00 , A/T=.204
3 - 600 secs , 2C/A=46.75 , A/T=.397
4 - 850 secs , 2C/A=86.47 , A/T=.848

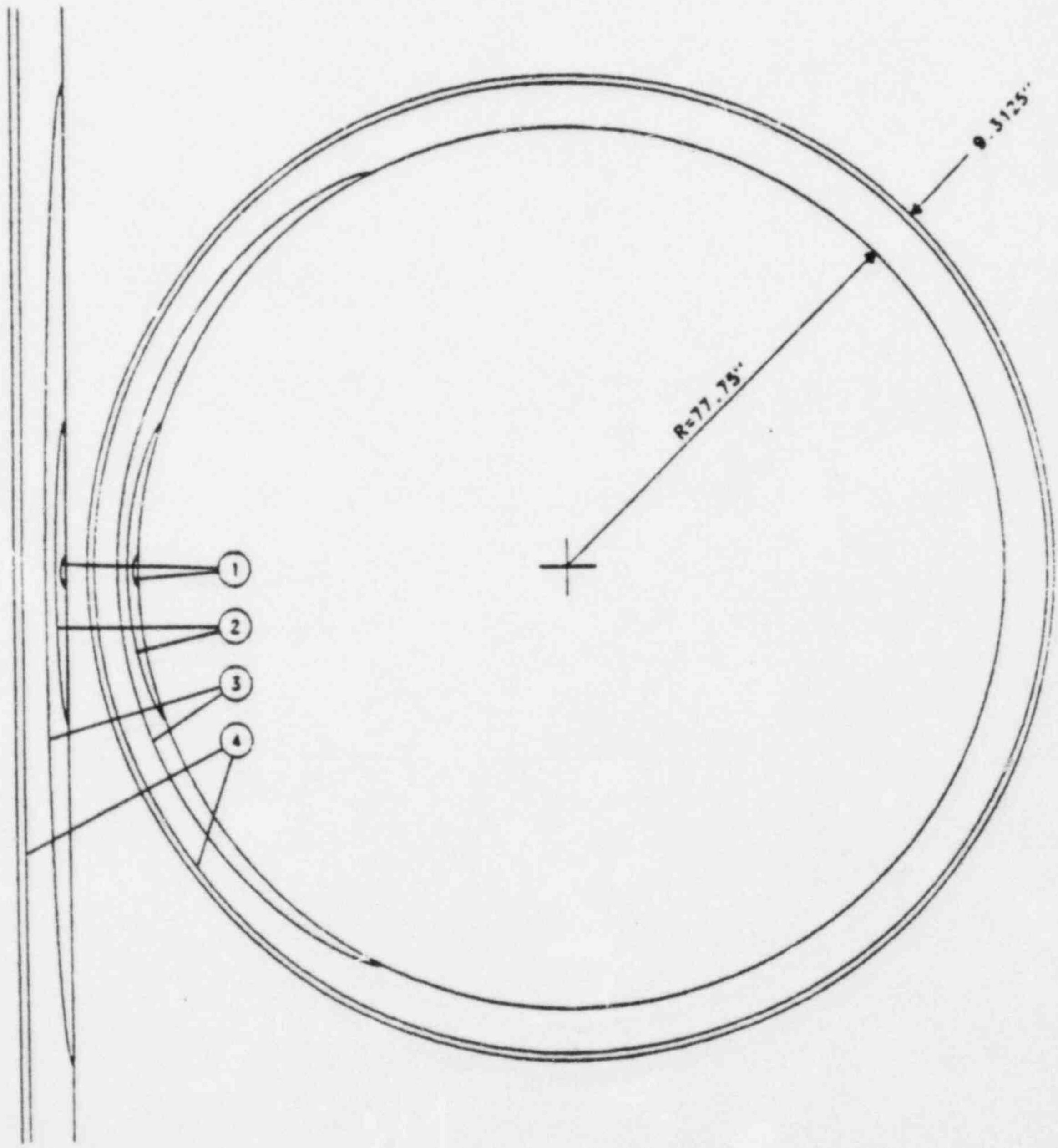
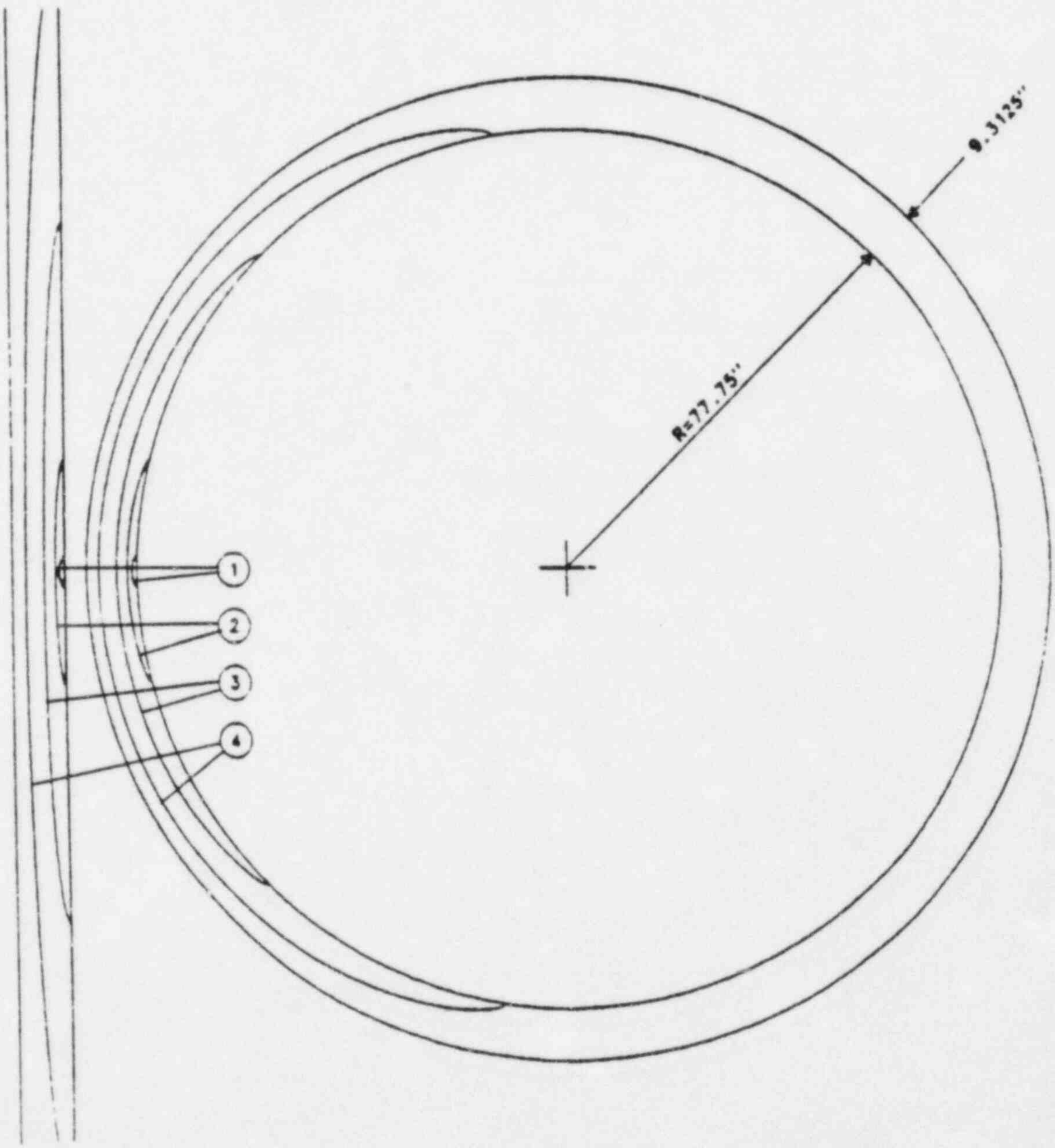


FIGURE C-3 SHAPE CHANGE EFFECT FOR CIRCUMFERENTIAL

FLAW USING 2-D NEUTRON FLUX VARIATION

1 -   0 secs , 2C/A= 6.00 , A/T=.097
2 - 200 secs , 2C/A=23.50 , A/T=.183
3 - 600 secs , 2C/A=30.22 , A/T=.397
4 -3000 secs , 2C/A=31.60 , A/T=.746



APPENDIX D
STEAM GENERATOR TUBE RUPTURE
DETERMINISTIC FRACTURE MECHANICS EVALUATION

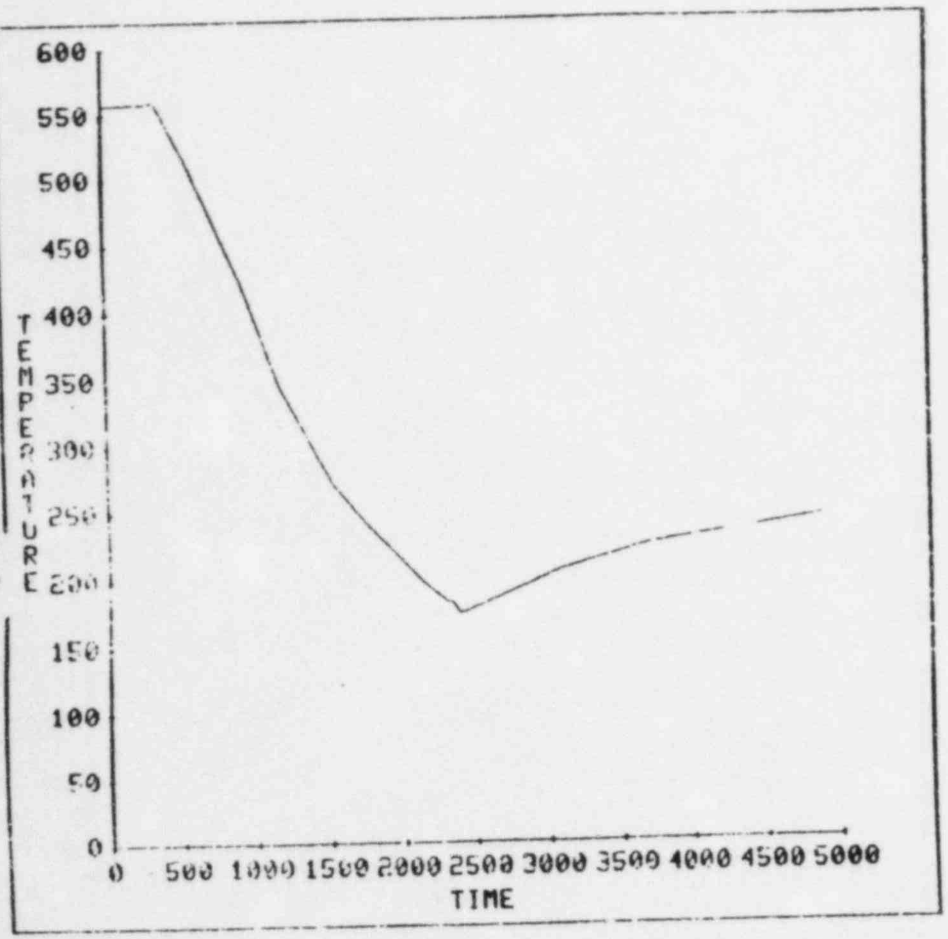
This appendix presents the results of a deterministic fracture mechanics SGTR transient evaluation performed on the H.B. Robinson Unit Number 2 reactor vessel beltline region. This transient was identified to be a dominant transient scenario from the results of the HBR risk assessment. The results are provided in terms of the flaw initiation and arrest criteria used by Westinghouse pertaining to nuclear reactor vessels. The methodology employed is similar to that described in WCAP-10309, "Fracture Mechanics Evaluation of Heating the Refueling Water Storage Tank for H.B. Robinson Reactor Vessel Beltline Following a Postulated Small LOCA Transient"[D.1]. The differences in the methodology along with a brief description are found below.

D.1 ANALYSIS

The postulated steam generator tube rupture transient was developed using a plant specific mixing cup model as described in Appendix B. The input to the code was consistent with that presented in Appendix B. The temperature history resulting from the LOFTRAN and MXGCUP computer codes is illustrated in Figure D.1-1; the pressure history is illustrated in Figure D.1-2. A free convection heat transfer coefficient is assumed throughout the evaluation.

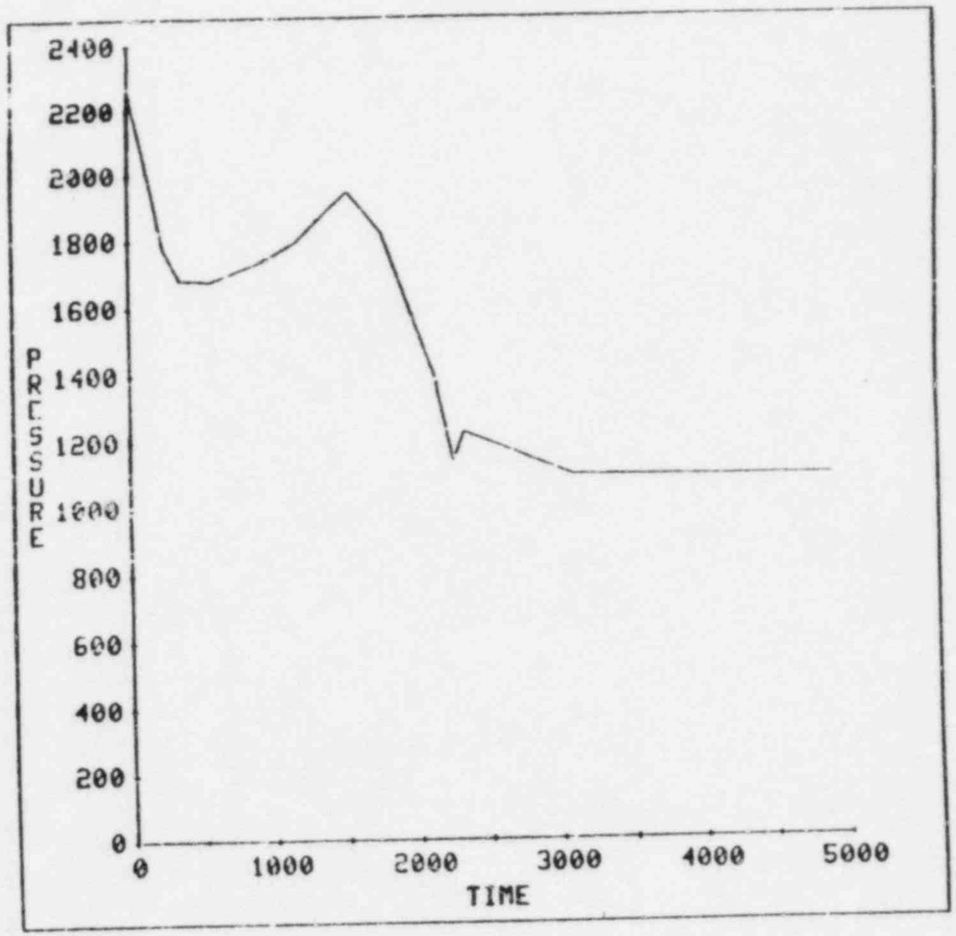
The methodology used in this analysis for the thermal and stress analyses, the neutron fluence attenuation through the vessel wall and the fracture mechanics analysis are identical to that described in WCAP 10309. The fracture mechanics analysis differs from the WCAP 10309 evaluation relative to the use of the warm prestressing benefit. The benefit of warm prestressing was not applied in the fracture mechanics analysis of the SGTR transient because of the high potential for multiple system repressurizations. (In the case of a small break LOCA transient, evaluated in WCAP 10309, the potential for repressurization is minimized by the inability of the ECC and charging systems to maintain high pressure values.)

FIGURE D.1-1



TEMPERATURE PROFILE FOR STEAM GENERATOR TUBE RUPTURE TRANSIENT

FIGURE D.1-2



PRESSURE PROFILE FOR STEAM GENERATOR TUBE RUPTURE TRANSIENT

The acceptance criteria used in this evaluation of the reactor vessel are as follows:

i. Minimum critical flaw depth* for crack initiation of a finite flaw is greater than 1.0 inch or,

ii. Crack arrest of a continuous flaw occurs within 75 percent of the vessel wall thickness.

The first initiation of a crack relies on the 6:1 finite flaw assumption, whereas crack arrest and potential subsequent reinitiation rely on the continuous flaw assumption.

D.2 RESULTS

The results of the deterministic fracture mechanics analyses for the postulated circumferential flaw in the limiting circumferential weld show an acceptable surface RT_{NDT} value of 335°F using the Guthrie/HEDL trend curves.

Figure D.2-1 illustrates the flaw growth and arrest behavior. Figure D.2-2 shows that for the critical crack size, the stress intensity drops below the arrest curve at about 74 percent of the wall depth. The results of this analyses are used in the PTS risk analysis found in Appendix B and Section IV of the main report.

D.3 REFERENCES

D.1 Turner, R. L., R. P. Ofstun, S. L. Anderson, "Fracture Mechanics Evaluation of Heating the Refueling Water Storage Tank for the H. B. Robinson Reactor Vessel Beltline Following a Postulated Small LOCA Transient," WCAP 10309, April, 1983.

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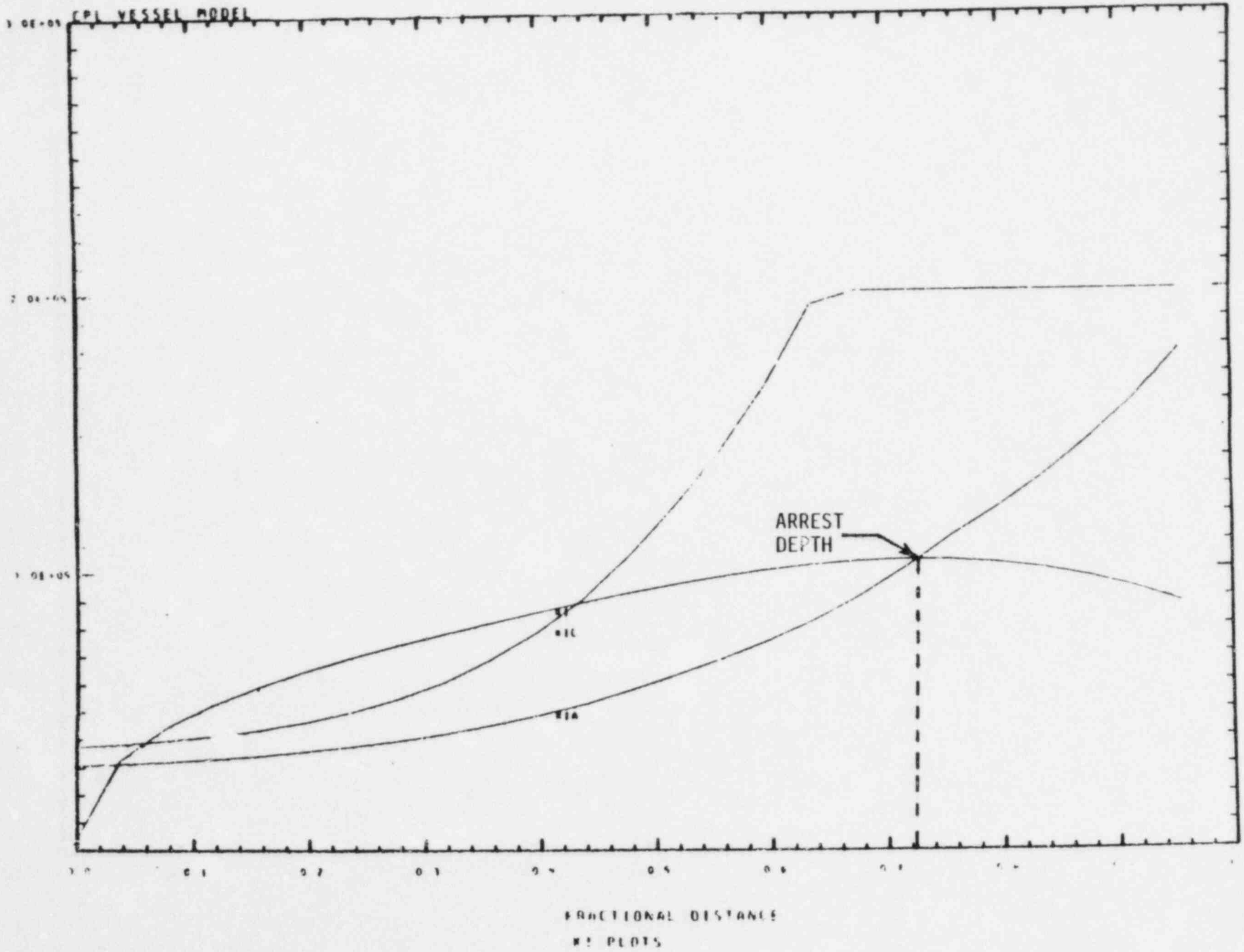
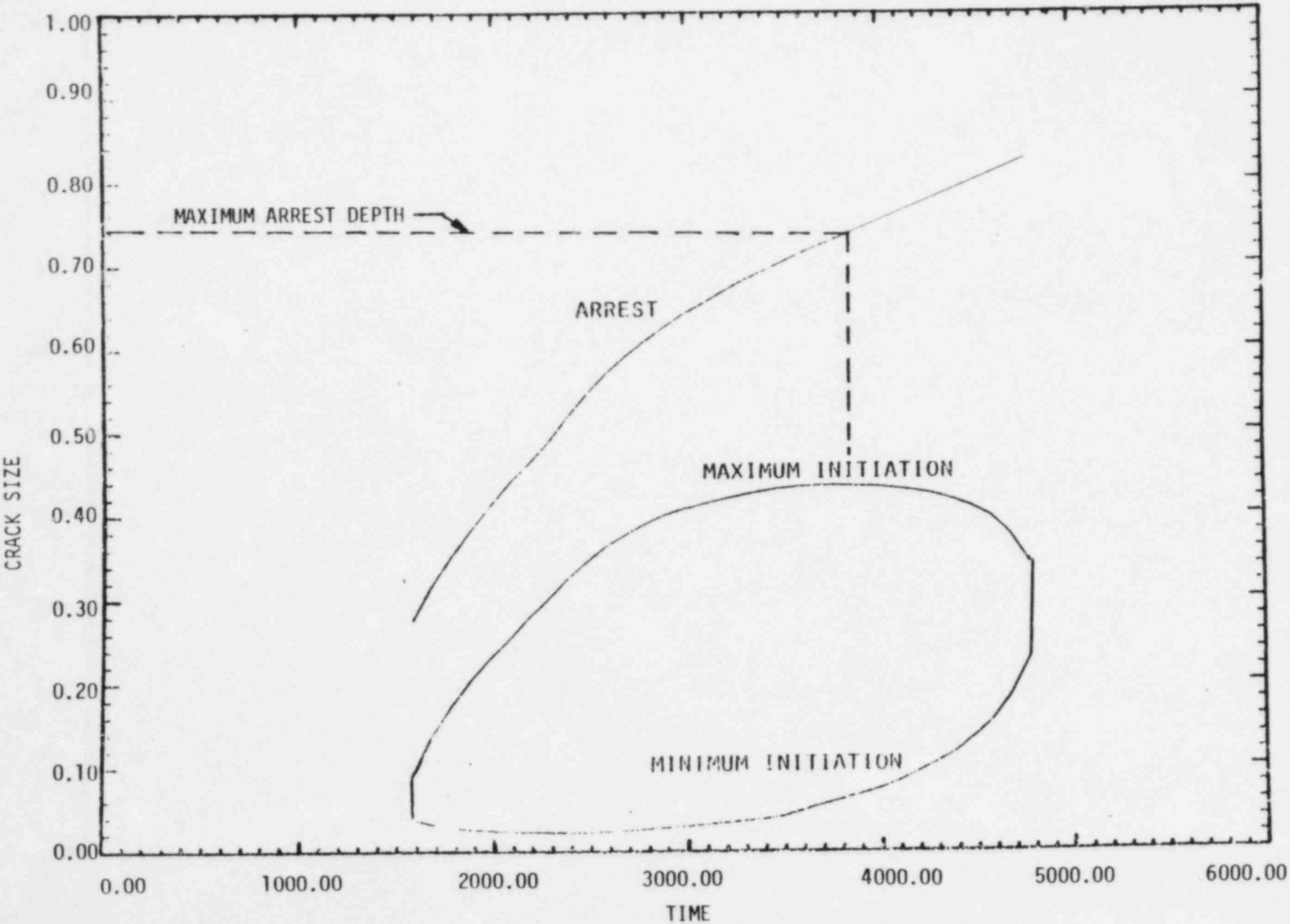
* Critical flaw depth is defined as the depth of flaw, on the inside surface of the reactor vessel that is required before flaw initiation will occur.

FIGURE D.2-1



STRESS INTENSITY FACTOR, INITIATION AND ARREST FRACTURE TOUGHNESS FOR THE STEAM GENERATOR TUBE RUPTURE.

FIGURE D.2-2



CRACK SIZE VERSUS TIME SHOWING ARREST, MAXIMUM AND MINIMUM INITIATION