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NUREG/CR-2837
PNL-4327
Supp. 1

PNL Technical Review of Pressurized Thermal Shock Issues

Supplement 1: Technical Critique of the
NRC Near-Term Screening Criteria

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Pacific Northwest Laboratory
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U.S. Nuclear Regulatory
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Manuscript Completed: November 1982
Date Published: May 1983

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NRC FIN B2510

ABSTRACT

Pacific Northwest Laboratory (PNL) provided a technical critique of the draft report, NRC Staff Evaluation of Pressurized Thermal Shock, dated September 13, 1982. This report provided the basis for the NRC near-term regulatory position on pressurized thermal shock (PTS) and recommended a generic screening criteria for welds in the vessel beltline region.

The PNL staff concluded that the screening criteria were adequate to meet the intent of the NRC safety goal and to retain past predictions of vessel reliability. The conclusion was based on selecting the plant-specific nil-ductility transition reference temperature (RT_{NDT}) in the conservative manner described within the staff report.

Conservative and unconservative factors were mentioned throughout the NRC staff report. The PNL staff has listed these factors together with unknown (may be either conservative or unconservative) factors and estimated, where possible, the range in $^{\circ}F$ RT_{NDT} . The unknown factors were so widespread that the PNL staff recommended that specific conservatisms not be reduced until the unknowns are further resolved.

EXECUTIVE SUMMARY

Pacific Northwest Laboratory (PNL) provided a technical critique of NRC's near-term NRC regulatory position on the unresolved safety issue of pressurized thermal shock (PTS). This position, which was described in a September 13, 1982 draft report, NRC Staff Evaluation of Pressurized Thermal Shock, proposed a generic screening criterion based on reactor vessel nil-ductility transition reference temperature (RT_{NDT}). Licensees with reactor vessels that are within three years of reaching the screening limit would be required to perform plant-specific analyses and specify corrective actions for NRC evaluation and approval within a yet unspecified time.

The PNL staff concluded that the screening criteria were adequate to meet the intent of the NRC safety goal and to retain past predictions of vessel reliability. This conclusion was based upon selecting the plant-specific RT_{NDT} in the conservative manner described within the NRC staff report.

The PNL staff recommended that the statistical trend curves be used for the high copper, nickel and fluence values. The NRC staff used the Regulatory Guide 1.99.1 curve for these upper bound values. It was further recommended that the plant-specific analyses be submitted to NRC for evaluation and approval two years before the screening criterion is exceeded.

The PNL staff believed that the fracture mechanics analyses used by the NRC staff to establish the screening limit were reasonable and conformed to the current state of analytical methods and data (e.g., ASME codes). However, the use of probabilistic fracture mechanics by NRC to justify licensing requirements represented a departure from traditional vessel safety evaluations, and this demanded the use of conservative fracture mechanics models. A number of significant unquantified conservatisms are present in the NRC fracture model which are probably sufficient to offset the uncertainties and unconservative features.

The conservative assumptions and methods used in establishing the screening criteria and plant-specific vessel RT_{NDT} are mentioned frequently throughout the staff report. Unconservative and unknown (may be conservative or unconservative) factors, however, are only briefly mentioned by the NRC staff. A listing of conservative, unconservative, and unknown factors, together with the estimated range in $^{\circ}F$ RT_{NDT} , was provided by PNL staff where possible. It seems evident that the unknowns are so widespread that it would be unwise to reduce specific conservatisms until the unknowns are further resolved.

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ACKNOWLEDGMENTS

This report contains an account of the work sponsored by the Division of Safety Technology, Office of Nuclear Reactor Regulation, of the U.S. Nuclear Regulatory Commission. The work was completed under the direction of Felix Litton and Roy Woods, Project Manager and Technical Monitor, respectively, Generic Issues Branch.

The dedicated efforts of Mary Kay Willis for typing, Jodie Smith, Bev Einarsson for word processing, and Carolyn Novich for editorial assistance are greatly appreciated.

1.0 INTRODUCTION

This supplement to NUREG-2837 evaluates the NRC staff's proposed near-term regulatory position on pressurized thermal shock issues presented to PNL staff and consultants on August 20, 1982, and described in an NRC staff report.⁽¹⁾ The NRC position recommends a generic vessel screening criteria of 270°F for the upper limit on RT_{NDT} for longitudinal welds and 300°F for circumferential welds. Licensees that have reactor vessels which are within three years of reaching the screening limit would be required to perform plant-specific analyses and propose corrective actions for NRC evaluation and approval within a yet unspecified time. The methods used to establish plant-specific vessel material RT_{NDT} values and probabilities of vessel failure should a PTS event occur are important issues in determining the acceptability of these screening criteria.

Also in this supplement is a preliminary value-impact analysis of the estimated cost and potential safety benefits of the proposed regulatory position.

2.0 CONCLUSIONS AND RECOMMENDATIONS

The proposed generic screening criteria described in the September 13, 1982 draft report (Chapter 4),⁽¹⁾ is judged to be adequate to meet the intent of the safety goal⁽²⁾ and to retain past predictions of the reliability of nuclear reactor vessels.⁽³⁾ This conclusion was based upon selecting the plant-specific RT_{NDT} in the conservative manner described within the NRC report (Chapter 5).⁽¹⁾

The PNL staff believes that the fracture mechanics analyses used by the NRC staff to establish the screening limit are reasonable and conform to the current state of analytical methods and data (e.g., ASME codes). However, the use of probabilistic fracture mechanics by NRC to justify licensing requirements represents a departure from traditional vessel safety evaluations, and this demands the use of conservative fracture mechanics models. The NRC fracture model contains a number of significant, unquantified conservatisms that should be sufficient to offset the uncertainties and unconservative features of the model.

Pacific Northwest Laboratory recommends that plant-specific RT_{NDT} determinations use only the statistical trend curves and not the Regulatory Guide 1.99.1 curve for the high copper, nickel, and fluence values. The adoption of an appropriate (non-zero) standard deviation for the initial RT_{NDT} of plate and forgings is also recommended to ensure that consistent generic and plant-specific RT_{NDT} criteria are applied to all vessel materials.

The conservative assumptions and methods used in establishing the screening criteria and plant-specific vessel RT_{NDT} are mentioned often throughout the NRC staff report. Only brief mention, however, is given to unconservative and unknown (i.e., conservative or unconservative) factors. Many of these factors, including the estimated range in °F RT_{NDT} that some of these factors may represent, are listed in Table 2.1. The PNL staff recommends that all conservative and unconservative factors and their possible effects be examined together rather than separately. Because the unknowns are so widespread, PNL does not recommend that specific conservatisms be reduced until the unknowns are further resolved.

The value-impact analyses in Chapter 9 provide a review of the potential costs and safety benefits for performing plant-specific analyses required by the regulatory position proposed by the NRC staff. Plant-specific analyses are a necessary step in resolving the PTS issue. The relative costs of such analyses are moderate; however, there is no net safety benefit until corrective actions are completed.

TABLE 2.1. Conservatisms, Unconservatisms and Unknown Factors and their Estimated °F RT_{NDT} Range

Conservative Factors (RT _{NDT} Range)	Unconservative Factors (RT _{NDT} Range)	Unknown Factors (RT _{NDT} Range)
<ul style="list-style-type: none"> • Surface flaws assumed rather than underclad cracks (0-35°F) • Warm prestress omitted (0-±100°F) • Worst case flaw description (long ID flaw, extending through clad and oriented normal to maximum stress) (unknown) • Suppression of lengthwise crack growth by clad neglected (unknown) • Upper limit values of chemistry assumed for poorly documented welds (unknown) • Simplistic, one-dimensional analysis applied to location of minimum toughness/maximum cooling (unknown) • High nickel content assumed in VISA calculation (0-15°F) • Use of higher pressure for evaluation of RPV failure probability than from event sequences from PRA (unknown) • Probable over-estimate of residual variance of ΔRT_{NDT} because of adjustment for uncertainty in fluence (~2°F) • Use of generic rather than plant-specific operating history for establishing screening criteria for CE facilities (unknown) 	<ul style="list-style-type: none"> • Safety factor omitted in fracture mechanics (30°F) • Residual stress omitted (0-25°F) • Reduction in apparent fracture toughness for long crack fronts not treated (~30°F) • Uncertainties in crack arrest model (<10-20°F) • Use of probabilistic fracture mechanics departure from traditional approaches (unknown) • VISA code not well documented, validated or subjected to extensive peer review (unknown) • Variability in initial RT_{NDT} not treated for plate materials (~10°F) • Best estimate of fluence used in deterministic fracture predictions (unknown) • Noncondensable gases not considered in thermal-hydraulic modeling (unknown) • Use of mean value without uncertainty on RT_{NDT} in VISA calculation (5-10°F) • Use of constant error term for screening ΔRT_{NDT} (5-10°F) • Use of generic rather than plant-specific operating history for establishing screening criteria for B&W facilities (unknown) 	<ul style="list-style-type: none"> • Simplistic treatment of heat transfer coefficient in VISA (±20°F) • ASME K_{1a} reference toughness not based on recent test methods and data from weld material (±50°F) • Simplistic stochastic model used in probabilistic fracture mechanics (unknown) • Distribution functions in VISA based on very limited data sets and engineering judgment (unknown) • Flaw distributions in VISA not based on consideration of underclad cracking mechanisms (unknown) • Reference toughness curves assumed to be independent of irradiation (unknown) • Sensitivity of irradiation damage to temperature not included in shift curves (unknown) • Temperature measured in cold leg (unknown) • Use of homogenous and equilibrium thermal-hydraulic models (unknown) • Cold leg-downcomer mixing (~10°F) • Break size, location causing SBLOCA flow stagnation (unknown) • Lack of adequate pressurizer condensation model (unknown) • Different trend curves for ΔRT_{NDT} in VISA and screening criteria calculation (unknown) • Assumption of independent uncertainties in VISA (unknown) • Use of generic rather than plant-specific operating history for establishing screening criteria for W facilities (unknown) • Use of past operating history and failure rates to postulate future event frequencies and consequences (unknown) • Prediction of vessel wall spectrum and fluence from surveillance measurements (unknown)

3.0 SELECTION OF SCREENING CRITERIA

The screening criteria were selected from frequency data based on operating history (Chapter 4).⁽¹⁾ Selection of the screening criteria using these data is not supportable by rigorous technical arguments. In fact, using the customary conservative assumptions employed in analyzing complex engineering safety issues would result in lower (more conservative) generic RT_{NDT} screening criteria than the NRC currently proposes. However, the determination of plant-specific RT_{NDT} as described in Chapter 5⁽¹⁾ provides the conservatism necessary to render these screening criteria acceptable. The PNL technical critique is based on three considerations:

1. The ability of the proposed generic screening criteria to provide an adequate safety margin against vessel failure from pressurized thermal shock events.
2. The applicability of the generic screening criteria to the various pressurized water reactor (PWR) nuclear steam supply designs (Westinghouse, Babcock & Wilcox and Combustion Engineering).
3. The ability of the plant-specific vessel RT_{NDT} criteria to provide an adequate safety margin against vessel failure probabilities from pressurized thermal shock events.

3.1 GENERIC SCREENING CRITERIA

The NRC generic screening criteria were primarily based on the frequency of PTS events (Chapter 4).⁽¹⁾ Frequency data provide little information relative to safety margin against failure probabilities from PTS events. The frequency of vessel failure probability curves provided in the NRC staff report (Figures 8-2 and 8-3)⁽¹⁾ provide needed supporting information. The Westinghouse Owners Group PRA results (Figure 8-2)⁽¹⁾ predict a considerably lower frequency of failure and are dominated by steam line break events, while the NRC staff PRA results (Figure 8-3)⁽¹⁾ predict nearly two orders of magnitude higher frequencies and are dominated by small-break LOCA events. Westinghouse staff reported⁽⁴⁾ that their PRA results did not consider 2-inch and larger small break LOCA events which could lead to loss of natural circulation and therefore potentially severe PTS consequences. The NRC staff data supported a higher frequency of small break events than that reported by Westinghouse. For these reasons, we consider the NRC staff curve (Figure 8-3) to be more complete in that it does contain the principal contributors to PTS events. Further, the calculated probabilities of failure are reported to be uncertain by plus or minus at least two orders of magnitude. In addition, the NRC staff cautions against using these curves for determining the PTS criteria. However, with this being the best available data and realizing that the screening criteria should be primarily a conservative triggering criteria for requiring further, plant-specific analysis, we used Figure 8-3 for a check on the indicated safety margin of the generic screening criteria. Predictions for a mean screening criterion of 270°F RT_{NDT} resulted in a probability of vessel failure by longitudinal crack extension of approximately 10^{-4} . The

uncertainty of two orders of magnitude would result in a 10^{-2} to 10^{-6} range for the expected frequency probability of vessel failure.

The margin of safety as based on a mean surface RT_{NDT} of 270°F provided by the generic screening criteria, which should be unquestionably conservative, apparently does not satisfy the Safety Goal⁽²⁾ reasoning described in the NRC draft report (Section 8.5).⁽¹⁾ In addition, the vessel integrity prediction of less than 10^{-6} ⁽³⁾ could be seriously compromised by PTS events. These objections to the generic screening criteria are, however, mostly rectified by the methods of selecting plant-specific vessel RT_{NDT} (see Section 3.3 below).

3.2 APPLICATION OF THE GENERIC SCREENING CRITERIA TO VARIOUS TYPES OF PRESSURIZED WATER REACTORS

The NRC staff PRA and operating history data analyses do not separately address each type of PWR. Therefore, the magnitude and conservatism or unconservatism of the generic screening criteria as applied to each reactor type are uncertain. It is reported that the three most severe PTS events took place in Babcock & Wilcox plants. Combustion Engineering plants reported no PTS events. Design improvements, training, and procedure changes could result in more consistent future event frequencies among reactor types. Significant differences, however, remain between major types of reactor systems and between specific design features, components and operating procedures within common reactor types. Plant-specific analyses, required after a plant reaches the screening criteria, should consider these variations.

The requirement that licensees initiate plant-specific evaluations within three calendar years of the time the vessel RT_{NDT} is projected to exceed the generic screening criteria will result in a vessel RT_{NDT} approximately 15°F lower than the screening criteria at that time. This conservatism, together with the improvements that have occurred since PTS became an issue, should be an adequate safety margin for specific plants which may be unconservative in regard to the generic vessel failure probability. However, the added conservatism will further penalize plants already on the conservative side.

3.3 PLANT-SPECIFIC VESSEL RT_{NDT}

The plant-specific vessel RT_{NDT} determined in the manner described in the NRC draft report (Chapter 5)⁽¹⁾ will provide approximately 60°F conservatism relative to the mean RT_{NDT} as used in the NRC staff PRA results (see Figure 8-3).⁽¹⁾ Therefore, for the 60°F standard deviation, an equivalent RT_{NDT} of 210°F should be used in Figure 8-3. This standard deviation results in a projected vessel failure probability of approximately 10^{-6} . This value, even with the uncertainty factor of plus or minus two orders of magnitude, is more closely reconcilable to the safety goal⁽²⁾ and to predicted vessel integrity⁽³⁾ values.

The conservatism used in establishing plant-specific RT_{NDT} values is required to satisfy the many unknowns and also to provide the necessary safety

margin to the generic screening criteria. For these reasons, identified conservatism should be evaluated during the plant-specific evaluations and relaxed only with due caution.

4.0 EVENT SCENARIOS

The following comments relate to the frequency and characterization of overcooling events discussed in Chapters 2 and 6 and Appendixes A, B, C, G, J, K and N of the September 13, 1982 NRC staff draft report.⁽¹⁾ The comments are referenced to the appropriate chapter or appendix of the NRC staff report.

4.1 FREQUENCY AND CHARACTERIZATION OF EXPERIENCED OVERCOOLING EVENTS

In Chapter 2 of the NRC draft report,⁽¹⁾ downcomer temperatures for the eight operating histories were based on measurements at cold legs. These temperature data may be conservative or unconservative, depending on where the resistance thermal detectors (RTDs) are located. In cases where the RTDs are located upstream from the injection port, the temperature measurements would be less useful for predicting downcomer temperatures.

4.1.1 Temperature Fit Using β and T_f

Temperature fit using β and T_f to characterize the temperature transient (Section 2.1)⁽¹⁾ is judged adequate for the purpose of determining the RT_{NDT} screening criterion. In particular, for the limiting case of small break LOCA with stagnant loop flow, a simplified analysis with the assumption of total mixing shows that the vessel downcomer temperatures can be approximated through exponential decay. Once the plant has failed the screening criterion, however, the PNL staff recommends that actual temperature and pressure curves as derived from more detailed analysis or plant data be used for plant-specific analyses.

4.1.2 Event Descriptions

The pressure and temperature conditions that NRC staff selected from the stylized event characterization should be revised to agree with the transient data. The following comments refer to particular events described in Section 2.2 of the NRC staff report:⁽¹⁾

- In Section 2.2.1, the most conservative (highest) pressure occurred during the low temperature period at about 80 min. PNL recommends the use of 2150 psig, not 2000 psig.
- In Section 2.2.2, justification for selecting 1000 psig was not provided nor documented by the transient data curves provided.
- In Section 2.2.3, justification for selecting 250°F and 500 psig was not provided, nor does it appear to be justified by data from the transient curves.
- In Section 2.2.4, the characterization pressure of 2300 psig was not justified; 2100 psig appears to be a more accurate choice.

4.2 CONSIDERATION OF LOWER PROBABILITY EVENTS

4.2.1 Initiatory Events

The events analyzed in Chapter 6 and Chapter 2 of the NRC staff report⁽¹⁾ included main steamline break (MSLB), small steamline break (SSLB), small break LOCA (SBLOCA), excess feedwater, steam generator tube rupture, and overcooling transients that actually happened in different plants. The staff report analyzed all of the dominant scenarios. There could be numerous variations of the events described above (e.g., combination of two or more single failure events). However, without the information on detailed event analysis, the staff's approach is judged adequate for screening criteria purposes.

4.2.2 Loss of Natural Circulation

The SBLOCA without natural circulation is the dominating scenario in the NRC probabilistic risk analysis. It is therefore important to obtain accurate knowledge about the conditions that will lead to loop stagnation. The time and break sizes are plant-specific. Therefore, it is possible that for certain plants, the probability of SBLOCA with no loop flow occurring may be such that the scenario is not a dominating one and the total PRA failure curves may be lowered.

Also, the single value of failure probability ($3 \times 10^{-4}/\text{RY}$) used by the staff for 2-inch to 6-inch SBLOCA should be investigated further. There would be a considerable difference in the probability between a 2-inch break and a 6-inch break. Further analyses should be performed as part of a long-term effort to quantify the break sizes that will lead to a loss of natural circulation.

For a Westinghouse 4-loop plant, a simplified analysis showed that under stagnant flow conditions, the final downcomer temperature, T_f , would reach 136°F (initial temperature = 560°F, $T_{\text{HPI}} = 60^\circ\text{F}$) and the decay constant, β , would be 0.13 min^{-1} . These values are in agreement with the values used by the NRC staff ($T_f = 125^\circ\text{F}$ and $\beta = 0.12 \text{ min}^{-1}$). Results of the simplified analysis are based on the assumption that the injected water completely mixes with the surrounding coolant, as suggested by Levy.⁽⁵⁾ This assumption may not be valid. On the one hand, it is conservative in that the mixing volume does not include the vessel lower plenum and the metal heating from the thermal shield. On the other hand, it may be unconservative in that thermal stratification may occur in the cold leg where the colder water stays at the bottom of the pipe. Detailed three-dimensional analyses and CREARE mixing tests are needed to address these uncertainties.

4.3 NRC STAFF CONCLUSIONS AND RECOMMENDATIONS

While it is clear that the corrective actions recommended in Chapter 10 of the NRC staff report⁽¹⁾ will help preclude or mitigate potential PTS events, it should be stated that further work is also required to quantify the "gain" in safety before credit can be given for improved margins (e.g., °F + RTNDT).

Additional work is necessary to support these values and should include a more systematic identification process for potential PTS scenarios, a more detailed consideration of operator error, and a more consistent treatment of multiple failures and potential dependencies.

4.4 REACTOR VESSEL FRACTURE ANALYSIS

A constant heat transfer coefficient of $300 \text{ Btu/hr ft}^2 \text{ }^\circ\text{F}$ was used in the staff fracture mechanics analysis (Appendix D).⁽¹⁾ The actual values in a scenario with stagnant flow decrease from about 500 to 150. The resulting effects on fracture mechanics may be significant because of the sensitivity of the vessel failure probability on the coefficients in the range of 150 to 400 (see Figure H-30).⁽¹⁾

4.5 OVERCOOLING EVENT SEQUENCES LEADING TO POTENTIAL PTS CONDITIONS

Appendix G of the NRC staff report⁽¹⁾ does not take into account that pipe breaks will become a more serious problem as plants get older. Current reliance on operating history, extrapolation of auxiliary steam line break data to main steam line break data, and correlation of primary and secondary piping failures as if they were the same are not justified. The determination of main steam line break frequency requires more technical and analytical support. Current assumptions that are difficult to defend include:

1. Steam line and primary coolant pipe-break failure rates are treated interchangeably. Both pressure-induced and flow-induced failures should be evaluated. Because design water flows are 4 to 15 ft/s and steam flows are 50 to 300 ft/s, steam piping has a much higher percentage of flow-induced failures. Also, length of pipe is not the most important variable. Straight-length pipe failures are almost unknown. What is important is the number of bends, fittings, stress points, and welds in the pipe, and the quality of the water or steam.
2. Use of operating history to establish steam line failure frequency is not accurate. The current projected frequency estimate for steam line breaks appears to be too low. In the last year, four failures of steam lines similar to the recent Oconee event have occurred. They are Vermont Yankee on 12-27-81; Trojan 1 on 1-9-82; Zion 1 on 2-12-82, and Browns Ferry 1 on 6-24-82. At Vermont Yankee, a leak occurred in the 12-inch-diameter drain line between the moisture separator and the heater drain tank. At Zion 1, a steam leak occurred in the 150 psig high-pressure exhaust steam line from the turbine (I and E Notice 82-22).

As these plants get older both the steam erosion problem and the potential for a PTS initiating event will increase. Attempts by utilities to reduce that frequency will be highly dependent on the

extent, criteria, and accuracy of the ultrasonic monitoring and evaluation programs at the various facilities.

4.6 SUMMARY OF NRC STAFF POSITION ON REVIEW OF CONTROL SYSTEMS

In addition to the instrumentation-oriented recommendations listed in Appendix J of the NRC staff report,⁽¹⁾ the following instrumentation-oriented recommendations⁽⁶⁾ should be considered:

1. Add temperature-based subcooling meter indication.
2. Add instantaneous RCS cooldown rate monitor.
3. Add integrated RCS cooldown monitor.
4. Improve RCS pressure readout.
5. Add RCS cooldown rate change indicator.
6. Add NDT margin meter.
7. Improve steam generator level instrumentation.
8. Add transient monitoring indicators.

4.7 EFFECTS OF HEATING ECCS WATER ON PRESSURIZED THERMAL SHOCK

A simple analysis by PNL⁽⁷⁾ showed that under stagnant flow conditions, heating the ECC water could have a significant effect on alleviating the PTS concern by effectively increasing the downcomer final temperature T_f . In scenarios where a continuous natural circulation is maintained, heating the ECC water would not increase T_f significantly. However, the cooldown in this case would not be severe either, because of the constant heat source of the loop flow. Therefore, where the dominating scenario is a transient without loop flow, heating the ECC water is beneficial.

4.8 FUTURE CONFIRMATORY STUDIES

In addition to those studies described in Appendix N of the NRC staff report,⁽¹⁾ confirmatory studies are required to better establish how generic nuclear plant operating history relates to plant-specific design restrictions, how pipe failures for older plants can be predicted from newer plant operating data, and how simulator exercises can be used to develop an "expanded" plant-specific operating history data bank. The timing of operator action also plays a critical role in determining whether a potential PTS scenario may cause crack initiation. As a part of the procedure upgrade implemented under Item I.C.I of the TMI task action plan, the following items should therefore be addressed:

1. PTS-related procedures should be audited using time-line charts to identify critical time constraints and ensure that such constraints are clearly noted to the operator.
2. Simulator and in-plant testing should be conducted to establish and/or validate generic estimates for expected operator response times for certain critical PTS evolutions. (Example: X percent of operators can perform evolution A within N minutes.)

3. Licensee Event Reports (LERs) should be reviewed to ascertain actual time-related information (including data on operator error) which can be used to supplement and validate information obtained in items (1) and/or (2) above.

5.0 DETERMINATION OF RT_{NDT}

The PNL staff generally agrees with the methods the NRC staff used to determine RT_{NDT} values provided in Chapter 5 and Appendixes E and P of the NRC staff report.⁽¹⁾ A few exceptions, however, can be identified. First, uncertainties should be applied uniformly to all forms of metal and irradiation conditions; hence, the Regulatory Guide 1.99.1 upper bound value should not be used to replace the statistical trend curves for the high-copper, high-nickel and high fluence welds. Second, an appropriate non-zero standard deviation for the initial RT_{NDT} of plate and forging metals should be used.

The statistically based trend curves should be used to predict irradiated steel ductile-to-brittle transition temperatures for all chemistry and irradiation conditions. Regulatory Guide 1.99.1 curves should not be used for the special conditions stated in the NRC staff report. The benefits of the statistical curves are that they describe vessel material properties as a function of chemistry and they are insensitive to the addition of new data to the data base. The lesser shift for lower-nickel steels is appropriately described by the statistical curves. Furthermore, the statistical analysis indicates that there are quantitative uncertainties in predicted values. The uncertainties are useful when evaluating conservatism and when performing probabilistic failure analyses. The upper bound method that is used for the Regulatory Guide 1.99.1 analysis can be sensitive to the addition of new data and does not provide standard deviations. Discovery of an outlying data point requires shifting the upper bound value. When statistically based trend curves are used, an outlying data point is added to the data base and is only given its appropriate statistical significance. The use of Regulatory Guide 1.99.1 instead of the statistical curve for high-copper, high-nickel and high-fluence welds means that a lesser RT_{NDT} is assumed for these extreme properties. A greater uncertainty for the extreme properties requires that RT_{NDT} estimates should be above, not below, the statistical trend curve values.

The statistical trend curves are less conservative for high values of copper, nickel and fluence. The curves are less conservative because only 6% of the trend curve data base includes data for copper content $>0.3\%$, nickel content $>0.5\%$ and fluence $>5 \times 10^{18}$ n/cm² ($E > 1$ MeV). Also, it should be noted that the worst five plants are in this extreme class of chemistry and fluence. It is important to use the best available data for estimating irradiated properties for these five plants. The trend curve represents the best available data. (See Chapter 7 for comments on the statistical treatments of RT_{NDT} .)

The PNL staff believes that conservatism used in the PTS evaluation should be applied uniformly to all forms of metal. The NRC staff assumes zero uncertainty for the initial RT_{NDT} of plates and forgings. The uncertainties for plates and forgings should be applied in the same fashion as for welds. The uncertainty that is used for each form of metal should be determined from a statistical analysis of data from appropriate materials. The appropriate data base should be established for the particular metal and then the standard deviation for that data base should be used for the estimate of uncertainty.

If material characteristics of particular welds are unknown, then a conservative data base should be used, e.g., by assuming high chemistry values or worst weld-flux conditions.

The one standard deviation for weld metal is assumed to be 17°F (Appendix E.2);⁽¹⁾ the one standard deviation for well-characterized plate metal should be developed and used, and should not be zero. When uncertainties are applied to new plant welds that are well characterized, new data bases for these welds could be used. These new distributions may have lesser standard deviations, in which case credit could be given for the better characterized welds.

A standard procedure for treating uncertainties would result in a clearer understanding of plant-specific analyses and a consistent treatment of variability in probabilistic fracture analyses. Comparison of PTS evaluations performed by different groups would be easier to interpret if consistent treatments of uncertainties are used throughout all analyses. Differences in treatments should be accommodated in the input data; namely, the assumed specific variability for each material.

6.0 FRACTURE MECHANICS

This Chapter reviews the fracture mechanics analyses of the NRC staff as summarized in Chapters 3 and 7 and detailed in Appendixes D, H and O of the NRC staff Report.⁽¹⁾ The review focuses on the probabilistic fracture mechanics predictions rather than on the deterministic predictions. Both analyses, however, use essentially the same fracture mechanics models, and thus much of the discussion applies to both types of analyses described in the staff report.

6.1 OVERVIEW OF FRACTURE MECHANICS

The main points of PNL's critique are stated briefly below. Selected topics are discussed in detail in Sections 6.2 through 6.8.

The PNL staff judges the 270°F screening criterion to be a reasonable selection which is based on the best available data and analytical methods. The NRC staff states (Chapter 1, Section H.4.1)⁽¹⁾ that the predicted failure probabilities for vessels are subject to possible errors or uncertainties of one to two orders of magnitude. Furthermore, the proposed screening limit provides essentially no margin between the predicted failure probability and the "desired" vessel failure probability of less than 10^{-6} per year. To provide a suitable margin, one could impose a simplistic requirement of a predicted failure probability of 10^{-7} to 10^{-8} for setting the RT_{NDT} screening limit. The corresponding screening criteria for longitudinal welds would then be 20° to 40°F less than the 270°F limit proposed by the NRC staff.

The PNL staff has identified a number of unquantified conservatisms inherent in the NRC fracture mechanics model which could readily offset uncertainties and unconservatisms in the probabilistic analysis. These are discussed below and are also listed in Table 2.1. A number of unconservatisms are also discussed below. These factors could offset the conservative features of the fracture mechanics model. Accordingly, the PNL staff does not recommend any relaxation of the proposed screening limit, particularly if the relaxation is justified on selective reductions of only conservative factors in the analysis rather than on a thorough review of all factors.

6.1.1 Application of Probabilistic Analyses

The use of probabilistic fracture mechanics by the NRC staff to justify licensing requirements represents a departure from traditional vessel safety evaluations. The PNL staff believes that the "newness" of the approach is a reasonable justification for expecting some extra margin or level of conservatism in the deterministic assumptions of analytical models. In particular, the VISA computer code is viewed as being in an early stage of development and application. It has not yet been well documented, validated by comparison with similar codes, subjected to extensive peer review, or seen widespread application by other organizations. The PNL staff has reviewed the VISA code in NUREG-2837⁽⁶⁾ and concluded that the code, while subject to limitations, can provide valuable insights into the PTS issue.

The PNL staff believes that the application of probabilistic fracture mechanics by the NRC staff in the current situation is a necessity since the use of design-basis accidents to describe the transients of concern to PTS was judged to be inadequate by the NRC staff. Once the severity transient is described by a probabilistic distribution, it is difficult to apply deterministic fracture mechanics in a logical manner.

6.1.2 Warm Prestress

The PNL staff believes that the NRC staff has introduced a significant conservatism into their fracture mechanics evaluations by assuming that warm prestress is never effective in preventing crack initiation. Section 6.5 below discusses the issue of warm prestress in detail, and lists justifications for neglecting warm prestress in the development of generic screening criteria for PTS.

6.1.3 Flaw Description

The fracture mechanics model as applied by the NRC staff makes a number of conservative assumptions about the nature of flaws (Sections D.1.3, H.2.1, and H.2.3).⁽¹⁾ Except for flaw depth, which is treated as a probabilistic parameter, worst-case assumptions are consistently made. For example, the flaw is located at the inside surface of the vessel wall, extends through the clad, is very long in the axial or circumferential direction, and is normal to maximum tensile stresses. The PNL staff believes that these assumptions introduce a substantial level of conservatism into the fracture predictions. While each of the assumptions may apply to a given flaw, only a very small fraction of cracks will encompass all the assumed features.

The PNL staff recommends that NRC abandon the assumption of a long flaw in favor of a flaw of 6:1 aspect ratio (length to depth), for reasons stated in Section 6.6 of this report. On the other hand, it is currently difficult to justify further relaxation of the other assumptions because of the uncertainties in the flaw size distributions used in the NRC probabilistic fracture mechanics analyses as discussed in Section 6.8. The major limitation identified in the PNL staff's evaluation is the lack of any consideration of flaws due to underclad cracking mechanisms in the development of the flaw size distribution.

6.1.4 Clad Effects

The NRC fracture mechanics model includes an enhancement of the crack-tip stress intensity factor due to the difference in the thermal expansion coefficient of the clad relative to the vessel base metal (Sections D.1.1 and H.3.2.9).⁽¹⁾ The PNL staff believes that this assumption introduces conservatism into the fracture predictions, since few flaws in the welds are expected to extend through the clad. The assumed extension of flaws through the clad is viewed largely as expeditious to simplify the calculation of stress intensity factors. However, the PNL staff does not necessarily recommend that the clad thermal expansion effect be disregarded. This conservatism regarding

clad effects will tend to balance the convenient neglect of weld residual stresses. These stresses could promote the growth of cracks in embrittled weld metal.

The NRC model does not include the possibility that a ductile clad could suppress the lengthwise growth of flaws. This effect has been the subject of considerable speculation, but only recently have experimental validations been undertaken (at ORNL). Results of preliminary tests at ORNL have been reported, and PNL fails to see in these test results a consistent trend in which cladding suppresses lengthwise flaw growth. Lacking consistent evidence, it is recommended that current assumptions be retained by NRC (i.e., short flaws can grow in length).

6.1.5 Residual Stress

In the NRC fracture mechanics analyses, residual stresses induced by welding and by other sources has been assumed to be zero (Section D.2).⁽¹⁾ Residual stresses are not readily predictable and are highly dependent on details of the welding procedure. The PNL staff suggests that these residual stresses could be tensile at the vessel inside wall, but that their magnitude should not exceed 10 to 20 ksi after heat treatment to relieve stress. The assumption of zero residual stress is viewed as a definite unconservatism in the NRC analysis. The assumption of a 20 ksi tensile residual stress would not be totally unrealistic in the context of other conservative assumptions in the fracture mechanics model (e.g., long flaws extending from the base metal through the clad). Nevertheless, the PNL staff recommends that residual stresses be specifically included only if sufficient data exist to avoid assigning an arbitrary value to residual stress. Even then, the resolved stresses should be used only if the overall fracture mechanics model is extensively refined through the deletion of other offsetting conservative assumptions in the analysis.

6.1.6 Safety Factor

The NRC fracture mechanics model generally follows traditional approaches (e.g., the ASME code), but it does not include a specific safety factor. Rationale for including safety factors to offset uncertainties in fracture mechanics models are stated in Section 6.3. The PNL staff, however, does not recommend the addition of a safety factor to the existing NRC model, since the model currently includes conservatisms (e.g., long flaws and clad thermal expansion effects) that exceed those used in traditional ASME code fracture mechanics models.

6.1.7 One Dimensional Models

The NRC fracture mechanics model is one-dimensional in nature (Section D.1.1).⁽¹⁾ The model does not include axial and circumferential gradients in stress and toughness. It assumes that minimum toughness and maximum stress occur at the same location in the vessel. The PNL staff believes that such a simplistic model is appropriate for development of

screening criteria. The model, however, may be excessively conservative for specific vessels and cooling transients. In particular, the inclusion of axial toughness gradients may result in models that will predict a much higher probability of crack arrest. The PNL staff believes that three dimensional fracture mechanics (and thermal hydraulic) models would be appropriate for plant-specific evaluations for those vessels that exceed the screening criteria.

6.1.8 Effect of Long Crack Fronts

Experiments at ORNL as part of the Heavy Section Steel Program have been widely reported (Section H.3.2.4).⁽¹⁾ Results indicate that very long cracks exhibit apparent fracture toughness levels that correspond to lower bound rather than mean toughness values as measured on standard toughness specimens. The ORNL staff have proposed mechanistic explanations for the observed trend.

The NRC fracture mechanics model does not include crack length as a parameter that governs apparent fracture toughness. The PNL staff does not recommend such effects be included for predicting crack initiation. The probable flaw lengths for initiation are expected to be relatively short. It is, however, recommended that the long flaw effects be included in predictions of arrest and reinitiation of arrested long flaws. Due to lengthwise flaw growth, most flaws will become long after initiation. Since the NRC model does not predict great benefit from crack arrest under PTS conditions, it is expected that the inclusion of length effects will result in little change in predicted vessel failure probabilities.

6.1.9 Crack Arrest

In NUREG-2837,⁽⁶⁾ PNL extensively reviewed state-of-the-art fracture mechanics methods for the prediction of crack arrest. Crack arrest calculations by NRC and others have generally followed the guidelines of the ASME code. The PNL staff's review identified possible questionable features of the ASME code which have been brought to light by results of recent developments in fracture mechanics. These developments are summarized in Section 6.7 of this report.

The PNL staff believes that inclusion of changes in crack arrest methodology discussed in NUREG-2837 would have had little effect on NRC's selection of the RT_{NDT} screening criteria (Section H.3.2.8).⁽¹⁾ In fact, some of PNL's recommendations are included in the NRC analysis (i.e., failure by net section plastic collapse). In the future, however, more sophisticated fracture models may predict much greater benefit from crack arrest, and it is essential that these models be based on up-to-date theories and appropriate arrest toughness data.

6.1.10 Simplistic Treatment of Heat Transfer

The NRC staff has used a constant value of surface heat transfer coefficient in its fracture mechanics analysis (Section D.1.1).⁽¹⁾ The PNL staff believes this is an unnecessary simplification, but can make no statements about whether the resulting fracture mechanics are or are not conservative. The constant heat transfer coefficient is viewed as part of the general level of uncertainty and simplification that exists in PTS thermal hydraulic calculations.

6.1.11 Simplistic Stochastic Model

The VISA code, like most probabilistic treatments, uses a default option and assumes that all errors and uncertainties are independent. Chapter 7 addresses PNL concerns with the VISA stochastic model. Without extensive study it is not possible to evaluate if a significant level of unconservatism is introduced into predictions of vessel failure probabilities. It should be noted, however, that the default option in VISA is in direct contrast to deterministic fracture mechanics methodology which assumes that worst-case toughness always occurs with the worst-case flaw size.

6.1.12 Uncertainties in Distribution Functions

In NUREG-2837,⁽⁶⁾ PNL reviewed the distributions used as input to the VISA fracture mechanics code. The distributions were based on limited data and engineering judgement. It was not possible within the scope of the PNL evaluation to determine the extent of any possible unconservatism introduced into the vessel fracture probabilities. The flaw size distributions were identified as having the greatest uncertainty. Section 6.8 and Chapter 7 address flaw distributions. Uncertainties in toughness and radiation-induced shift are discussed in Chapters 5 and 7.

6.2 UNQUANTIFIED CONSERVATISMS

The NRC staff position uses both deterministic and probabilistic fracture mechanics calculations to predict whether or not a vessel will fracture during a given cooling transient. In general, the conservatism inherent in traditional engineering evaluations of structural reliability has come from three sources:

1. The analytical models are based on conservative physical assumptions.
2. Where inputs to the analyses are based on data subject to statistical scatter, bounding or low probability values are used in calculations.
3. Factors of safety are used in the calculations to account for unidentified and/or unquantifiable uncertainties that cannot be addressed by items (1) and (2) above.

The NRC position requires no explicit safety factor but relies solely on physical assumptions and bounding type inputs to ensure that deterministic predictions are conservative. The physical assumptions of the NRC probabilistic fracture mechanics calculations essentially parallel the assumptions of the deterministic calculations. Thus, failure probability estimates provide no insight into the conservatism of physical assumptions, but indicate only the level of conservatism in the deterministic calculations arising from placing bounding values on all inputs simultaneously. If the unquantified physical assumptions are conservative, the results of the probabilistic fracture mechanics should give a conservative estimate of the "true" probability of failure.

A number of conservative factors and assumptions used in both the NRC and industry analyses are listed in Table 6.1. These analyses are reviewed in detail in NUREG-2837.⁽⁶⁾ For comparison, the assumptions suggested by the ASME code are also given. Except for flaw length and clad thermal expansion, the NRC analyses generally follow the guidance of the ASME code. Clearly the NRC fracture mechanics analyses have a number of unquantified conservatisms not addressed by the probabilistic model (e.g., worst-case flaw assumed, warm prestress not included, clad beneficial effects neglected).

TABLE 6.1. Unquantified Conservative Factors

Factor	Analytical Model		
	NRC Staff	ASME Code	Industry
Safety Factor	None	1.414	None
Clad Thermal Expansion	Yes (K Increased 17%)	No	Yes (CE) No (<u>W</u> , B&W)
Flaw Length	Very Long (K Increased 20%)	6 x Depth	Very Long (CE) 6xDepth (<u>W</u> , B&W)
Flaw Description	At Vessel ID Normal to Stress Extends Thru Clad	Same	Same
Warm Prestress	No	No	Yes
Crack Arrest	Yes	LOCA, Yes PTS (uncertain)	Yes
Suppression of Crack Growth by Clad	No	No	No
Simplified One Dimensional Analysis	Yes	Not Required	Yes

The PNL staff believes that the conservatisms listed in Table 6.1, if quantified, would contribute to a substantial reduction in calculated failure probabilities below the values estimated by the NRC staff. The additional conservatism, however, is appropriate to offset the uncertainties in the probabilistic treatment and other unconservative assumptions.

6.3 SAFETY FACTORS

The PNL staff recommended the use of a modest safety factor (see NUREG-2837, Section 6.6) in accordance with the approach of the ASME code. The NRC fracture mechanics analyses include no safety factors, but in other respects the analyses are somewhat more conservative than those recommended by PNL. The NRC model assumes a long surface flaw (as opposed to a finite-length surface flaw) and includes the enhancement of K due to the presence of clad. These two factors together are roughly equivalent to the $\sqrt{2}$ safety factor recommended by PNL. In effect, the NRC analysis is consistent with the ASME code.

6.4 COMMENTS ON PROBABILISTIC FRACTURE MECHANICS

It appears that NRC's probabilistic analyses include a comprehensive treatment of the known random variables that govern the irradiated fracture toughness of the vessel (fluence, copper content, initial RT_{NDT}). The treatment of these variables was intended to be realistic rather than conservative. The PNL staff has yet to evaluate all possible unconservatisms in NRC's estimates of vessel failure probability, but has noted the following:

- NRC has sought to avoid excessive conservatism in estimating the variability in irradiated fracture toughness (Sections H.2.1 and H.2.3).⁽¹⁾ Accordingly, it was assumed that all randomness in measured shift was due to uncertainties in copper, fluence and initial RT_{NDT} , and not due to any randomness in the response of the material to irradiation (e.g., effects of chemical elements other than copper and nickel). Consistent with this assumption, a decision was made in NRC's shift calculation to use a mean shift curve in a deterministic sense. While this was consistent with a realistic analysis, it does introduce a possible unconservative uncertainty into the vessel failure probability calculation.
- It is implied that the existence of a crack at a given location does not imply either a relatively low or high toughness at that location. Such lack of correlation between variables is consistent with "conventional wisdom" on material properties, but may introduce some unknown level of unconservatism into the probabilistic analyses.
- As has been noted, the flaw distributions are subject to great uncertainty. They are probably conservative for volumetric type weld flaws, but are of questionable relevance to underclad type cracks.

6.5 WARM PRESTRESS (WPS)

The NRC staff has elected to ignore possible beneficial effects of warm prestress in the fracture mechanics analyses which form the basis of the RT_{NDT}

screening limit (Sections 3.1, and D.3).⁽¹⁾ The following discussion indicates the appropriateness of this position and proposes a rationale for excluding warm prestress.

Exclusion of warm prestress from the RT_{NDT} screening limit evaluation can be justified for the following reasons:

1. The generic treatment of overcooling transients does not permit adequate identification of rapid pressure fluctuations that can result in violation of the requirements for the warm prestress effect.
2. In viewing the fracture mechanics analysis as a whole, it is necessary to ensure that the various physical assumptions are clearly balanced on the side of conservatism. It is appropriate and convenient to neglect warm prestress to offset possible unconservatisms of an unquantifiable nature.

Other concerns that could discourage the use of warm prestress are listed below:

1. There are specific mechanistic reasons to believe that warm prestress applies under conditions of decreasing K_I . A consistent collection of experimental evidence supports this conclusion. Nevertheless, rare exceptions could occur for specific materials (e.g., compositions, welds, heat treatments) and conditions (e.g., temperatures, loading rates, irradiation effects, prior strain history). Hence, the level of reliability that can be assigned to the concept may be less than 100 percent.
2. The PNL staff has noted some borderline transients treated in the owners group responses. The predicted values of K decrease, but at a very slow rate, and never fall much below the maximum value. Therefore, one should be assured that uncertainties in cooling rates will not lead to increasing rather than decreasing K . For example, a range of possible surface heat transfer coefficients, mixing assumptions, flow rates, etc., should be considered for borderline transients before warm prestress is accepted as a beneficial factor. In this regard, one may wish to impose limitations on WPS to require that K drop some (small) fraction below its maximum value, and that the rate of decrease in K continue at some minimum rate (either an absolute rate or at a specified rate relative to the corresponding rate of decrease in fracture toughness).

While warm prestress has been excluded in the NRC staff's RT_{NDT} screening limit evaluations, its application in detailed plant-specific evaluations may be appropriate with certain limitations. For example, there may be classes of transients that cannot be clearly excluded from fracture concerns for the particular plant. Physical constraints specific to the plant on the resurization rates for these transients may limit the rate of increase in K

during the critical periods of the transient. On such a selective basis, inclusion of warm prestress could be justified since the above concern for rapid pressure fluctuations would have been addressed.

6.6 CRACK SHAPE

The postulated flaws in the NRC staff's fracture mechanics analyses have been assumed to be axial flaws of infinite length and 360° complete circumferential cracks. The "long" flaw assumption is convenient because published stress intensity factor solutions are available for long flaws. Applicable solutions for finite length flaws are often more difficult to obtain. The analyses for long flaws also give higher stress intensity factors for a given flaw depth. Hence, the long flaw assumption can be further justified since it introduces conservatism into the analyses.

The PNL staff believes that there are three reasonable bases for assuming finite length flaws in the fracture mechanics analyses:

1. There is an established precedent for a 6:1 aspect ratio for postulated flaws in Section III, Appendix G of the ASME Boiler and Pressure Vessel Code.
2. The long flaws are much less likely to be present than flaws of shorter length. Estimates of flaw aspect ratios in a recent NRC piping reliability study⁽⁸⁾ suggest that more than 90 percent of weld defects will have an aspect ratio of less than 6:1.
3. The stress intensity factors for deep axial flaws of infinite length can have unrealistically high values. If the length of the vessel beltline is imposed as a physical constraint on flaw length, then significantly lower values of calculated stress intensity factors result.

In NUREG-2837,⁽⁶⁾ the PNL staff recommended the adoption of a 6:1 aspect ratio for postulated flaws. This aspect ratio requires somewhat more effort to calculate stress intensity factors. It is believed that the finite length flaw is justified because it will tend to balance the various conservative features of the postulated flaw, in particular the assumption that the flaw is located at the inside surface of the vessel, extends through the clad, and is oriented normal to the maximum stresses.

It is recommended that the 6:1 aspect ratio be used only for predicting the initiation of flaw growth in vessels and not for predicting the subsequent propagation and arrest of these growing flaws. After crack growth initiates, the crack aspect ratio changes because the growth rates in depth and length differ. Certain ORNL tests have shown that a rapid growth in crack length could transform a crack of relatively small aspect ratio into a long flaw. Accordingly, the acceptance criteria proposed by PNL in NUREG-2837⁽⁶⁾ recommended that the 6:1 postulated flaw be treated as a long axial or circumferential flaw once crack growth initiates.

6.7 COMMENTS ON CRACK ARREST

The fracture mechanics calculations in the NRC staff report⁽¹⁾ have included predictions of crack arrest. Results of these calculations indicate that arrest has a relatively small effect on critical values of RT_{NDT} , and that arrest is likely only for relatively low pressure levels. The NRC probabilistic calculations also indicate that increasing the upper shelf fracture toughness above a 200 ksi $\sqrt{\text{in.}}$ has little effect in reducing failure probability.

A PNL review of crack arrest calculations is described in NUREG-2837.⁽⁶⁾ The review identified a number of unconservative features in these calculations. Notably, recent data were discussed that showed that the ASME K_{Ia} reference toughness curve was unconservative for weldments in the critical upper portion of the transition temperature range.

The upper shelf toughness of 200 ksi $\sqrt{\text{in.}}$ has been extensively used in NRC and industry fracture evaluations. In addressing the possible over-conservatism introduced by this limit on upper shelf toughness, the following points must be considered:

1. The upper shelf arrest toughness apparently does not generally govern predictions of vessel failure as shown by NRC calculations. Evidently, small surface flaws seldom initiate for transients that maintain the outer portion of the vessel wall at temperatures corresponding to upper shelf toughness levels.
2. Irradiation induced decreases in upper shelf toughness are in themselves a safety issue of concern to NRC⁽⁹⁾ and should be addressed in detail for specific vessels before upper shelf toughness greater than 200 ksi $\sqrt{\text{in.}}$ is used for PTS evaluations.

The question of postulated flaw shape has been reviewed in Section 6.6 of this report. The PNL staff recommends that a 6:1 aspect ratio flaw be postulated for calculating crack initiation, and that a "long" flaw be postulated for crack arrest calculations. One feature of this physical model is a trade-off between reduced conservatism in the initiation calculations with possible over-conservatism in arrest calculations. This combination of crack shape assumptions should help to minimize the impact of alternate assumptions regarding the K_{Ia} toughness curve. With the assumed immediate increase in flaw length after initiation, crack arrest will seldom be predicted regardless of alternate K_{Ia} curves used for the fracture evaluations.

6.8 FLAW DISTRIBUTIONS

The distribution of flaws is no doubt the weakest input to probabilistic fracture mechanics analyses, and short-term improvements in these distributions are not expected. The VISA code has used the OCTAVIA flaw distribution. This distribution is based on a combination of engineering judgement and reports of flaws that have been discovered in welds during fabrication. The OCTAVIA distribution differs somewhat from other estimates, such as that in the Marshall

Report,⁽¹⁰⁾ but generally shows the same trends. Other distributions have been based, at least in part, on destructive examination of welds.

Dr. S. H. Bush at PNL has reviewed various sources of data on weld defects for the NRC, including recently obtained data from foreign sources. Information obtained to date does not clearly support any new trends to support changes in previously developed flaw size distributions.

In evaluating the consequences of PTS events, PNL believes that two types of flaws should be considered. One type is random flaws distributed throughout the weld. These flaws will develop during production of the weld. The second type of flaw is underclad cracks, produced during the cladding process. Mechanisms of both hot and cold underclad cracking have been observed and reported.

The PNL staff believes that the VISA code models volumetric weld flaws in a conservative manner. It is recognized that the assumed distributions of the number and depth of flaws may be subject to uncertainty. However, VISA describes all other characteristics of the flaws very conservatively. In particular, the flaws are always at the inside surface of the vessel, are cracks rather than less serious flaws such as porosity, the flaws are very long rather than approximately circular, and the flaws are oriented normal to maximum stresses rather than parallel to these stresses. These factors are likely to offset any possible unconservatism in depth estimates.

The VISA code does not directly model the more serious type of underclad cracks, since its flaw distributions are not based on data from observed underclad cracks. By assuming that all the flaws are at inside surface cracks, the VISA analysis does indirectly model underclad cracks. The distributions describing the number and depth of these flaws, however, is based on data for volumetric weld flaws. As such, the description of possible underclad cracks in VISA can be viewed only as a rough approximation that lacks any specific base of supporting data.

7.0 STATISTICS

The evaluation of PTS requires the synthesis of a collection of theory, data, and considered judgment. Each of these elements has some associated uncertainty. Uncertainty can be due to real variation in physical properties or it can represent a measure of the lack of knowledge. Throughout the evaluation of PTS, uncertainty has been treated through a combination of statistical methods, probability theory, and conservative judgment. PNL's conclusion is that an appropriate combination has been used. Some specific areas where additional attention is warranted are discussed below. These comments apply to Chapters 5, 6, and 8 and Appendixes G and H of the NRC draft report.⁽¹⁾

7.1 EVENT SEQUENCES AND PROBABILITY OF VESSEL FAILURES

Event sequences that may result in a significant PTS risk are described and characterized in Chapter 6 and Appendix G.⁽¹⁾ Chapter 8⁽¹⁾ summarizes the results of combining the probability of an event sequence with the conditional probability of reactor pressure vessel failure.

There appear to be some inconsistencies between Chapter 6 and Appendix G in regard to SGTR characterizations in Table 6.5, page G-16, and Table G-6. Chapter 8⁽¹⁾ states that 23 groups of transients were used to calculate vessel failure probability. Chapter 6 lists a total 22 groups (8 MSLB, 10 SSLB, 1 SBLOCA and 3 SGTR) in Tables 6.2 through 6.5. Appendix G shows 23 groups (10 MSLB, 10 SSLB and 3 SBLOCA) in Tables G.1, G.3, and G.5, but does not include SGTR events. These inconsistencies need to be corrected, although they are not expected to change the selected screening criteria.

7.2 RPV FAILURE PROBABILITY STUDY

The calculation of the probability of vessel failure is a synthesis of information from many diverse sources. The overall calculation uses the logical theories of PRA and Monte Carlo simulation to draw together several physical theories, recorded data on measured variables, data on event frequencies and considered judgement. Each of these parts has some associated uncertainty which arises either from real variation or from lack of knowledge. The final result expressed as the probability of vessel failure, is obtained by multiplying the conditional probability of failure given a transient group T, $P(F/T)$, by the probability of the transient group, $P(T)$, and then summing the probabilities over all transient groups. At the point where this calculation is carried out, the only uncertainty associated with $P(F/T)$ is sampling variability due to the use of Monte Carlo. All of the uncertainties in material properties, flaw size, fracture mechanics, and material chemistry have been incorporated into the statement of probability given by $P(F/T)$. The sampling uncertainty is small, generally less than a factor of 2 for 500,000 passes through the simulation. The major component of uncertainty in the final result arises from the uncertainty in the estimate of $P(T)$.

All groups of transients in Chapter 6⁽¹⁾ except MSLB have uncertainty factors of 10. For the MSLB, the uncertainty factor is between 20 and 100. These factors lead to an uncertainty factor on the cumulative probability of just over 10. It is likely, however, that the NRC staff estimates of frequency are too low. (See comments on Appendix G⁽¹⁾ in Chapter 4 of this report.)

The probabilistic analyses performed using the VISA code described in Chapter 8 and Appendix H⁽¹⁾ were generally carried out using best estimate of uncertainty distributions. Additional sensitivity studies tended to provide confidence that even conservative estimates of uncertainty did not lead to large changes in probability estimates. The VISA code represents an accurate or conservative estimate of the current state of knowledge regarding the probability of failure of a vessel subject to pressurized thermal shock. However, the estimates of vessel failure probability should not be accepted without qualification.

The primary shortfall of the VISA code, and indeed, of our current state of knowledge, is a lack of a definitive stochastic structure for the system simulated by VISA. The current structure is the default that arises from assuming that all errors or uncertainties are independent. The effect of this assumption is to make unfavorable combinations appear infrequently in the simulation. This is as it should be if in fact the errors are independent. However, if an unknown dependence exists then the estimated probabilities may be much too low. At present, the relationship of all the uncertainties considered in VISA is not sufficiently well characterized to preclude such a possibility.

The VISA code uses a trend curve for ΔRT_{NDT} with a fluence term of $[f/10^{19}]^{0.218}$. The trend curve used in the determination of the screening criterion uses a fluence exponent of 0.27. The screening criterion on trend curve will predict a greater ΔRT_{NDT} for $Cu \geq 0.25\%$, and $Ni > 0.65\%$ and fluence greater than 5×10^{18} . This would appear to be a conservative factor.

A potential unconservatism in VISA is the lack of an uncertainty on ΔRT_{NDT} . The uncertainty of ΔRT_{NDT} is assumed to be entirely due to uncertainty in Cu, Ni, and fluence. Although the major portion of the uncertainty in ΔRT_{NDT} is probably due to these sources, some of the variation about the trend curve is due to variation in material properties. Thus, the total uncertainty in ΔRT_{NDT} is probably understated.

In order to accommodate the weakness of the stochastic structure of VISA and the possibly low estimates of transient frequency noted above, PNL recommends an additional factor of at least 10. In combination with the NRC staff estimates, this gives a total uncertainty of at least 2 orders of magnitude on the probability of vessel failure. This would then agree with the staff estimate of ± 2 orders of magnitude for PRA results given in Section 8.3.⁽¹⁾

7.3 DETERMINATION OF ΔRT_{NDT} FOR SCREENING

The use of a " 2σ " uncertainty term for RT_{NDT} probably does not provide as high a level of confidence as was intended by the staff. An interval of the "mean $\pm 2\sigma$ " covers 95% of a population if the population has a normal distribution, and if the mean and standard deviation are known exactly, not estimated from data. Neither of these conditions are satisfied.

The RT_{NDT} trend curve described in Chapter 5⁽¹⁾ was fitted using a modified least squares technique that allows for uncertainty in the recorded value of fluence as well as in ΔRT_{NDT} . This technique is described in the statistical literature as the "error in variables." From a mathematical standpoint, the data violate the usual assumption of regression analysis; that is, that the independent variables are known without error and that all uncertainty is associated with the dependent variable. For the linear regression problem, standard least squares techniques will produce biased estimates of the parameters in the model. For a purely empirical model as contained here it is questionable if the additional complexity introduced by treating independent variables as random is worth the additional effort.

The error in the predicted ΔRT_{NDT} is the difference between the actual shift in RT_{NDT} and its predicted value. The error can be broken down into two components: (1) error due to variation of the measured ΔRT_{NDT} about the true value and (2) error due to variation of the predicted mean value about the true mean value. Component 1 arises because of measurement error and real variation in material properties. It would still be present even if the fluence, Cu and Ni were exactly known, and the true relationship between ΔRT_{NDT} and fluence, Cu and Ni were known. Component 1 would generally be considered to have a constant variance over the range of independent variables. The variance of error in Component 1 is estimated from the residual sum of squares. This is the only component that was considered in Chapter 5.⁽¹⁾

Error Component 2 is usually regarded as stemming primarily from sampling error. The mathematical model of the relationship between the independent and dependent variables includes parameters estimated from data. The data are a (random) sample from the universe of all possible groups of independent and dependent variable values. The parameters, then, are a sample from the population of all sets of parameter values that could arise from different samples of the independent and dependent variables.

If the trend curve ΔRT_{NDT} were fit to a new set of data, different estimated values of the parameters would result, even if the data arose from experiments that were to all intent exact duplicates of the previous experiments. Error component 2 accounts for the variation in the parameter estimates.

The variance of error Component 2 is not constant over the range of the independent variables. Generally, the variance tends to be smallest near the centroid of the range and larger near the extremes.

The implication of the above discussion is that the margin of $\pm 60^{\circ}\text{F}$ probably does not include 95% of RPV RT_{NDT} 's. However, the concern is to obtain an upper bound on RT_{NDT} . The actual confidence level of the upper bound is not especially important, as long as the margin is large enough to provide an acceptably low probability of failure at the best-estimate RT_{NDT} .

8.0 PLANT-SPECIFIC ANALYSES AND CRITERIA

The NRC staff's evaluation of analyses and acceptance criteria required of licensees (Chapter 9)⁽¹⁾ is overly definitive for the current extent of investigations. It also seems erroneous to establish the time when plant-specific analyses will start rather than time when analyses and proposed corrective action plans are required to be submitted for evaluation and approval by NRC. It is recommended that the plant-specific analyses and plans be received two years before the screening criterion is reached. This will allow enough time for NRC evaluation and approval, as well as time for implementing corrective actions, if required. It is not clear from the staff's description whether NRC will require full plant-specific analyses (i.e., plans that cover the full expected life of the plant) or whether analyses and corrective actions can cover an interim period such as 3, 5, or 10 years. A plan covering the full expected plant life is recommended to enable evaluation of nil-ductility temperature during all phases, including end-of-life.

8.1 EVALUATION OF OVERCOOLING EVENT SEQUENCES

Section 9.2⁽¹⁾ outlines the methods used by the NRC staff to determine the proposed screening criteria, and the use of similar methods by licensees is inferred. The use of simulators to quantify the important operator reaction and timing following PTS precursor events on a plant-specific basis would seem to be an important requirement and should be included in plant-specific analyses.

8.2 VESSEL MATERIALS PROPERTIES

Pacific Northwest Laboratory recommends that other material property evaluations be made in addition to those recommended by NRC in Section 9.3.⁽¹⁾ Tensile strength data should be examined to validate chemistry and fluence dependencies observed using charpy testing. Furthermore, the PNL staff recommends that steels irradiated in surveillance programs be characterized using microscopic and submicroscopic methods and, if possible, that chemical measurements be made of irradiated pressure vessels.

Tensile strength data are obtained in surveillance test programs; therefore, an evaluation of composition and fluence effects on tensile properties could be readily performed. Because brittle fracture of irradiated pressure vessel steels is controlled by critical stress, the tensile strength data should add confidence to the charpy data.

Testing the microhardness of material irradiated in surveillance programs also could provide supplementary information that would support parametric dependencies assumed for trend curves. Microhardness is a mechanical property measurement that could be made remotely on plant-specific welds and could be correlated to yield stress. Metallurgical tests have shown that when irradiation increases both the 0.2% offset yield stress and the ultimate stress, then

a valid microhardness/yield stress correlation can be established. A demonstration that irradiation has the necessary effect on yield and ultimate stress would indicate the usefulness of the microhardness test.

During the last decade several material characterization techniques have been developed and perfected. Some of these techniques could be used to evaluate irradiation effects on pressure vessel steels. Field-ion microscopy has been used to evaluate the interactions between chemical impurities and radiation damage. Position annihilation and small angle x-ray and neutron scattering are other characterization techniques that could be used to establish confidence in assumed parametric dependencies that more accurately reflect the real properties of pressure vessel steels.

The critical plant welds often have poorly documented chemistries and mechanical properties. Conservative estimates of embrittlement of these welds are made by assuming a maximum credible copper and nickel content. The specific plants are penalized by assuming worst possible weld chemistries. X-ray fluorescence and laser excitation techniques could be used for remote weld chemical analysis and for improving reliability of predictions of irradiation embrittlement.

8.3 FLUENCE DETERMINATION

The methods used to determine fast fluence value in Appendix F⁽¹⁾ do not adequately examine uncertainties in extrapolating surveillance specimen environments to vessel wall environments. Temperature, neutron flux, and neutron spectrum, in addition to fluence, affect the degree of embrittlement induced by irradiation. The trend curves establish only the effect caused by neutron fluence. Because secondary parameters are neglected in the material property evaluation, there is uncertainty in applying the generalized trend curves to plant-specific analysis. The total uncertainty in estimating pressure vessel wall irradiation environments should be discussed in Appendix F⁽¹⁾ to provide guidance for future plant-specific evaluations.

8.4 FLUX REDUCTION

The evaluation of flux reduction in Appendix I⁽¹⁾ does not emphasize the superior benefit of flux reduction at lower fluences. Newer plants not yet significantly embrittled but having a high potential for embrittlement should be included in Appendix I. Embrittlement rate decreases with increasing fluence; therefore, flux reduction has a minimal benefit at high fluences and a maximum benefit at beginning-of-life.

The diminishing benefits of flux reduction with increasing age of a plant demands a strong recommendation for immediate flux reduction, especially for newer plants. The advantage of early flux reduction could be demonstrated by calculating the end-of-life RT_{NDT} versus the calendar year that a specific plant implements flux reduction. The calculations would clearly show the costs of delayed action in terms of future RT_{NDT} values.

8.5 DETERMINISTIC FRACTURE MECHANICS EVALUATIONS

In Section 9.4,⁽¹⁾ the NRC is overly definitive in defining licensee submittal requirements for the current state of plant-specific analyses and acceptance criteria investigations. For example, crack shape, location and orientation at initiation, and arrest assumptions may be established by the NRC staff rather than the licensee to provide a degree of conformity.

8.6 EFFECTS OF HEATING ECC WATER

We agree with the Appendix K⁽¹⁾ treatment of this potentially important corrective action on a generic basis. However, it is premature to conclude that, "Small-break LOCAs (less than two inches equivalent diameter) also are not a PTS problem...." On a plant-specific basis, and with certain contributing events, breaks of less than two inches may still result in serious PTS scenarios.

8.7 NONDESTRUCTIVE EVALUATION METHODS

We agree with the nondestructive evaluation methods discussed in Appendix L⁽¹⁾ for inservice inspection of plant-specific vessels. It should be emphasized that procedure qualifications, including methods for replication of actual surface conditions, need to be developed for inservice vessels in order to determine the expected probability of detecting various flaw sizes.

8.8 IN SITU ANNEALING

Pressurized thermal shock is viewed as a serious problem because it contributes to vessel embrittlement (i.e., high RT_{NDT}). Annealing is the only proposed corrective action that directly reduces embrittlement. Annealing provides no added procedural burden to reactor operators nor does it require changes in operating procedures or hardware modifications that may compromise core cooling or other safety concerns.

The NRC staff should further promote research directed at qualifying the feasibility of annealing. The staff report⁽¹⁾ largely cited preliminary results of industry-supported research which, understandably, did not favor annealing; however, the NRC staff should also consider independent evaluations of real and/or alleged practical problems in accomplishing annealing.

Value-impact analyses of annealing should be used to compare benefits of annealing with other proposed corrective actions. For example, an anneal can result in an improvement of about 200°F in RT_{NDT} . By comparison, inservice inspection and flux reduction schemes, which give improvements of 10° to 30°F in RT_{NDT} , have been given considerable attention by NRC.

The PNL staff believes that anticipated risks from annealing have not been adequately evaluated and may be less than implied by the limited studies

to date. Further studies are needed to identify potential risks of damage from annealing to vessels, piping, support structures and adjacent concrete. The results should establish whether or not these concerns are serious relative to the reduction in risk associated with the restoration of vessel toughness.

The PNL staff does not believe that testing of a few "reconstituted" charpy specimens would provide adequate predictions of plant-specific rates of re-embrittlement as described in Appendix M.⁽¹⁾ Testing of recovery and reirradiation embrittlement should focus on establishing predictions based on the knowledge of radiation embrittlement described by the current statistically based trend curves. Specific plants should be required to establish a credible prediction of vessel embrittlement after annealing and reirradiation. The plant-specific tests by themselves will not provide the necessary confidence that the plant will not be re-embrittled to an unacceptable level. Therefore, methods need to be developed that allow use of the statistically based trend curves. In this manner, the confidence inherent in the trend curves could be used to enhance the confidence in annealing and subsequent embrittlement rates.

8.9 FUTURE CONFIRMATORY STUDIES

We agree with the Appendix N⁽¹⁾ treatment of the confirmatory studies, but note that it is incomplete in treating all future studies providing confirmatory or new technical data directed at providing solutions to PTS issues. Little information is provided on ongoing thermal hydraulic studies to model plant-specific transients following a PTS event. Future work on the probabilities of vessel failure using the Vessel Integrity Simulation Analysis (VISA) code is not mentioned. Research being done on human factors which apply to expected operator response following PTS precursor events are not covered. Finally, work being done to quantify the mitigating actions to be expected through specific operator training using simulators is not mentioned.

9.0 VALUE-IMPACT ANALYSIS

This chapter presents PNL's value-impact analysis of NRC's proposed screening criteria. The basic objective of a value-impact analysis is to provide adequate information concerning the need for and the consequences of a proposed regulatory action so that cost-effective regulatory actions can be identified. An outline for a value-impact analysis is provided in Section 9.1. The NRC proposed near-term regulatory position is briefly summarized in Section 9.2, and the cost and estimated benefits to be derived from the NRC's proposed position are discussed in Section 9.3. An analytical procedure for examining the radiation dose reduction that may be derived from proposed longer-term corrective actions is described in Section 9.4.

More detailed value-impact information will be provided by PNL staff in subsequent phases of the Pressurized Thermal Shock Project. Information will include examination of generic and plant-specific corrective actions.

9.1 OUTLINE OF A VALUE-IMPACT ASSESSMENT

Requirements and guidelines for value-impact assessments are currently being developed at NRC. An information paper, "Revised Guidelines for Preparing Value-Impact Analysis," was prepared by NRC staff in May 1982.⁽¹⁰⁾ The NRC Division of Risk Assessment is developing a value-impact handbook to provide potential NRC users with a set of instructions to serve as a guide in performing value-impact analyses.

The staff information paper⁽¹¹⁾ indicates that a basic outline would include the following elements: 1) statement of the problem, 2) objectives, 3) proposed action/alternatives, 4) consequences, 5) decision rationale, and 6) implementation.

The actual values and impacts would be analyzed in the consequences section. Some potential measures of value and impact include:

- public health risk - accidental events and routine operations
- public welfare - accidental events
- occupational exposure - routine operations and accidental events
- risk to on-site property
- replacement power costs
- industry implementation costs
- NRC development, implementation, and operation costs
- NRC cost related to accidents
- industry and NRC resource impacts
- regulatory efficiency
- increased understanding of the PTS issue.

The value or impact is the difference between the before and after conditions for each of these measures.

9.2 PROPOSED ACTION

The NRC proposes to establish an RT_{NDT} screening limit of 270°F for axial welds and 300°F for circumferential welds. Three years before any vessel is predicted to reach these limits, plant-specific analyses would be required. These analyses will be used by the licensee and NRC on a plant-specific basis to determine what actions are required to maintain risks from potential PTS events at an acceptably low level through end-of-plant-life. These analyses must consider: plant-specific system features and possible improvements; events which contribute significantly to PTS risk; better identification of vessel material properties; deterministic and probabilistic fracture mechanics evaluations; plant procedures and operator training improvements; inservice inspection for cracks of the vessel inner walls; flux reduction methods to slow future embrittlement; and detailed studies of the feasibility of in-place annealing of the pressure vessel.

The proposed NRC screening criteria determines when analyses and studies are required but do not specify hardware or system modifications, procedure or training changes, or other such actions.

9.3 CONSEQUENCES

The consequences of the RT_{NDT} screening limits and required analyses are discussed in terms of the potential costs and benefits.

There are no direct safety benefits for performing plant-specific analyses. The safety benefits will be derived from implementing the required corrective actions identified in these analyses. Plant-specific analyses are a necessary step in this process. The operating experience record and probabilistic risk assessment studies described in the NRC staff report indicate that potential safety benefits are large for initiating PTS corrective actions.

The major costs of performing plant-specific analyses are the cost to industry to perform the indicated analyses and the cost to NRC to develop guidelines and to review the results. The costs of NRC research to support the PTS regulatory actions was not included in this current analysis but will be considered in the value-impact statement of required corrective actions.

The NRC staff report provides RT_{NDT} data on 40 PWRs. Based upon a preliminary evaluation of this data using the ΔRT_{NDT} formulas in the staff report, (1) seven PWRs are anticipated to reach the screening limit within ten years. Approximately 16 to 17 PWRs are anticipated to reach the screening limit before end-of-life. The cost to industry for performing the plant-specific analyses is estimated as follows:

Identification and quantification of PTS events	\$ 500K
Better identification of vessel material properties	75K
Deterministic and probabilistic fracture mechanics analyses	100K
Flux reduction program analyses	100K
Inservice inspections and nondestructive evaluation study	75K
Plant modification study	200K
Operating procedures and training study	150K
In-situ annealing study	50K
Total cost	<u>\$1250K</u>

After uncertainties are incorporated, a range of \$1-2 million is estimated per plant to perform the required analyses. For the seven PWRs which are expected to reach the screening limit within ten years, this implies an industry analysis cost of \$7 to 14 million. The NRC analysis guideline development and review costs are assumed to be 10 percent of this, or \$0.7-1.4 million.

9.4 RADIATION DOSE REDUCTION

This section outlines input information and an analysis framework that may be useful for determining what reduction in risk could be derived from proposed longer-term corrective actions. Thermal annealing of Robinson 2 is discussed as an example. More detailed evaluations of annealing and other longer-term corrective actions will be provided by PNL staff as part of subsequent phases of the Pressurized Thermal Shock Project.

No detailed risk studies of Robinson 2 are available. In the absence of detailed studies, risk can be estimated by selecting a PWR for which an adequate risk study has been performed. The PTS event frequencies and consequences should be added to this risk study. The results will be assumed to be representative of Robinson 2. Two cases are then examined:

- Case 1: Risk of Robinson 2 to end-of-life without annealing
- Case 2: Risk of Robinson 2 to end-of-life assuming annealing at NRC circumferential screening RT_{NDT} of 300°F.

The Oconee 3 RSSMAP study⁽¹²⁾ is used as a basis for determining risk. This study gives the following frequencies for the WASH-1400 PWR core melt categories:

<u>PWR Release Category</u>	<u>Total Accident Sequence Frequency (per reactor year)</u>	<u>Whole Body Dose (man-rem)</u>
1	1.1E-7	6.4E+6
2	1.0E-5	5.6E+6
3	2.9E-5	6.0E+6
4	9.7E-8	3.1E+6
5	4.6E-7	1.1E+6
6	7.3E-6	1.5E+5
7	3.5E-5	2.3E+3

Public dose consequences were estimated for these release categories by PNL (Prioritization of Safety Issues Project) using the CRAC2 code. The total risk (expected dose) for Oconee 3 is obtained by multiplying the frequency and dose for each consequence category and summing the resultant values. A value of 2.3E2 man-rem per reactor year is obtained.

The Oconee 3 study did not consider PTS events. An overall PTS event sequence is therefore added. The conditional probability of core melt, given a PTS event sequence, is assumed to be unity. The consequences are assumed to be analogous to large LOCA sequences in the RSSMAP study. Four containment failure modes are considered:

- α - vessel steam explosion
- β - penetration leakage
- γ - overpressure due to hydrogen burning
- ϵ - base mat melt through.

Four PTS scenarios are analyzed in the following release categories:

PTS- α (PWR1)	$\alpha = 0.01$
PTS- γ (PWR3)	$\gamma = 0.2$
PTS- β (PWR5]	$\beta = 0.0073$
PTS- ϵ (PWR7)	$\epsilon = 0.8$

These sequences are added to the RSSMAP results and the following is obtained for PTS frequencies of 1.0E-6, 1.0E-5, 1.0E-4, 1.0E-3, 1.0E-2:

PTS Frequency (per reactor year)	Total Oconee 3 Risk (including PTS) (man-rem per reactor year)
1.0E-6	2.3E2
1.0E-5	2.5E2
1.0E-4	3.6E2
1.0E-3	1.5E3
1.0E-2	1.3E4

Now consider the Robinson 2 plant. The PTS risk is increasing because of increased vessel embrittlement throughout the reactor life. The following table was constructed for the case of no annealing (assumes $t = 0$ at Dec. 1981; Robinson 2 is 10 years old and has a remaining life of 25 years):

T Year	EFPY	Best Est. $^{\circ}\text{F}$ (a) R_{TNDT}	PTS Frequency per year (b)	Total Risk man-rem/year
0	7	208	E-6	230
5	10.5	232	1.5E-5	249
10	14	250	7E-5	319
15	17.5	266	1.5E-4	520
20	21	279	3E-4	610
25	24.5	290	4E-4	740

(a) Corrected for 30°F circumferential difference;
 $2\sigma = 34^{\circ}\text{F}$ is subtracted.

(b) Taken from NRC staff report Figure 8-3.(1)

For the case of no annealing, the total risk must be integrated over the 25-year period. This is approximated by:

0-5 years	5(240)	=	1200 man-rem
5-10 years	5(284)	=	1420 man-rem
10-15 years	5(420)	=	2100 man-rem
15-20 years	5(565)	=	2825 man-rem
20-25 years	5(675)	=	3375 man-rem
Total		=	10920 manrem

For the case of annealing, it is assumed that the annealing criterion is 300°F. This results in a best estimate (corrected for 30°F circumferential difference; $2\sigma = 34^{\circ}\text{F}$ for Robinson 2) $300 - 34 - 30 = 236^{\circ}\text{F}$. This implies annealing after five years. Risk is now

0-5 years	5(240)	=	1200
5-25 years	20(230)	=	4600
Total		=	5800 man-rem
Risk reduction = $10920 - 5800 = 5120$ man-rem.			

This example calculation did not include the large uncertainties and should only be used for preliminary comparative purposes. Additional sensitivity and uncertainty calculations should be performed. Key areas which should be investigated further are the probability of PTS-initiated reactor pressure vessel failures leading to core melt and the mode of containment vessel failure.

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NRC FORM 335 <small>(11-81)</small>		U.S. NUCLEAR REGULATORY COMMISSION BIBLIOGRAPHIC DATA SHEET		1. REPORT NUMBER (Assigned by DDC) NUREG/CR-2837, Supp. PNL-4327	
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) PNL Technical Review of Pressurized Thermal Shock Issues: Technical Critique of the NRC Near-Term Screening Criteria				2. (Leave blank)	
7. AUTHOR(S) L. T. Pederson, W. J. Apley, S. H. Bian, P. J. Pelto, E. P. Simonen, F. A. Simonen, D. L. Stevens				3. RECIPIENT'S ACCESSION NO.	
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Pacific Northwest Laboratory Richland, WA 99352				5. DATE REPORT COMPLETED MONTH YEAR November 1982	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Division of Safety Technology Office of Nuclear Reactor Regulation U. S. Nuclear Regulatory Commission Washington, D.C. 20555				DATE REPORT ISSUED MONTH YEAR May 1983	
				6. (Leave blank)	
				8. (Leave blank)	
13. TYPE OF REPORT Technical				PERIOD COVERED (Inclusive dates)	
15. SUPPLEMENTARY NOTES				14. (Leave blank)	
16. ABSTRACT (200 words or less) Pacific Northwest Laboratory (PNL) provided a technical critique of the draft report, NRC Staff Evaluation of Pressurized Thermal Shock, dated September 13, 1982. This report provided the basis for the NRC near-term regulatory position on pressurized thermal shock (PTS) and recommended a generic screening criteria for welds in the vessel beltline region. The PNL staff concluded that the screening criteria were adequate to meet the intent of the NRC safety goal and to retain past predictions of vessel reliability. The conclusion was based on selecting the plant-specific nilductility transition reference temperature (RT _{NDT}) in the conservative manner described within the staff report. Conservative and unconservative factors were mentioned throughout the NRC staff report. The PNL staff has listed these factors together with unknown (may be either conservative or unconservative) factors and estimated, where possible, the range in F RT _{NDT} . The unknown factors were so widespread that the PNL staff recommended that specific conservatisms not be reduced until the unknowns are further resolved.					
17. KEY WORDS AND DOCUMENT ANALYSIS			17a. DESCRIPTORS		
17b. IDENTIFIERS/OPEN-ENDED TERMS					
18. AVAILABILITY STATEMENT Unlimited			19. SECURITY CLASS (This report) Unclassified		21. NO OF PAGES
			20. SECURITY CLASS (This page) Unclassified		22. PRICE \$