November 23, 1982

SECY-82-465

POLICY ISSUE
(Notation Vote)

FOR: The Commissioners

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: PRESSURIZED THERMAL SHOCK (PTS)

PURPOSE: To request Commission approval of recommended near-term actions related to protection against pressurized thermal shock events.

DISCUSSION: The issue of pressurized thermal shock (PTS) arises because in pressurized water reactors (PWRs) transients and accidents can occur that result in severe overcooling (thermal shock) of the reactor pressure vessel, concurrent with or followed by repressurization. In these PTS events, rapid cooling of the reactor vessel internal surface results in thermal stress with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature profile across the reactor vessel wall as a function of time. The effects of this thermal stress are compounded by pressure stresses if the vessel is pressurized.

Severe reactor system overcooling events which could be accompanied by pressurization or repressurization of the reactor vessel (PTS events) can result from a variety of causes. These include system transients, some of which

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SECY NOTE: This subject is scheduled for consideration at an open meeting on December 1, 1982.

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are initiated by instrumentation and control system malfunctions including stuck open valves in either the primary or secondary system, and postulated accidents such as small break loss-of-coolant accidents (LOCAs), main steam line breaks (MSLBs), and feedwater pipe breaks. Eight overcooling events have already occurred in which the primary coolant system water temperature rapidly decreased by 200°F or more.

As long as the fracture resistance of the reactor vessel material is relatively high, such events are not expected to cause vessel failure. However, the fracture resistance of reactor vessel materials decreases with exposure to fast neutrons during the life of a nuclear power plant. The rate of decrease is dependent on the metallurgical composition of the vessel walls and welds. If the fracture toughness of the vessel has been reduced sufficiently by neutron irradiation, severe PTS events could cause propagation of fairly small flaws that might exist near the inner surface. The assumed initial flaw might initiate and propagate into a crack through the vessel wall of sufficient extent to threaten vessel integrity and, therefore, core cooling capability.

The PTS issue is a concern only for operating PWRs. Boiling water reactors (BWRs) are not a significant PTS concern. BWRs operate with a large portion of water inventory inside the pressure vessel at saturated conditions. Any sudden cooling will condense steam and result in a pressure decrease, so simultaneous creation of high pressure and low temperature is improbable. Also contributing to the lack of PTS concerns for BWRs is the lower fast neutron fluence at the vessel inner wall, and the use of a thinner vessel wall which results in a lower stress intensity for a postulated crack.

Evaluations of Pressurized Thermal Shock by the NRC staff in the spring of 1981 concluded that no immediate licensing actions were required at that time, but that since the consequences of overcooling events increase as the vessels accumulate additional neutron irradiation, extensive further investigations were needed to determine whether and when corrective actions will be needed to provide assurance of vessel integrity throughout the intended service life of a reactor vessel.

On March 31, 1981, the NRC staff held the first of many meetings that were to occur over the following sixteen months with licensees, reactor manufacturers, and owners groups to discuss pressurized thermal shock concerns and exchange technical information.

The Commission was briefed regarding PTS on June 11, 1981 (See SECY 81-286, May 4, 1981).

In letters dated August 21, 1981, the staff requested the licensees of eight plants representative of older reactor vessels to provide more detailed information on the present and projected pressure vessel materials properties, on the probability and possible severity of events that could cause failure of embrittled vessels, and on the efficacy and feasibility of several potential corrective actions.

In Commission briefings on September 15, 1981 (SECY 81-286A, September 8, 1981) and on November 24, 1981, the staff reported on the progress of these activities and indicated its plans to develop a near-term position on PTS by the summer of 1982.

On December 28, 1981, the Commission approved the staff's proposal to designate pressurized thermal shock as an Unresolved Safety Issue (SECY 81-687, December 8, 1981). The Task Action Plan for the PTS issue (A-49) comprises both the development of the near-term position and completion of longer-range studies by mid-1984.

On March 9, 1982, the staff briefed the Commission on the status of the staff reviews of the responses to the August 21, 1981 letters (SECY 82-97, March 5, 1982). Many of the event-sequence analyses provided by licensees in response to the August 21, 1981 letter can be characterized as design-basis event analyses of the type generally submitted in Safety Analysis Reports (SARs) in support of license applications. Such analyses tend not to be of much help in evaluations of PTS. Many of the assumptions in such analyses were developed and accepted for licensing purposes without regard to PTS concerns. While SAR analyses appear to be appropriately conservative for calculations of reactor core thermal performance, PTS evaluations are most usefully performed using best-estimate calculations of pressure and temperature behavior. In addition, some potential event sequences that are not generally analyzed in detail in Safety Analysis Reports, because their consequences for core cooling are bounded by the design-basis event analyses, can be of greater significance for PTS evaluations. Thus, it became clear to the staff that plant-specific PTS evaluations must include a systematic examination of many potential events, with particular attention to the probability and consequences of various possible operator actions and omissions, and equipment malfunctions. At the March 9 briefing, the staff indicated that in June 1982 it expected to provide a reassessment of the PTS issue directed at the question of whether corrective actions are required at any plant that must be initiated before the longer term PTS program provides generic resolution and acceptance criteria.

In early June 1982, the staff described to the ACRS, and to the PWR owners groups, the status of the staff's work and the approach it expected to take in its June reassessment. The owners' groups took issue with the bases for several of the staff's recommendations and indicated that recently developed information would be provided to the staff that would support different conclusions. The ACRS noted in its letter of June 7, 1982, that it had not been provided sufficient information to evaluate the adequacy of the staff's approach but expressed the belief that there is no need for any immediate plant modifications to permit continued operation.

As a result of these comments, and since the staff, the ACRS, and the industry had concluded that there was no need for immediate action, the staff decided that an additional effort should be made to review the forthcoming additional industry information, attempt to resolve technical differences, and modify the staff position, if appropriate, before moving forward to the CRGR and the Commission. This reevaluation raised a number of significant technical problems to be resolved as part of the re-formulation of the staff recommendations. As a result of intensive staff efforts and many discussions with the owners groups and consultants, the staff developed an initial draft evaluation report on September 13, 1982. Subsequent to additional staff and management reviews, review by the ACRS on October 8, and review by CRGR on October 6 and October 28, the draft staff report has been revised (Enclosure A) and forms the technical basis for the staff's recommendations to the Commission.

Summary of the Staff Report

The staff PTS report (Enclosure A) describes the nature of the PTS concern, the staff's approach to assessment of the risk to the public from PTS events, and the staff's technical conclusions. This paper contains the staff's recommendations to the Commission for future actions that we believe are needed to maintain that risk at an acceptable level as further vessel irradiation is incurred.

Section 1 of Enclosure A gives a general background and introduction to the PTS issue.

Section 2 discusses the frequency and characterization of overcooling events that have actually been experienced at operating domestic PWRs. These events are used as the primary basis for determining the frequency of potential PTS related events that may reasonably be expected in the future.

Section 3 summarizes deterministic fracture mechanics calculations performed for these experienced events and parametric studies of crack growth potential as a function of the event characteristics and RT_{NDT} values. For each of the eight most significant experienced events, a fracture mechanics parametric study was performed to determine the critical material condition that would have resulted in vessel failure (due to the event) if a flaw were present of whatever size would be most likely to propagate. This single critical material condition (Critical Reference Temperature for the Nil-Ductility Transition, or RT_{NDTc}) was then used to characterize the severity of each of the experienced events.

In Section 4, the frequency of PTS events, and the severity of such events are combined to determine a value of RT_{NDT} below which the PTS risk is acceptable. That RT_{NDT} is recommended as a screening criterion.

Section 5 presents the staff's proposed conservative method for estimation of vessel-specific values of RT_{NDT} for comparison with the screening criterion.

Section 6 describes an evaluation of the frequency and character of potential lower probability overcooling events. These events have not occurred and so are evaluated to have a frequency lower than the eight events discussed in Section 2. The frequency and severity of several such groups of less frequent but potentially more severe events is estimated, using probabilistic methods.

Section 7 summarizes sensitivity studies performed using a probabilistic treatment of PTS fracture mechanics calculations. In this approach, significant PTS parameters are estimated in a more realistic, probabilistic way (for example, materials properties, and flaw size probability distributions). The result is an estimate of reactor vessel failure probability, given the occurrence of a specified overcooling transient.

In Section 8, the probabilistic fracture mechanics calculations are used in combination with the overcooling sequence results of Section 6 to estimate probabilities of vessel failure due to PTS.

Section 8 also considers how all of the above analyses affect our perception of PTS risk for a vessel with RT_{NDT} at the proposed screening criterion, and why those risks are believed to be acceptable.

Section 9 presents an outline of the plant specific safety evaluations proposed to be furnished well before a plant reaches the screening criterion. The analyses would be to determine risk due to PTS at the specific plant, and to define corrective actions needed, if any, to reduce that risk to an acceptable value.

Section 10 of the staff PTS report (Enclosure A) presents the staff conclusions.

Summary of Technical Conclusions and Recommendations

As a result of evaluations performed thus far on the issue of pressurized thermal shock, the NRC staff has reached the following conclusions:

(1) The risk from PTS events for reactor vessels with RT_{NDT} values less than the proposed screening criterion (270°F for axial welds, and 300°F for circumferential welds) is acceptable. On the basis of presently available information, no reactor vessel will exceed the screening criterion for the next few years, therefore there is no need to shutdown or anneal any operating PWR in the next few years.

(2) Most plants can avoid reaching the screening criterion throughout their service life by timely implementation of flux reduction programs. Such

flux reduction programs should be implemented on a time schedule that will avoid foreclosure of this option.

(3) Any plant for which the value of RT_{NDT} is projected to reach the screening criterion before the end of service life, using the conservative method of RT_{NDT} determination described in Section 5 and Appendix E of Enclosure A, should submit plant-specific evaluations (of the type described in Section 9 of Enclosure A) to determine what, if any, modifications to equipment, systems and procedures should be required to provide acceptable protection against vessel failure from PTS events for the remainder of plant life. These evaluations should be submitted three years before the vessel is projected to reach the screening criterion.

(4) In the near future, the staff should develop more detailed guidance for these evaluations and acceptance criteria for determining whether plant modifications are needed based on the evaluations.

(5) Some of the Commission's regulations (Appendix G to 10 CFR Part 50, 10 CFR 50.46, and possibly others) may not appropriately reflect current understanding of the state of reactor vessel embrittlement and the potential for vessel failure as a result of PTS. Timely consideration should be given to the need for amendments to the regulations (as discussed in Section 8.6 of Enclosure A).

Discussion of Proposed Near-Term Actions

(1) Proposed Rulemaking: The staff proposes to develop a notice of proposed rulemaking. The proposed rule would: establish an RT_{NDT} screening criterion of 270°F for axial welds, and 300°F for circumferential welds; require that licensees of all operating PWRs submit a determination of the present RT_{NDT} values

for their reactor vessels, and the estimated date at which the RT_{NDT} value will exceed the screening criterion; require licensees to implement such flux reduction programs as are feasible and necessary to avoid reaching the screening criterion before expiration of the operating license; and require licensees of operating PWRs for which the RT_{NDT} value is projected to exceed the screening criterion before the expiration of the operating license to submit a plant-specific PTS safety analysis, of the type outlined in Section 9 of Enclosure A, three years before the screening criterion is exceeded, or one year after the effective date of the regulation, whichever is later. For purposes of comparison with the screening criterion, the rule would require calculation of RT_{NDT} values in the manner described in Section 5 of Enclosure A. During the public comment period on the proposed rulemaking, the staff would develop more detailed guidance on the plant-specific analyses to be required and on the acceptance criteria to be used in judging the acceptability of the results. Since most of the technical basis for such a rulemaking has already been assembled in Enclosure A and the documents referenced therein, the staff believes that the notice of proposed rulemaking could be prepared for Commission approval in about six months or less.

(2) Flux Reduction Programs

On the basis of information currently available to the staff, it appears that although some plants will require no remedial actions for their vessel RT_{NDT} to remain below the screening criterion throughout their service life, a substantial number of PWR vessels are now predicted to exceed the criterion well before the end of life. Flux reduction programs can reduce the rate of increase of RT_{NDT}, but the effectiveness of flux reduction depends on early implementation. The staff has developed information on the costs and benefits of near-term flux reduction measures in Appendix I of Enclosure A. Those plants for which the vessel RT_{NDT} is expected to exceed the screening criterion before the end of service life can generally be grouped into three categories.

For several plants, action within the next few years to reduce the flux at critical welds by a factor of two or less will ensure that they do not exceed the screening criterion throughout their service life.

It appears that such a reduction can be achieved through installation of a low leakage core (installation of partially burned fuel assemblies in the periphery of the core in place of fresh fuel assemblies). This fuel management option is already being implemented by some licensees at reportedly little or no additional cost. For plants in this category, the staff believes that the rulemaking proposed above will provide a mechanism for ensuring prompt consideration of appropriate flux reduction measures.

There is a group of about eight plants for which near-term action to reduce the flux at critical welds by factors of two to five would ensure that they do not exceed the screening criterion throughout their service life. It appears to the staff that such flux reduction factors could be attained through the installation of a low leakage core and the replacement of a few (8 - 12) peripheral fuel assemblies by dummy assemblies. These measures seem practical and cost-effective to the staff, based on present knowledge, with some loss in margin to core overheating limits in certain postulated transients. We estimate that there would be an engineering cost (redesign and safety analyses) of about $20 million per plant, and some small increase in fuel cycle or operating costs. For such plants, the staff proposes to meet with licensee management when the notice of proposed rulemaking is issued to discuss the importance of prompt detailed evaluation of near-term flux reductions that would avoid foreclosure of that option as an effective means of reducing PTS risk, and to determine the licensee's plans for such a program. Should these meetings not result in effective actions by the licensees, the staff would propose to issue letters pursuant to 10 CFR 50.54(f) requesting information "to enable the Commission to determine whether or not the license should be modified, suspended or revoked." The scope of the information request is planned to include the amount of flux reduction needed to stay within the screening criteria, the flux reduction attainable, safety evaluation for non-PTS aspects of flux reduction, other actions proposed by the licensee, and value/impact information.

One plant, H. B. Robinson Unit 2, is so close to reaching the screening criterion that the fuel management options described above could not reduce the flux at the critical welds sufficiently to prevent reaching the screening criteria before the end of service life. The staff proposes that Carolina Power and Light Company be ordered to submit a comprehensive plan showing what actions will be taken to resolve the PTS issue for the Robinson plant and to show cause why the license should not be modified to provide adequate protection for PTS for this plant.

Discussion of Longer-Term Actions

(1) The ongoing program to improve procedures and operator training regarding prevention and mitigation of PTS events should continue, as described in Appendix C of Enclosure A.

The staff has audited training and procedures at eight older PWRs. Emphasis during these audits was on procedural adequacy and the operators' understanding of PTS events and the potentially conflicting requirements of avoiding PTS situations while at the same time assuring adequate cooling to the core. The audit reports are summarized in Appendix C to Enclosure A. Generally, it was found that adequate procedures and training exist at the eight plants, although longer range improvements are desirable. The exceptions were noted, and were corrected promptly by the licensees.

The industry is pursuing a major revision of emergency operating procedures as part of TMI Action Plan Item II.C.1.

Longer range improvements to the PTS related procedures and training will result from the integrated, long range reassessment of procedures in this program which is aimed at adopting "symptom oriented" in addition to "event oriented" procedures. That program is also discussed in Appendix C to Enclosure A.

The staff believes that it is important to avoid quick and/or frequent changes to procedures with consideration being focused on a particular concern.

Therefore, only the urgent and necessary PTS-related procedure changes are being implemented before the integrated program results.

(2) Industry and NRC programs are needed to provide additional confirmatory PTS information, to decrease the large uncertainty of current PTS analyses, to apply the analysis to B&W and CE plants, and to investigate more thoroughly the alternatives to reduce and mitigate PTS risks. In particular, the analytical and experimental studies underway as part of the NRC research program, as described in Appendix N of Enclosure A, should continue on a high-priority basis. These programs should improve the staff's capability for independent audits and assessments of licensee evaluations, confirm or improve calculations methods and assumptions, and aid in further assessments of safety margins.

As discussed in Section 2, of Enclosure A, it is particularly important to obtain B&W analyses to confirm the acceptability of the screening criteria for B&W plants, since B&W plant design is significantly different from W and CE. Also as noted in Section 2, operating experience shows a significantly different history of PTS precursors for B&W plants.

(3) The best available methods should be used for periodic in-service inspection of high-RT_{NDT} vessels, to maximize the likelihood of detecting any flaws that may be present relevant to PTS. In addition to the necessary assurance of vessel integrity provided by such inspections, it would be useful to investigate how inspection results could be used to improve the flaw probability distributions to be used in plant-specific analyses.

(4) The staff and the Commission should give timely consideration to the possible need to amend certain of the regulations to better reflect the potential for PTS.

Value-Impact

Enclosure B provides a preliminary value-impact assessment of the recommended near-term requirements for a screening criterion and analysis.

The major costs involved in the presently proposed action 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 to implement actions found to be necessary have not been estimated and will have to be determined on a plant-specific basis.

The proposed requirements for plant-specific analyses are but the first of two stages in the resolution of pressurized thermal shock. Their costs are modest and appear well justified since the consequence of not performing the recommended analyses is that certain reactor vessels would become increasingly embrittled and the risk of pressure vessel failure would continue to increase. The second stage of resolution, implementation of any facility modifications needed to reduce the risk of vessel failure from pressurized thermal shock, could result in large costs, and result in significant risk reduction benefits. These costs can be minimized by the early implementation of flux reduction measures, as discussed above. Value-impact assessment for such implementation will have to be performed on a plant-specific basis.

There are no direct safety benefits from performing the plant specific analyses. The safety benefits will be derived from implementing any required corrective actions identified in these analyses. The plant specific analyses are a necessary step in this process. The operating experience record and probabilistic studies described in Enclosure A indicate that the potential safety benefits are large for initiating PTS corrective regulatory action for plants approaching the screening criterion.

COORDINATION: In a letter to the Chairman on June 7, 1982, (Enclosure C) the ACRS commented that it had not been given sufficient information to evaluate the adequacy of the NRC Staff's approach to develop regulatory requirements concerning pressurized thermal shock (PTS). In response to the comment, the Staff prepared and sent a draft report to the ACRS entitled, "Draft NRC Staff Evaluation of Pressurized Thermal Shock" dated September 13, 1982. This report, with supporting information in the Appendices, addressed the numerous technical aspects of the PTS issue.

During the 270th meeting of the ACRS, October 7-8, 1982, the ACRS reviewed this draft report and provided comments in a letter to the Chairman dated October 14, 1982, (Enclosure D). The staff responded to the ACRS in a letter dated November 12, 1982 (Enclosure E). We believe that the actions proposed in this paper, and the technical information in Enclosure A are consistent with the ACRS recommendations.

The ACRS "believes that if due consideration is given to the items mentioned above, and if regulatory actions are based on the proposed screening criteria, the pressurized thermal shock matter should not present an undue risk to the health and safety of the public."

The Committee recommended that an orderly, comprehensive research program is needed and suggested some program elements. The staff agrees that such a program is needed, and is re-evaluating the ongoing program to define any changes or additions that should be made.

The Committee recommended five items for special attention:

- Careful assessment of the uncertainties and special circumstances related to each data point used in the correlation of vessel properties with material composition and fast neutron irradiation, in the range of high embrittlement. The staff intends to do this.

- Improvements in PTS-related operator training and procedures. These have been accomplished for the seven plants already audited (see Appendix C of Enclosure A), and are planned as part of the comprehensive procedures improvement program for all plants.

- Better characterization of PTS initiating events and subsequent operator actions. This is planned to be a part of recommended Long Term Action (2).

- Consideration of heating the ECCS water. This is included in the plant-specific analysis program outlined in Section 9 of Enclosure A. At least two plants have already announced their intention of heating the ECCS water to alleviate potential PTS problems.

- Fast neutron fluence reduction. The staff proposes in this paper to pursue a vigorous program for all

(3) Note that the staff will meet with licensees of plants for which near-term flux reductions of factors of two to five would ensure that the screening criterion would not be exceeded throughout service life, to determine the licensees' plans for such programs, and that the staff will propose issuance of 10 CFR 50.54(f) letters to such licensees if appropriate following the meetings;

(4) Note that the staff will prepare additional guidance for the plant-specific PTS safety analyses referred to in the proposed rulemaking, during the public comment period on the proposed rule;

(5) Note that the staff will consider the possible need to amend certain of the regulations to better reflect the potential for PTS, as discussed in Section 8.6 of Enclosure A.

(6) Note that the staff will continue the ongoing program to improve procedures and operator training regarding prevention and mitigation of PTS events, as described in Appendix C of Enclosure A; and

(7) Note that the analytical and experimental programs described in Appendix N of Enclosure A, and in the Task Action Plan for USI A-49, "Pressurized Thermal Shock," will continue on a high-priority basis.

William J. Dircks
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Enclosures:
*A. NRC Staff Evaluation of Pressurized Thermal Shock (November 17, 1982)
B. Value-Impact Assessment
C. Letter, P. Shewmon to Chairman Palladino, June 7, 1982
D. Letter, P. Shewmon to Chairman Palladino, October 14, 1982
E. Letter, W. J. Dircks to P. Shewmon, November 12, 1982
F. Minutes of CRGR Meeting No. 21, October 30, 1982
G. Minutes of CRGR Meeting No. 23, November 12, 1982

*Commissioners, SECY, OGC, OPE and EDO only.

The Commissioners

Commissioners' comments should be provided directly to SECY by c.o.b. Friday, December 10, 1982.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT December 3, 1982, with an information copy to SECY. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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ERRATA SHEET FOR ENCLOSURE A (PTS Report and Appendices)

(Enclosure A was distributed seperately from SECY-82-465)

1. Insert the attached page numbered 4-3 at the end of Section 4.
2. Insert the attached page numbered 5-7 at the end of Section 5.
3. Note that references to RT_{NDT} on Figure 7.3 are to mean values of RT_{NDT}.
4. Insert the corrections shown on attached markup of pages 8-10 and 8-16 (Corrections have already been made by hand on most copies).
5. Replace page D-6 in Appendix D with the attached corrected page numbered D-6.
6. Replace all 5 pages of Table P-1 at the end of Appendix P with the attached 5 page "Rev. 1" Table P-1

plants. The data also show that the CE experience (zero events in 60 reactor years) does not statistically justify use of a different, less restrictive criterion for CE plants. Such justification, if presented in the near future, could not be based on statistics alone.

The use of an experience base of only eight events to develop an expected frequency distribution clearly yields values with large uncertainties and does not take account of low frequency events that have not occurred. In addition, the temperature histories used in the fracture mechanics calculations were measured in the cold leg piping, whereas the temperature of interest (at highly irradiated welds) is in the reactor vessel downcomer which might have been colder.

The fracture mechanics calculations assume that flaws of critical size are present at the limiting welds (those with highest RT_{NDT}). This is clearly a conservatism in the analysis, which is considered in an approximate manner in the probabilistic fracture mechanics analysis (Section 7).

Because the intent is to select a screening criterion generically, covering a wide range of transient sequences, the analysis does not take credit for the warm prestress phenomenon which would be beneficial in many actual transient sequences. No account has been taken of the effects of weld residual stresses, which would generally add to the thermal and pressure stresses.

Perhaps the most significant uncertainty in the treatment described thus far is that there are potential low frequency overcooling events much more severe than those that have been observed. Because these events have not occurred, they have not been taken into account in the frequency distribution used.

Because of all of the nonquantified uncertainties noted above, the staff has also examined what insights can be gained from calculations of the characteristics of various postulated overcooling events and estimates of their expected frequency of occurrence; and from a probabilistic study of the fracture mechanics calculations. These considerations are described in Sections 6, 7 and 8.

5.4 "Evaluation of Irradiation Response of Reactor Pressure Vessel Materials" Prepared by Combustion Engineering, Inc. (J. D. Varsik principal investigator) for Electric Power Research Institute, Research Project RP 1553-1, Final Report, to be published.

reactor accidents. For analyzing how PTS events contribute significantly to the risk to the public, the following logic applies:

1. PTS event sequences leading to RPV failure have overall frequency F per reactor-year. Figures 8-2 and 8-3 provide a very approximate estimate of F. A plant evaluated (as described in Section 5 or 9 and Appendix E) to be at the 270°F screening criterion is likely to have a true RT_{NDT} of 150-270°F (two sigma ≅ 60°F). For the mean of 210°F, F≅ 6 x 10^{-6} per reactor-year on the NRC curve (Figure 8-3), and much smaller on the WOG curve (Figure 8-2).

2. A fraction X<1 of RPV failure sequences leads to core melt, giving an expected value of XF core melts per reactor-year.

3. A fraction Y of failures leading to core melt leads to significantly early radioactive releases, so the expected value of the frequency of significant early releases due to PTS is XYF, which is therefore the expected value of the frequency of events involving non-zero early deaths due to PTS.

4. THe risk of early deaths to the average individual within one mile of the site is given by XYFD, where the factor D includes the effects of dispersion and wind direction.

5. To show PTS risk to be lower than 10% of the safety goal guidelines would involve showing

$$XF < 10^{-5} \text{ per reactor-year}$$

and

5×10^{-8}

$$XYFD \lesssim \cancel{10^{-5}} \text{ per reactor-year}$$

The limiting value for XYFD is approximate, for an averge site.

Flaws and Cracks - The deterministic calculations assume the presence of a long through-clad flaw of critical depth--a substantial conservatism. The probabilistic calculations use a through-clad crack probability that is highly uncertain and that some people believe is conservative. No account is taken of actual in-service inspection results in these generic calculations.

The crack growth/arrest model used by the staff assumes long initial flaws that grow uniformly over their length. This initial flaw shape is conservative. The growth/arrest shape is discussed in Section 3; we believe that, once a crack initiates, the long crack is a more realistic description than less conservative shapes used in other models.

Stresses - the models include no residual stresses, which is non-conservative. The NRC model includes cladding effect, which is realistic for through-clad cracks. The probabilistic analysis of small-break LOCA used in Section 8 took credit for warm prestressing.

None of the models used for generic events currently includes a warm prestress (WPS), which is a conservatism for transients satisfying the WPS conditions. WPS is included in calculations specifically limited to the small-break LOCA, where it has been demonstrated that WPS is applicable.

Material Properties - The estimation of RT_{NDT} as described in Section 5.4 and 8.5, is a substantial conservatism.

Fracture Mechanics - The use of linear elastic fracture mechanics in the upper-shelf temperature region is not appropriate. Assuming vessel failure when the stress intensity factor for a crack reaches the upper-shelf toughness is believed by many people to be conservative because considerable ductility exists in the remaining ligament. Until we have validated applicable elastic-plastic models, however, the degree of conservatism cannot be determined.

Uncertainties in Probabilistic Calculations - Substantial uncertainties exist in probabilistic calculations as discussed in Section 8.3. The characterization of event sequences by T_f, β, and P is an oversimplification that may or may not be on the conservative side.

D.1.2 Stress Algorithms

The total peak stresses (thermal plus pressure plus residual plus any other stresses) are assumed to be less than, or at least not significantly larger than, the material yield strength so that components of stress can be added and that linear-elastic fracture mechanics procedures can be utilized. For rapid thermal transients, high stresses usually occur locally at the inner vessel wall and acceptable stress distributions (total stress below yield) over the remaining section can still be obtained if the overstressed region is relatively thin.

D.1.3 Postulated Initial Cracks

Long through-clad cracks, either axial or circumferential, are assumed to exist in the welds of limiting (highest) RT_{NDT}. In this case, the cladding effect is conservatively applied in that the stresses due to the different expansion coefficients of the clad and base metal are added to the nominal thermal stresses. For short through-clad cracks or underclad cracks it is conceivable that the cladding can have a beneficial effect if the cladding is sufficiently tough, that is, it is less affected by irradiation damage than the base material. In this case, it could deter crack elongation or could even prevent crack initiation depending on the specific transient. At present, there are differences of opinion as to clad toughness after irradiation, and further research is needed as to the behavior of short or underclad cracks in an overcooling event. Also, analyses to date omit consideration of weld residual stresses and in the case of circumferential cracks, the effect of dead weight stresses. ~~An uneven temperature distribution in the azimuthal direction increases K_I value for circumferential cracks.~~ Therefore, the NRC concludes that the more conservative assumption of long through-clad cracks should be used at least for scoping calculations, until further information s developed to permit a relaxation of this assumption.

D.1.4 Fracture Mechanics Algorithms

Fracture mechanics analyses utilize the linear-elastic boundary integral methods of Heliot, Labbens and Pellissier-Tanon (References D.1 and D.2).

Rev. 1

Table P.1 RT_{NDT} Values for All Plants(1) Calculated Per the Recommendations of the Working Group on RT_{NDT} (2) for the Vessel Inside Surface.

Plant NSSS/Vessel Fabricators	EFPY as of 12/31/81	Fluence n/cm^2 $\times 10^{18}$	Copper %	Nickel %	Mean Initial RT_{NDT}	Mean ΔRT_{NDT}	$2\sqrt{\sigma_0^2+\sigma_\Delta^2}$ (5)	RT_{NDT}, °F, as of Dec 31, 1981(6) Circum.	Axial	Licensee's RT_{NDT}, °F
Robinson 2 W/CE	7.10	(14.1)(3)(8) 14.8 (3)(8)	(0.35) 0.27	(1.20) 0.20	(-56) -56	(303)(4) 151	34 (4) 59	281	154	290 220
Turkey Point 4 W/B&W	5.67 No Axial Welds	9.1 (9)	(0.32)	(0.57)	(0)	(200)	59	259		211
Turkey Point 3 W/B&W	5.67 No Axial Welds	(9.1)(9)	(0.32)	(0.57)	(0)	(200)	59	259		
Fort Calhoun CE/CE	5.07	(7.04) 5.1 (10)	(0.35) 0.35	0.99 0.99	(-56) -56	(264)(4) 248 (4)	34(4) 34 (4)	242	226	209 (239)(7)
Maine Yankee CE/CE	5.90	(5.02) 4.14	(0.36) 0.36	(0.99) 0.99	(-56) -56	(248)(4) 238(4)	34(4) 34(4)	226	216	170 (198)(7)
Indian Point 3 W/CE	2.98 Plate Governs	(1.67)	(0.24) 0.24	(0.52) 0.52	(+74) +74	(90) 90	48 48	212	212	
Yankee Rowe W/B&W/B&W	14.56 Plate Governs	(11.35)	(0.20) 0.20	(0.63) 0.63	(+30) +30	(133) 133	48 48	211	211	
Rancho Seco B&W/B&W	3.54	(2.33) 2.05	(0.31) 0.35	(0.59) 0.59	(0) 0	(135) 148	59 59	194	207	
Three Mile Island 1 B&W/B&W	3.52	(1.87) (1.87)	(0.31) 0.35	(0.68) 0.60	(0) 0	(133) 145	59 59	192	204	145
Oconee 2 B&W/B&W	4.71 No Axial Welds	(2.87)	(0.35)	(0.71)	(0)	(172)	59	231		

(1)See footnote(s), last page of table.

These values are subject to change when plant-specific analyses yield better information.

Rev 1

Table P-1 (Continued)

Plant NSSS/Vessel Fabricators	EFPY as of 12/31/81	Fluence n/cm^2 x10^{18}	Copper %	Nickel %	Mean Initial RT_{NDT}	Mean ΔRT_{NDT}	$2\sqrt{\sigma_0^2+\sigma_\Delta^2}$ (5)	RT_{NDT}, °F, as of Dec 31, 1981(6) Circum.	Axial	Licensee's RT_{NDT}, °F
Point Beach 1	8.07	(10.01)	(0.24)	(0.57)	(0)	(151)	59	210		
W/B&W		7.34	0.24	0.57	0	139	59		198	
Oconee 1	5.04	(2.32)	(0.26)	(0.61)	(0)	(113)	59	172		
B&W/B&W		2.73	0.31	0.55	0	138	59		197	160
Zion 1	4.97	(3.13)	(0.35)	(0.59)	(0)	(166)	59	225		
		0.99	0.31	0.61	0	108	59		167	
Indian Point 2	4.40	No Circum Data								
W/CE		2.2	0.34	1.2	-56	211 (4)	34 (4)		189	
Arkansas ANO-1	4.42	(2.70)	(0.31)	(0.59)	(0)	(140)	59	199		
B&W/B&W		1.99	0.31	0.59	0	129	59		188	
Point Beach 2	7.54	(9.35)	(0.25)	(0.59)	(0)	(156)	59	215		
W/B&W, CE		No Axial Welds								
Ginna	8.18	(9.49)	(0.25)	(0.56)	(0)	(154)	59	213		
W/B&W		No Axial Welds								
San Onofre	9.04	(33.45)	(0.27)	(0.20)	(-56)	(188)	59	191		203
W/CE		27.12	0.27	0.20	-56	178	59		181	
Zion 2	4.49	(2.83)	(0.26)	(0.61)	(0)	(119)	59	178		
B&W/B&W		0.90	0.35	0.59	0	118	59		177	
Palisades	4.12	(4.78)	(0.25)	(1.2)	(-56)	(174)	59	177		
CE/CE		4.78	0.25	1.2	-56	174	59		177	
Crystal River 3	2.49	(1.44)	(0.35)	(0.59)	(0)	(134)	59	193		
B&W/B&W		1.36	0.31	0.61	0	118	59		177	

Rev. 1

Table P-1 (Continued)

Plant NSSS/Vessel Fabricators	EFPY as of 12/31/81	Fluence n/cm^2 x10^{18}	Copper %	Nickel %	Mean Initial RT_{NDT}	Mean ΔRT_{NDT}	$2\sqrt{\sigma_0^2+\sigma_\Delta^2}$ (5)	RT_{NDT}, °F, as of Dec 31, 1981(6) Circum.	Axial	Licensee's RT_{NDT}, °F
Surry 1	4.88	(7.61)	(0.25)	(0.51)	(0)	(141)	59	200		
W/B&W		1.66	0.21	0.59	0	81	59		140	
Cook 1	4.56	(2.87)	(0.40)	(0.82)	(-56)	(222) (4)	34(4)	200		
W/CE		1.55	0.13	0.99	-56	58	59		61	
North Anna 1	2.41	(4.42)	(0.14)	(0.80)	(+38)	(76)	48	162		
W/RD		No Axial Welds		Forging Governs		48			162	
Beaver Valley	1.87	(3.16)	(0.37)	(0.62)	(-56)	(179)	59	182		
W/CE		0.47	0.36	0.62	-56	104	59		107	
North Anna 2	0.77	(1.38)	(0.13)	(0.83)	(+56)	(52)	48	156		
W/RD		No Axial Welds		Forging Governs		52	48		156	
Salem 1	2.26	(1.49)	(0.24)	(0.51)	(+51)	(87)	48	150		
W/CE			0.24	0.51	Plate Governs	87	48		150	
Oconee 3	4.78	(2.92)	(0.24)	(0.63)	(0)	(112)	59	(171)		
B&W/B&W		No Axial Welds								
Surry 2	4.83	(7.54)	(0.19)	(0.56)	(0)	(108)	59	167		
W/B&W, RD		1.64	0.21	0.59	0	81	59		140	
Calvert Cliffs 1	4.65	(6.84)	(0.30)	(0.18)	(-56)	(135)	59	138		
CE/CE		6.84	0.21(11)	0.85	-56	136	59		139	205(244)(7)
St. Lucie	3.52	(2.22)	(0.31)	(0.11)	(-56)	(98)	59	101		
CE/CE		2.22	0.30	0.64	-56	132	59		135	

Rev. 1

Table F-1 (Continued)

Plant NSSS/Vessel Fabricators	EFPY as of 12/31/81	Fluence n/cm^2 $x10^{18}$	Copper %	Nickel %	Mean Initial RT_{NDT}	Mean ΔRT_{NDT}	$2\sqrt{\sigma_0^2+\sigma_\Delta^2}$ (5)	RT_{NDT}, °F, as of Dec 31, 1981(6) Circum.	Axial	Licensee's RT_{NDT}, °F
Calvert Cliffs 2 CE/CE	3.63	(5.34)	(0.30) 0.30	(0.18) 0.18	(-56) -56	(127) 127	59 59	130	130	
Trojan W CBI	3.00	(2.07)	(0.16)	(0.62) Plate Governs	(+10)	(65) 65	48 48	123	123	
Davis Besse 1 B&W/B&W	1.68	(1.11) No Axial Welds	(0.24)	(0.61)	(0)	(85)	59	144		
Haddam Neck W/CE	10.92	(14.30) 11.90	(0.22) 0.22	(0.10) 0.10	(-56) -56	(111) 106	59 59	114	109	
Kewaunee W/CE	5.87	(7.86) No Axial Welds	(0.20)	(0.77)	(-56)	(129)	59	132		
Farley 1 W/CE	2.19	(3.70) 0.83	(0.24) 0.27	(0.60) 0.60	(-56) -56	(117) 89	59 59	120	92	
Millstone 2	3.91	(2.19) No Data for Axial Welds	(0.37)	(0.06)	(-56)	(114)	59	117		
Prairie Island 2 W/SFAC	5.62	(7.53) No Axial Welds	(0.19)	(0.13)	(-56)	(81)	59	84		
Prairie Island 1 W/SFAC	5.90	(7.90) No Axial Welds	(0.14)	(0.17)	(-56)	(60)	59	63		

Rev 1

Footnotes

(1) Arranged in descending order of the RT_{NDT} as of December 31, 1981 considering circumferential to be 30°F less severe than axial orientations.

(2) Memorandum, M. Vagins to S. Hanauer, August 30, 1982.

(3) Values shown in parentheses on top line are for circumferential welds, bottom line is for axial welds. When plate governs--both lines.

(4) Determined by Reg. Guide 1.99, Rev. 1, Upper Limit Line, $\sigma_\Delta = 0$.

(5) σ_o (17°F) and σ_Δ (24°F) are the standard deviations of the initial RT_{NDT} and ΔRT_{NDT}, respectively. If plate or forging governed, actual initial RT_{NDT} was available and $\sigma_o = 0$

(6) The sum of the Mean Initial RT_{NDT}, the mean ΔRT_{NDT} and $2\sqrt{\sigma_o^2 + \sigma_\Delta^2}$, as of Dec. 31, 1981.

(7) Initial RT_{NDT} assumed by licensee to be -50°F and by CE to be -20°F. Values in parentheses are by CE.

(8) Fluence is per letter from CP&L Co., Sept. 24, 1982.

(9) Fluence reduced from 11.16 n/cm² per letter from FPL Aug. 31, 1982, on TP 4. TP 3 tentatively assumed to be the same as TP 4.

(10) Fluence reduced to 0.73 x peak per letter from Omaha PPD, Sept. 1, 1982.

(11) Cooper and Nickel values reduced per letter from Baltimore G&E, Oct. 28, 1982.

Enclosure B

VALUE-IMPACT ASSESSMENT

INTRODUCTION

The revised guidelines for preparing value-impact analyses to satisfy the requirements of Executive Order 12291 are stated in a memorandum dated May 7, 1982 (SECY-82-187) to the Commissioners from William J. Dircks. This Value-Impact Assessment is based on the staff's technical evaluation in Enclosure A and follows a simplified outline based on the revised guidelines.

Proposed Regulatory Actions

1. Establish an RT_{NDT} screening criterion of 270°F for axial welds and 300°F for circumferential welds.

2. Three years before the reactor vessel is predicted to reach the screening criterion, plant specific analyses must be started. These analyses will be used by the licensee and NRC on a plant specific basis to determine the regulatory action required to maintain risks from potential PTS events at an acceptably low level during continued plant operation. The analyses should consider: plant specific system features and possible improvement; identification of events which

contribute significantly to PTS risk; reliable estimate of the reactor vessel material properties; deterministic and probabilistic fracture mechanics evaluations; plant procedures and operator training improvements; inservice inspection for cracks at the vessel inner walls; flux reduction methods to slow further embrittlement; and detailed studies of the feasibility of in-place annealing of the pressure vessel.

3. Obtain information from licensees of a few plants for which near-term flux reduction would ensure that the screening criteria are not exceeded throughout plant life, to provide for early evaluation of the need, safety benefit, and cost of such programs.

Consequences of Action

There are no direct safety benefits for performing the plant specific analyses. The safety benefits will be derived from implementing any required corrective actions identified in these analyses. The plant specific analyses are a necessary step in this process. The operating experience record and probabilistic studies, as described in the Enclosure A, indicate that the potential safety benefits are large for initiating PTS corrective regulatory action for plants approaching the screening criterion. A Value-Impact Assessment pursuant to SECY-82-187, will be prepared after receipt of the plant specific analysis and the determination of the particular corrective regulatory action necessary and expedient for a given operating facility.

The major costs involved in the presently proposed action (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 to implement actions found to be necessary have not been estimated and will have to be determined and evaluated on a plant-specific basis. They could be large if system modifications or annealing are found to be needed.

Appendix P of Enclosure A provides RT_{NDT} data for forty pressurized water reactors. Based on our preliminary evaluation of the data using the proposed screening criteria, four reactors are estimated to reach the screening criterion within ten years. Nineteen plants are estimated to reach the screening criterion before end-of-life.

An estimate of the cost for performing the plant specific analyses was made for the staff by engineering consultants at Battelle Pacific Northwest Laboratories. The items considered in the cost estimates are identified below. The items were reviewed and collectively discussed with reactor specialists in each of the engineering areas. A judgment of cost was then made based upon an estimate of the time and facility requirement.

(1)	Identification and quantification of PTS events	$ 500K
(2)	Better identification of vessel material properties	70K
(3)	Deterministic and probabilistic fracture mechanics analyses	100K
(4)	Flux reduction program analyses	100K
(5)	Inservice inspections and nondestru ive evaluation study	75K
(6)	Plant modification study	200K
(7)	Operating procedures and training study	150K
(8)	In-situ annealing study	50K
	TOTAL	$1250K

RES experience with an ORNL/INEL study of PTS events similar to what will be needed in item 1 of the PNL estimate indicates that contractor costs for that item are about $800K plus significant support from the utility.

Incorporating uncertainties and contingency, the staff estimates a cost of approximately two million dollars per plant to perform the required plant-specific analyses. For the four PWRs which are expected to reach the screening criterion within ten years, this implies an industry cost of eight million dollars. The NRC guideline development and review evaluation costs are estimated to be about two million dollars. These costs are modest and appear well justified since the consequence of not performing the recommended analyses is that certain reactor vessels would become increasingly embrittled and the risk of pressure vessel failure would continue to increase.

ENCLOSURE C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 7, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

Subject: ACRS REPORT ON PRESSURIZED THERMAL SHOCK

During its 266th meeting, June 3-5, 1982, the Advisory Committee on Reactor Safeguards completed its review of the current status of the pressurized thermal shock problem (PTS). The NRC Staff is developing a regulation based on a combination of deterministic and probabilistic analyses to establish regulatory requirements concerning pressurized thermal shock. The ACRS has not been provided sufficient information to evaluate the adequacy of this approach.

The ACRS does not believe there is a need for any immediate plant modifications to permit continued operation of the plants which have been identified up to now as having potential PTS problems.

The most beneficial actions for these plants in the short term would be to:

1. Make certain that the metallurgical properties of the vessel beltlines are established adequately with respect to fracture toughness.

2. Determine which is the most effective in-service inspection capability for the beltline that current technology can provide. For those welds of principal concern, inspection should be accomplished, if practical, at the next refueling shutdown using such techniques, if such inspection has not previously been accomplished.

3. Provide effective operator training to avoid pressurized thermal shock and provide capability to diagnose events that could cause it.

4. Examine the depressurization capability for these plants and train operators when and how to use it.

5. Provide a demonstration of pressure vessel annealing to recover fracture toughness.

There are many intricacies associated with evaluation of pressurized thermal shock consequences that deserve attention, but the above actions would be the most effective contributors to assuring that pressurized thermal shock does not create public safety problems.

The ACRS plans to continue its review of pressurized thermal shock and the related NRC Staff program. The Committee will report further at an appropriate time.

Sincerely,

P. Shewmon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

ENCLOSURE D

October 14, 1982

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON DRAFT NRC STAFF EVALUATION OF PRESSURIZED THERMAL SHOCK DATED SEPTEMBER 13, 1982

During its 270th meeting, October 7-8, 1982, the Advisory Committee on Reactor Safeguards reviewed the Draft NRC Staff Evaluation of Pressurized Thermal Shock dated September 13, 1982. This matter was also considered at a Subcommittee meeting on September 30, 1982 in Washington, D.C. In its review the Committee had the benefit of discussions with representatives of the NRC Staff, the Westinghouse, Combustion Engineering, and Babcock and Wilcox Owners Groups, and the Southwest Research Institute. The Committee reported previously to you regarding this matter on June 7, 1982.

The NRC Staff is developing a regulation based on a combination of deterministic and probabilistic analyses to establish requirements concerning pressurized thermal shock (PTS). The NRC Staff proposes to use RT_{NDT} screening criteria for reactor vessels as the basis for further action concerning PTS. A value of 270°F for longitudinal welds and base material and a value of 300°F for circumferential welds have been selected. These proposed criteria are reasonable on the basis of current knowledge and provide adequate time for licensees to demonstrate plant-specific capability or planned actions in order to avoid unacceptable public safety consequences from PTS. For reactor vessels that are expected to be the earliest to exceed the screening criteria, we wish to be kept informed about PTS control actions under consideration.

The NRC Staff report indicates that several years are available before the first plant will exceed the screening criteria limits. This provides adequate time to conduct an orderly, comprehensive research program concerning measures needed to protect against pressurized thermal shock if a diligent effort to implement the program is applied. The NRC Staff has not yet defined a suitable program. We believe that the following should be elements of such a program: improved nondestructive examination capability; a more complete study of in situ reactor vessel annealing; improved fracture mechanics analysis methods that will account for realistic crack geometry, cladding effects, and crack arrest phenomena; use of three-dimensional and elastic-plastic techniques where appropriate; and potential improvements in diagnostic capability to help the operator recognize and control thermal shock events.

In accord with your desire to obtain our views on short-term steps regarding the NRC Staff's program on PTS, the recommendations in our letter of June 7, 1982 still apply. The following items deserve special attention:

- The reactor vessels with the greatest potential increase in RT_{NDT} are those having relatively high copper content. Only a small fraction of the available irradiation test data can be fully correlated with composition effects at high fluence levels. The correlations relating the rise in RT_{NDT} to metallurgical composition would benefit from a careful assessment of the uncertainties and special circumstances related to each data point used in the correlation.

- Improvements in PTS-related operator training and procedures should be carried out by all licensees with special emphasis on those plants indicated to have high RT_{NDT} vessels. Operational problems that need to be dealt with include the conflicting need to maintain adequate pressure for core cooling purposes while avoiding PTS, the control of feedwater and auxiliary feedwater to provide adequate heat removal while avoiding overcooling, and the recovery after a transient event which has caused violation of the cooldown rate limits.

- The range of initiating events and subsequent operator actions which are most likely to cause PTS need to be better characterized. A more extensive evaluation of operational events, including operator errors of commission as they apply to specific plant designs, will improve our understanding.

- The ACRS has considered the value of heating the ECCS water as a means of reducing public safety risk from PTS. Heating ECCS water may be helpful in the specific set of small break LOCAs that result in primary loop flow stagnation. If this case is an important contributor to the overall thermal shock risk and if heating the water does not unacceptably diminish containment integrity margins, then it can be a useful measure.

- Fast neutron fluence reduction is being implemented in some plants by using revised fuel management techniques. Further fluence reduction can be achieved by changing the core design. The value of such measures and the penalties involved must be determined for each plant individually.

The Committee believes that if due consideration is given to the items mentioned above, and if regulatory actions are based on the proposed screening criteria, the pressurized thermal shock matter should not present an undue risk to the health and safety of the public.

Additional comments by ACRS Member David Okrent are presented below.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

Additional comments by ACRS Member David Okrent

I generally support the ACRS recommendations in this report and have no problem with use of the proposed screening criteria on an interim basis. The comments which follow are made in no small part because of the generic implications to the regulatory process of how an issue like pressurized thermal shock is handled.

I believe it has been useful for the NRC Staff to attempt to examine the PTS issue probabilistically. The preliminary probabilistic studies reported thus far should be made more comprehensive, reported in depth, and subjected to extensive independent review.

In Section 8.4 of the September 13, 1982 Draft NRC Staff Evaluation of PTS, the NRC Staff compares its proposed PTS requirements to the Commission's, proposed policy statement on "Safety Goals for Nuclear Power Plants: A Discussion Paper," NUREG-0880.

If, for purposes of discussion, I accept the NRC Staff PRA results in Figure 8-3, as well as its statement that at the $270^{\circ}F$ screening criterion, RT_{NDT} is likely to have a mean value of $210^{\circ}F$ (RT_{NDT} of $270^{\circ}F$ representing a 2σ upper confidence bound), I have trouble agreeing with some significant statements made by the Staff in Section 8.4.

On page 8-5, the NRC Staff says the following:

> "The core melt Safety Goal guideline states, 'The likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of reactor operation.' This suggests that the core melt frequency ascribable to one sequence, for example PTS, should not exceed approximately 10^{-5} per reactor-year.

"Because of the unusually large uncertainty in the risk estimation for PTS, compared to other sequences, a value of less than 10^{-5} might well be assigned for a safety goal for PTS. We have not done this in the discussion in this section, but have used 10^{-5}. The reader should keep in mind that the risk numbers for PTS given in the following discussion are highly uncertain.

"We have no technical analysis of the course and consequences of a PTS sequence that involves RPV failure. Determination of the RPV failure mode (better, estimation of the probabilities of the various failure modes) has not been done and is dependent on the details of the scenario. Moreover, the outcome would likely be dependent also on the plant design details. In particular, ice condenser containments would be expected to have different failure modes, with different probabilities, than large dry containments."

I disagree with the use of 10^{-5} per reactor-year for at least two reasons. First, there are many more than ten potential initiators of large-scale core melt. Allocating 10^{-5} per reactor-year to a single initiator is questionable. It is even more questionable in view of the large uncertainty. Most importantly, until one knows with considerable confidence that a PTS failure has only a small likelihood of leading to early containment failure or otherwise leading to one of the large radioactivity release categories, the assignment of 10^{-5} per reactor-year (best estimate, with very large uncertainties) is probably unacceptable.

The NRC Staff states it has no technical analysis of the course and consequences of a PTS sequence that involves reactor pressure vessel failure. However, on page 8-6 the NRC Staff defines a quantity Y as the fraction of PTS-caused failure leading to core melt which leads to significant radioactive release. In the September 13, 1982 draft report, the NRC Staff used values of Y between 5×10^{-2} and 5×10^{-3} in suggesting that its proposed screening criterion is compatible with a PTS risk less than 10% of the proposed safety goal guidelines. In its oral presentation during the 270th ACRS meeting, the NRC Staff modified its approach such that a value of $Y = 8\times10^{-2}$ is compatible with meeting the safety goal.

The NRC Staff provides no basis for these values of Y, which are much less than unity. In the absence of any reasonable justification for the suggested range for Y, this aspect of the logic supporting the choice of the screening criterion becomes weak.

I recommend that, before the proposed screening criteria are adopted for the long term, the relevant information on containment behavior, given a PTS failure, be developed and included in an expanded probabilistic study which attempts to deal quantitatively with all contributions to uncertainty and states which uncertainties cannot be addressed meaningfully and why. The sensitivity studies reported in Appendix H are useful but do not take the place of a critical examination and evaluation of uncertainties, which the NRC Staff currently estimates are a factor of 100 in either direction.

The NRC Staff should then state a recommended position and include reasons for whatever approach is recommended in light of the uncertainties.

I might note the incongruity illustrated by the NRC Staff's comment on page 8-7 that "For scenarios involving core melt without significant releases, the core melt guideline will govern and ALARA is not a consideration." This conclusion by the NRC Staff may be compatible with NUREG-0880. However, I find it hard to believe that the Commission would not credit an appropriate benefit to some measure which significantly reduced the likelihood of pressure vessel failure, even if such failure were estimated to lead to core melt without significant release of radioactivity.

Finally, I should like to observe that PTS is a real issue in which the NRC Staff, the industry, and others are using probabilistic considerations coupled with ad hoc safety criteria as one input into engineering judgment. PTS is a significant issue which is subject to such large uncertainties that a plausible set of confidence bounds may well encompass a risk band which separates the clearly acceptable and clearly unacceptable areas and that these confidence bounds may also extend into those areas. Quite aside from any Commission action on safety goals, it seems to me important that the Commission take steps to assure that the probabilistic aspects of this issue are done as well as practical and that the appropriate review processes are established and accomplished. I also recommend that the Commission participate actively in establishing the criteria to be used on this issue for decision making under uncertainty. This includes the basis for action if and when flaws or indications of flaws in the size range of interest are found during forthcoming inspections of reactor vessels.

References:
1. Draft NRC Staff Evaluation of Pressurized Thermal Shock, dated September 13, 1982, including Appendices A-P, dated September 15, 1982.
2. Letter from Demetrios L. Basdekas, NRC to P. G. Shewmon, ACRS concerning comments on the September 13, 1982 Draft NRC Staff Evaluation of Pressurized Thermal Shock, dated October 6, 1982.

ENCLOSURE E

NOV 1 2 1982

MEMORANDUM FOR: P. S. Shewmon, Chairman
Advisory Committee on Reactor Safeguards

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: ACRS COMMENTS ON DRAFT NRC EVALUATION OF PRESSURIZED THERMAL SHOCK

Your letter of October 14, 1982, to Chairman Palladino provided the comments of the ACRS regarding the draft NRC Staff Evaluation of Pressurized Thermal Shock, dated September 13, 1982. The Committee indicated that the RT_{NDT} screening criteria proposed in the draft staff evaluation "are reasonable on the basis of current knowledge and provide adequate time for licensees to demonstrate plant-specific capability or planned actions in order to avoid unacceptable public safety consequences from PTS." As you know, the draft staff evaluation is currently under management review prior to the submittal of recommendations to the Commission. The ACRS comments will be addressed by the staff in that Commission paper.

You requested that the Committee be kept informed about PTS control actions under consideration for reactor vessels that are expected to be the earliest to exceed the screening criteria. At the ACRS meeting on November 5, 1982, the staff informed the Committee that further consideration is being given to the need for actions to assure the early implementation of flux reduction programs for those plants that are currently projected to exceed the screening criteria before the end of design life. The staff will continue to keep the ACRS informed of the progress of these deliberations.

The Committee indicated that there is adequate time to conduct an orderly, comprehensive research program concerning measures needed to protect against pressurized thermal shock and recommended some elements of such a program. The staff agrees and is developing a better defined program. We will arrange a subcommittee briefing on the research program early next year.

The Committee recommended that the staff's program on PTS give special attention to improvements in PTS-related operator training and procedures; better characterization of initiating events and subsequent operator actions; the value of heating the ECCS water; and plant-specific evaluations of the value and costs of fast neutron fluence reduction programs. The staff agrees and consideration of these items is part of the planned program.

The Committee also recommended an additional careful assessment of uncertainties in the irradiation test data used to develop correlations for

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prediction of RT_{NDT} increases with fluence for various materials compositions. The staff will include additional studies of RT_{NDT} shift correlations in the research program under development.

The staff is also considering the additional comments of ACRS member David Okrent in its preparation of the Commission paper on PTS.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Note: See previous concurrences.*
Revised to reflect 11/5/82 ACRS meeting.

PPAS*
10/28/82

OFFICE	GRO	DST	RES	NRR	NRR	EDO	
SURNAME	FSchroeder/jm	SHanauer*	GArlotto*	ECase*	HDenton*	WDircks	
DATE	10/27/82	10/27/82	10/26/82	10/28/82	10/29/82		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE F

OCT 20 1982

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: Victor Stello, Jr., Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NUMBER 21

The Committee to Review Generic Requirements met on Wednesday, October 6, 1982, from 1-6 p.m. A list of attendees is enclosed.

S. Hanauer (NRR) presented the technical background and the recommendations proposed by NRR to address the issue of Pressurized Thermal Shock (PTS). These recommendations were included in a draft report that is intended by NRR to form the basis for a Commission paper. The Committee noted that the staff did a thorough job in examining the various technical aspects of this complicated issue. The draft report reflects a good balance between deterministic engineering analyses and probabilistic risk assessment (PRA) techniques. While acknowledging the large uncertainties involved (the staff estimated as much as two orders of magnitude uncertainty in the vessel failure probability estimate) the Committee believes the PRA analysis is a valuable supplement to the deterministic analysis in arriving at a balanced engineering judgment on this issue.

NRR proposed screening criteria for the vessel reference temperature RT_{NDT}, a parameter that characterizes the state of embrittlement of reactor vessels. The Committee agreed that the proposed screening criteria (270°F for longitudinal welds and 300°F for circumferential welds) seem appropriate. NRR proposes that, whenever the value of RT_{NDT} for a given vessel is projected to exceed either of the screening criteria within the next 3 calendar years, the licensee would be required to submit a plant specific analysis, the scope of which has yet to be specified. NRR also proposes that a number of long term actions be required to ameliorate the PTS problem.

Demetrios Basdekas attended the CRGR meeting and summarized the comments in his memorandum of September 24, 1982 to Carl Johnson in RES. In his memo, Mr. Basdekas noted the short time available for him to offer comments on the NRR draft staff report and the fact that he "had not participated in PTS related activities...for quite some time..." Mr. Basdekas agreed with the thrust of CRGR discussions on the need for prompt decisions on plant modifications, such as low leakage fuel loadings. Mr. Basdekas expressed dissatisfaction with the screening criterion recommended by NRR but was unable to make specific recommendations for its alteration. Mr. Basdekas reiterated what he called his long standing concern that there is insufficient information available to him or to the NRC generally to properly address the safety implications of reactor control systems (unresolved safety issue A-47 which Mr. Basdekas has responsibility to support through his research tasks). He urged more attention to obtaining better control system data from representative licensees. It is his belief that the findings of A-47 may eventually have an important influence on the decisions on the PTS issue. He did not offer any specific examples of this influence nor did he hold out any hope that the A-47 input would be available in any reasonable time frame to support the short term PTS decisions. He also did not have any specific criticism concerning the proposition that the draft staff report takes into account the control system contributions to PTS event sequences that have occurred or were covered by the PRA.

The Committee finds Mr. Basdekas' arguments that PTS decisions should depend on the resolution of USI A-47 to be generally lacking in substance; i.e., we see no reason that A-47 cannot follow A-49. Furthermore, it would appear that requiring resolution of A-47 before deciding on PTS would be contrary to the desire for reaching an early resolution of PTS.

Mr. Basdekas brought to the Committee no new technical information or unique insight on PTS not otherwise available to the staff and already utilized in the development of the draft report on PTS. This is not to fault Mr. Basdekas - he was asked to comment on the staff report in a very short time period and he is not normally assigned to work in that area. The Committee feels it is counterproductive to efficient and effective staff work for NRC management to seek his reactions to staff proposals on PTS in this manner.

Mr. Sanford Israel also cautioned the CRGR that there could be more severe overcooling transients than considered in the PRA analysis. The Committee agreed with his observation and suggested that NRR continue to evaluate the probabilities and consequences of the full range of potential overcooling transients in their ongoing PTS work.

Based on the briefing by NRR and review of the extensive background material, the following recommendations are made:

1. The draft report should be modified to make clear that a rule change will be required to finally resolve the PTS issue.

2. It was noted that, because the pressure vessel embrittlement increases with irradiation exposure, the risk from PTS increases with time. In the absence of remedial actions, some PWR vessels are estimated to have RT_{NDT} well in excess of 300°F at the end of their service life. Since this indicates that some remedial action will be required for those vessels, the Committee requested that NRR develop further information on the costs and benefits of requiring near term flux reduction measures such as replacing outer row fuel assemblies with dummy assemblies. The CRGR stressed that this action, if implemented, was needed not because the PTS risk is unacceptably high at this time, but because the passage of time forecloses the flux reduction option as an effective remedy.

3. The Committee agreed that improved operator training and emergency operating procedures are needed. However, it was emphasized that these improvements must be done in an integrated manner and must not deemphasize the importance of maintaining adequate core cooling in the event of a loss-of-coolant accident by an over emphasis of the PTS issue.

4. The Committee suggested that NRR add a short term task to investigate whether the frequency of overcooling transients for B&W plants may be higher than the average, based on operating experience to date.

The CRGR will continue its review of the PTS issue on October 28, 1982, at which time NRR will present information on the costs and benefits of requiring near term flux reduction measures.

Victor Stello, Jr., Chairman
Committee to Review Generic Requirements

Enclosure: List of Attendees

cc: CRGR Members
Commission (5)
Office Directors
Regional Administrators
G. Cunningham
S. Hanauer

CRGR MEETING #21

LIST OF ATTENDEES

October 6, 1982

CRGR MEMBERS

Vic Stello
R. Bernero
Roger Mattson (for D. Eisenhut)
Ed Jordan
Joe Scinto

OTHERS

Tom Murley
Walt Schwink
Steve Hanauer
Roy Woods
Mat Taylor
L. F. Litton
Karl Kniel
Carl Johnson
Frank Schroeder
Harry Boulden
Richard Donovan
James Popelarski
Demetrois Basdekas
E. D. Throm
Alan Rubin
Bill Shields
Jack Strosnider
Larry Shao
Jesse Funches
Steve Stern
Sanford Israel
Norm Lauber
Neil Randall
Ray Klecker
Hugh Thompson
Lambros Lois
Jim Clifford
Ed Abbott
Tom Dorian
John Austin
Milt Vagins

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 26, 1982

MEMORANDUM FOR: Victor Stello, Jr.

FROM: Joseph F. Scinto

SUBJECT: COMMENTS ON MINUTES OF CRGR MEETING NUMBER 21

On page 1 of the Minutes, dated October 20, 1982, I would note that the way in which RTNDT for the screening criterion is to be computed (by including a 2 sigma value to the mean values, p.5-5 of the draft) makes the 270°F screening criterion more protective than figure 8.3 in the draft report might suggest. That figure provides longitudinal crack extension frequencies for mean surface RTNDT, rather than for RTNDT computed as mean plus 2 sigma.

Joseph F. Scinto

cc: R. Bernero
Ed Jordan
R. Cunningham
C. Heltemes
D. Eisenhut

CRGR MEETING #23
LIST OF ATTENDEES
October 28, 1982

CRGR MEMBERS

Vic Stello
Ed Jordan
Joe Scinto
Jack Heltemes
Bob Bernero
Bob Purple (For Darrell Eisenhut)

OTHERS

Tom Murley
Steve Hanauer
Walt Schwink
Lambros Lois
Guy Vissing
Norm Lauben
Les Rubenstein
Ed Case
Steve Stern
Bill Shields
Ed Throm
Ray Klecker
Neil Randall
Rainer Rantala
Jack Strosnider
Frank Schroeder
Roy Woods
Jim Milhoan
Tom Cox
Mat Taylor
Ed Abbott
Felix Litton

ENCLOSURE G

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 12 1982

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: Victor Stello, Jr., Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NUMBER 23

The Committee to Review Generic Requirements met on Thursday, October 28, 1982, from 1-4 p.m. A list of attendees is enclosed.

S. Hanauer (NRR) presented further technical information on the issue of Pressurized Thermal Shock (PTS) in response to questions raised at CRGR Meeting No. 21.

Material was presented which disaggregated the overall PWR operating experience according to reactor manufacturer. This analysis of operating experience suggested that the frequency of overcooling transients for B&W plants may be higher than that for Westinghouse and Combustion Engineering plants. However, because the sparcity of data leads to large uncertainty bands, the Committee felt that there was not a sound basis for establishing a different value of the RT_{NDT} screening criterion for B&W plants. The Committee agreed with NRR that more detailed analysis of B&W plants will be required.

NRR presented further analyses of small break loss-of-coolant accidents (SBLOCA), which the staff's PRA results had shown to be the dominant risk sequence for Westinghouse plants. If, for example, the small break LOCA were to occur in a location where the break were isolatable by operator action to close a valve, then the threat to the vessel would be greater due to (a) repressurization to full system pressure and (b) no credit could be taken for the ameliorating effect of warm prestress. On the other hand, recent data from ECC mixing tests show that there is a better mixing of cold ECC water with hot water in the cold leg pipes and vessel downcomer and, as a result, the vessel would not cool down as fast during a SBLOCA than previous analyses had indicated. The net result of all these factors is that the conditional vessel failure probability for an isolatable SBLOCA would be increased by a factor of 10 over earlier estimates for an nonisolatable SBLOCA. NRR did not have an estimate for the relative probabilities of isolatable vs. nonisolatable SBLOCAs, but in light of the large uncertainties in the overall risk analyses (the staff estimated at least a factor of 100 uncertainty) the Committee did not believe this new information would significantly alter the engineering judgments on this issue. The Committee recommended that NRR continue to evaluate probabilities and consequences of the full range of potential overcooling transients for each reactor manufacturer in their ongoing PTS work.

At Meeting No. 21, the CRGR had noted that, because the pressure vessel embrittlement increases with irradiation exposure, the risk from PTS increases with time. In the absence of remedial actions, some PWR vessels were estimated to have RT_{NDT} well in excess of 300°F at the end of their service life. Since this indicates that some remedial action will be required for those vessels, and since the passage of time reduces the effectiveness of flux reduction as a remedy, the Committee had asked NRR to develop further information on the costs and benefits of near term flux reduction measures.

NRR presented information which showed that there are some plants which will need no remedial actions for their vessel RT_{NDT} to remain below the screening criteria throughout their service life, based on current information. The remaining plants can generally be grouped into three categories.

(1) There are several plants for which near term action to reduce the flux at critical welds by a factor of 2 or less will ensure that they do not exceed the screening criteria throughout their service life. Information available to NRC from reactor manufacturers indicates that a flux reduction factor of 2 can be attained through installation of a low leakage core, which is simply the installation of partially burned fuel assemblies in the periphery of the core in place of fresh fuel assemblies. This fuel management option is already being implemented by some licensees at reportedly little or no additional cost.

(2) There is a group of about nine plants for which near term action to reduce the flux at critical welds by factors of 2 to 4.5 will ensure that they do not exceed the screening criteria throughout their service life. NRR presented analyses which showed that these flux reduction factors can be attained through the installation of a low leakage core and the replacement of some peripheral fuel assemblies (estimated 4-12) by dummy assemblies. There would be an estimated engineering cost of $20 million per plant, but no substantial increase in fuel cycle costs or operating costs.

(3) There is one plant, H. B. Robinson 2, which is close to reaching one of the screening criteria and for which the fuel management options described above could not reduce the flux at the critical welds enough to prevent reaching the screening criteria. It was suggested at the meeting that there were no practical fuel management schemes or operating modes that would prevent Robinson 2 from reaching the screening criteria within the next few years, and this was confirmed by NRR. It is possible, however, to reduce the flux at the critical weld by a factor of about 9 by replacing the entire outer row fuel assemblies with dummy assemblies. This option would entail an engineering cost of $20 million and probably additional operating costs due to the need to derate the power level of the plant.

The Committee discussed the technical options open to NRC to resolve the pressurized thermal shock issue.

(a) NRC could continue to refine the analyses, including plant specific risk analyses, in the expectation that they would show that the plants could meet their service life without remedial actions. The Committee felt that this option would be costly and not likely to produce convincing arguments for no remedial actions.

(b) NRC could require near term actions to reduce the flux levels for those plants where such actions would ensure that the vessel RT_{NDT} would remain below the screening criteria throughout the service life.

(c) NRC could establish a regulatory limit on vessel embrittlement and permit plant operation until that limit is reached, at which time the vessel would have to be thermally annealed or the plant shut down. The Committee noted that vessel annealing appears to be technically feasible, although unproven on a large scale, and costly in terms of engineering, plant down time and occupational exposure.

The initial NRR proposal would establish screening criteria for the vessel reference temperature, RT_{NDT}, for critical welds. Whenever the value of RT_{NDT} for a given vessel would be projected to exceed the screening criteria within the following 3 calendar years, the licensee would be required to submit a plant specific analysis. The staff would develop acceptance criteria for determining whether plant modifications would be required after the staff review of the plant specific analyses. The initial NRR proposal also included a number of long term actions intended to ameliorate the PTS problem.

Discussions with the Committee made clear that for those plants that the staff estimates are currently near the screening criteria, the two-step process above would result in delays which could foreclose options that currently appear to be feasible at little additional cost, particularly flux reduction options. Such delays could mean that, at the point of decision, there may be few if any alternatives to annealing the vessel.

For those reasons, NRR proposed and the Committee agreed that the staff should take steps to initiate flux reductions for those plants in categories (1) and (2) above to ensure that RT_{NDT} for critical welds does not reach the screening criteria before the end of service life. The Committee believes that the PTS risk is not unacceptably high at this time, but by taking these relatively low cost actions in the near term, the PTS risk can be maintained at acceptable levels for these plants and the need for requiring costly and unproven actions in the future would thereby be obviated for these plants. NRR staff indicated agreement with this course of action at the meeting.

Although various methods for initiating flux reduction requirements were discussed, including rulemaking, generic letters and orders, the Committee concluded that selection of the appropriate procedural method should be left to the staff.

For the case of H. B. Robinson 2, the Committee recommends that the staff promptly have the licensee submit a plan showing what actions they intend to take to resolve the PTS issue for their plant. It is expected that this plan could include consideration of heating ECC water, safety systems to prevent repressurization, flux reduction methods, annealing, or some combination thereof, but the Committee judges that some remedial actions will be needed in the next few years to ensure that the PTS risk for Robinson 2 remains within acceptable levels throughout its service life.

The CRGR reemphasized that staff actions to improve operator training and emergency operating procedures must be done in an integrated manner and must not deemphasize the importance of maintaining adequate core cooling in the event of a loss-of-coolant accident by an over emphasis of the PTS issue in the training and emergency procedures.

In summary, the Committee recommends that NRR develop a program to inform licensees of the need to modify plant operations through flux reduction programs to ensure that RT_{NDT} values do not reach unacceptable levels. For plants in categories (1) and (2) above, flux reduction appears adequate. In one plant, H. B. Robinson 2, a more comprehensive plan is needed.

Victor Stello, Jr., Chairman
Committee to Review Generic Requirements

Enclosure: List of Attendees

cc: Commission (5)
Office Directors
Regional Administrators
CRGR Members
G. Cunningham

Secy-82-465

ENCLOSURE A

NRC STAFF EVALUATION

OF

PRESSURIZED THERMAL SHOCK

November 1982

CONTENTS

CONTENTS (Continued)

CONTENTS (Continued)

LIST OF APPENDICES

TABLE OF CONTENTS OF APPENDICES

Table of Contents of Appendices
(continued)

LIST OF FIGURES

LIST OF TABLES

1. INTRODUCTION

1.1 Background

Reactor pressure vessels (RPVs) in nuclear power plants have traditionally been considered extremely reliable structural components. Indeed, studies completed in the United States and Europe have concluded that the disruptive failure rate (loss of the pressure retaining boundary) for nuclear pressure vessels is less than 10^{-6} per year at a 99% confidence level for RPVs designed, fabricated, inspected, and operated in accordance with the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers. However, recent results from surveillance and research programs and operating experience suggest that the issue of RPV failure probability should be reassessed. The renewed interest in RPV failure probability is due to the observation that thermal hydraulic transients occurring in commercially operating nuclear power plants are subjecting RPVs to unanticipated loadings which could contribute significantly to the failure probability of the RPV. In addition, operating experience and research programs over the past few years have provided additional information that more clearly defines both material property variations in RPVs and the effect of neutron irradiation on the material's resistance to fracture.

As a result of operating experience, it is now recognized that transients can occur in pressurized water reactors (PWRs) characterized by severe overcooling causing thermal shock to the vessel, concurrent with or followed by repressurization (that is, pressurized thermal shock, PTS). In these PTS transients, rapid cooling of the reactor vessel internal surface causes a temperature distribution across the reactor vessel wall. This temperature distribution results in thermal stress with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature profile across the reactor vessel wall. The effects of this thermal stress are compounded by pressure stresses if the vessel is pressurized.

Severe reactor system overcooling events which could be accompanied by pressurization or repressurization of the reactor vessel (PTS events) can result from a variety of causes. These include instrumentation and control system malfunctions including stuck open valves in either the primary or secondary system, and postulated accidents such as small break loss-of-coolant accidents (LOCAs), main steam line breaks (MSLBs), and feedwater pipe breaks. As long as the fracture resistance of the reactor vessel material remains relatively high, such events are not expected to cause failure. After the fracture toughness of the vessel is reduced by neutron irradiation (and this occurs at a faster rate in vessels fabricated of materials which are relatively sensitive to neutron irradiation damage), severe PTS events could cause propagation of fairly small flaws that are postulated to exist near the inner surface. The assumed initial flaw might initiate and propagate into a crack through the vessel wall of sufficient extent to threaten vessel integrity and, therefore, core cooling capability.

The PTS issue is a concern only for operating PWRs. Boiling water reactors (BWRs) are not a significant PTS concern. BWRs operate with a large portion of water inventory inside the pressure vessel at saturated conditions. Any sudden cooling will condense steam and result in a pressure decrease, so simultaneous creation of high pressure and low temperature is improbable. Also contributing to the lack of PTS concerns for BWRs is the lower fluence at the vessel inner wall, and the use of a thinner vessel wall which results in a lower stress intensity for a postulated crack.

The PTS issue is being handled separately from the "cold overpressurization" issue that was investigated (and for which corrective actions were required) several years ago. The "cold overpressurization" problem and corrective actions are described in Reference 1.1. "Cold overpressurization" refers to events such as isolation of letdown flow and/or inadvertent starting of a high pressure pump, while the reactor is in cold shutdown condition. No thermal stresses are involved, as in PTS, because the plant is uniformly cold. All of the stresses result from the pressure, which alone can cause vessel rupture (if a flaw is present) due to the much colder temperature present during shutdown (compared to PTS events where the vessel is warmer).

Corrective actions were applied, for cold shutdown conditions, involving disabling of pumps that could cause overpressurization, changing the setpoint of the pressurizer pressure relief valve, and incorporating the shutdown heat removal system's relief valves into the system.

Since the condition leading to "cold overpressurization" events, and the corrective actions for such events, are different from the PTS concern, "cold overpressurization" was handled separately and is not discussed further in this report.

1.2 Staff Reviews of PTS Information Provided by Licensees and Industry

Evaluations of Pressurized Thermal Shock by the NRC staff in the spring of 1981 concluded that no immediate licensing actions were required at that time, but that since the consequences of overcooling events increase as the vessels accumulate additional neutron irradiation, extensive further investigations were needed to determine whether and when corrective actions will be needed to provide assurance of vessel integrity throughout the intended service life of a reactor vessel.

On March 31, 1981, the NRC staff held the first of many meetings that were to occur over the following 16 months with licensees, reactor manufacturers, and owners groups to discuss pressurized thermal shock concerns and exchange technical information.

Subsequently, the NRC, in letters dated August 21, 1981, requested the licensees of eight plants representative of older reactor vessels to provide more detailed information on the present and projected pressure vessel materials properties, on the probability and possible severity of events that could cause failure of embrittled vessels, and on the efficacy and feasibility of several potential corrective actions.

Many of the event-sequence analyses provided by licensees in response to the August 21, 1981 letter can be characterized as design-basis event analyses of the type generally submitted in Safety Analysis Reports (SARs) in support of license applications. Such analyses tend not to be of much help in evaluations of PTS. Many of the assumptions in such analyses were developed and accepted for licensing purposes without regard to PTS concerns. While SAR analyses appear to be appropriately conservative for calculations of reactor core thermal performance, PTS evaluations need best estimate calculations of pressure and temperature behavior. In addition, some potential event sequences that are not generally analyzed in detail in Safety Analysis Reports, because their consequences are bounded by the design-basis event analyses, can be of greater significance for PTS evaluations. Thus, it is clear that plant-specific PTS evaluations must include a systematic examination of many potential events, with particular attention to the probability and consequences of various possible operator actions and omissions, and equipment malfunctions.

Appendix A to this report summarizes the meetings that have been held with industry, licensee responses to the August 21 letters, and the NRC staff audits of operating procedures, operator qualifications, and training with respect to the PTS issue. Appendix B lists significant events and meetings concerning PTS. Appendix C is a more detailed discussion of the procedures and training audits.

As a result of the review of the extensive information provided by the industry, and of studies and analyses performed by the staff, assisted by contractors and consultants (see particularly the fracture mechanics calculations performed by the Oak Ridge National Laboratory described in Appendix O, and the report of a technical review of PTS issues performed by Pacific Northwest Laboratory, Reference 1.2), in the spring of 1982, the staff reaffirmed its previous assessment that no immediate plant modifications were needed to protect against PTS events (other than improvements in procedures and operator training already underway, and flux reduction programs to reduce the rate of vessel embrittlement). However, the staff concludes that some plants will require hardware and procedural modifications in the near future. The experience of the past 18 months in generic evaluations of the PTS concerns has made it clear that decisions on the need for, nature of, and timing of, such modifications will require plant-specific, rather than generic evaluations.

1.3 Procedure and Training Reviews

Operators have a significant impact on the course of events following an initiating transient or accident, as shown by operating experience throughout the nuclear power industry. In most cases, operator actions have led to the mitigation of severe consequences that could have resulted from the initial event. There have been cases, however, where the operator has taken an action that was improper for the condition he was facing, which has led in some of those cases to severe consequences to plant components. Improper actions by the operators could have been the result of operating procedures with incorrect technical information for the specific event in progress, cognitive errors, operative errors, a lack of understanding of the fundamental concepts of plant design, or a myriad of other reasons that affect the range of human capabilities. Regardless of the reason, it is important that the operator be given information that is technically correct and useable, and that this information and the operator's concept of the plant be consistent.

Audits of procedures and training at seven of the plants used for the short-term evaluation of PTS revealed that the mitigative capabilities of the industry for dealing with PTS were highly inconsistent, and that the knowledge gained from studies of the thermal-hydraulic phenomena and materials properties must be more closely tied to its application in plant operations. The mitigative capabilities were evaluated generally on the basis of the actions specified in plant operating procedures, with the evaluation of operator understanding of both PTS and mitigative actions used to determine if the procedural actions were understood, as well as to determine if the operators understood thermal-hydraulic phenomena and material properties well enough to overcome the procedural deficiencies suspected by the audit teams. The findings of each of these audits are included in Appendix C.

A review of the event-specific Westinghouse Owners' Group generic technical guidelines by Westinghouse resulted in a recommendation that 11 modifications be made to the guidelines. These changes were recommended based on a reevaluation of the basis for each step in light of the recent knowledge gained for PTS. The recommendations included strengthening cautions to maintain pressure-temperature limits, and lowering the target pressure for stabilizing plant conditions during the recovery phases to a pressure consistent with the Technical Specification limits.

Upgrading of operator's ability to deal with PTS is contingent upon having sound technical knowledge of the phenomena that are involved, a clear understanding of how the phenomena relate to equipment operation or malfunction, a program to translate the technical information into operational information, and a program for presenting the information to the operators in a coherent manner. This information must be integrated with other technical information to ensure that all relevant technical concerns are considered in the actions that the operator is instructed to take.

The only effective way of ensuring this integrated approach to accident mitigation is to provide the operator with procedures that are developed using an integrated analytical base, and to provide in the procedures training program the background information necessary for the operator to have confidence in the procedures, and thus make them useable. The approach the staff is taking is to have the industry use technical guidelines being developed for TMI Action Plan Item I.C.1 as a basis for integrating procedural actions to be used for dealing with PTS. By using the procedures developed from those guidelines, the industry will gain a consistent and improved ability for dealing with PTS. An additional benefit to this approach is that by having the analytic base documented, knowledge of PTS gained from the long-term program can be effectively integrated into the existing analytic base.

1.4 Proposed Approach for Future Evaluations

For the reasons noted above, there is a need for a disciplined technical basis to select plants for which detailed evaluations are required and to determine the timing of such evaluations. The approach proposed by the staff is to select a screening criterion that characterizes the present or projected state of embrittlement of reactor vessels as a function of neutron fluence. Licensees of plants with vessels that are projected to reach the screening criterion within 3 calendar years would be required to submit detailed, plant-specific evaluations of: the vessel condition; the expected frequency, course, and consequences of experienced and postulated overcooling events; plant procedures and operator training related to prevention or mitigation of PTS events; possible modifications of plant equipment, systems, and procedures that could reduce the probability and/or severity of overcooling events; possible improvements in in-service inspection methods that could provide increased assurance of the detection of existing flaws in critical regions of the pressure vessel; and possible modifications to decrease the rate of vessel embrittlement or actions to recover ductility.

These licensees would also be required to provide a technical basis for the acceptability of continued operation of the plant for the remainder of its design life taking into account the risk of pressure vessel failure from pressurized thermal shock events, based on the above plant-specific evaluations and such remedial actions as are proposed.

The screening criterion proposed by the staff is based on a parameter that characterizes the toughness state of the reactor vessel beltline region. This parameter is the reference temperature for nil ductility transition, RT_{NDT}. At normal operating temperatures, vessel materials are quite tough and resistant to crack propagation. As the temperature decreases, however, the metal gradually loses toughness over a temperature range of about 100°F. RT_{NDT} is used to identify where this toughness transition occurs. Its value depends on the specific material in the vessel wall and it increases with neutron irradiation. These effects are determined by destructive tests of material specimens.

The RT_{NDT} used for screening purposes is the value at the inner surface of the vessel wall. Fracture mechanics analyses, however, take into account the tougher material within the wall due to attenuation of neutron fluence and generally higher temperatures during a cooldown transient. Equations, based on tests of irradiated specimens, have been developed to calculate the shift in RT_{NDT} as a function of neutron fluence for various material compositions. The value of RT_{NDT} at a given time in a vessel's life is used in fracture mechanics calculations to determine whether assumed pre-existing flaws would propagate as cracks when the vessel is subjected to overcooling events.

The staff's approach to selection of an RT_{NDT} screening criterion has been to consider the overcooling events that have occurred in U.S. PWRs and, using a deterministic fracture mechanics algorithm, calculate the value of RT_{NDT} for which assumed pre-existing flaws in the reactor vessel would be predicted to initiate (grow deeper into the vessel wall). These "critical" values of RT_{NDT} were related to the expected frequency of the initiating events, based on the limited data base (only eight events in 350 reactor-years). Specific values of RT_{NDT} are selected for use as the screening criterion for axial and circumferential flaws.

In addition, the staff considered a wide spectrum of postulated overcooling events that have not occurred. These events were grouped into types, estimates were made of their expected frequency, and stylized characterizations of the temperature and pressure time-histories were developed for each event type. A probabilistic treatment of the fracture mechanics calculations was developed that permitted performance of studies to gain insights into the sensitivity of the fracture mechanics calculations to uncertainties in the various input parameters. By combining the calculated frequencies and characteristics of postulated events with the probabilistic fracture mechanics results, some very approximate estimates of the probability of vessel failure resulting from PTS eents were developed and used by the staff to provide some insight into the residual risks inherent in use of the screening criterion approach for further evaluations and resolution of the issue of pressurized thermal shock.

1.5 Structure of this Report

This report provides the NRC staff's technical basis for the selection of the screening criterion, and a brief description of the type of plant-specific analyses that would be required for plants with pressure vessels that are projected to exceed the criterion.

Section 2 of the report discusses the frequency and characterization of overcooling events that have actually been experienced. Section 3 summarizes deterministic fracture mechanics calculations performed for these experienced events and parametric studies of crack growth potential as a function of the event characteristics and RT_{NDT} values. Section 4 combines the results of Sections 2 and 3 and proposes values of RT_{NDT} for use as a screening criterion.

Section 5 presents the staff's proposed method for estimation of vessel-specific values of RT_{NDT} for comparison with the screening criterion.

Section 6 describes an evaluation of the frequency and character of potential lower frequency overcooling events.

Section 7 summarizes sensitivity studies performed using a probabilistic treatment of the fracture mechanics calculations that can be used in combination with the results of Section 6 to estimate probabilities of vessel failure. Consideration of these results is presented in Section 8.

Section 9 indicates the nature and timing of the plant-specific evaluations that would be requested for plants approaching the screening criterion.

Section 10 presents the conclusions and recommendations of the NRC staff regarding near-term actions and future programs for resolution of the pressurized thermal shock issue.

References

1.1 "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," NUREG-0224, September 1978.

1.2a) L. T. Pedersen, W. J. Apley, S. H. Bian, L. J. Defferding, M. H. Morgenstern, P. J. Pelto, E. P. Simonen, D. L. Stevens and T. T. Taylor, Pacific Northwest Laboratory "PNL Technical Review of Pressurized Thermal Shock Issues," USNRC Report, NUREG/CR-2837, Jul 1982

b) Supplement 1 to above, September 1982

c) L. T. Pedersen, PNL, letter to R. Woods, NRC, regarding the September 13, 1982 Draft NRC PTS report, October 7, 1982

d) L. T. Pedersen, PNL, letter to F. Litton, NRC, regarding issues reviewed for reference 1.2b), October 22, 1982

e) Later revised version of Supplement 1, November 5, 1982

2. FREQUENCY AND CHARACTERIZATION OF EXPERIENCED OVERCOOLING EVENTS

2.0 Preface

The data provided in this section were compiled by the staff as an attempt to quantify overcooling events that occurred at commercial U.S. PWRs which appeared to be of concern for PTS.

In reviewing the available data, a conscious decision was made to use the data in its elemental or raw form, to the extent deemed reasonable. Temperature measurements, without consideration of instrumentation errors, taken in the reactor coolant system cold leg upstream of the safety injection location are, in general, used to characterize an event.

It is recognized that use of the data in this manner may be somewhat nonconservative as the temperature of concern is that at the coldest location in the vessel downcomer. The actual downcomer temperature is dependent on a number of parameters which are not available, at least to the degree required, to quantify the actual temperature. These time-dependent parameters are: (1) cold leg fluid temperature upstream of the safety injection location, (2) cold leg loop flow rate, (3) system pressure, (4) safety injection fluid temperature, (5) safety injection flow rate, (6) heat addition from metal walls, (7) vent valve flow rate for B&W plants, (8) vent valve fluid temperature for B&W plants, and (9) local flow rates in the vessel downcomer.

Thermal-hydraulic models are just now beginning to be developed to the depth necessary to obtain the information required to determine these parameters and obtain a downcomer temperature profile. Current efforts by both industry and the NRC are also, just now, obtaining the experimental data required to verify the new analytical models needed to fully understand these events.

The presentation of an event as a PTS event, in this section, is speculative. The data used address the question "What if?" this pressure-temperature history actually occurred at the most limiting location.

We do not, at this time, have the necessary analyses to adjust the cold leg measurements to the downcomer temperature for any given event. We recognize that use of cold leg temperature measurements in this way is subject to criticism and cannot be completely justified. However, we believe these data are representative of operational occurrences and define an appropriate data set for insight into the PTS issue.

2.1 Introduction

This section of the report describes the staff's review of eight actual overcooling transients of interest as potential PTS initiators. The event descriptions were reviewed and plots of pressure and temperature as functions of time were developed, based on plant data available. These actual pressure and time histories were used, as described in Section 3, in deterministic fracture mechanics calculations for each event to determine critical RT_{NDT} values.

In addition, the actual temperature versus time data for each event were fit to a simple stylized characterization of the temperature transients that could be used conveniently in parametric fracture mechanics studies. For this purpose, the fluid temperature at the reactor pressure vessel inner surface is assumed to decrease exponentially from the initial temperature. The equation used is:

$$T = T_0 - (T_0 - T_f)\ (1 - \exp(-\beta t))$$

Where T_0 = initial temperature, °F
T_f = final temperature, °F
β = cooldown parameter, min^{-1}
t = time, min

For each of the operating experience events, the actual event sequences were reviewed and values of T_f and β were selected to characterize the event. The selection of T_f and β required some engineering judgment. In general, the final temperature is selected to characterize the observed value when a temperature plateau is reached. The cooldown temperature profile is the best fit of the cooldown or is adjusted somewhat for cases where the temperature increases following termination of the uncontrolled cooldown. The adjusted cooldown used is based on the Westinghouse approach, which considered the fracture mechanics response to the actual temperature transients and the fracture mechanics response to the stylized formulation with an adjusted β value. The adjusted β* is obtained from

$$\beta^* \quad 2/t^*$$

where t* = the time of lowest temperature,
and
β* is never less than the "natural" cooldown β.

A representative constant value of the pressure was also selected for each event. These stylized representations of the experienced events were then used for comparison with parametric fracture mechanics calculations as described in Section 3.

Finally, the eight events of interest were used to construct a cumulative frequency distribution of observed events as a function of T_f which is considered in Section 4 in selecting a screening criterion.

The eight events represent very limited statistical data, with five events on Westinghouse plants, three on Babcock and Wilcox plants, and none on Combustion Engineering plants. The data represent the best information available, however, and were therefore used to define generic screening criteria. The same screening criteria were recommended for all three vendors' plants because of the limited statistical significance of indicated differences between the three groups and (in CE's case) because of their plants' similarity to Westinghouse plants, where data is available. Use of the same screening criteria for B&W plants needs further near-term justification, as discussed in Section 4. In addition, if there are real differences in PTS risk between the three vendors plants, those differences will be identified in the plant-specific analyses that are required when a plant "trips" the screening criteria, and account will be taken of those differences before corrective actions resulting from those analyses are required. In the immediate future, however, we recommend use of the same screening criteria for all PWRs.

2.2 Event Descriptions

2.2.1 H. B. Robinson Steam Line Break (04/28/70)

On April 28, 1970, during hot functional testing (no fuel loaded), one of the steam generator safety valve connections failed due to overloading. A 360° circumferential break allowed the safety valve to blow off the main steam line. The plant conditions were:

- 533°F, 2225 psi primary
- 900 psi secondary
- 3 reactor coolant pumps (RCPs) running
- 45 gpm charging/letdown
- no feedwater to the steam generator

As a result of the 6-in. schedule 80 pipe break, and with no decay heat, the plant cooled down 213°F in ≅½ hour to a 320°F cold leg temperature. The operator immediately tripped the RCPs (30 seconds) and started the remaining two coolant charging pumps (70 seconds). The minimum primary system pressure was 1880 psi; with the safety injection (SI) setpoint at 1715 psi, no safety injection occurred. The plant was recovered to a normal no-load condition of 2050 psig and charging/letdown reestablished prior to shutdown.

A post-event review of the data indicated that the pressurizer surge line did not empty. An analysis was performed for the event. In addition, a sensitivity analysis was performed without RCP trip, with only one charging pump, and with a primary heat source. The analysis showed that the pressurizer would drain and the primary system pressure would fall below the SI setpoint in about 3 minutes. The cooldown was less and the pressures were lower than the base case analysis. It is expected that the operator actions, based on current procedures, would be similar to this sensitivity analysis. The safety valve stand-off piping was redesigned to prevent any similar occurrences.

The transient data for this event are provided in Figure 2-1. For the stylized characterization of the event the staff selected T_f = 295°F, β = 0.08 min^{-1} and a pressure of 2000 psig. This exponential temperature curve is compared with the broken loop cold leg temperature data in Figure 2-2.

2.2.2 H. B. Robinson Stuck Steam Generator Relief Valve (11/05/72)

While at nominal full power operation conditions, the operator was using steam generator relief valves to provide reactor coolant system (RCS) temperature control. One valve would not reclose, resulting in the equivalent of a small steam line break. The secondary side blowdown resulted in a reactor trip and safety injection. The overall cooldown rate was 200°F over a 3-hour period, to 340°F during the course of the event. Insufficient information is currently available to address operator actions taken during this event.

The transient data for this event are provided in Figure 2-3. For the stylized characterization of the event the staff selected T_f = 340°F, β = 0.015 min^{-1} and a pressure of 1000 psig. The exponential temperature curve is compared with the cold leg temperature data in Figure 2-4.

2.2.3 H. B. Robinson RCP Seal SBLOCA (05/01/75)

During full power operation, RCP "C" seal number one leakage exceeded the technical specification limit of 6 gpm. A load reduction was commenced at a rate of 10% per minute to 36% power and pump "C" was deenergized. Reactor trip occurred due to a turbine trip resulting from the load reduction. The decision was made to restart pump "C" when seal injection could not be restored to pumps "A" and "B." Shortly after restarting the pump, while at 1700 psig and 480°F, seals number two and three failed on pump "C" and the pressurizer level began to decrease.

Safety injection pumps were manually started, charging flow was diverted to the auxiliary pressurizer spray to reduce pressure and the SI accumulators partially injected when the pressure dropped to 500 psig.

The cooldown for this event was from 450°F to approximately 310°F in one-half hour, with the pressure decreasing from 1700 psig to about 1150 psig over the

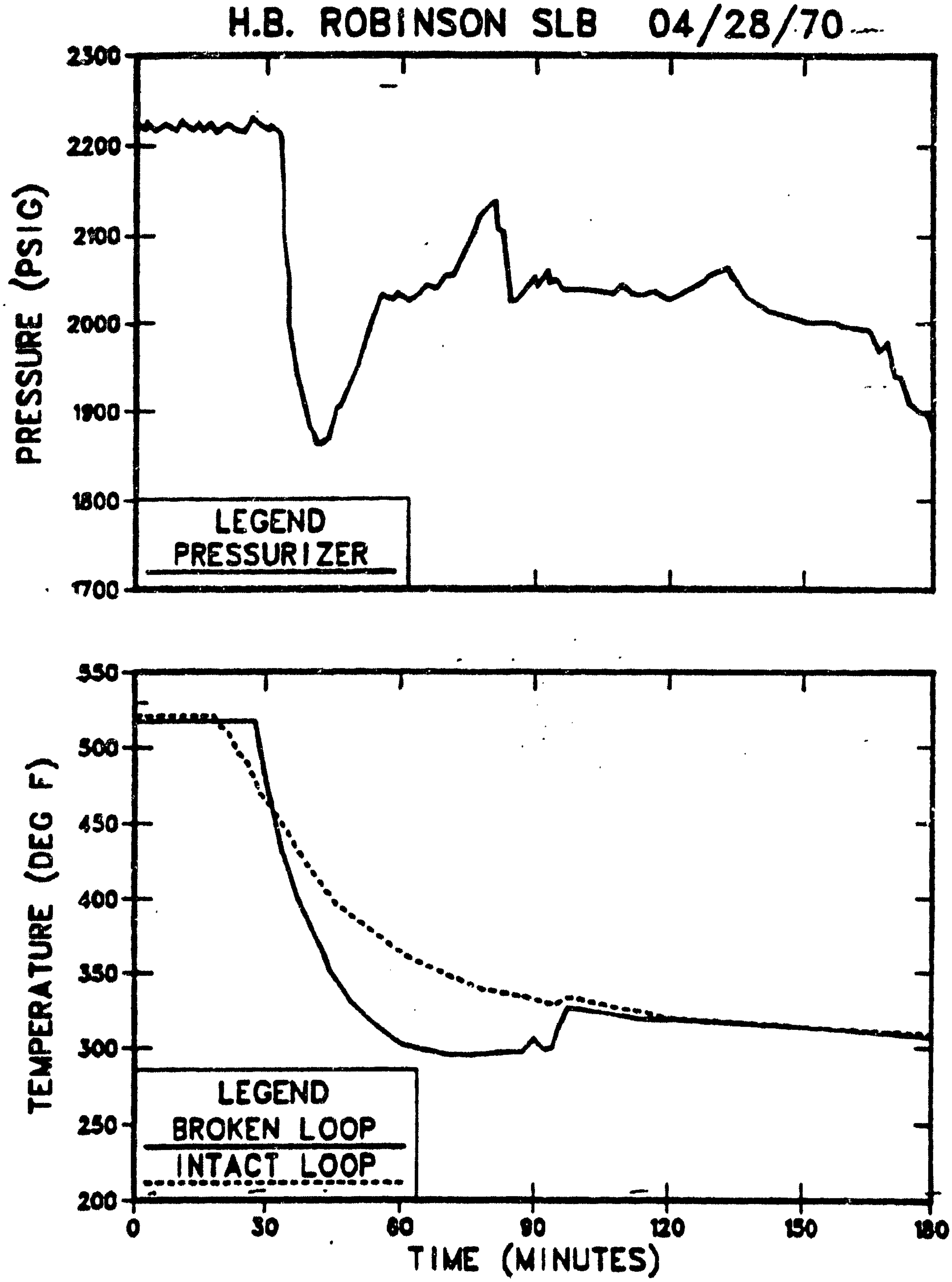
H.B. ROBINSON SLB 04/28/70
PRESSURE (PSIG)
2300
2200
2100
2000
1900
1800
1700
LEGEND
PRESSURIZER
TEMPERATURE (DEG F)
550
500
450
400
350
300
250
200
LEGEND
BROKEN LOOP
INTACT LOOP
0
30
60
90
120
150
180
TIME (MINUTES)

FIGURE 2-1

H.B. ROBINSON SLB 04/28/70

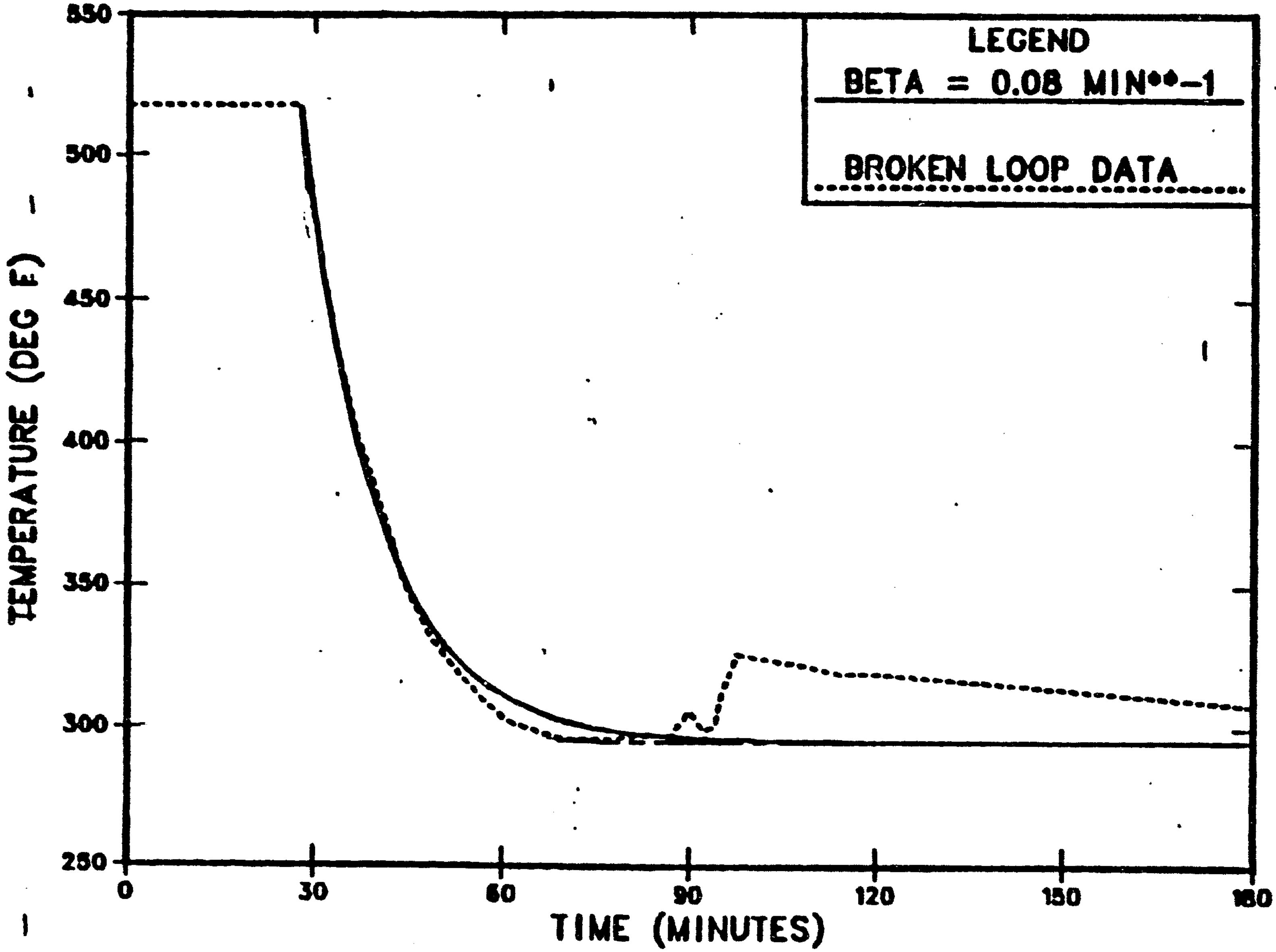

FIGURE 2-2

note: 17 and 18 deleted

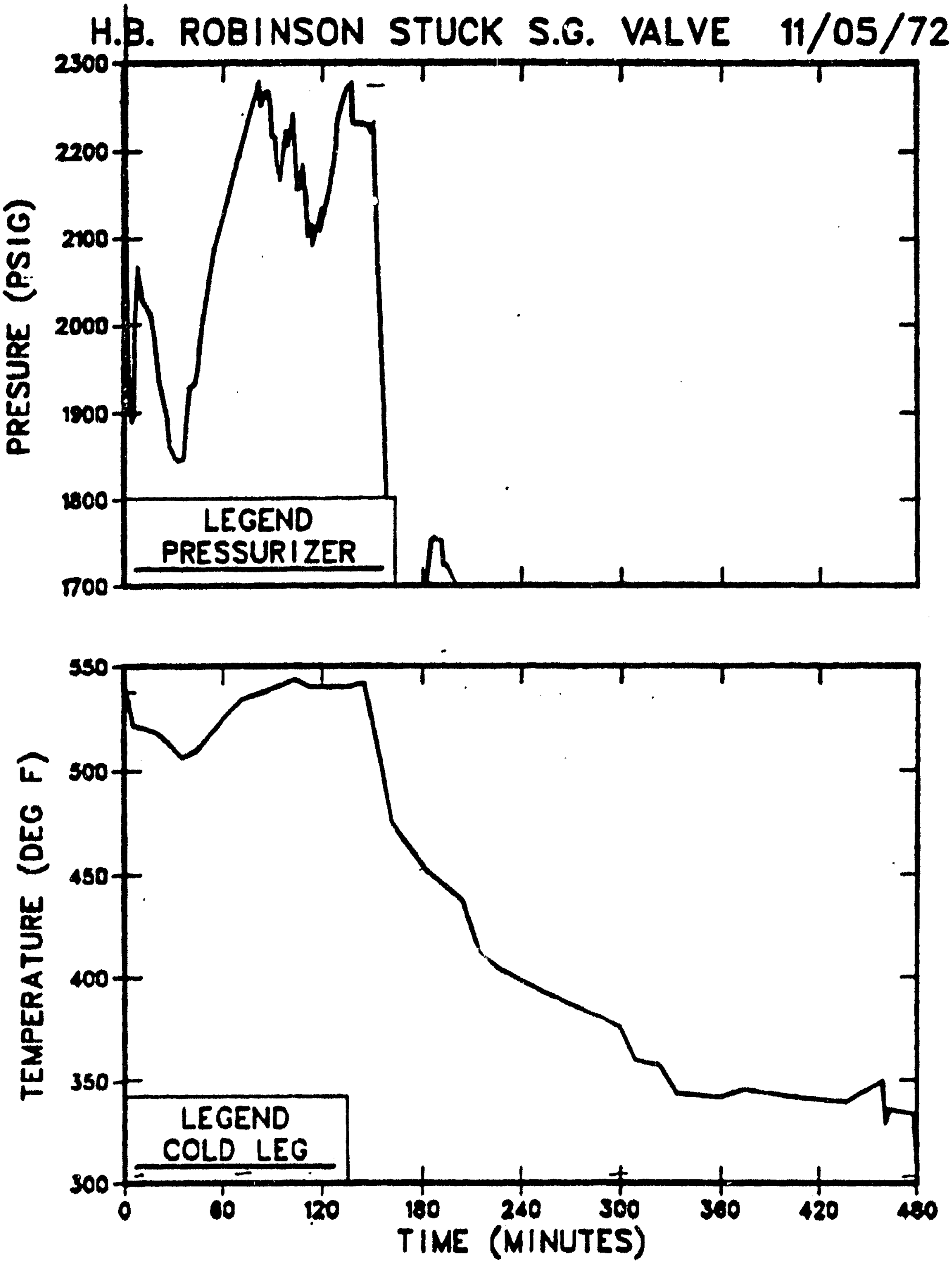

FIGURE 2-3

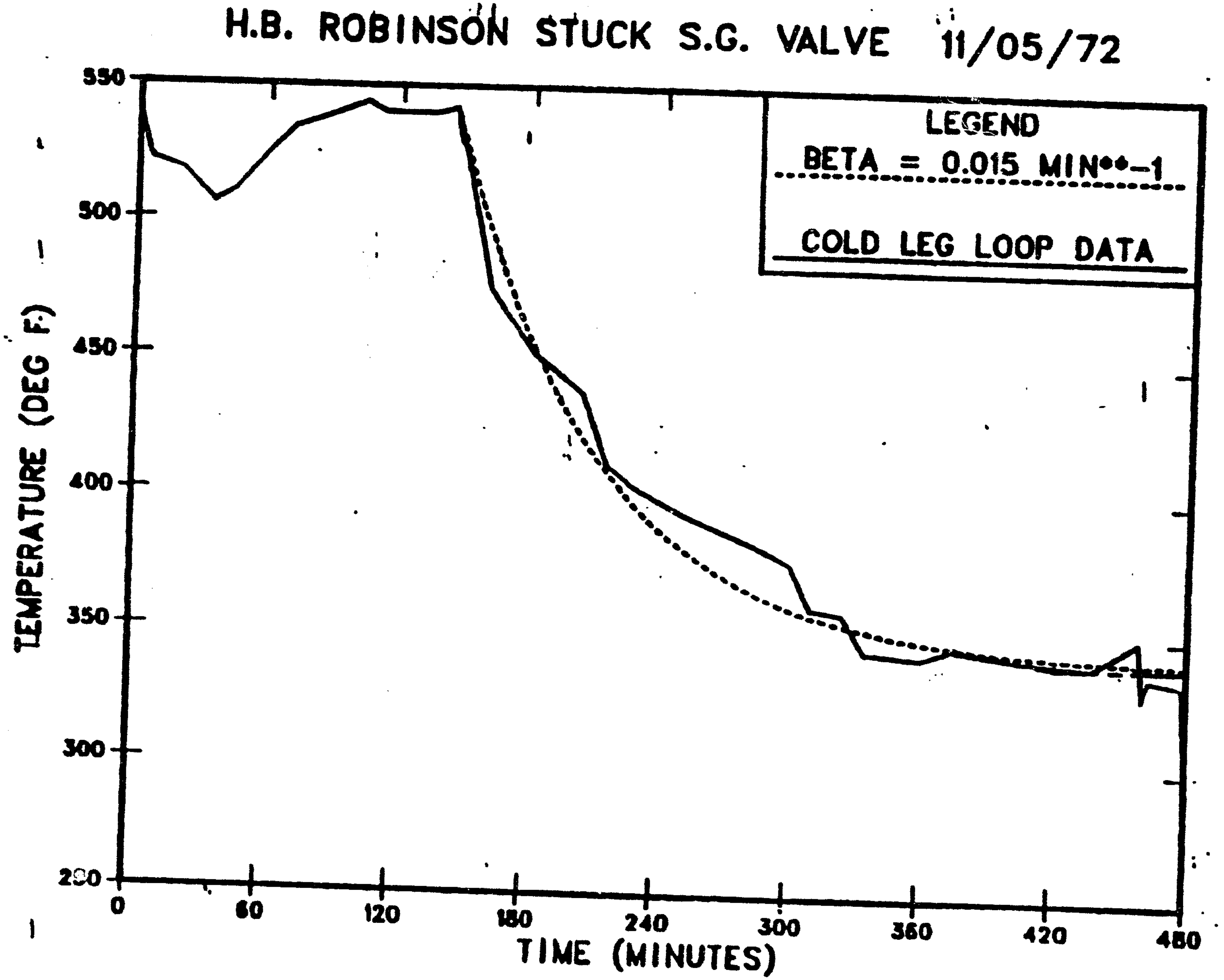
H.B. ROBINSON STUCK S.G. VALVE 11/05/72
LEGEND
BETA = 0.015 MIN**-1
COLD LEG LOOP DATA
TEMPERATURE (DEG F)
550
500
450
400
350
300
250
0
60
120
180
240
300
360
420
480
TIME (MINUTES)

FIGURE 2-4

period of interest. The use of the auxiliary pressurizer spray rapidly reduced the pressure to 500 psig.

The operator used SI to stabilize pressurizer level and pressure while using the main condenser to cool down the plant for RHR entry.

There is no indication that SI was used to repressurize the plant.

The transient data for this event are provided in Figure 2-5. For the stylized characterization of this event, the staff selected T_f = 250°F, β = 0.02 min^{-1} and a pressure of 500 psig. The exponential temperature curve is compared with the broken loop cold leg temperature data in Figure 2-6.

2.2.4 Rancho Seco NNI/ICS (03/20/78) (excess feedwater transient)

On March 20, 1978, the Rancho Seco plant RCS was cooled from 582°F to about 285°F in slightly more than one hour (approximately 300°F/hr), while RCS pressure was about 2000 psig. The transient was initiated by an inadvertent short in a DC power supply causing a loss of power to the plant's non-nuclear instrumentation (NNI). Loss of NNI power caused the loss of most control room instrumentation and the generation of erroneous signals to the plant's Integrated Control System (ICS). The ICS reduced main feedwater, causing the reactor to trip on high pressure. The cooldown was initiated when feedwater was readmitted to one steam generator by the ICS (auxiliary feedwater was restored). The cooldown caused system pressure to drop to the setpoint (1600 psig) for the safety features actuation system, which started the high pressure injection pumps and auxiliary feedwater to both steam generators. High pressure injection flow restored pressure to 2000 psig. With control room instrumentation either unavailable or suspect for one hour and ten minutes (until NNI power was restored), operators continued auxiliary feedwater and main feedwater to the steam generators while maintaining RCS pressure with the high pressure injection pumps.

The transient data for this event are provided in Figure 2-7. For the stylized characterization of this event the staff selected T_f= 285°F, β = 0.10 min^{-1} and

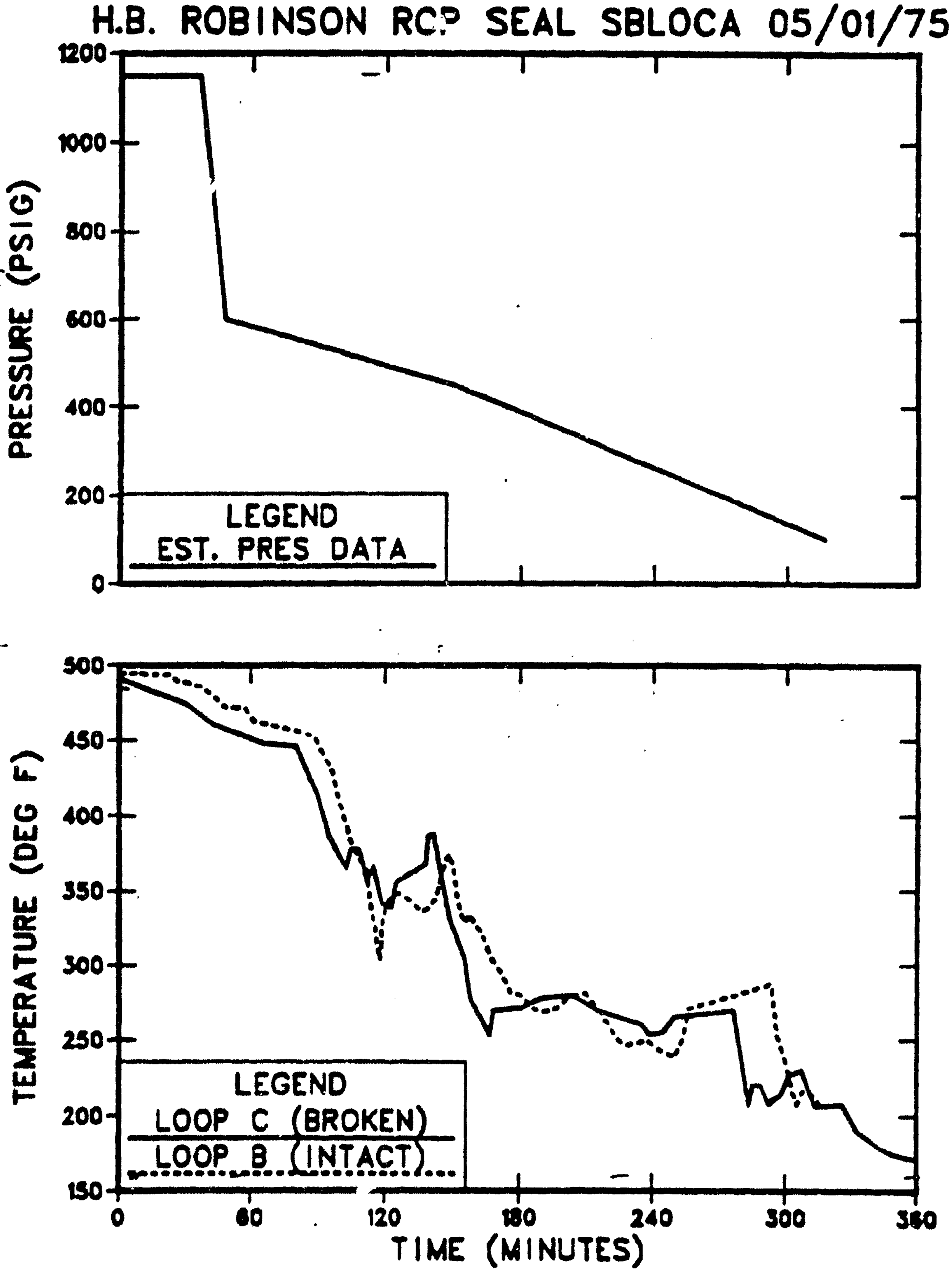
H.B. ROBINSON RCP SEAL SBLOCA 05/01/75
PRESSURE (PSIG)
1200
1000
800
600
400
200
0
LEGEND
EST. PRES DATA
TEMPERATURE (DEG F)
500
450
400
350
300
250
200
150
LEGEND
LOOP C (BROKEN)
LOOP B (INTACT)
0
60
120
180
240
300
360
TIME (MINUTES)

FIGURE 2 - 5

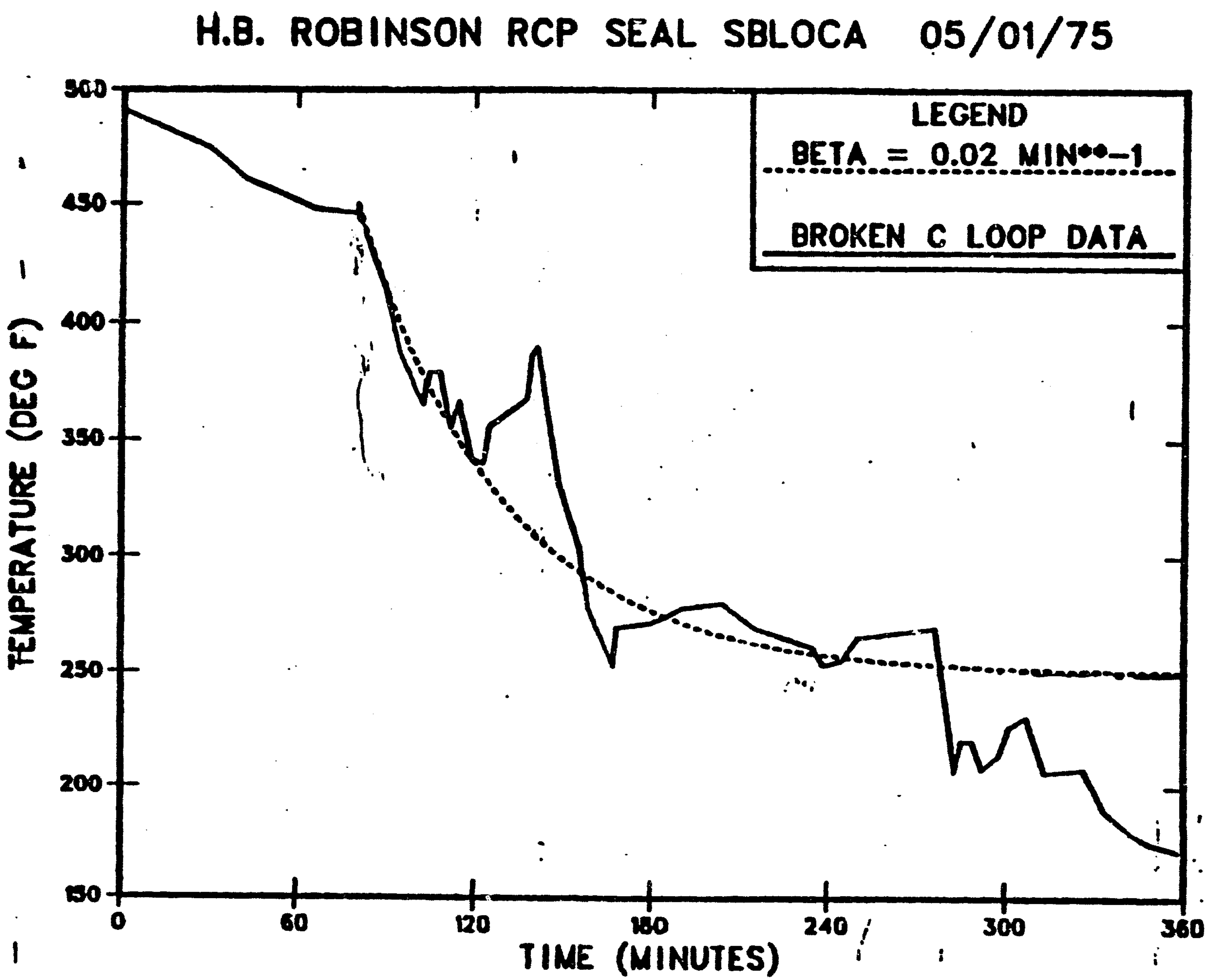
H.B. ROBINSON RCP SEAL SBLOCA 05/01/75
LEGEND
BETA = 0.02 MIN**-1
BROKEN C LOOP DATA
TEMPERATURE (DEG F)
500
450
400
350
300
250
200
150
0
60
120
180
240
300
360
TIME (MINUTES)

FIGURE 2-6

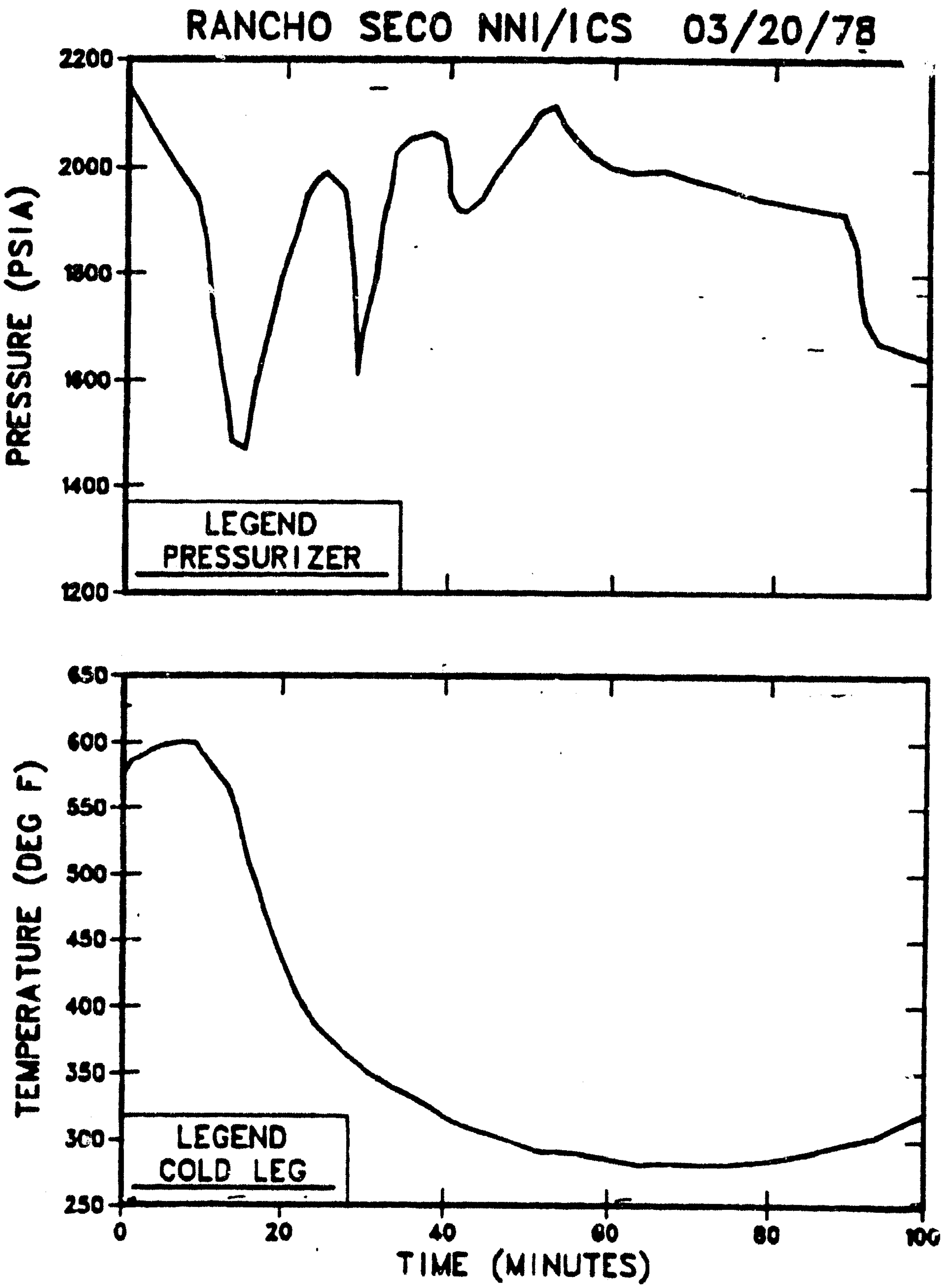
RANCHO SECO NNI/ICS 03/20/78
PRESSURE (PSIA)
2200
2000
1800
1600
1400
1200
LEGEND
PRESSURIZER
TEMPERATURE (DEG F)
650
600
550
500
450
400
350
300
250
LEGEND
COLD LEG
0
20
40
60
80
100
TIME (MINUTES)

FIGURE 2-7

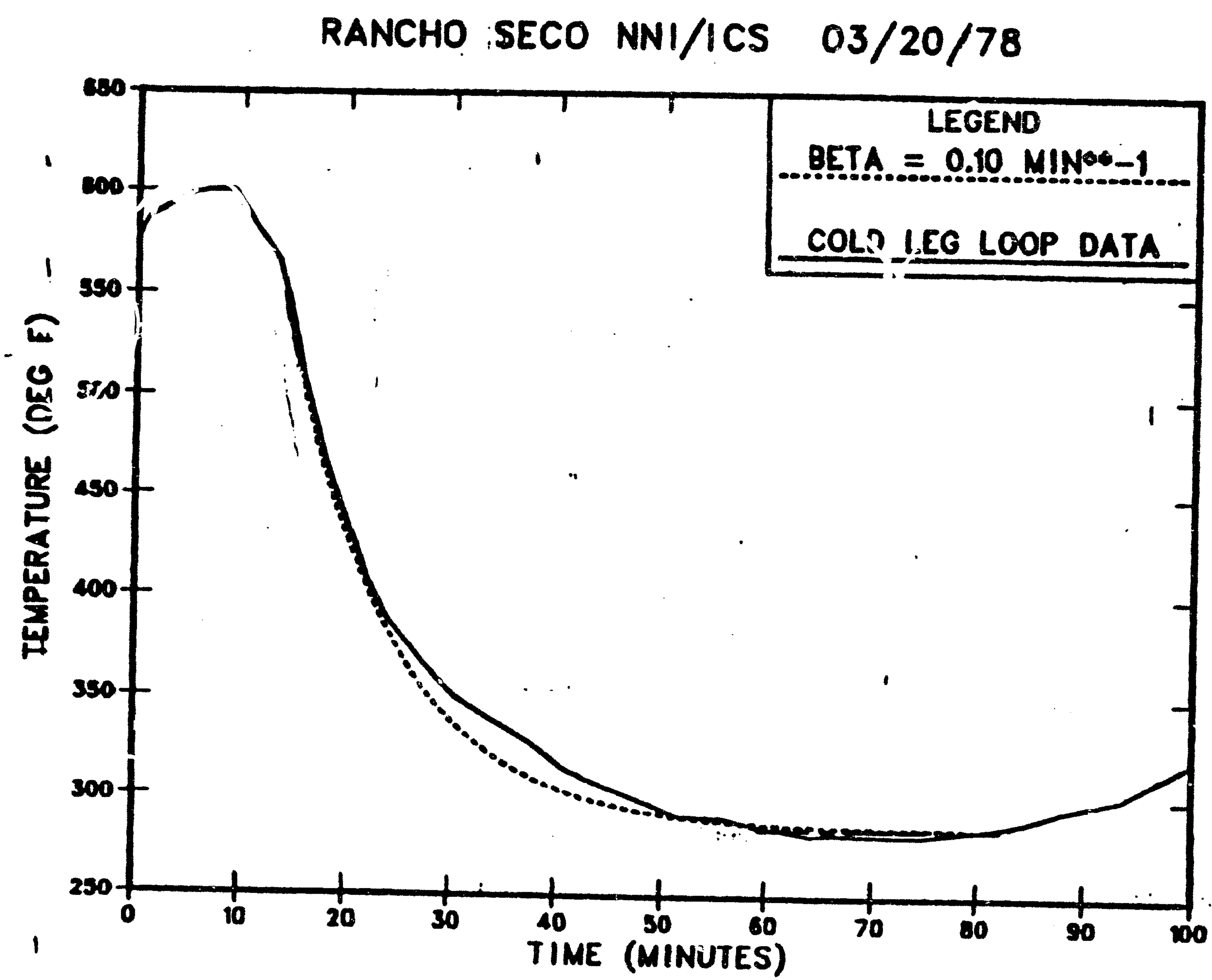
RANCHO SECO NNI/ICS 03/20/78
LEGEND
BETA = 0.10 MIN**-1
COLD LEG LOOP DATA
TEMPERATURE (DEG F)
650
600
550
500
450
400
350
300
250
TIME (MINUTES)
0
10
20
30
40
50
60
70
80
90
100

FIGURE 2-8

a pressure of 2300 psig. The exponential temperature curve is compared with the cold leg temperature data in Figure 2-8.

2.2.5 Three Mile Island 2 (03/28/79)

This accident was initiated by a loss of normal feedwater to the steam generators resulting in a turbine trip. As a result of the loss of heat sink, the RCS overpressurized and the power-operated relief valve (PORV) opened, which is a normal response and in accordance with the design. The PORV stuck open and remained open for about 2.4 hours, unnoticed by the operator. High pressure injection (HPI) was actuated on low pressure. However, at about 3 minutes into the event an operator bypassed the injection actuation signal. One HPI pump was turned off, and the remaining flow was reduced as a result of a high-level indication in the pressurizer. HPI was automatically actuated again at about 3.3 hours into the event. For the first 73 minutes the RCPs were running. After this time the pumps were turned off due to excessive vibration.

The transient data for this event are provided in Figure 2-9. For the stylized characterization of this event, the staff selected $T_f = 225°F$, $\beta = 0.04\ min^{-1}$ and a pressure of 2300 psig. The exponential temperature curve is compared with the cold leg temperature data in Figure 2-10.

2.2.6 R. E. Ginna SGTR & PORV (01/25/82)

The plant was operating at 100% power with normal pressure and temperature prior to the steam generator tube rupture (SGTR). The SGTR resulted in automatic reactor trip and automatic actuation of safety injection. On the SI signal, automatic containment isolation occurred and the charging pumps were tripped. Both RCPs were tripped by the operator in accordance with plant procedures. The operators attempted to equalize the primary and faulted SG pressure, in accordance with plant procedures, by opening the PORV. The PORV failed open, and the operator manually closed the block valve to stop the coolant loss.

The transient data for this event are provided in Figure 2-11. For the stylized characterization of this event, the staff selected $T_f = 325°F$, $\beta = 0.12\ min^{-1}$,

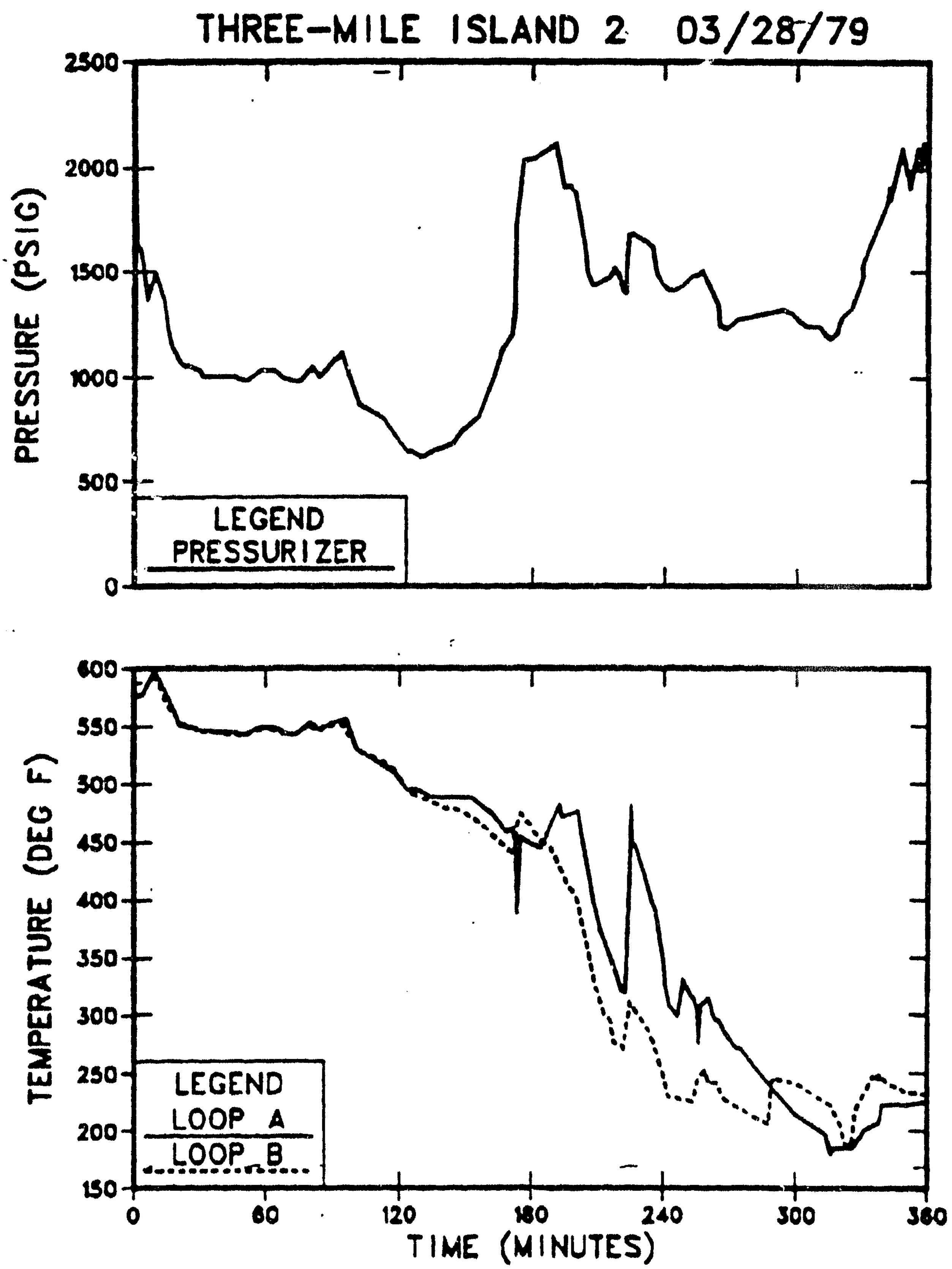
THREE-MILE ISLAND 2 03/28/79
PRESSURE (PSIG)
2500
2000
1500
1000
500
0
LEGEND
PRESSURIZER
TEMPERATURE (DEG F)
600
550
500
450
400
350
300
250
200
150
LEGEND
LOOP A
LOOP B
0
60
120
180
240
300
360
TIME (MINUTES)

FIGURE 2-9

THREE-MILE ISLAND 2 / 03/28/79

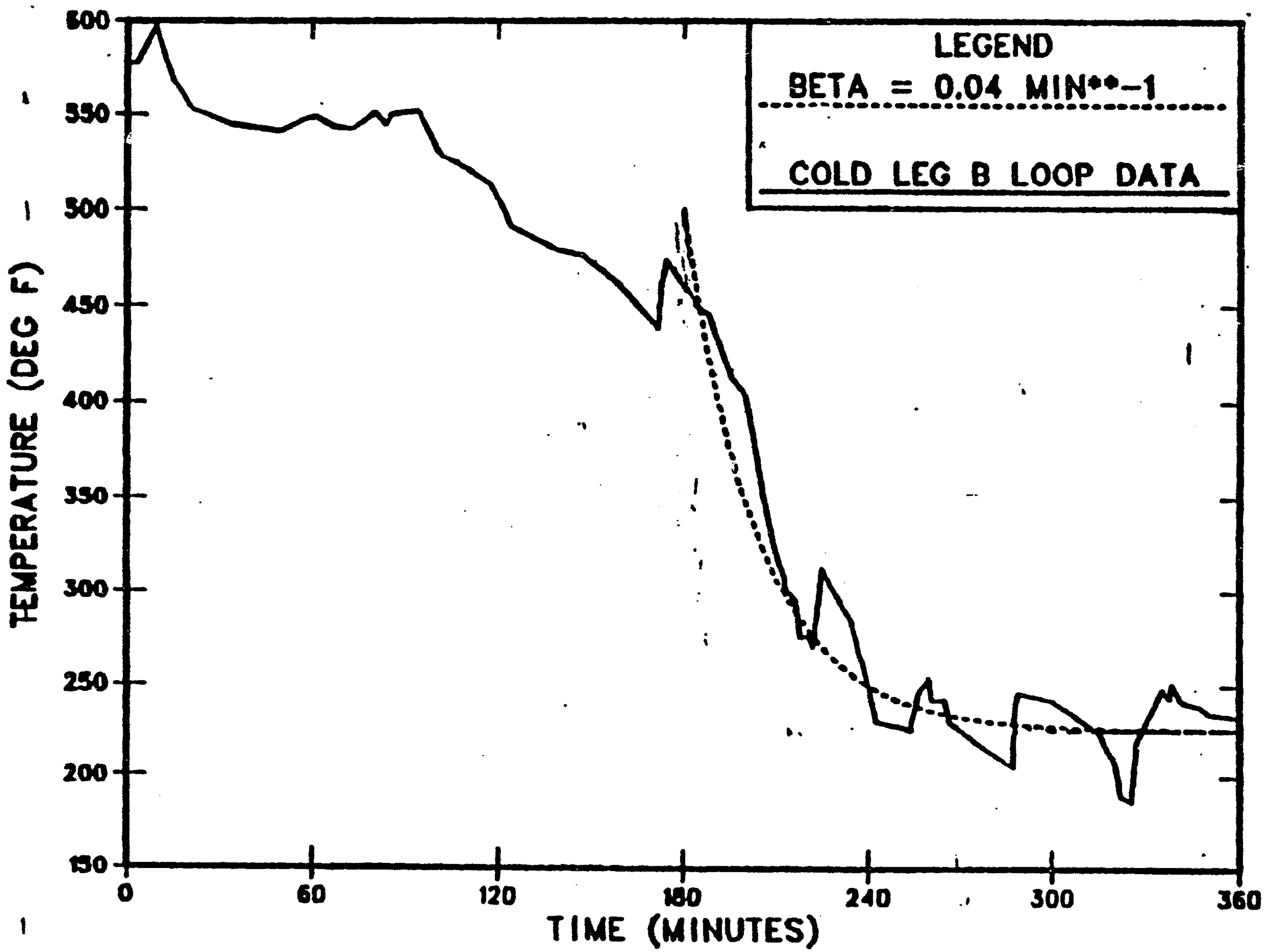

FIGURE 2-10

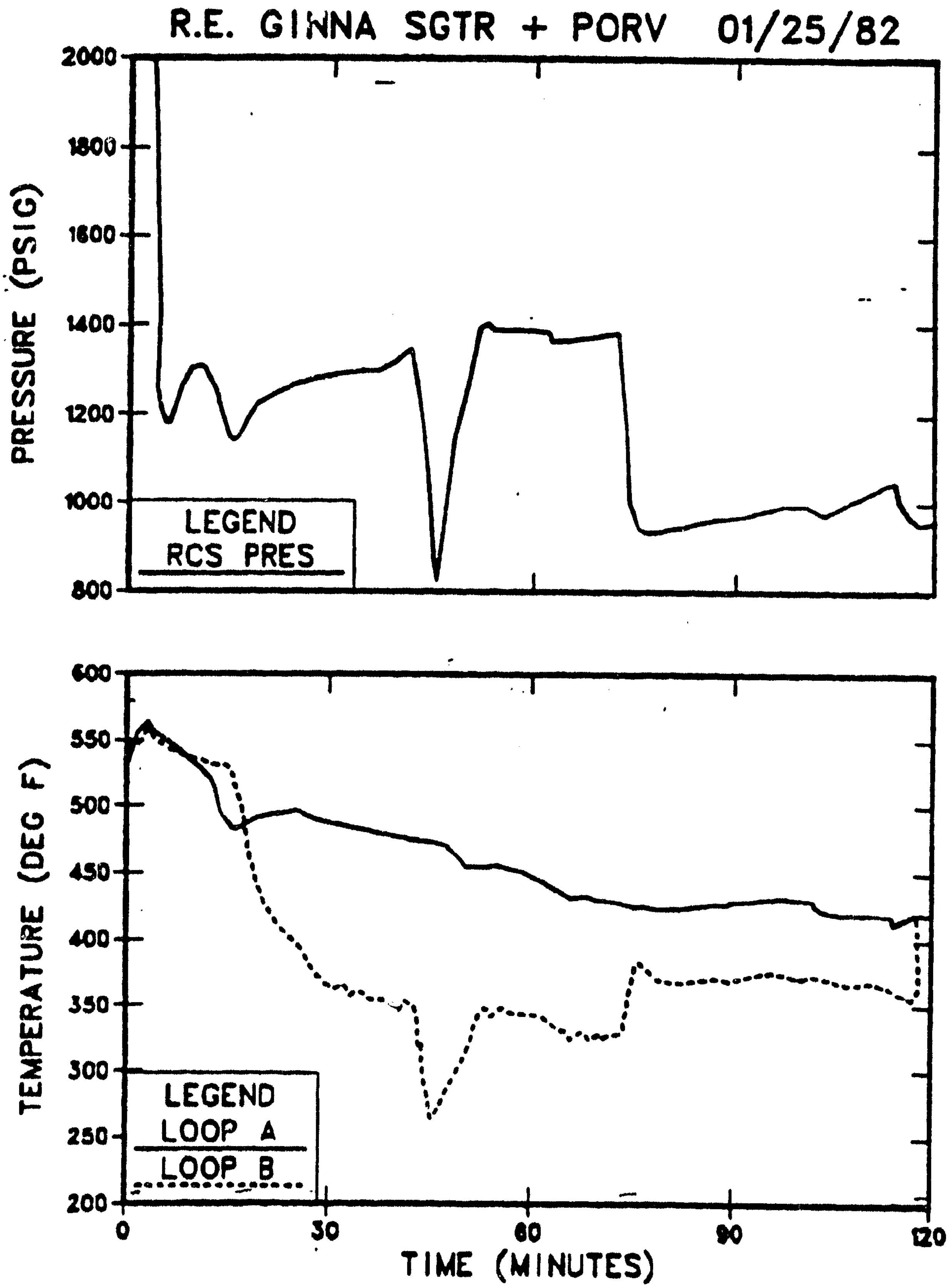
R.E. GINNA SGTR + PORV 01/25/82
2000
1800
1600
1400
1200
1000
800
PRESSURE (PSIG)
LEGEND
RCS PRES
600
550
500
450
400
350
300
250
200
TEMPERATURE (DEG F)
LEGEND
LOOP A
LOOP B
0
30
60
90
120
TIME (MINUTES)

FIGURE 2-11

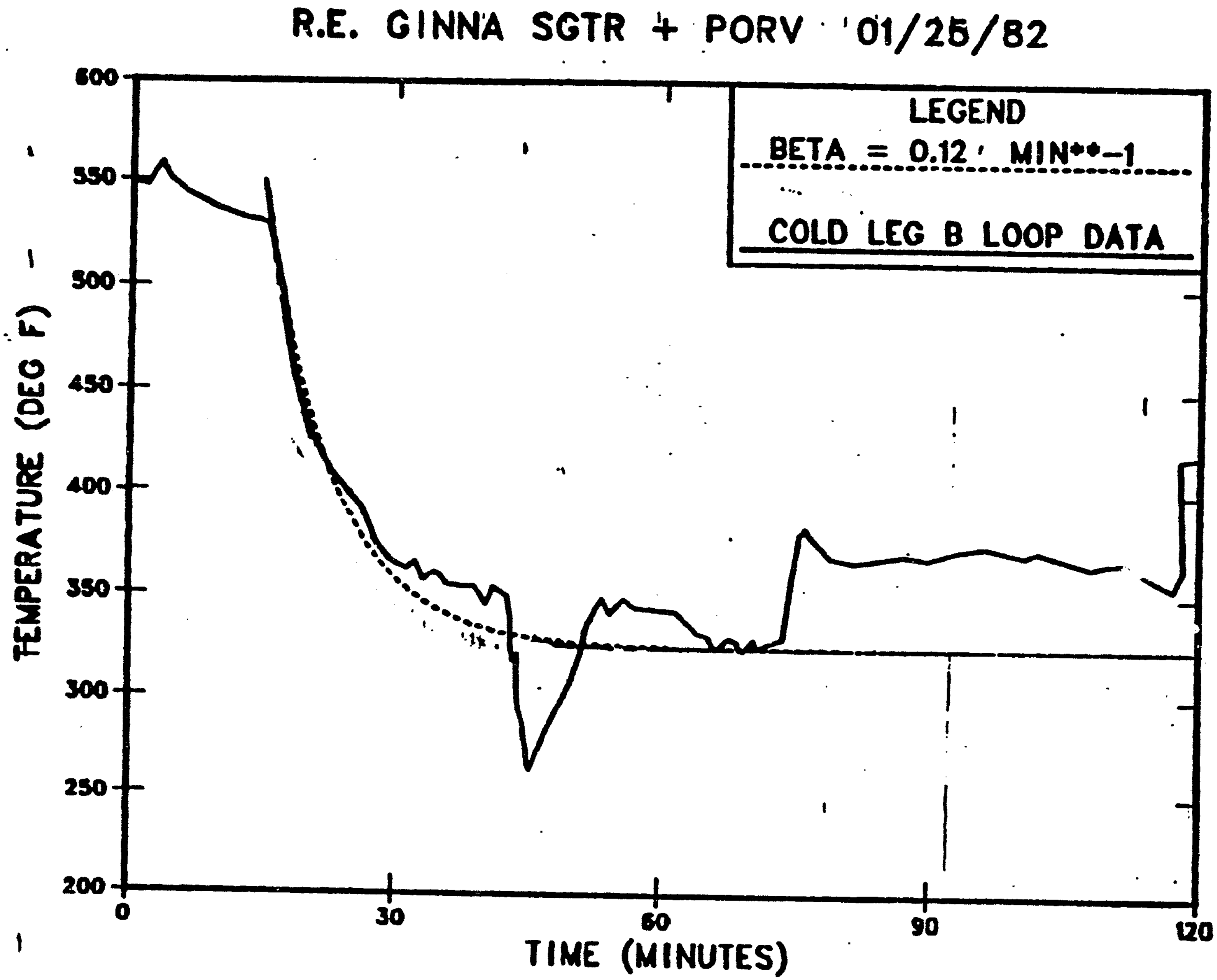
R.E. GINNA SGTR + PORV 01/25/82
LEGEND
BETA = 0.12 MIN**-1
COLD LEG B LOOP DATA
600
550
500
450
400
350
300
250
200
TEMPERATURE (DEG F)
0
30
60
90
120
TIME (MINUTES)

FIGURE 2-12

and a pressure of 1400 psig. The exponential temperature curve is compared with the cold leg temperature data in Figure 2-12.

The sudden temperature dip at about 45 minutes has been shown not to be significant in the fracture mechanics analysis, and has been ignored in characterizing this event.

2.2.7 Crystal River 3 NNI/ICS (02/26/80) (small-break LOCA transient)

On February 26, 1980, the Crystal River 3 plant experienced a small-break LOCA transient when a power-operated relief valve (PORV) was opened inadvertently. The resulting transient caused a decrease in RCS temperature (whose magnitude is discussed below) with a system pressure of about 2400 psig. The transient was initiated when an electrical short in a DC power supply for the plant's NNI caused a pressurizer PORV to open, a loss of most control room instrumentation, and the generation of erroneous signals to the plant's ICS. The ICS caused a reduction in feedwater flow and a withdrawal of control rods. RCS pressure initially increased, tripping the reactor on high pressure, and then decreased as coolant discharged through the open PORV. The high pressure injection pumps started at 1500 psig and repressurized the RCS to about 2400 psig. The PORV block valve was closed, but flow out of the RCS continued through the pressurizer safety valves. After approximately 30 minutes, the high pressure injection pumps were throttled back, but RCS pressure was maintained at about 2300 psig for the next one and a half hours while shutdown to cold shutdown conditions by normal operating procedures was initiated.

Since temperatures in the downcomer are not measured, and since many of the temperature measurements normally available were lost when instrumentation power was lost, minimum temperatures were calculated.

For the purpose of this evaluation, the minimum downcomer temperature is based on calculated mixing in the downcomer of the HPI with the minimum vent valve flow (1 vent valve), using the TRAC code and Creare (Ref. 2.1) data for thermal mixing. The mean mixed value for T_f is approximately 250°F (the same value indicated by B&W). A cooldown β of 0.10 min^{-1} is used, based on a preliminary review of the TRAC analysis, and an approximate time span of 20 minutes prior to the operator regaining control of the transient. For the stylized characterization of this event, the staff selected T_f = 250°F, β = 0.10 min^{-1}, and a pressure of 2300 psig.

2.2.8 Prairie Island SGTR (10/02/79)

This event was similar to the Ginna SGTR; however, the minimum temperature was 350°F with a β of 0.1 per minute. β is estimated from the adjusted β^* value for a cooldown period of approximately 20 minutes. A pressure of 1000 psig was selected. No plots of temperature and pressure data were available.

2.3 Summary of Operating Experience

In addition to the eight events described in Section 2.2, 24 other events which could have led to PTS concern have been identified. The data sources are the work performed by Phung (Ref. 2.2) and the various licensee submittals on PTS. The Phung study utilized over 16,000 licensee event reports (LERs) covering 329 reactor years of operating experience at 47 PWR units. The search was for events that had one or more of the characteristics of a severe PTS event, which are rapid cooling of the pressure vessel to a low temperature and maintenance of the low temperature and/or rapid cooling rate for several minutes (typically greater than 10 minutes), plus maintenance of a high vessel pressure or vessel repressurization. A list of significant PTS events was presented and the most significant PTS event from the list and from the NRC staff's own experience are presented in Figure 2-13, where final temperature is shown for 24 events. It is noted that the CE submittals did not identify the Millstone-2 and St. Lucie-1 events as PTS events of concern. It is also noted that 2 of the 3 San Onofre-1 events were not identified by Phung. By vendor there are 21 Westinghouse events, 4 CE events, and 7 B&W events. Only the eight events discussed above which resulted in final temperatures of 350°F and less with significant pressure, fast cooling and sustained low temperature are of interest for the detailed PTS analyses. These eight events selected for detailed analyses are underlined in Figure 2-13. The values of T_f, β and pressure that have been selected to characterize these events are summarized in Table 2-1.

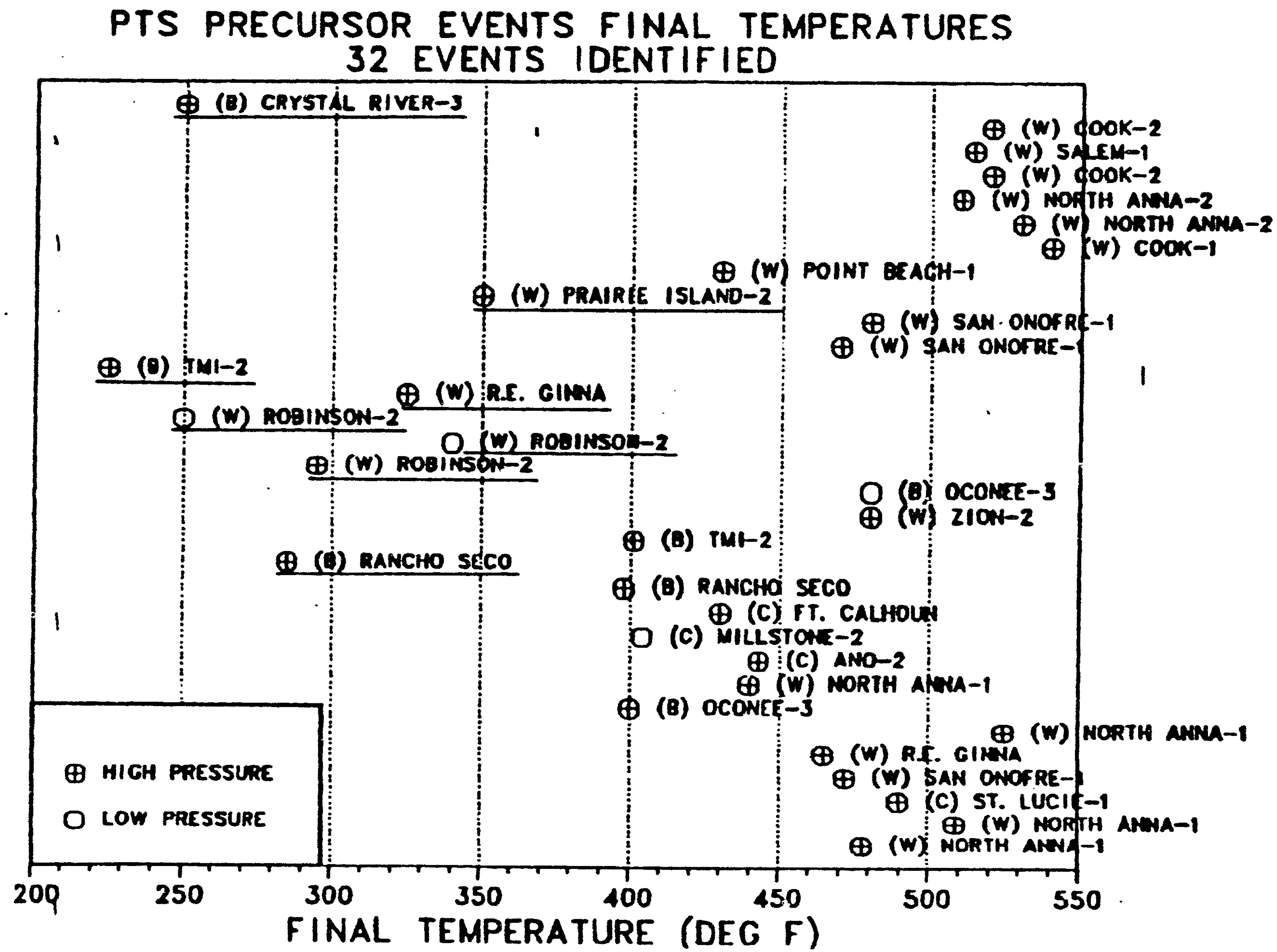
PTS PRECURSOR EVENTS FINAL TEMPERATURES
32 EVENTS IDENTIFIED
(B) CRYSTAL RIVER-3
(W) COOK-2
(W) SALEM-1
(W) COOK-2
(W) NORTH ANNA-2
(W) NORTH ANNA-2
(W) COOK-1
(W) POINT BEACH-1
(W) PRAIRIE ISLAND-2
(W) SAN ONOFRE-1
(W) SAN ONOFRE-1
(B) TMI-2
(W) R.E. GINNA
(W) ROBINSON-2
(W) ROBINSON-2
(W) ROBINSON-2
(B) OCONEE-3
(W) ZION-2
(B) TMI-2
(B) RANCHO SECO
(B) RANCHO SECO
(C) FT. CALHOUN
(C) MILLSTONE-2
(C) ANO-2
(W) NORTH ANNA-1
(B) OCONEE-3
(W) NORTH ANNA-1
(W) R.E. GINNA
(W) SAN ONOFRE-1
(C) ST. LUCIE-1
(W) NORTH ANNA-1
(W) NORTH ANNA-1
HIGH PRESSURE
LOW PRESSURE
200
250
300
350
400
450
500
550
FINAL TEMPERATURE (DEG F)

FIGURE 2-13

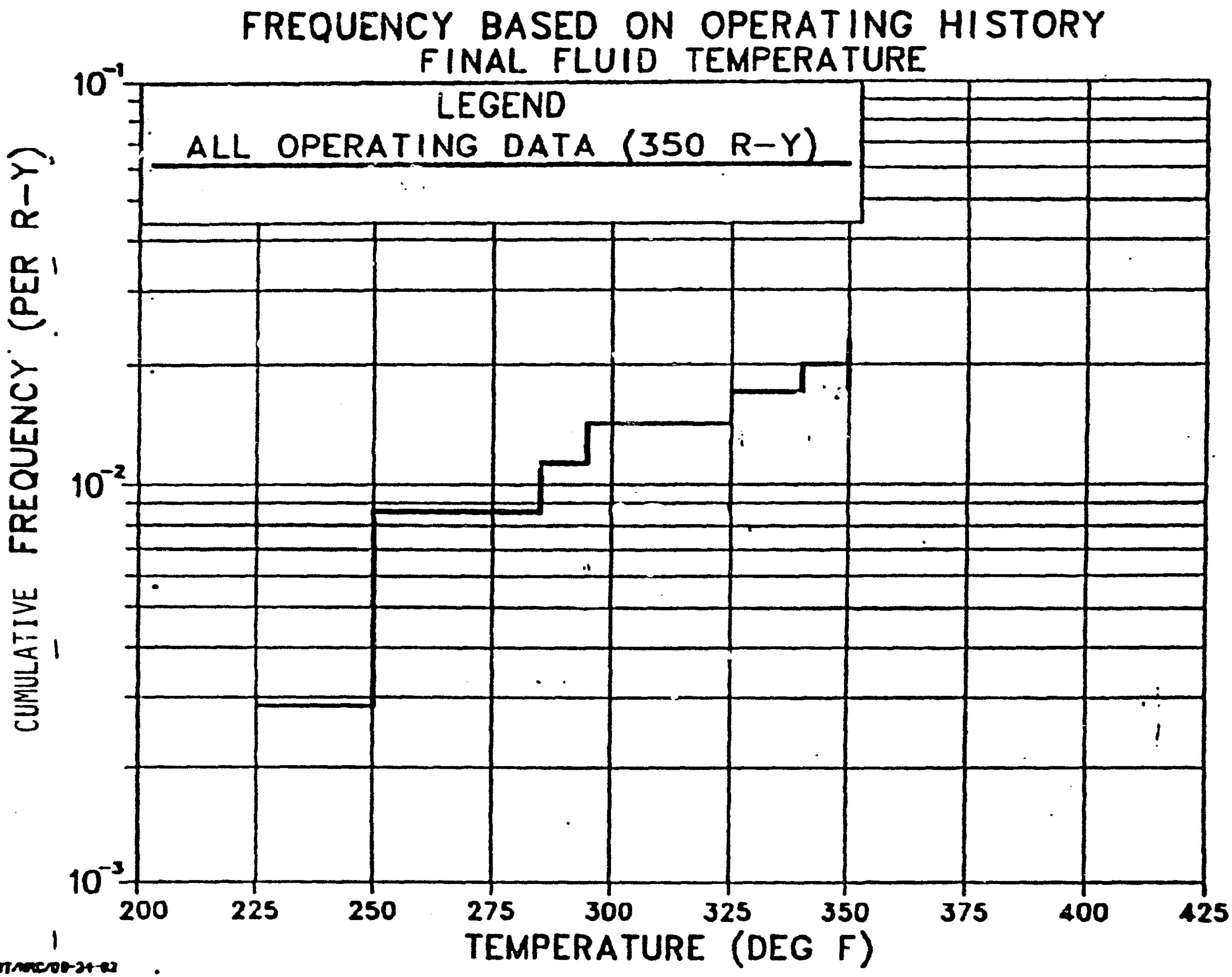
FREQUENCY BASED ON OPERATING HISTORY
FINAL FLUID TEMPERATURE
LEGEND
ALL OPERATING DATA (350 R-Y)
CUMULATIVE FREQUENCY (PER R-Y)
10⁻¹
10⁻²
10⁻³
200
225
250
275
300
325
350
375
400
425
TEMPERATURE (DEG F)

FIGURE 2-14

The eight events characterized in Table 2-1 above occurred during approximately 350 reactor-years of PWR operating experience. On that basis, a cumulative frequency distribution has been plotted as a function of the final temperature of the event, T_f, as shown in Figure 2-14.

2.4 Comparison with Westinghouse Characterization of Operating Experience

Westinghouse believes the operational events referred to in this section that occurred in Westinghouse-designed plants should be characterized somewhat differently. (Their most recent discussion is contained in Reference 2.3)

The comparison is as follows:

Event	NRC T_f	W T_f
HBR '75	250	327
Ginna	325	300
HBR '70	295	295
HBR '72	340	400
Prairie Island	350	390

The differences are due to three causes, according to Westinghouse.

First, they state that we plotted the cumulative distribution of events incorrectly. We agree, with respect to a much earlier curve we used. We now plot T_f correctly in Figure 2-14.

Second, they state that one of the events (HBR-'70) was a pre-fuel loading event that occurred during testing conducted to detect weaknesses exactly like the one that was found and, therefore, should not be included. We agree that inclusion of the event is somewhat on the conservative side, but note that deletion of the event would make no significant difference in our conclusion.

Third, they state that we should terminate an event for PTS consideration when the opertor gets the plant within Appendix G cooldown limits. We do not agree. Certainly it is true that a shutdown under normal conditions within Appendix G limits is not a PTS concern. However, a cooldown (whether deliberate or

Table 2-1 Parameters for stylized representation of experienced events

Event	T_f(°F)	β(min^{-1})	P(psig)
Robinson SLB (W) ('70)	295	0.08	2000
Robinson Stuck SG Valve (W) ('72)	340	0.015	<1000
Robinson RCP Seal SBLOCA (W) ('75)	250	0.02	<500
Rancho Seco (B&W)	285	0.10	2300
TMI-2 (B&W)	225	0.04	2300
Ginna SGTR (W)	325	0.12	1400
Crystal River-3 (B&W)	250	0.10	2300
Prairie Island SGTR (W)	350	0.10	1000

uncontrolled) within Appendix G limits immediately following a more rapid cool-down of PTS concern can very well exacerbate the PTS concern and must be considered.

References:

2.1 "Fluid and Thermal Mixing in a Model Cold Leg and Downcomer With Vent Valve Flow," Creare Incorporated, EPRI Report NP-2227, March 1982.

2.2 Phung, D. L., "Pressure Vessel Thermal Shock at U.S. Pressurized Water Reactors: Events and Precursors, 1963 to Mid-1981," ORNL, Interim Report, May 1982, and final report ORNL/NSIC-112, pre-print version dated August 15, 1982.

2.3 Letter from O. D. Kingsley, Jr. to Harold R. Denton dated Sept. 2, 1982 regarding Westinghouse Owners Group Activities related to Pressurized Thermal Shock.

3. DETERMINISTIC FRACTURE MECHANICS ANALYSES

3.1 Fracture Mechanics Discussion

The calculations reported in this section are used to analyze the response of a reactor pressure vessel (RPV) to an overcooling transient. The input information includes (1) pressure and temperature of the reactor coolant as a function of time, obtained from thermal-hydraulic calculations; (2) materials properties, including temperature and irradiation effects; and (3) an actual or assumed initial flaw. Vessel integrity analyses, the results of which are reported in this document, include a determination of the temperature distribution across the vessel wall versus time, the thermal stresses as a consequence of this temperature distribution, as well as fracture mechanics results. Thus, the term "fracture mechanics (FM) analysis" used in this section really means vessel integrity analysis because it includes heat transfer and stress analysis. The stresses considered are those as a result of pressure and differential thermal expansion between the clad and the base metal as well as thermal stresses.

Once the stress distribution is determined as a function of time and position, FM examines the behavior of preexisting cracks (postulated or real) in this stress field. For specific crack geometries, a stress intensity factor, K_I, is calculated and compared to a material toughness property, K_{IC}. When K_I exceeds K_{Ic} for a specified crack, the crack will initiate, i.e., grow deeper into the metal. K_I for the crack then increases until it reaches a value equal to K_{Ia} which is another material property. The crack then arrests, i.e., stop growing larger. The material properties (K_{Ic} and K_{Ia}) vary with temperature and degree of irradiation damage and hence are a function of time and depth into the vessel wall.

FM algorithms consider these factors. For pressurized thermal shock (PTS) evaluations, linear elastic fracture mechanics (LEFM) is used because, at the temperatures involved, the metal is at less than its maximum or upper shelf toughness. Illustrations of typical temperature, stress and stress intensity factor distributions within the vessel wall at different times during the transient are shown in Figure 3-1 (a), (b) and (c) respectively.

The quantities K_{Ic}, the vessel toughness that determines crack initiation, and K_{Ia} the toughness at crack arrest, also vary with position and time, since they are functions or irradiation and temperature. When K_I exceeds the value of K_{Ic} at the location of the tip of the flaw, crack initiation is expected, if warm prestressing is not effective (warm prestressing is discussed below and in Appendix D). The crack would then grow to a depth where K_I equals the value of K_{Ia} at the tip of the growing crack. For some transients, metals properties, and flaws, K_I will remain above K_{Ia} and the crack is assumed to go through the vessel wall without arrest if the system is pressurized.

Similar results would occur for a circumferentially oriented crack except that arrest will generally occur at shallower depths because the stress intensity factor, K_I, for long axial cracks is higher than for long circumferential cracks, especially for cracks that extend relatively deep into the vessel wall.

Equivalent calculations are made as a function of time in the transient, and the results cross-plotted on a critical crack depth diagram. From this diagram, the behavior of a crack versus time for a particular PTS scenario can be determined. Such a diagram is shown in Figure 3-2. See additional discussion in Section 3.4.4.

Warm prestressing (WPS) is a phenomenon that can inhibit or prevent crack initiation even though the calculated stress intensity factor, K_I, becomes greater than the material toughness parameter, K_{Ic}, at the time and location of the flaw tip. For WPS to be effective, K_I must be at less than a previous maximum value at the time K_I becomes equal to or greater than K_{Ic}. This can occur if K_I is monotonically decreasing as the metal cools causing K_{Ic} to decrease. When the course of a PTS transient can be described with confidence, the time behavior of K_I can be evaluated for determining whether WPS occurs. For generic studies, however, wherein the pressure variation versus time cannot be unambiguously defined, the NRC does not assume the benefit of WPS. In general, K_I will increase after its initial peak only due to an increase in pressure.

3.2 Description of FM Computer Analysis Programs

The NRC staff utilized its own in-house FM program in performing heat transfer, stress and fracture mechanics analyses related to pressurized thermal shock (PTS) and has also relied on ORNL to supplement the staff analyses by use of the OCA program as reported later in this section. The NRC program is also utilized as the deterministic portion of the VISA program in performing the probabilistic fracture mechanics analyses discussed in Section 7. The NRC and ORNL programs are equivalent, as described in Appendix D. Analytical results utilizing these programs for specific PTS scenarios have been compared and found to be in close agreement. Similar comparisons have been made with results of industry analyses. We conclude that the analytical methods used by the NRC, ORNL and the vendors yield essentially the same results if all input assumptions are the same. Differing conclusions result primarily from various assumptions regarding input parameters.

When material properties and the transient are known, fracture mechanics procedures can predict crack behavior quite well as demonstrated by comparison with a wide variety of experiments. The Heavy Section Steel Technology research program has included hundreds of irradiated test samples, plus model vessels tested at low temperatures to include brittle and transition behavior. Tests have included thermal shock, and plans for the near future include combined pressure and thermal stresses.

When a crack initiates and grows deeper into a reactor vessel wall, the shap it becomes depends on its initial shape, the stress intensity factor along t e crack front and the relative toughness of the metal in which it is growing. Thermal stress analyses for typical PTS transients result in higher tensile stresses at the cooled surface where the metal is colder and hence less toug than deeper into the wall. Based on analyses where cladding effects are neglected and on thermal shock experiments, cracks tend to grow in length prior to growing deeper. In other words, the cracks become relatively long. For this reason, the NRC postulates long cracks at the time of arrest regardless of the original postulated crack geometry.

Discussions with Westinghouse personnel indicated that their analyses assumed a self-similar crack shape with a length-to-depth ratio of six during crack growth and at arrest. The staff does not accept the Westinghouse assumption for the reasons discussed above. Subsequently, the Westinghouse Owners' Group has verified that if they utilized the same assumptions as the staff then their results are essentially the same as those of the NRC. (Ref. 3.1) The original differences in the models resulted in significant differences in critical RT_{NDT} at crack arrest.

In view of the importance of this matter, the staff has consulted with recognized experts in this field who have agreed that, although the NRC model is somewhat conservative, it is more realistic than models that include the assumption of a six-to-one ellipse.

3.3 Determination of K_{Ic} and K_{Ia}

The fracture analyses performed by utilities, vendors and the NRC have all utilized the values of K_{Ic} and K_{Ia} given in Section XI of the ASME Code and reproduced in Appendix D. The Code values are bounds on the conservative (low) side of experimentally determined toughness values. They have been correlated using the relative temperature, T minus RT_{NDT}, which is the reference temperature, nil-ductility transition.

RT_{NDT} is defined in Appendix D. It is a reference temperature that is used to characterize the transition in material properties, from ductile to brittle, that takes place as the temperature is decreased. Actually, the transition in properties is gradual, taking place over a temperature range of about 100°F. The use of the relative temperature, $T-RT_{NDT}$ has been shown to allow correlation of experimental toughness data in RPV materials at various temperatures, and irradiation states. The Heavy Section Steel Technology experimental data also show the $T-RT_{NDT}$ correlation.

The initial value of RT_{NDT} in a new, unirradiated vessel is quite low, but increases with irradiation. The NRC staff's method for estimating the initial RT_{NDT} and the change in RT_{NDT} caused by irradiation for a given vessel are given in Section 5 and Appendix E of this report. Estimates are given for RT_{NDT} at the inside surface of the vessel wall (at the clad-base metal interface) for the critical locations, which are almost always the welds, either a longitudinal weld or a cirumferential weld in the beltline. The attenuation of RT_{NDT} through the vessel wall is then calculated to get K_{IC} and K_{Ia} at the tips of postulated cracks (see Appendix D).

3.4 Generic Deterministic Studies of Crack Initiation

Using the models described in the preceding sections and in Appendix D, NRC and ORNL have performed a variety of deterministic FM analyses. The results are given in Appendix D and are summarized here.

3.4.1 Stylized Transients

The stylized transients used are described in Section 2.1, characterized by constant pressure, P, initial water temperature of 550°F, final water temperature, T_f, and exponential decay constant β, minutes^{-1}. The water temperature is assumed to be uniform over the inner surface of the vessel. A constant heat transfer coefficient, h, is used for the water-metal interface. An infinitely long through-clad flaw is assumed to exist on the inner surface of the vessel wall.

3.4.2 Crack Initiation for Stylized Transients

At the request of the NRC staff, ORNL performed a series of analyses with different assumed values of T_f, β, and P assuming that crack arrest and WPS were not effective. The results are plotted as a series of curves of pressure versus T_f - RT_{NDT}, an example of which is Figure 3-3. Other examples are provided in Appendix D. Note that from these diagrams, the thresholds of crack initiation can be determined. Thus, for a specific vessel RT_{NDT} and a given β and T_f, it is possible to determine the limiting pressure to avoid crack initiation.

Utilizing Figure 3-3, it is possible to relate approximately the RT_{NDT}, that a vessel must possess to avoid crack initiation for a given transient, to the final temperature of the transient. For conservatism when considering a generalized event, it is assumed that a moderately fast cooldown has occurred (β = 0.15 min^{-1}) and that full pressure (2300 psig) exists in the vessel since there is no assurance that it will be possible to take credit for automatic or manual pressure reduction. Thus, the upper right-hand portion of the figure is used, and it is seen that, for T_f of 250 to 300°F, and for longitudinal flaws, final temperatures approximately 5°F above RT_{NDT} are acceptable, but as one proceeds to more severe cooldown events (T_f = 150°F) the final temperature must stay as much as 20°F above RT_{NDT}.

3.4.3 Sensitivity Studies

In addition to the many uncertainties regarding PTS scenarios such as the temperature and pressure profiles versus time, the degree of mixing of cold with warm water, etc., parametric uncertainties in the stress and fracture mechanics analysis become significant when the cooldown temperature, T_f, is approximately equal to RT_{NDT} because small changes in assumptions can influence whether or not crack initiation is predicted. The staff performed analyses similar to those by ORNL with various assumptions as to crack shape and orientation with and without cladding-induced stresses and for different models for fluence attenuation through the wall in order to determine the effects of these assumptions. (Cladding stresses are induced because of the different coefficients of expansion of the stainless steel cladding and the carbon steel of the vessel wall.)

Sensitivity studies used a base case with T_f = 250°F, β = 0.15 min^{-1}, and considered various values of P. Some results are shown in Figure 3-4. The threshold value of RT_{NDT} for crack initiation is given.

The importance of the pressure (assumed constant in these stylized transients) is shown in Figure 3-4. For axial welds, the critical RT_{NDT} value of 245°F for a 2250 psig transient is increased to 290°F if the pressure can be limited to 500 psig during the time interval of high thermal stresses.

Cladding-no cladding comparisons (Figure 3-4) show a decrease of about 10°F in critical RT_{NDT} when the cladding effect is included.

For this reference transient, with RT_{NDT} = 294°F, the pressure has to be reduced to near saturation within about 30 minutes to avoid crack initiation. However, if the pressure remains constant after an initial drop or monotonically decreases with time for this stylized transient, WPS would be effective at about 18 minutes and crack initiation would not occur after that time. The measured temperatures and pressure experienced in actual overcooling transients (Section 2.2) show ups and downs, some of which would be predicted to negate WPS.

The orientation of postulated cracks affects their behavior during a PTS event. For a specified thermal transient and the same shape and depth of a pre-existing crack, the thermal stress intensity factor for a circumferential orientation is less than that for an axial orientation. The difference is minimal for shallow cracks but becomes significant for deep cracks. The reason for this difference is the relative stiffness of the vessel wall in the two directions which is accounted for in the fracture mechanics analytical model. For typical reactor vessels, the axial and circumferential thermal stresses are essentially equal in magnitude. Axial pressure stresses, on the other hand, are about a factor of two lower than tangential stresses; the axial stresses affecting circumferential cracks and tangential stresses affecting axial cracks. Thus, the total axial PTS stresses are equal to or less than the tangential stresses depending on the system pressure. For the above reasons, circumferential cracks are more tolerant of PTS events.

The difference between the two orientations in terms of critical RT_{NDT} depends on the specific PTS scenario. A limited number of examples described in Appendix D show that for relatively severe postulated transients, the RT_{NDT} difference is about 30°F for crack initiation and the order of 100°F for crack arrest situations. The higher RT_{NDT}s are for circumferential cracks.

Detailed comparison of Westinghouse and NRC calculations show the following sensitivities:

Assumptions	Change in critical value of RT_{NDT}, °F
(a) Cladding vs. no clad stress	10
(b) Continuous flaw for initiation vs. elliptical flaw (a/c = 1/3)	20
(c) h = 300 BTU/hr-ft^2-°F vs. Westinghouse free convection correlation	15

The above tabulated assumption differences account for a total variation of about 45° in critical RT_{NDT} between staff analyses and those of Westinghouse, with the Westinghouse assumptions giving higher values of critical RT_{NDT} than the NRC assumptions. The NRC staff is inclined to accept the Westinghouse assumptions (b) and (c) as more nearly realistic than the NRC staff assumptions, but believes that the cladding effect should be included in accordance with the NRC assumption.

Such "fine tuning" details are relevant to all calculations but are believed by the NRC staff to be within the error band of such calculations. Only for limiting transients like the small break LOCA with loss of loop flow (Section 6 and Appendix G) are these minor corrections important; they are taken into consideration there.

3.4.4 Crack Arrest

For much more severe thermal transients, crack initiation may occur due to high thermal stresses. In this case it is appropriate to consider the potential for crack arrest. Figure 3-2 is a schematic representation of a critical crack depth diagram to illustrate the analytical model used by the staff for determing acceptable arrest criteria. For a small crack, the path of the transient is shown by the dotted line in Figure 3-2. An initial flaw of critical depth is shown; smaller or larger flaws would initiate later. After initiation, the crack runs until $K_I = K_{Ia}$ as shown, then arrests.

Although the K_{Ia} arrest value becomes quite high at larger times, the model in its simple form does not include ductile tearing. For this reason, a maximum allowable value of K_{Ia} is imposed, the "upper shelf" value. For NRC calculations, an upper shelf toughness of 200 ksi $(in.)^{1/2}$ is assumed; however, higher or lower values may be more appropriate for a specific material.

The vessel remains intact if WPS prevent crack initiation or, if a crack initiates, and it arrests, for crack depths where K_I is lower than the upper shelf value.

Since the total stress intensity is the sum of pressure and thermal contributions, if the thermal value is known at the time of WPS, a diagram like Figure 3-2 gives the maximum pressure allowable for crack arrest. When the thermal stress intensity factor is known at the time of WPS, the maximum pressure is determined such that arrest will occur at or before the time of WPS and for crack depths such that K_I is below the upper shelf curve. The limiting case is shown as point "A" in the figure.

3.5 Prediction of Critical RT_{NDT} (RT_c) for Actual Events

For transients that have actually occurred, it is not necessary to make assumptions of the stylized transients of Section 2.1 and the preceding sections of this chapter. Rather it is possible to perform fracture mechanics calculations for the pressure and temperature history as it actually occurred. These calculations were performed assuming a range of RT_{NDT} values, for the eight overcooling transients experienced to date and described in Section 2.2. No credit was taken for "warm prestressing" in these calculations. Thus, it was possible to predict the limiting vessel material condition (critical RT_{NDT} or RT_c) necessary to prevent vessel failure for each of these experienced transients. The results are shown in Table 3.1, for longitudinal cracks, together with results from Section 2.2 of estimating T_f, β, and P for stylized transients to approximate the course of the events actually experienced.

It is seen that, with the TMI exception (where cooldown must stop 16°F above RT_{NDT}) cooldowns that have actually occurred to estimated values of T_f shown in Table 3.1 would not be predicted to fail the vessel unless they (T_fs) were from 10°F to 100°F below the RT_{NDT} of the vessel at the time of the cooldown.

When compared with the results of the stylized procedure presented in Section 3.4.2, which showed that relatively severe cooldown (β=0.15 min^{-1}) should stop 5°F to 20°F above RT_{NDT}, this result shows some of the conservatism generally present in the stylized procedure compared to direct calculations of critical RT_{NDT} for experienced events. Figure 3-3 presents results showing how T_f is related to RT_{NDT} (for deterministic prediction of vessel failure) at two cooldown rates and at any pressure.

Reference:

3.1 Letter (OG-79) form O. Kingsley, Jr., Chairman Westinghouse Owner's Group, to H. R. Denton dated September 16, 1982.

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Table 3.1

Critical RT_{NDT} Values for Experienced Events

Plant and Year	T_f(°F)	β(min^{-1})	P(psig)	RT_c*
TMI	225	0.04	2300	209
HBR "75	250	0.02	500	354
Ginna	325	0.12	1400	378**
Rancho Seco	285	0.10	2300	295
HBR '70	295	0.08	2000	321
HBR '72	340	0.015	1000	381
Crystal River	250(?)	0.10	2300	(250)
Prairie Island	350	0.10	1000	(400)

*RT_c is the RT_{NDT} that is necessry to prevent crack initiation based on actual pressure and temperature variations with time. Stylized values of T_f, β, and P are shown from Section 2 but were not used in these calculations to determine RT_c.

**Applies to circumferential weld. All others apply to axial welds.

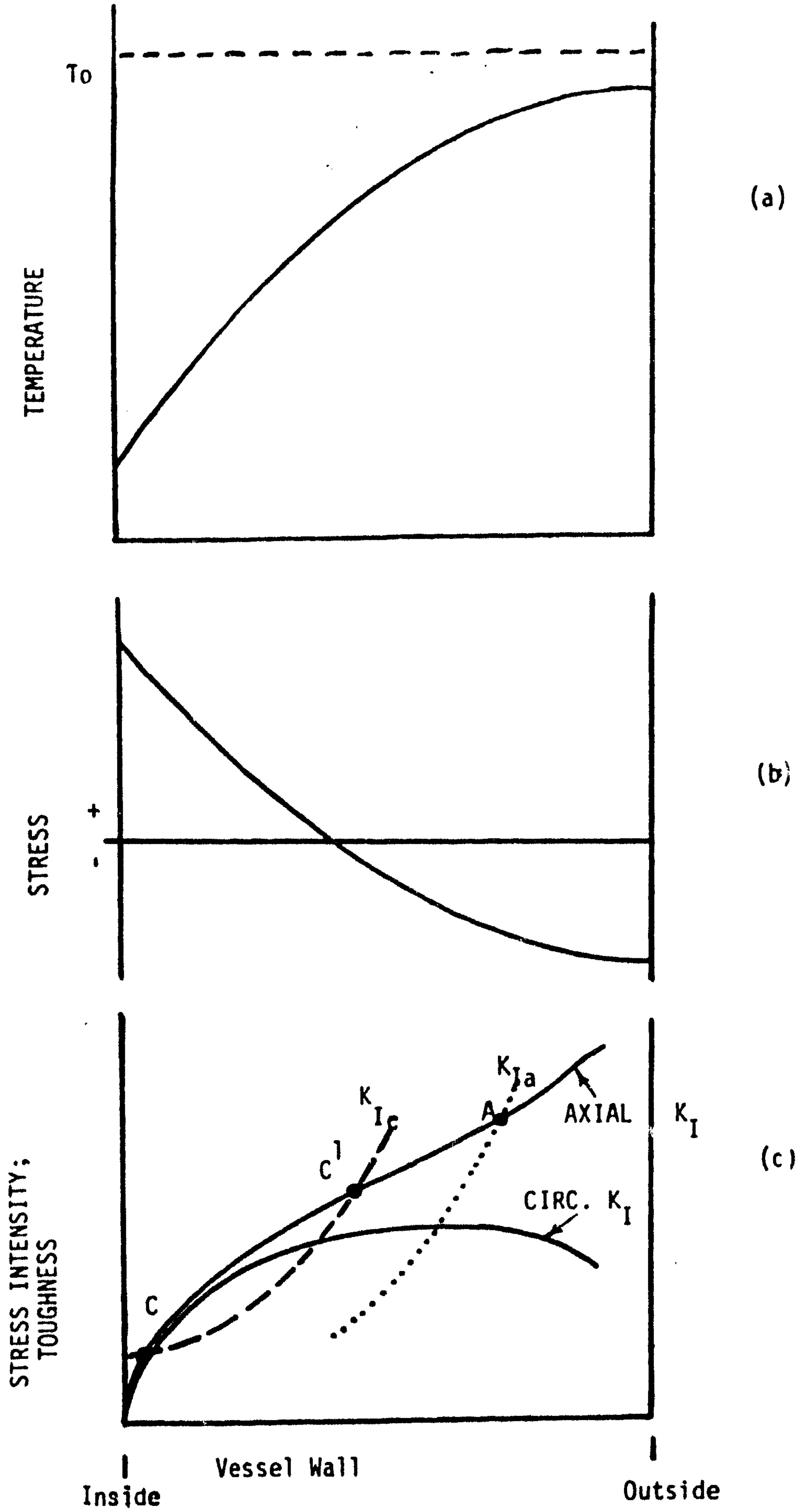


FIGURE 3-1 RESULT OF PTS ON VESSEL WALL

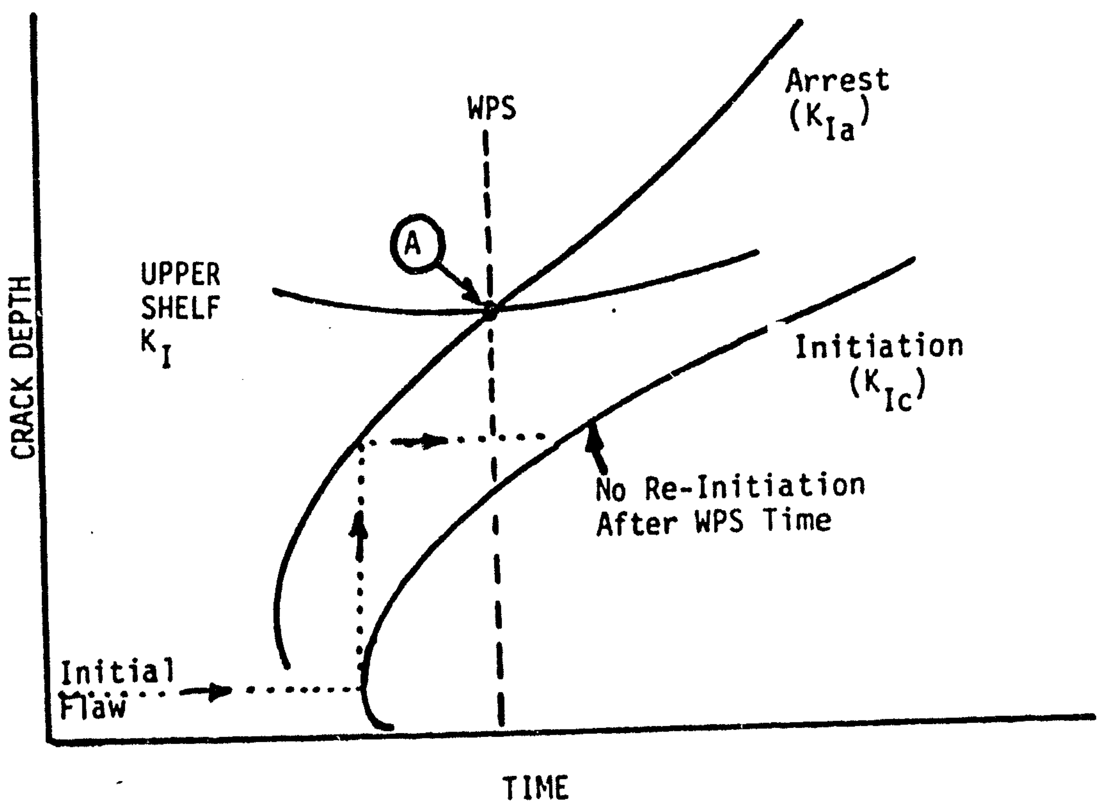
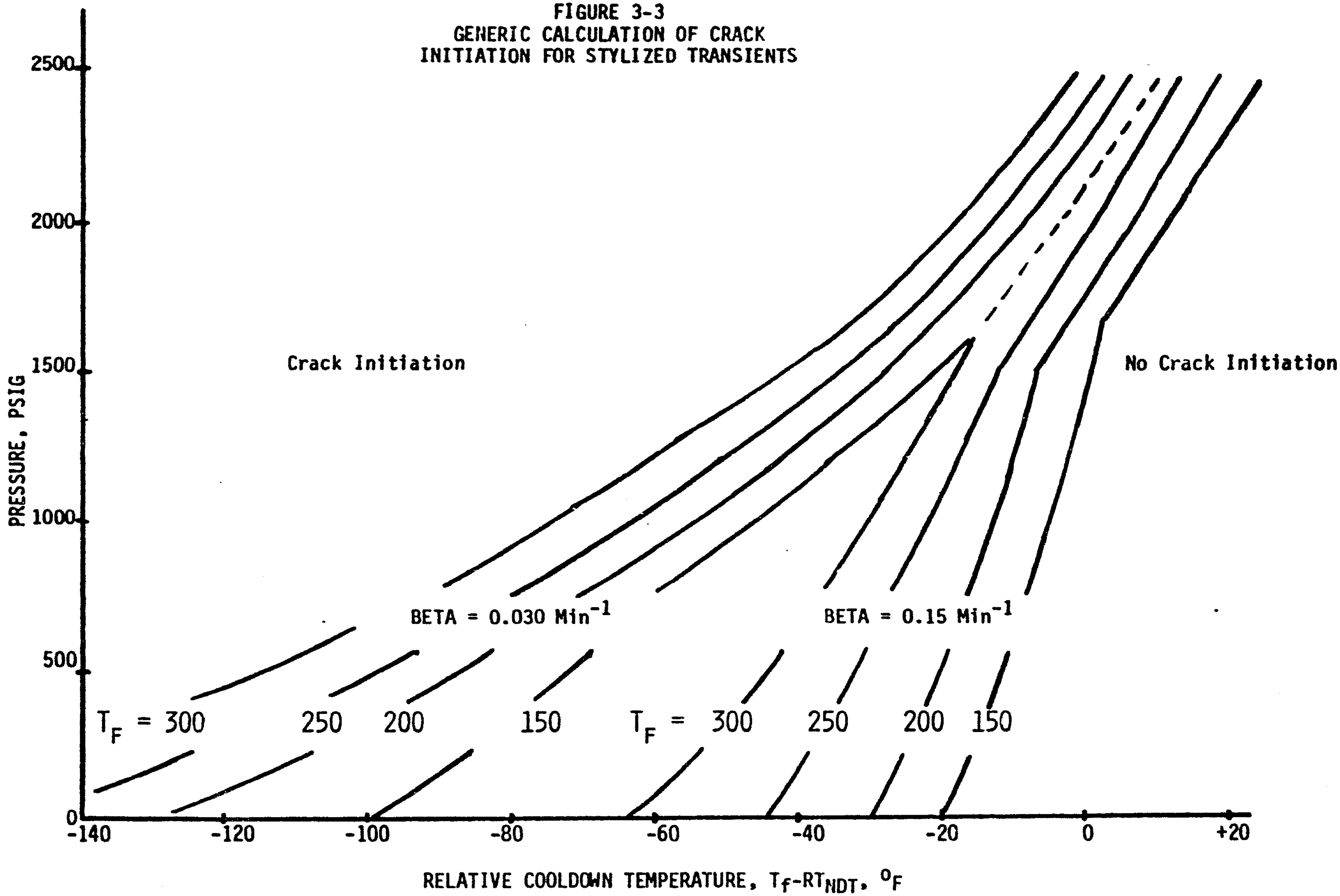


FIGURE 3-2, NRC CRACK INITIATION AND ARREST MODEL

FIGURE 3-3
GENERIC CALCULATION OF CRACK
INITIATION FOR STYLIZED TRANSIENTS



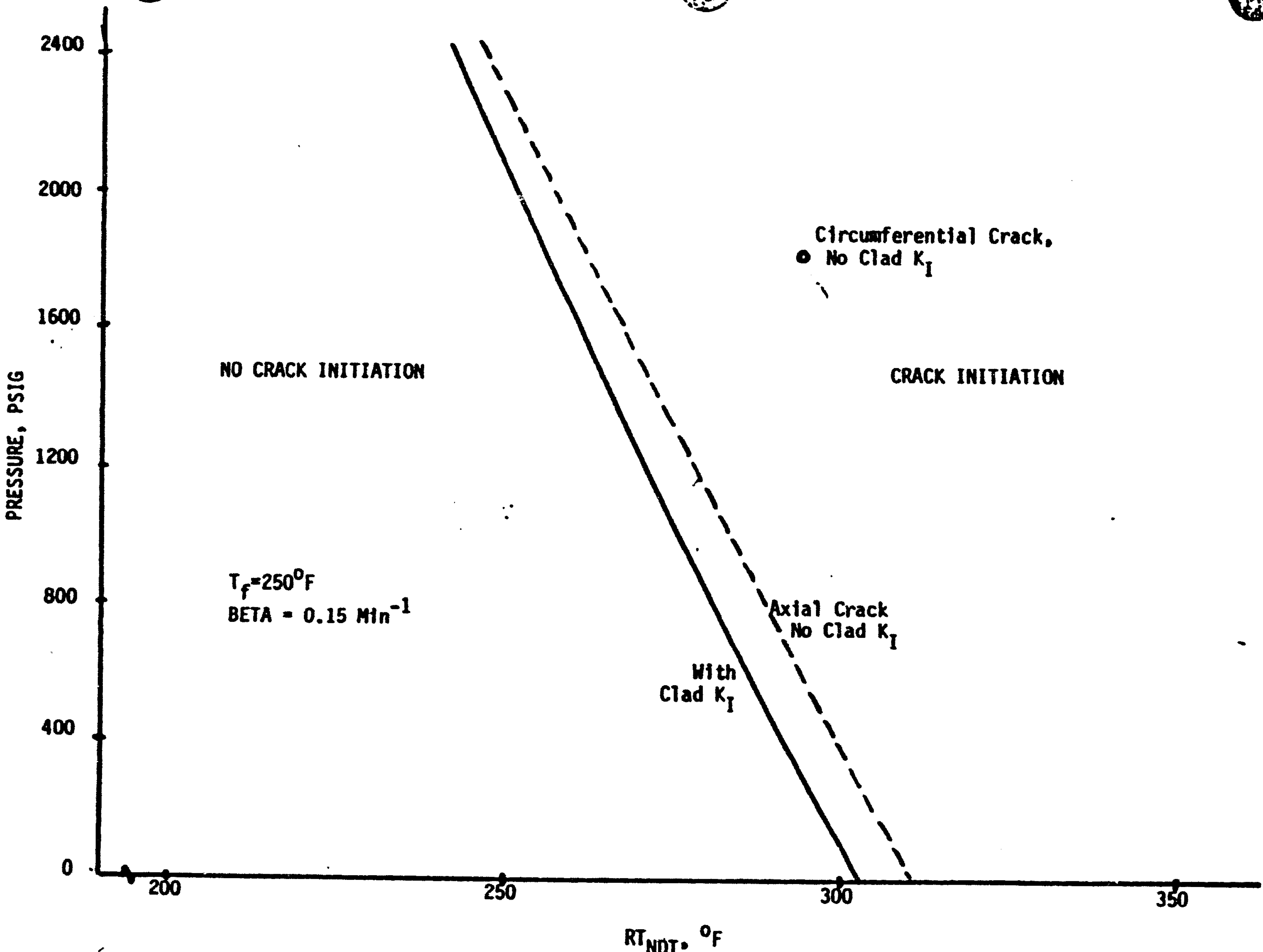
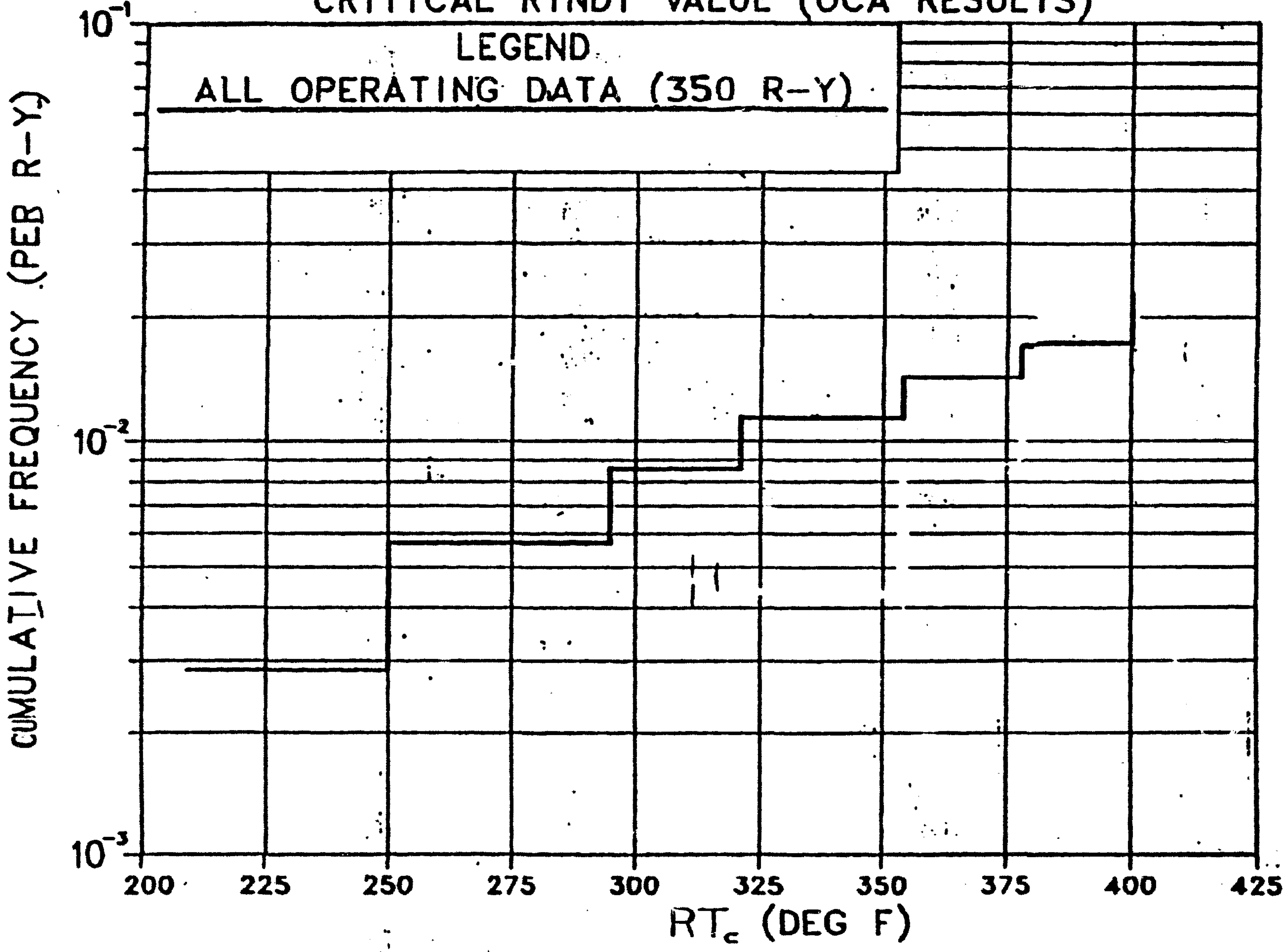
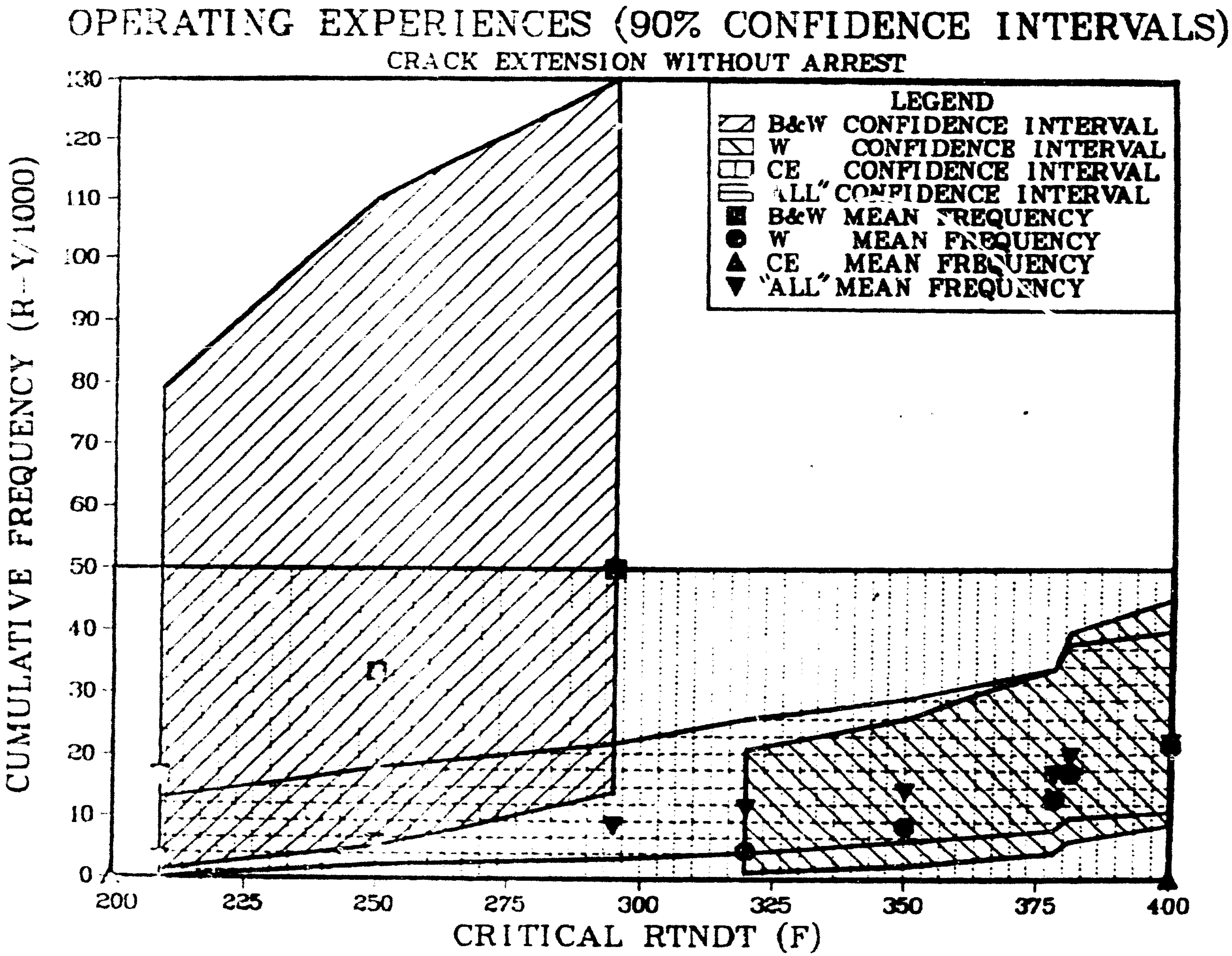


FIGURE 3-4, SENSITIVITY STUDIES

FIGURE 4-1

FREQUENCY BASED ON OPERATING HISTORY
CRITICAL RTNDT VALUE (OCA RESULTS)





OPERATING EXPERIENCES (90% CONFIDENCE INTERVALS)
CRACK EXTENSION WITHOUT ARREST
LEGEND
B&W CONFIDENCE INTERVAL
W CONFIDENCE INTERVAL
CE CONFIDENCE INTERVAL
"ALL" CONFIDENCE INTERVAL
B&W MEAN FREQUENCY
W MEAN FREQUENCY
CE MEAN FREQUENCY
"ALL" MEAN FREQUENCY
CUMULATIVE FREQUENCY (R-Y/1000)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
200
225
250
275
300
325
350
375
400
CRITICAL RTNDT (F)

Figure 4-2

4. SELECTION OF SCREENING CRITERION

The experienced events discussed in Section 3 were used as the basis for selecting a RT_{NDT} screening value, as described in this section.

The events were listed on Table 3.1 in terms of the cool down temperature (T_f, first column) and in terms of the critical RT_{NDT} described in Section 3.5 (RT_C, last column). Based on about 350 total PWR reactor years operating experience in the United States, the T_f values for the eight events can be used to develop a plot of the cumulative frequency per reactor year of events with final temperatures lower than the temperature shown. This was done in Figure 2-14. Similarly, the RT_C results of Table 3.1 were used to develop a plot of the cumulative frequency of events versus the RT_C for which the deterministic fracture mechanics calculations predict crack extension will occur (Figure 4-1).

The value selected, 270°F, for longitudinal cracks, was based on earlier curves similar to Figures 2-14 and 4-1, which gave valves of approximately 260°F and 280°F, respectively, for a nominal event frequency of 10^{-2} per reactor-year. The justification for choosing 10^{-2} is only that this is comfortably lower than the range of "anticipated operating occurrences."

Since the 270°F screening criterion was chosen, the curves of Figures 2-14 and 4-1 have been corrected for earlier errors in the interpretation of the experienced events, so that the 10^{-2} frequency now corresponds to T_f = 285°F and RTc = 320°F. The original 270°F screening criterion is now believed to correspond to a frequency of 9×10^{-3} for T_f and 6×10^{-3} for RT_C. The staff has not readjusted the screening criterion for this change.

The RTc evaluation is, in many ways, a better way to characterize an event than using T_f alone. Calculating RT_C includes the actual time variation of temperature and pressure, and is preferable to the stylized constant pressure and simple exponential temperature behavior approximation inherent in the T_f evaluation of Figure 2-14. Moreover, characterization of events by T_f alone requires neglect of the effect of different pressure and different time decay constants on PTS severity. Therefore, if selecting a frequency of 10^{-2} had any real justification, a screening criterion of 320°F would be given by examination of Figure 4-1.

But in fact using 10^{-2} was only an intuitively attractive place to start. The probabilistic analyses reported in Section 8 of this report show a moderate factor of safety -- a moderate degree of conservatism -- for a screening criterion of 270°F, so the staff has decided not to increase this value as Figure 4-1 and a reference event frequency of 10^{-2} might otherwise suggest.

The staff therefore proposes that RT_{NDT} values of 270°F for axial welds, and 300°F for circumferential welds be used as screening criteria to determine when plant-specific evaluations should be performed for operating plants. It is recognized that the choice of a criterion for action on the basis of generic deterministic fracture mechanics analyses and the limited number of overcooling events that have occurred is subject to many uncertainties and assumptions, some of which are conservative, and some are nonconservative.

The experience base used for these evaluations comprises the eight overcooling events experienced so far in U.S. power plants (Section 2). Non-U.S. experience (and non-U.S. reactor-years) have not been included for lack of detailed knowledge.

U.S. experience has been aggregated for the three PWR reactor manufacturers and various balance-of-plant designers, thus neglecting any differences in experience that may exist among the different designs. The data, however, suggests that B&W plants have a history of more frequent and more severe PTS precursor events. Figure 4-2 shows that the difference appears to have statistical significance. On the other hand, the primary cause of two of the three B&W events discussed in Section 2 of the staff PTS report has been corrected. Nevertheless, the staff believes that although at present the same screening criterion should be used for all plants, near-term further justification is needed from the B&W owners for longer term use of this criterion for B&W

5. DETERMINATION OF RT_{NDT}

5.1 Introduction

If as is recommended in this report, a value of RT_{NDT} is selected to serve as a screening criterion to determine the timing of plant specific evaluations of possible needed modifications to provide protection against pressurized thermal shock events, then it is important for the staff to select a suitably conservative and uniform method for determining the plant-specific values of RT_{NDT} at a given time. During the service life of the reactor vessel the intitial value of RT_{NDT} (RT_{NDTo}) increases because of neutron irradiation by an amount ΔRT_{NDT} which depends on fluence and materials properties. The initial value, RT_{NDTo}, is determined from materials tests made at the time the vessel was fabricated. The change, ΔRT_{NDT}, is determined from fluence measurements and calculations and from trend curves, based on tests of irradiated specimens, that predict the effects of neutron irradiation. There are a number of uncertainties in the estimation of both RT_{NDTo} and ΔRT_{NDT}, and it is important to establish a prescribed method for calculation with a degree of conservatism appropriate for use in connection with the screening criterion. The methods described in this section were selected based on the recommendations of an NRC Working Group of staff members and consultants (Reference 5.1). The methods and the bases for them are presented in greater detail in Appendix E of this report. The uncertainties in estimates of fluence are discussed in Appendix F.

5.2 Estimation of Initial RT_{NDT}

The summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code contained the first requirements for measurements to be made at the time of fabrication of RT_{NDT} for the plates, forgings, and welds that make up the reactor vessel. Two types of tests are required--drop weight tests and Charpy tests. However, most of the vessels of concern regarding PTS were fabricated in the 1960's when only Charpy tests were required.

Typically, the data available comprise three Charpy tests at 10°F for each plate, forging and weld, complete Charpy curves for the surveillance weld and base materials, and in cases where the base material was controlling, some drop weight data on archive or surveillance material. In the past, the NRC has used the guidelines given in the Standard Review Plan Branch Technical Position MTEB 5-2, to obtain an estimate of initial RT_{NDT}. In summary, those guidelines were to use the temperature corresponding to a Charpy 30 ft-lb level, but not lower than 0°F. The Charpy curves from the surveillance tests were used to guide any extrapolation needed to get the 30 ft-lb temperature from the three test results at +10°F. Such estimates are not very satisfactory, however. They are overly conservative for some cases.

For compilations of data obtained subsequent to the time the vessels in question were made, it is possible to divide the welds into two groups according to the weld flux used, and to develop a mean value and a standard deviation (sigma) for the generic data. One must then decide if it is prudent to use the mean generic value as the best estimate for the vessel welds in question. Except for some archive material, the welds that are represented in the data based were made at a later time than the vessel welds. There may have been some differences in weld chemistry or welding practice. Furthermore, even if there were actual RT_{NDT} values for the vessel weld in question, the samples would come from weld metal qualification welds, not from actual vessel weld prolongations and not from full thickness test pieces.

The staff has concluded that a suitably conservative method for estimating the initial value of RT_{NDT} for use in comparisons with the screening criteria proposed in Section 4 is to use the mean value as described above with an adjustment for the standard deviation as discussed in Section 5.4 below. Additional discussion and details regarding the estimation of the initial RT_{NDT} are presented in Appendix E and in Reference 5.1.

5.3 Estimates of the Shift in RT_{NDT} Due to Radiation (ΔRT_{NDT})

Two methods are generally used to estimate the shift in RT_{NDT} caused by neutron irradiation of the pressure vessel: (1) tests of metallurgical surveillance specimens irradiated in the reactor vessel, and (2) "trend curves" of ΔRT_{NDT} as a function of weld chemistry and neutron flux developed from analyses of a large number of irradiated specimens.

Surveillance specimens have been withdrawn and tested for many older operating plants. However, there are problems associated with using individual surveillance results as the sole source of information about a plant. First, the surveillance weld often does not match the critical vessel weld exactly, i.e., the weld wire heat numbers are different. A broader problem is that caused by scatter in the ΔRT_{NDT} data. This results in part from the fact that ΔRT_{NDT} is the difference between the curves for irradiated and unirradiated material, both of which were fitted to data that typically show considerable scatter. Thus, there is a preference for the use of trend curves, instead of individual surveillance data.

Regulatory Guide 1.99 Rev. 1 published in April 1977 contains the procedure recommended at that time by the NRC to obtain ΔRT_{NDT} as a function of chemistry and neutron fluence. Copper was the dominant residual element in the chemistry term (the other was phosphorus).

Critics of Regulatory Guide 1.99 have asserted that (a) the curves are too conservative at high fluences, especially for low-nickel materials, and (b) the phosphorus term is not supported by recent studies such as that of the Metal Properties Council (Reference 5.2) and EPRI (Reference 5.4). Evidence has been accumulating for several years that low-nickel materials are less sensitive to neutron radiation. When the PWR surveillance data base was analyzed by the NRC in October 1981, the difference between high- and low-nickel content material was apparent. Westinghouse and CE reported similar findings and presented empirical equations for the low-nickel material. (B&W has no plants with low-nickel materials in the reactor vessel.)

The PWR surveillance data have now been fitted by a multiple regression analysis technique. The work was done at HEDL by George Guthrie (Ref. 5.3). The Guthrie mean curve is as follows:

$$\Delta RT_{NDT} = (-10 + 470\ Cu + 350\ Cu\ Ni)\ (f/10^{19})^{0.27}$$

where: ΔRT_{NDT} = adjustment of reference temperature, degrees F

Cu = weight percent copper

Ni = weight per nickel

f = fluence, n/cm² (E>1 MeV)

The standard deviation obtained from the analysis is 24°F.

As shown in Appendix E, the Guthrie mean curve has been compared with a mean curve developed by the Metal Properties Council (MPC) for ASTM Committee E-10 on Nuclear Technology and Applications (Ref. 5.2) and EPRI (Reference 5.4). The MPC data based contains all of the test reactor and surveillance data that were available in November 1977, and that fit the criteria for material type and irradiation temperature. There is reasonably good agreement between the MPC trend curves and the Guthrie curves, considering that the MPC curves were for a range of nickel content, but were without a nickel term in the equation.

The MPC trend curve did not contain a phosphorus term, because in the regression analysis the addition of a phosphorus term did not produce any significant decrease in the residual variance. In a further study of this finding, the MPC Task Group found a statistically significant relationship of phosphorus content to copper content, i.e., high phosphorus was found with high copper. Thus, their combined effects were represented in the MPC trend curve formulation by a copper term alone.

For high values of copper and nickel contents, the Guthrie mean curve described above gives values higher than those predicted by the part of the Upper Limit Curve of RG 1.99, given by the equation:

$$\Delta RT_{NDT} = 283\ (f/10^{19})^{0.194}$$

Experience has shown that the latter equation bounds the available data.

Therefore, in developing the method for estimating RT_{NDT} values to be compared with the screening criteria proposed in Section 4, the staff recommends that ΔRT_{NDT} be calculated using a combination of the Guthrie mean curve and the RG 1.99 upper bound curve, with adjustments for the standard deviation as discussed in Section 5.4.

5.4 Recommended Method for Calculation of RT_{NDT}

An NRC Working Group of staff members and consultants reviewed the available information regarding RT_{NDT} determinations and recommended that the following method for calculating RT_{NDT} values for specific reactor vessels be used for comparison with the screening criteria of Section 4 (Ref. 5-1).

The value of RT_{NDT} at the inside surface of the vessel should be taken as the lesser of:

$$RT_{NDT} = RT_{NDT}(0) + \Delta RT_{NDT}\,(\text{mean}) + (\sigma^2 + \sigma^2)^{\frac{1}{2}}$$

$$RT_{NDT} = RT_{NDT}(0) + \Delta RT_{NDT}(\text{RG}) + 2\sigma_o$$

where: $RT_{NDT}(0)$ = the mean value of the initial RT_{NDT} determined as described in Section 5.2 above and in Appendix E.

$\Delta RT_{NDT}(\text{mean})$ = the mean value of RT_{NDT} based on the Guthrie trend curve

$$= (-10 + 470\ \text{Cu} + 350\ \text{CuNi})\ (f/10^{19})^{0.27}$$

$\Delta RT_{NDT}(\text{RG})$ = the upper bound curve of Regulatory Guide 1.99 for high values of copper fluence extended to low fluence if necessary

$$= 283\ (f/10^{19})^{0.194}$$

σ_o = the standard deviation value from the $RT_{NDT}(0)$ analysis (see detailed discussion in Appendix E)

σ_Δ = the standard deviation for the Guthrie mean curve = 24°F

Cu = weight percent

Ni = weight percent nickel

and f = fluence, n/cm^2 (E>1 MeV) (See discussion of fluence uncertainty in Appendix F.)

Note that the second of the two equations above does not include a standard deviation term for $\Delta RT_{NDT}(RG)$ since the Regulatory Guide term used is an upper bound equation.

This formulation is plotted in Figure 5-1 for three values of copper content and a nickel content of 1%.

5.5 Plant-Specific RT_{NDT} Valves

The results of applying the above methodology to all operating PWRs to calculate RT_{NDT} now, projected three effective-full-power-years into the future, are presented in Appendix P, Table P-1.

REFERENCES

5.1 Report of the Working Group on RT_{NDT}, Memo, M. Vagins to S. Hanauer, August 30, 1982.

5.2 Prediction of the Shift in the Brittle-Ductile Transition Temperature of LWR Pressure Vessel Materials, Edited by J. J. Koziol, Prepared by a Task Group of Subcommittee 6 of the Metal Properties Council, April 6, 1981 as a Report to ASTM Committee E10. To be published by ASTM.

5.3 LWR Pressure Vessel Irradiation Surveillance Dosimetry Quarterly Progress Report, Jan.-Mar. 1982, HEDL-TME 82-18. To be published.

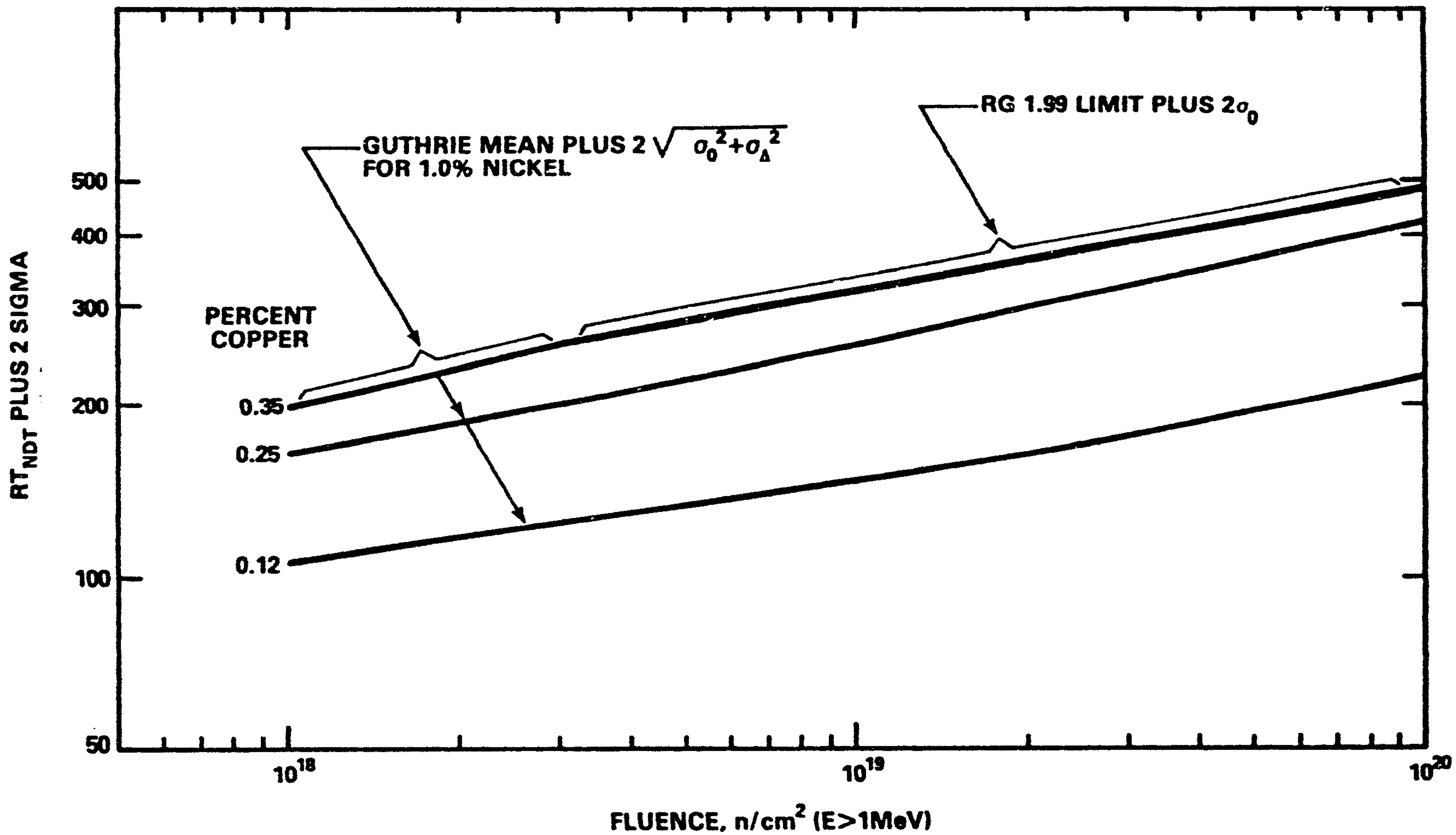


FIGURE 5-1: EXAMPLE OF NRC PRESCRIPTION FOR RT_{NDT} (FOR ASSUMED $RT_{NDT}(0)$ = 0°F)

6. CONSIDERATION OF LOW FREQUENCY EVENTS

6.1 Identification of Event Sequences with Significant PTS Risk

In order to determine the potential significance of challenges to reactor vessel integrity due to pressurized thermal shock from low frequency events, a systematic approach which identifies all relevant sequences of single and multiple failures from all pertinent initiating events is needed. Event tree techniques are an orderly approach for performing this quantification. Such a study using probabilistic methods and mostly existing PRA data bases was performed by Westinghouse for the Westinghouse Owners Group (WOG). The description and results of this study were submitted to the NRC by the WOG as Reference 6.1. The staff accepts the methodology used in the study to identify event sequences which contribute to risk from pressurized thermal shock and important portions of the discussion presented below have been adapted from Reference 6.1. Although there is agreement with WOG on the structure of the events that should be considered, the staff differs with the WOG in the resulting frequencies for many of the event sequences significant to PTS.

"The approach taken is to first identify the set of all the initiating transients or events which either by themselves or along with succeeding failures could lead to potential challenges to vessel integrity. The sequence of accompanying branching chains of events including component failures and their probabilities is logically traced out in the event trees. The output of the event tree is a set of end states and their frequencies. These end states can then be evaluated for potential challenges to the vessel from pressurized thermal shock. The sum of the frequencies of the end states which are potential challenges is the total frequency per reactor year of vessel integrity challenges summed over all types of initiating events" (Reference 6.1).

Initiating events used in this study, based on the WOG PRA, are presented in Table 6.1 and include those which either directly or through consequential failures may lead to PTS. These events are the same as those used in recent risk studies. The first eight of these initiating events do not in themselves result in PTS, however, consequential events postulated as a result of these first eight initiators do result in transients with PTS. The events which could lead directly to a PTS challenge are small LOCAs, excessive feedwater, steamline rupture, and steam generator tube rupture. Consequential failures for these initiators can also enhance their seriousness as a cooldown challenge.

We believe that the WOG study has been sufficiently general and thorough to identify the event sequences of greatest significance to PTS risk. For the purposes of this study, we have adopted the significant event sequences that they have identified in modified form after staff review. All of the significant event sequences have been characterized by the staff with respect to frequency per reactor year, the temperature reciprocal time constant, β, and the final reactor coolant system temperature at the pressure vessel wall. The staff review and evaluation of these event sequences included important changes to the initiating and consequential event frequencies based on the staff's PRA studies including Reference 6.2. Some changes in the reciprocal time constant and the final reactor coolant system temperature at the pressure vessel wall were also based on what we believe to be better themal hydraulic analyses for some of the events considered. The bases for our selection of event frequencies, reciprocal time constants and final reactor coolant system temperatures and how they differ from the values presented by the WOG are further discussed in Appendix G. The event sequences determined by the WOG as reviewed and evaluated by the staff are further addressed in separate categories below.

We have chosen to apply the WOG PRA results to all three PWR vendors' plants for purposes of recommending the generic screening criteria. This is because: (1) the WOG results are the most complete available; (2) we believe that the PTS risk at all three vendors's plants is acceptably covered by meeting those recommended criteria; and (3) if there are significant differences in PTS risk at the different vendors' plants, these differences will be identified in the plant-specific analyses that are required when a plant "trips" the screening criteria, and account will be taken of those differences before corrective actions resulting from those analyses are required.

The design of the Combustion Engineering (CE) nuclear plants is sufficiently similar to Westinghouse plants to conclude that the event sequences identified below and based on the WOG study are applicable to CE plants. Babcock and Wilcox (B&W) plants which all include the Once Through Steam Generator (OTSG) and the Integrated Control System (ICS) are known to behave somewhat differently in responding to transients. One of the conclusions of a task force which studied and reported on the transient response of B&W designed reactors was that the B&W plants are more susceptible to overcooling transients (Reference 6.3). For the B&W OTSG the effective heat transfer area increase rapidly with overfeed since the water level which is normally at about two thirds the height of the tubes increases, and thereby substitutes boiling heat transfer surface for steam superheating surface causing more rapid cooling. Also, with overfeed in the OTSG the average secondary system water temperature is reduced more quickly because of the smaller OTSG water inventory causing more rapid cooling.

A compensating factor, with respect to severity of overcooling transients, is the presence of vent valves in the pressure vessel of B&W reactors. Although vent valves do not reduce the frequency of overcooling transients they do reduce minimum temperature obtained in many overcooling transients by allowing mixing of the warmer core exit water with the incoming cooler cold leg flow in the vessel downcomer.

B&W plants are now in the process of installing equipment to reduce sensitivity to overcooling. Modifications which should be completed in 1983 and 1984 include:

- Main feedwater overfill protection
- Auxiliary feedwater (AFW) overfill protection
- Cavitating venturies in AFW lines (TMI-1)

- Programmed AFW fill rate

- Safety grade AFW system with automatic actuation

- AFW flow indication in the control room

Evaluations by the Argonne National Laboratory and the NRC staff based on evaluations of B&W plant operating history and plant analysis with the proposed changes concluded that the likelihood of overcooling will be greatly reduced by the above changes.

For the purposes of this report we are using the event sequences based on the WOG study as modified by the staff and discussed below as being applicable to the B&W plants. It would be desirable to verify this approach by a study of PTS event sequences for B&W plants specifically.

6.2 Characterization of Specific Groups of Event Sequences Identified as Contributors to PTS Risk

As a result of the above more general approach to the problem of identifying event sequences reported in Reference 6.1, we believe that the events of significance to the PTS issue have been identified as secondary (steam side) depressurization, small-break loss-of-coolant accidents, and steam generator tube ruptures (special case of small-break LOCA). In order to characterize individual event sequences within each of these groups, certain additional parameters have been identified which determine the significance of these sequences as a PTS challenge.

The level of decay heat present during an initiating event is an important parameter in the cooldown from a given transient. The level of decay heat is related principally to the operational history (full power operation, hot zero power, other) immediately preceding the transient. The frequencies of challenging event sequences are thus differentiated by the operational status of the plant.

The time allowed for initiation of proper operator action is another parameter that is important in some sequences. This variable has been used as a parameter in the results which characterize certain sequences that are presented below.

6.2.1 Secondary (steam side) Depressurizations

This group of cooldown events which involves some type of opening of the steam system includes steamline rupture of all sizes, inadvertent safety relief valve open to atmosphere, inadvertent steam dump valve open to the condenser, reactor trip without turbine trip, or operator error which results in any of these malfunctions. The transient is characterized by a rapid cooldown of the primary coolant system with shrinkage and consequential rapid depressurization until safety injection is actuated providing additional cooling and eventual primary repressurization. Natural circulation and, therefore, good mixing conditions are maintained in this transient for greater than 30 minutues unless low decay heat levels exist.

Parameters which are important with respect to severity of reactor coolant system cooldowns are (1) plant operational status (decay heat level), (2) operator action time to isolate auxiliary feedwater flow, (3) break size, (4) reactor coolant pump operation, and (5) location of the depressurization opening with respect to the main steam isolation valves.

Main Steam Line Break (MSLB)

An MSLB with break area larger than 6 inches equivalent diameter results in a rapid cooldown of the primary system. The final temperature can be as low as around 200°F, depending on the plant operating status (decay heat level) and operator action to terminate auxiliary feedwater. The system will repressurize as a result of safety injection and may reach a pressure in excess of 2000 psig depending on when operator action is taken to terminate safety injection. The MSLB results in a signal to close the main steam isolation valves (MSIVs) so that leaks downstream of the MSIV are bounded by leaks upstream of the MSIVs. The event frequency, reciprocal time constant, β, and final reactor coolant system temperature for the parameters of initial power and time for operator action to isolate feedwater are presented in Table 6.2.

The staff results presented in Table 6.2 show that the frequency of this event for hot full-power conditions is significant to PTS risk whereas the frequency for hot full-power conditions determined by the WOG study is extremely low. The staff's final reactor coolant system temperature for this transient is, however, much higher than the WOG result based on what we believe to be more realistic thermal hydraulic analyses for this transient.

Small Steam Line Break (SSLB) or Stuck Open Steam Generator Safety/Relief Valve

The SSLB or stuck open SG safety/relief valve can result in an overcooling transient similar to the MSLB but of longer duration due to the smaller break size. This event has a much higher frequency than the MSLB. The event frequency, reciprocal time constant, β, and final reactor coolant system temperature for the parameters of initial power and time for operator action to isolate feedwater are presented in Table 6.3. The staff's results indicate a somewhat higher frequency for this event than the WOG study.

6.2.2 Small-Break Loss-of-Coolant Accident (SBLOCA)

The cooldown transient from an SBLOCA of the reactor coolant system includes reactor coolant pump seals, primary power-operated relief valve or safety valve failure or leakage as well as actual piping breaks of various sizes in hot or cold legs. For breaks less than a critical break size of about two inches (equivalent diameter, i.e., equal in area to a circle of this diameter), natural circulation is maintained and mixing occurs and the resulting cooldown rates are not expected to exceed Appendix G limits of less than 100°F per hour. Both the reactor coolant pump seal leak (break equivalent to 0.5 inches) and stuck-open power-operated relief valve (break equivalent to 1.4 inches) are included in that category.

Break sizes greater than two inches up to six inches are also included as SBLOCAs. For these breaks, the safety injection flow is less than the break flow, resulting in a net mass loss from the primary system. Loop flow (natural circulation) can be lost for this range of breaks, resulting in a rapid cooldown due to the cold safety injection flow. The exact break size where loss of flow ocurs is dependent on the safety injection flow rate (and makeup flowrate), the break location, the decay heat level, and the SG (heat sink) performance. Because of the stagnation of flow, mixing of the safety injection water is poor and rapid cooldown of the vessel could result.

We have reviewed the frequency of events that may result in stagnated loop conditions such as SBLOCAs in the 2 to 6 inch equivalent diameter range. There are several small diameter pipes in the range of 2 to 4 inches connected to bypass lines, pressurizer spray lines, power-operated relief valve lines, and the main primary system piping. These include charging and letdown lines, RTD safety injection lines. SBLOCA events in the 2- to 6-inch range are dominated by non-isolatable breaks and, therefore, operator action is not a major parameter. The event frequency, reciprocal time constant, β, and final reactor coolant system temperature are presented in Table 6.4. The final temperature selected is sensitive to the mixing model assumed, control volumes, etc. A discussion is presented in Appendix G to this report (G.2.3, NRC Staff Characterization of SBLOCA) and in Table G.5 and References G.8 through G.11.

As discussed above, these LOCAs can be differentiated by breaks smaller than about 2 inches where loop circulation is maintained (and good mixing of the cold safety injection water is therefore achieved) and breaks larger than about 2 inches where loop circulation is lost (and poor mixing of safety injection water results). The WOG judged LOCAs with effective diameters greater than 2 inches to have a negligible probability of causing vessel failure, so they only included LOCAs with breaks less than two inches. We agree that breaks in the size range less than two inches have small β's and appear similar to slightly accelerated shutdown transients where the operator can be expected to control the pressure.

The significance of breaks in the 2- to 6-inch range to PTS risk has been separately analyzed. Fracture mechanics analyses performed with a more exact representation of this cooldown transient show that the PTS risk from this transient is less than could be anticipated and consistent with the screening criteria proposed (see Section 8).

We have also considered other events that could lead to extended HPSI operation with stagnated loop(s) conditions and subsequently repressurize. These conditions are similar to those for a SBLOCA in the 2- to 6-inch range except that they may repressurize to >2000 psi. The potential sequences discussed in Appendix G appear to be dominated by a stuck-open safety valve that subsequently recloses. The event frequency, reciprocal time constant, β, and final reactor coolant system temperature are presented in Table 6.4.

6.2.3 Steam Generator Tube Rupture

The response of the reactor coolant system to a variety of steam generator tube failure events, up to the complete severance of a single tube, has been analyzed in Reference 6-1. These analyses simulate automatic protection systems, such as reactor trip and emergency core cooling systems, as well as operator actions. A steam generator tube failure should not result in a rapid cooldown of the primary system or excessively high reactor coolant system pressures if current plant operating procedures are used. In general, natural circulation should develop in all primary loops and mix with incoming safety injection flow to preclude local temperature depressions if RCPs are stopped. However, the subsequent operator actions to terminate primary-to-secondary leakage may rapidly cool the reactor coolant system for short periods and may stagnate the faulted loop. In that case, local temperature depressions due to continued safety injection flow may occur. The period of this temperature depressions is expected to be short and should not represent a significant PTS challenge to vessel integrity. The event frequency, reciprocal time constant, β, and final reactor coolant system temperature are presented in Table 6.5. The frequency for this event estimated by the staff is significantly greater than that determined by the WOG.

6.3 Low Frequency Event Sequences Contributing to PTS Risk

Tables 6.2, 6.3, 6.4, and 6.5 summarize the groups of postulated event sequences that appear to be significant PTS initiators in terms of their estimated expected frequency, and the parameters for a stylized transient (T_f, β and P). Calculations also have been made of typical temperature and pressure behavior for each class of events. In principle, each of the sequences could be used to perform deterministic fracture mechanics calculations for a range of vessel RT_{NDT}

values to determine the limiting value, RT_c, necessary to prevent crack initiation for each type of event, as was done for actually experienced events in Section 3.5 above. However, such calculations have not been made. Alternatively, the data of Tables 6.2, 6.3, 6.4, and 6.5 can be used to construct a cumulative frequency versus T_f distribution similar to that done for experienced events in Figure 2-14. This distribution is shown in Figure 6-1. In Section 3.5 above, it is shown that based on the deterministic fracture mechanics parametric studies, for relatively fast cooldown event ($\beta = 0.15$ min^{-1}) with final temperatures in the 250-300°F range and high system pressures (~2300 psig), crack initiation (in longitudinal welds) is not predicted if T_f is about 5-10°F higher than RT_{NDT}. Thus, the distribution curve of Figure 6-1 suggests that vessels with an RT_{NDT} of 270°F (the suggested screening criterion discussed in Section 4) would not be expected to experience longitudinal crack extension for events with frequencies greater than about 6 x 10^{-3} per reactor-year. This conclusion is similar to that obtained in Section 4 considering actual experienced events.

However, the frequency distribution of Figure 6-1 extends to low frequency events with low values of T_f. This low frequency "tail" on the distribution indicates that there are postulated events with estimated frequencies as high as 10^{-4} per reactor-year for which the final temperature is substantially below 270°F that must be considered. This is discussed in Section 8.

The discussion above is subject to the same large uncertainties as are described in Section 4 above. To gain additional insights into the conservatisms in the deterministic fracture mechanics treatment and to gain some notion of the risk of vessel failure considering low probability events, the data of Tables 6.2, 6.3, 6.4, and 6.5 are used in combination with a probabilistic treatment of fracture mechanics described in Section 7. The results are discussed in Section 8.

REFERENCES - SECTION 6

6.1 "Summary of Evaluations Related to Reactor Vessel Integrity Performed for the Westinghouse Owners Group," Westinghouse Electric Corporation, Nuclear Technology Division, May 1982.

6.2 "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014) October 1975.

6.3 "Transient Response of Babcock and Wilcox Designed Reactors" NUREG-0667, May 1980.

Table 6.1 Initiating events

EVENT		FREQUENCY PER REACTOR-YEAR
1.	Loss of Main Feedwater (LOFW)	3.41
2.	Closure of One Main Steam Isolation Valve (MSIV)	6.00×10^{-1}
3.	Loss of Primary Flow (LOPF)	3.21×10^{-1}
4.	Core Power Increase (POWIN)	4.77×10^{-2}
5.	Turbine Trip (TT)	4.00
6.	Spurious Safety Injection Activation (SSI)	1.59×10^{-1}
7.	Reactor Trip (RT)	4.11
8.	Turbine Trip Due to Loss of Offsite Power (TT/Loop)	1.01×10^{-3}
9.	Steam Generator Tube Rupture (SGTR)	3.92×10^{-2}
10.	Small LOCA, <1.5-inch diameter (LOCA-1)	9.07×10^{-3}
11.	Small LOCA, >1.5-inch diameter (LOCA-2)	6.11×10^{-4}
12.	Large LOCA, >6-inch diameter (LOCA-3)	3.88×10^{-4}
13.	Excessive Main Feedwater (EX FW)	2.50×10^{-1}
14.	Steamline Rupture Inside Containment (STM BRK In)	3.88×10^{-4}
15.	Steam Rupture Outside Containment (STM BRK OUT)	3.87×10^{-2}

Table 6.2 Event parameters for the main steam line break (MSLB)

Plant Status	Hot Full Power				Hot Zero Power			
Operator isolates auxiliary feedwater min.	0-5	5-10	10-30	30-60	0-5	5-10	10-30	30-60
Event frequency per reactor-year	8×10^{-5}	6×10^{-5}	1.5×10^{-5}	3×10^{-7}	8.5×10^{-6}	6.8×10^{-6}	1.6×10^{-6}	3×10^{-8}
Reciprocal time constant, β min^{-1}	0.4	0.2	0.09	0.09	0.4	0.2	0.2	0.2
Final reactor coolant system temperature at vessel wall, °F	450	390	300	250	212	212	210	190

Event frequency (MSLB) = $\sim 1.7 \times 10^{-4}$ per reactor-year

Table 6.3 Event parameters for the small steam line break (SSLB)

Plant Status	Hot Full Power					Hot Zero Power				
Operator isolates auxiliary feedwater min.	0-5	5-10	10-20	20-30	30-60	0-5	5-10	10-20	20-30	30-60
Event frequency per reactor-year	4.5×10^{-3}	3.6×10^{-3}	8.10^{-4}	6.3×10^{-5}	1.8×10^{-5}	4.5×10^{-4}	3.6×10^{-4}	8×10^{-5}	6.3×10^{-6}	1.8×10^{-6}
Reciprocal time constant, β min^{-1}	0.4	0.2	0.1	0.06	0.06	0.4	0.2	0.1	0.06	0.06
Final reactor coolant system temperature at vessel wall, °F	385	320	250	220	200	375	310	235	200	175

Event frequency (SSLB) = $\sim 10^{-2}$ per reactor-year

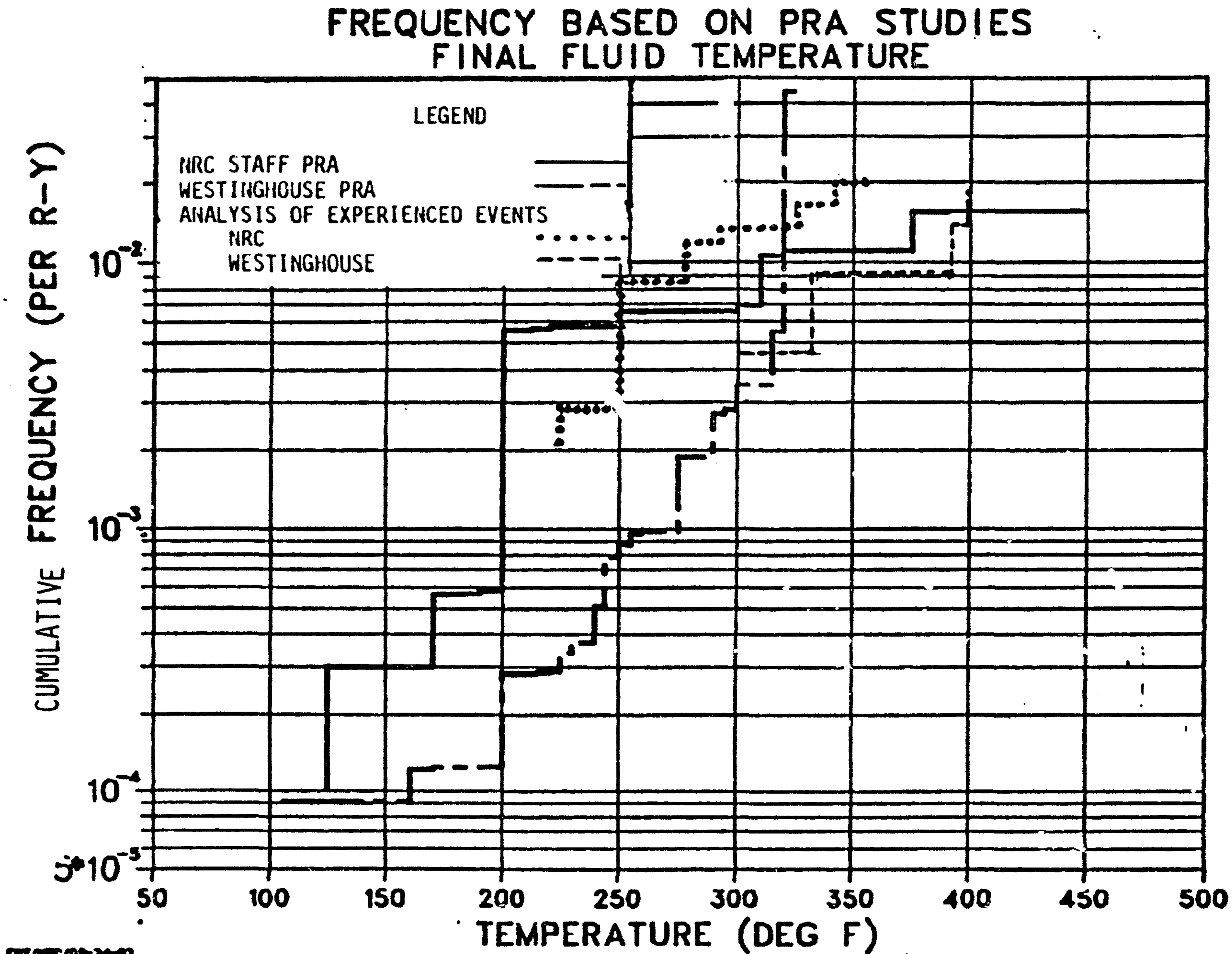

FREQUENCY BASED ON PRA STUDIES
FINAL FLUID TEMPERATURE
LEGEND
NRC STAFF PRA
WESTINGHOUSE PRA
ANALYSIS OF EXPERIENCED EVENTS
NRC
WESTINGHOUSE
CUMULATIVE FREQUENCY (PER R-Y)
10^{-2}
10^{-3}
10^{-4}
$5*10^{-5}$
50
100
150
200
250
300
350
400
450
500
TEMPERATURE (DEG F)

FIGURE 6-1

Table 6.4 Event parameters for the small-break loss-of-coolant accident (SBLOCA); and extended HPSI operation with a stagnated loop and high pressure extended HPSI operation

	SBLOCA	Operation
Event frequency per reactor-year	1.5×10^{-3}	10^{-4}
Reciprocal time constant, β, min^{-1}	0.05	0.05
Final reactor coolant system temperature at vessel wall, °F	125	125

Table 6.5 Event parameters for steam gener- or tube rupture

Without steam line break or stuck-open SRV		
Event frequency per reactor-year	5×10^{-3}	
Reciprocal time constant, β, min^{-1}	0.04	
Final reactor coolant system temperature at vessel wall, °F	200	
With steam line break or stuck-open SRV		
	Stuck open SRV outside containment	SLB inside containment*
Event frequency per reactor-year	2×10^{-4}	1×10^{-5}
Reciprocal time constant, β min^{-1}	0.04	0.04
Final reactor coolant system temperature at vessel wall, °F	170	170

*Excluded from these evaluations

7. PROBABILISTIC TREATMENT OF FRACTURE MECHANICS

7.1 Introduction

The deterministic fracture mechanics analyses discussed in Section 3 assume specific values for all the input parameters necessary to predict crack initiation, growth and/or arrest. However, many of these parameters are not known precisely. In order to quantitatively analyze the effect of a large number of uncertainties, a probabilistic approach can be taken to estimate the failure probability of a reactor pressure vessel. A Vessel Integrity Simulation Analysis (VISA) code was developed to gain insight into reactor pressure vessel failure probability due to pressurized thermal shock. Appendix H discussed in detail the probabilitistic fracture mechanics analysis and the VISA code. A brief description of the VISA code is present in Section 7.2.

7.2 Description of VISA Code

The VISA code consists basically of two parts. The first is a deterministic fracture mechanics analysis for a specified pressure/temperature transient. This analysis is similar to that discussed in detail in Appendix D (D.1) and includes heat transfer, thermal and pressure stress, and applied stress intensity value calculations for a range of crack depths. The second part uses Monte Carlo techniques to assess failure probability based on a very large number of deterministic calculations in which the input parameters are varied.

Certain parameters are treated as random variables, and their values are sampled from a statistical distribution defined as an input to the program. In each calculation, values of the random variables (crack depths, copper content, initial mean value of RT_{NDT}, fluence, critical stress intensity factor, K_{Ic}, and stress intensity factor for crack arrest, K_{Ia}) are selected from the specified probability distributions, and deterministic calculations are made using these values. Each calculation results in one of three outcomes: (1) no crack initiation, (2) crack initiation followed by arrest, or (3) pressure vessel failure.

For each iteration of the simulation, values of fluence, flaw size, and copper content are selected from their respective distributions. The means value of RT_{NDT} at the inner wall is calculated as a function of fluence and copper content. With these values fixed for the iteration, the code steps through the time history of the transient. For each time step, the stress intensity at the crack depth is taken from the deterministic portion of the code. A value of K_{Ic} is simulated to determine fracture initiation. If initiation does not occur, the simulation moves to the next time step. If initiation does occur, the crack is extended 1/4 in., and the crack arrest toughness (K_{Ia}) is simulated. If arrest occurs, the simulation moves to the next time step; if not, the crack is extended another 1/4 in. and a new value of K_{Ia} is simulated. This process is continued until either the vessel fails or the duration of the transient is reached. Each pass through the simulation loop represents a single computer calculation conducted to determine if RPV failure will occur. Up to a million passes through this loop can be made. The code keeps track of the number of crack initiations and RPV failures. The probabilities of crack initiation and RPV failure then are estimated by dividing these values by the total number of trails. Thus, the VISA code actually performs millions of deterministic calculations with each set of calculations based on a different set of values selected from the appropriate statistical distributions for the significant variables. This is the calculational equivalent to subjecting a population of up to a million operating reactor pressure vessels to the pressurized thermal shock transient of interest and then inferring the failure probability based on the number of observed failures.

7.3 Probabilistic Fracture Mechanics Sensitivity Studies

Section 3 and Appendix D of this report discuss the sensitivity of crack initiation and vessel failure to the various PTS parameters. This section discusses the same sensitivities based on probabilistic fracture mechanics. Results are portrayed in Figures 7-1,7-2, 7-3 and 7-4 for the stylized thermal transient discussed in Section 2 above:

$$T_w = T_f + (T_o - T_f)e^{-\beta t}$$

Figure 7-1 illustrates the sensitivity of the conditional vessel failure probability to T_f minus mean value of RT_{NDT}, for three values of β and a pressure of 1000 psig. A total of 225 cases were run, using three values of T_f -- 150, 225 and 300°F. It is seen that the relative risk is low for mean value of RT_{NDT} less than T_f, but if cooldown drops temperature below the vessel mean value of RT_{NDT}, then the risk rises quite rapidly.

Figure 7-2 illustrates the sensitivity of conditional failure probability to pressure for several values of T_f minus mean value of RT_{NDT} and a β of 0.15 reciprocal minutes.

Figure 7-3 illustrates the sensitivity to the decay parameter, β, with several values of T_f minus mean value of a RT_{NDT} and the other parameters held constant.

Figure7-4 illustrates the sensitivity of two postulated transients to the heat transfer coefficient used. For relatively low heat transfer coefficients, i.e., at low flow conditions, the risk is quite sensitive to the value (or correlation) used.

Note that Figures 7-1 through 7-4 are given in terms of failure probability per weld. Since there are six axial welds in the beltline region of the core, these values times six will yield the vessel conditional failure probability.

Appendix H includes more information regarding the sensitivity of relative failure probability to parametric assumptions. Although these studies assumed somewhat different input, assumptions regarding the relation of mean value of RT_{NDT} to fluence and fluence attenuation through the wall than were used for the deterministic fracture mechanics studies (Section 3 and Appendix D), the same trends are found.

These probabilistic sensitivity studies do not include the effect of cladding stresses. Based on the conclusions stated in Section 3, it is estimated that inclusion of the cladding stresses would shift the curves of Figures 7-1 approximately 10°F to the right, thus increasing the risk about a factor of 2 or 3 for that assumed transient.

Because the probabilistic fracture mechanics studies were conducted for only a limited range of parameters, the results should not be extended beyond these ranges. For instance, if T_f were only a few tens of degrees below 550°F, the thermal shock to the vessel would be significantly less severe than say for a cooldown to 200°F or lower. Thus, in terms of probability verus T_f minus mean value of RT_{NDT}, the results are expected to be considerably different.

The technology regarding probabilitistic fracture mechanics are related to PTS scenarios has evolved only during the past few years and perhaps is still some way from reaching maturity. It is, however, believed to be a useful tool. The NRC plans to develop the technology further, and the industry is encouraged to do the same. Future work is expected to include consideration of warm prestressing effects for a variety of postulated transients, the effect of cladding and perhaps other crack shapes.

The discussion in Appendix H suggests that the results which have been presented are most appropriately used in a relative sense for identifying significant variables and variable interactions. Because of the uncertainties associated with the calculated failure probabilities, use of the results in an absolute sense to establish an RT_{NDT} screening limit would be inappropriate. Nonetheless, there does exist a tendency to view the results in an absolute sense when evaluating proposed regulatory requirements. Furthermore, there is a desire to view the results in an absolute sense when performing a probabilistic risk assessment. Utilization of the results in these manners is useful in evaluating a regulatory position, but the limitations of the analysis as discussed in Appendix H must be kept in mind.

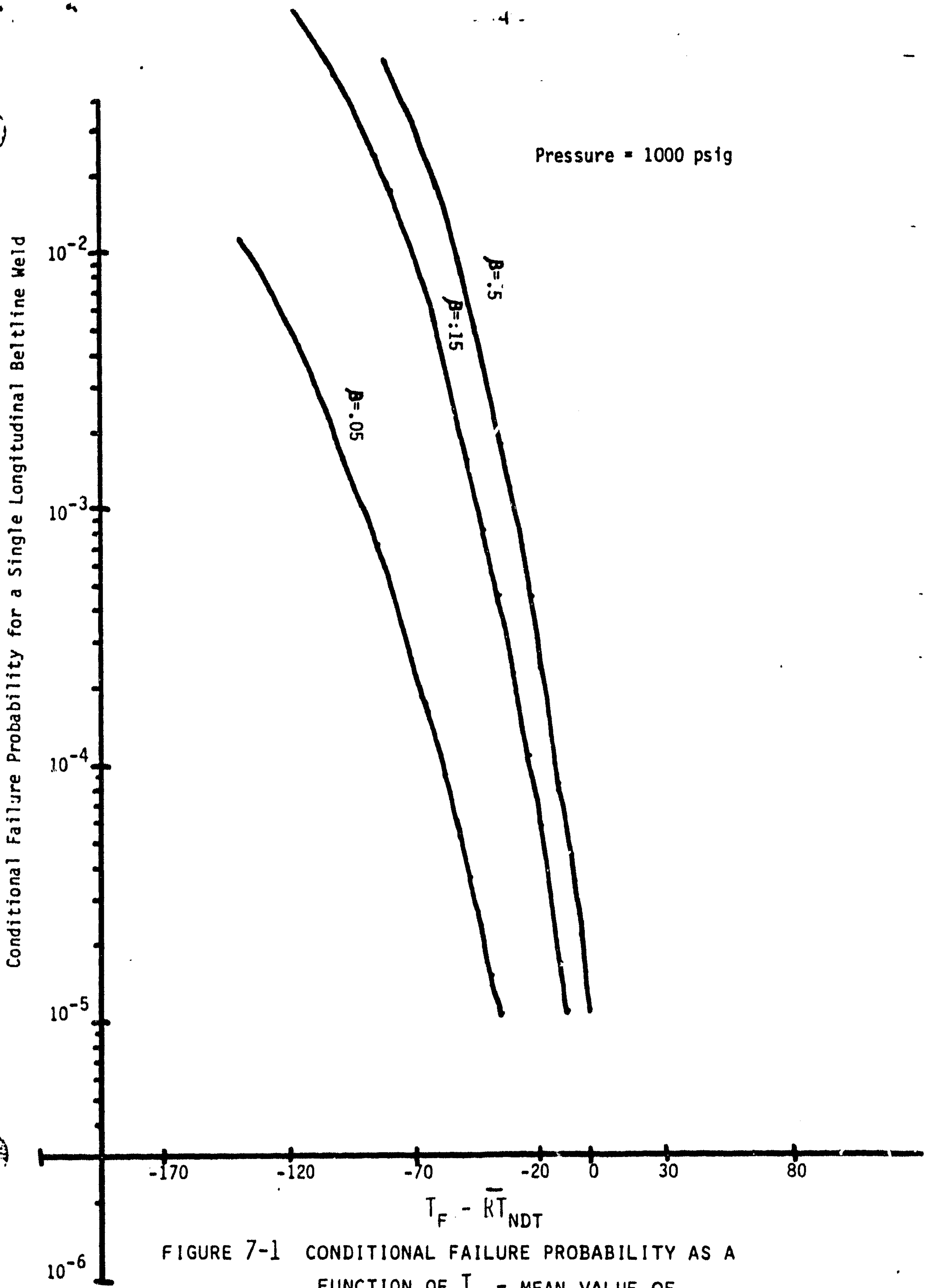


FIGURE 7-1 CONDITIONAL FAILURE PROBABILITY AS A FUNCTION OF T_F - MEAN VALUE OF RT_{NDT}

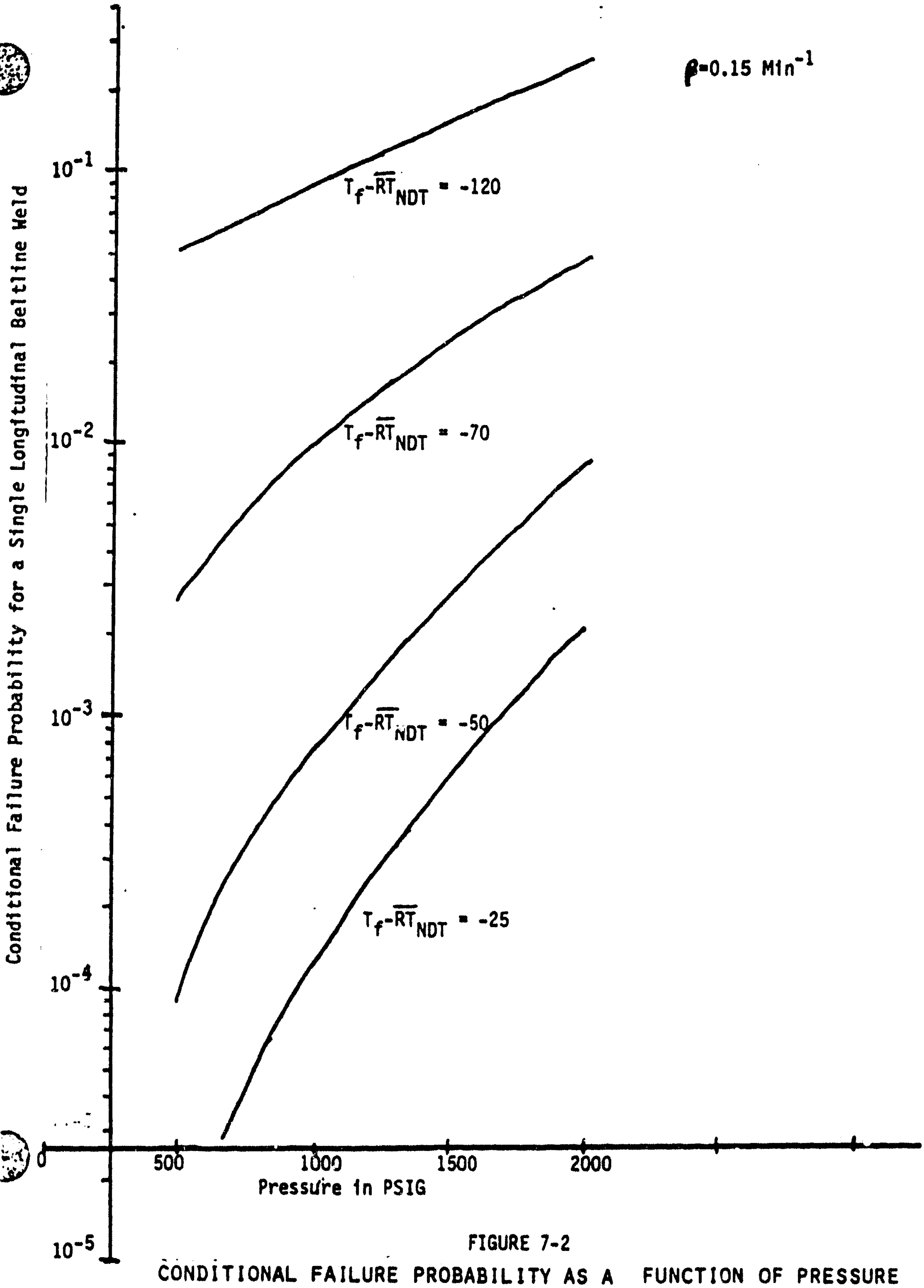


FIGURE 7-2
CONDITIONAL FAILURE PROBABILITY AS A FUNCTION OF PRESSURE

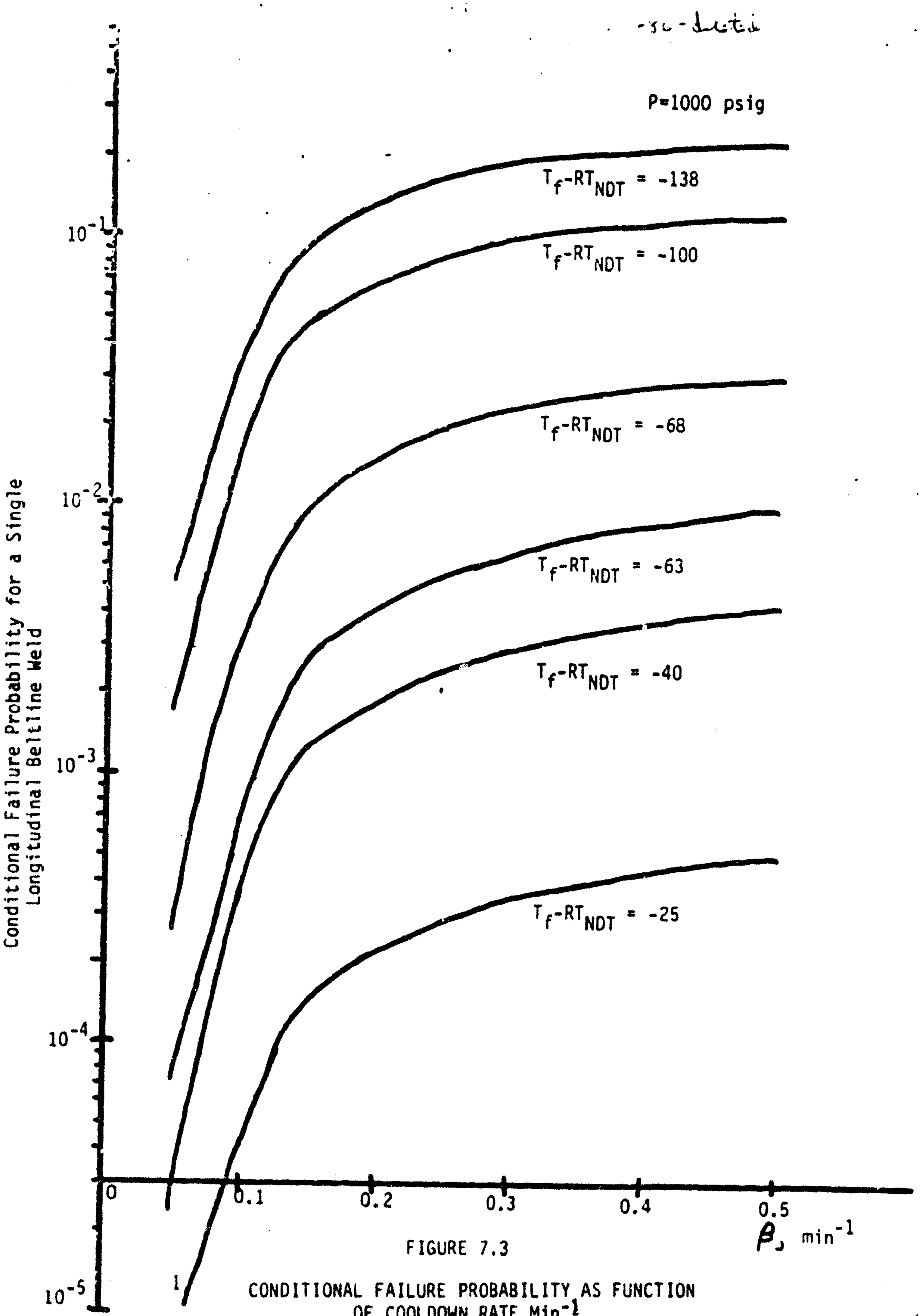


FIGURE 7.3

CONDITIONAL FAILURE PROBABILITY AS FUNCTION OF COOLDOWN RATE Min^{-1}

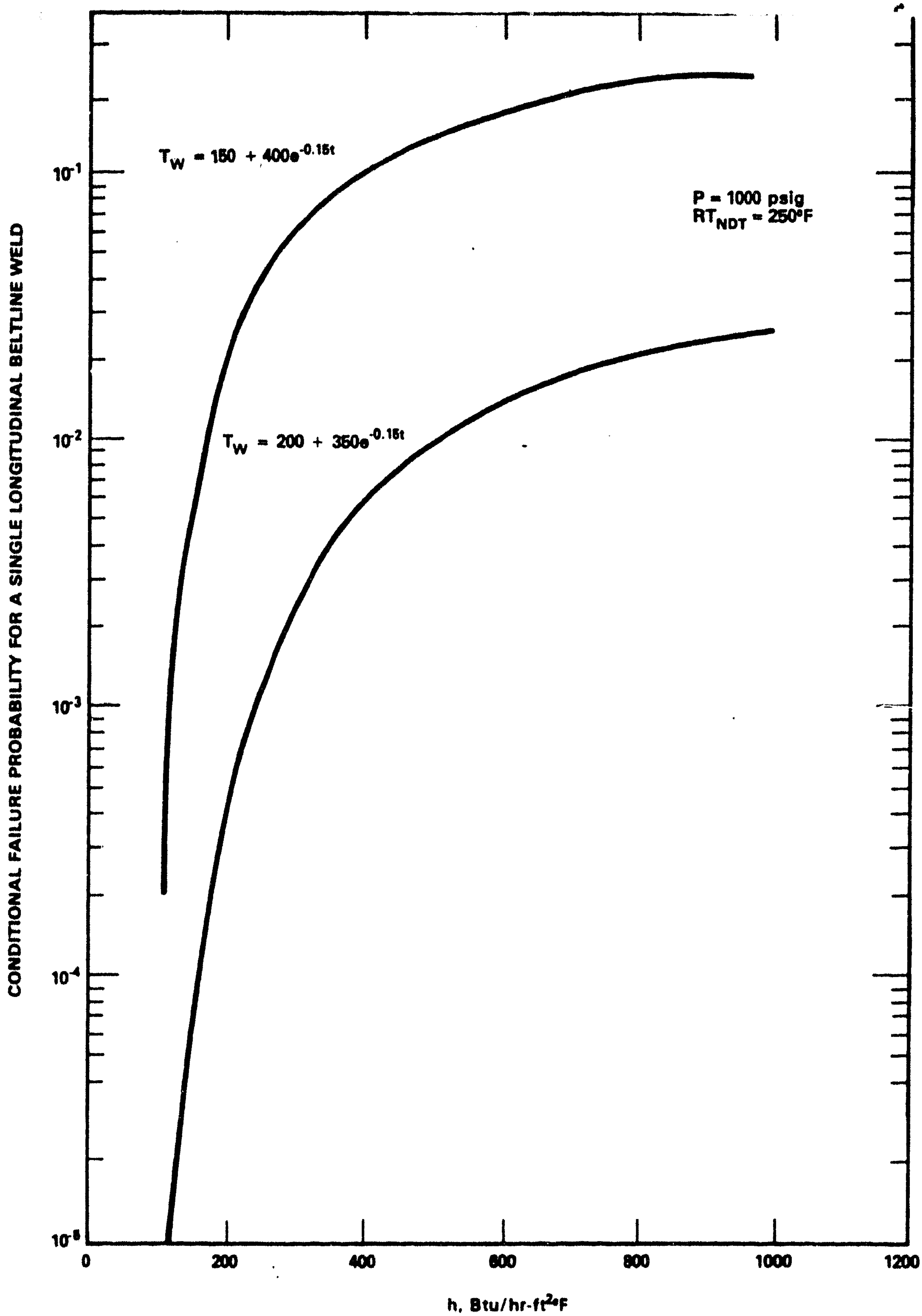


FIGURE 7-4: CONDITIONAL FAILURE PROBABILITY AS FUNCTION OF HEAT TRANSFER COEFFICIENT, h

8 PROBABILITY OF VESSEL FAILURE

8.1 Introduction

This Section summarizes a probabilistic study of reactor pressure vessel (RPV) failure as a result of pressurized thermal shock (PTS). The calculational method combines the frequencies of overcooling transient sequences (Sections 2 and 6) with the probabilistic treatment of RPV failure (Section 7). The results are expressed in terms of the probability, per reactor-year, of RPV failure due to PTS. Some risk considerations are also discussed and also the relationship of PTS to the current regulations.

8.2 Methodology

The basic approach is essentially a combination of (1) a probabilistic analysis of overcooling transients, plus (2) a probabilistic analysis of the consequences of such transients to the RPV, and the probability of RPV failure, given the transients.

In order for this procedure to be valid, the transient sequences must be separated, and analyzed in groups with similar properties. The course and severity of each transient group can then be used as the input transient for analysis of RPV behavior. In the work reported here, the transient sequences were obtained from calculations furnished by the Westinghouse Owners Group (Ref. 8.1), and also from transient analyses based on the WOG analysis but revised by the NRC staff as described in Section 6 and Appendix G.

The transient groups were derived from consideration of the various possible sequences following each of the initiating events--excess feedwater, small-break LOCA-etc.--given in Appendix G. The analyses of the frequencies and courses of the different sequences are also reviewed in Appendix G.

The VISA code, in its present state of development, can accept only stylized input transients characterized by T_f, β, and P or T and P described by simple polynomial functions of time. Therefore, each transient group was stylized. Cumulative frequency distributions of the T_f values used in this study are given in Fig. 8-1.

For each transient group, the values of T_f, β, and P were used, with the VISA code, to establish a probability of RPV failure, given the occurrence of a transient with the specified characteristics. Multiplication of this conditional RPV failure probability by the frequency of occurrence of the transients comprising the group then gives the frequency of RPV failure caused by this group of transients.

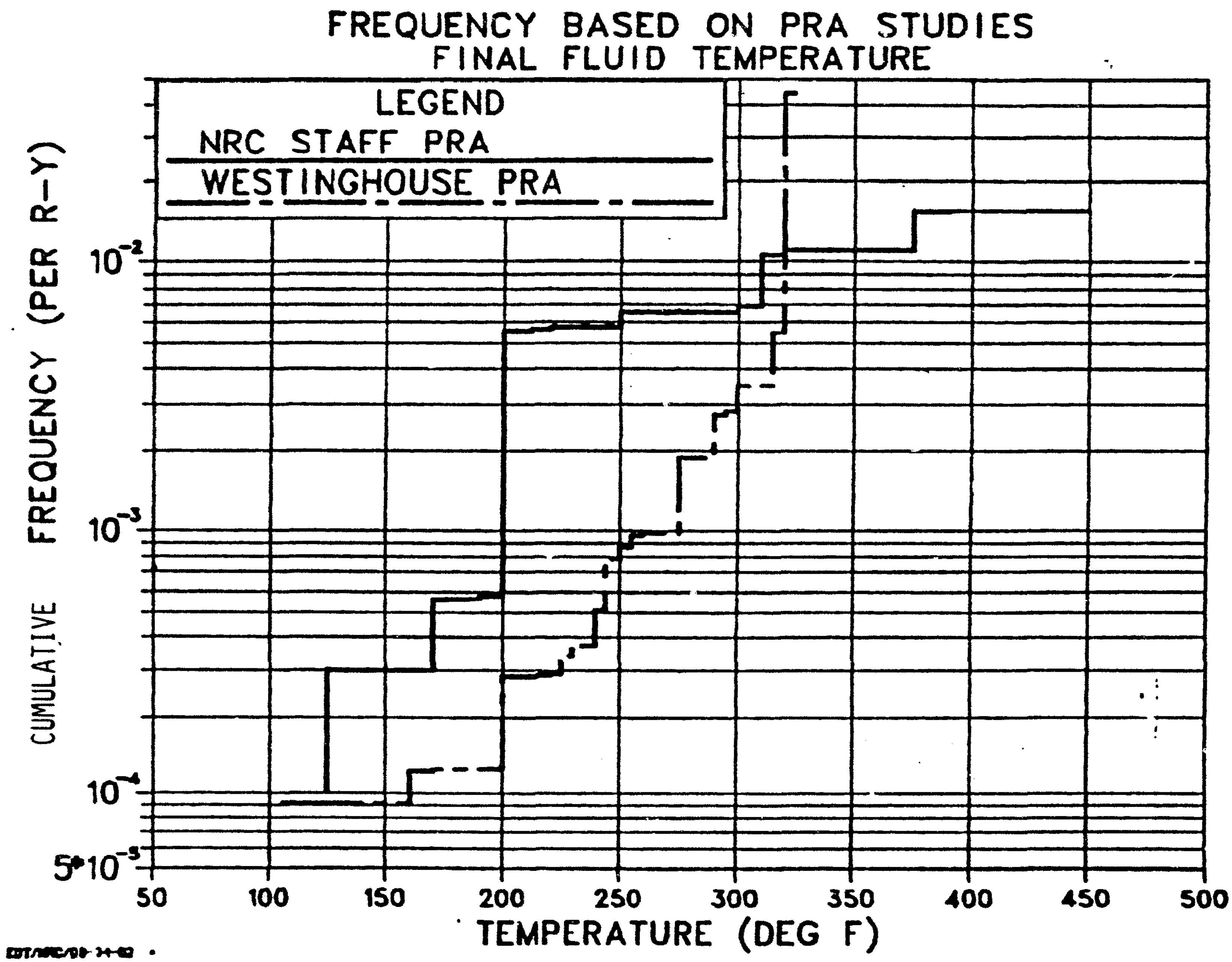
The sum of these failure frequencies gives the RPV failure frequency (per reactor-year) as caused by the ensemble of transients of all the groups considered.

8.3 Accuracy and Completeness

In order for the RPV failure frequency so calculated to be correct, the ensemble of transients must be complete. That is, all transaient sequences capable of inducing RPV failure due to PTS must be included.

The WOG analysis included consideration of several hundred candidate sequences, not all of which turned out to be PTS precursers. The array of initiating events and event sequences is given in Ref. 8.1 and summarized in Appendix G. Variations in reactor power level, break size (for LOCA and steamline break sequences), and operator actions were included. The staff review concentrated on the transient groups shown in the WOG analysis to be dominant, but also considered other candidates not significant in the WOG analysis; see again Appendix G. In all, some hundreds of possible sequences were reviewed by WOG, staff, or both.

Like all probabilistic analyses based on event sequences, the probabilistic PTS analysis presented here cannot be proved complete. The differences between the



FREQUENCY BASED ON PRA STUDIES
FINAL FLUID TEMPERATURE
LEGEND
NRC STAFF PRA
WESTINGHOUSE PRA
CUMULATIVE FREQUENCY (PER R-Y)
10^-2
10^-3
10^-4
5*10^-5
50
100
150
200
250
300
350
400
450
500
TEMPERATURE (DEG F)

FIGURE 8-1

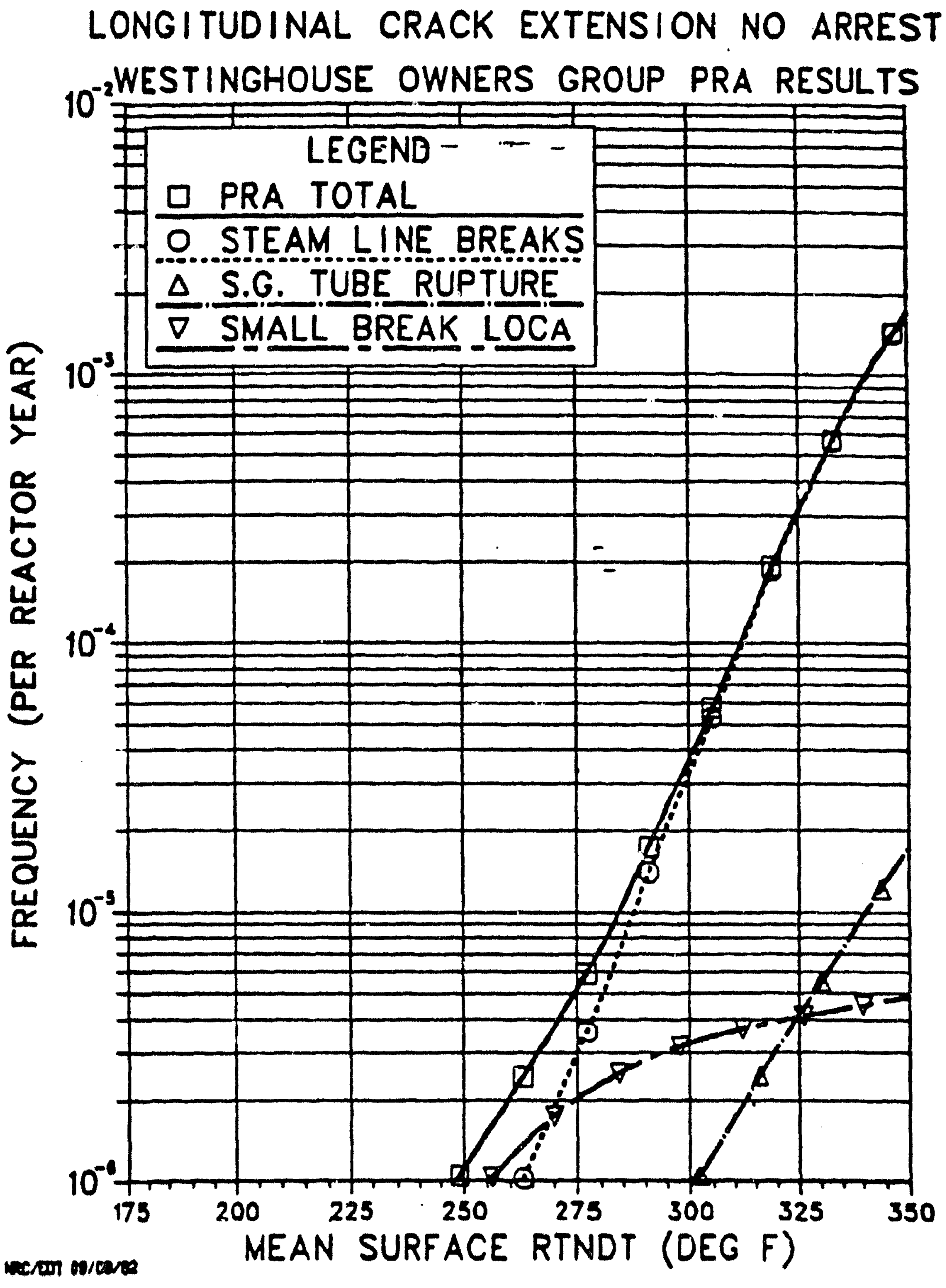

LONGITUDINAL CRACK EXTENSION NO ARREST
WESTINGHOUSE OWNERS GROUP PRA RESULTS
LEGEND
PRA TOTAL
STEAM LINE BREAKS
S.G. TUBE RUPTURE
SMALL BREAK LOCA
FREQUENCY (PER REACTOR YEAR)
10⁻²
10⁻³
10⁻⁴
10⁻⁵
10⁻⁶
175
200
225
250
275
300
325
350
MEAN SURFACE RTNDT (DEG F)
NRC/CDT 09/08/82

FIGURE 8-2

WOG and staff analyses show that more work can and should be done to investigate additional candidate sequences and to validate some of the approximate sequence analyses. However, the work done to date suggests that the principal contributors have probably been identified as well as can be done in approximate, generic analyses. Improved approaches to completeness should be sought in connection with the plant-specific analyses we recommended for plants soon to exceed the screening criterion (Sections 4 and 9). Some sequences not included are discussed in Section 2.6 of Appendix G.

Over the past few months the staff, with some consultation with the Westinghouse Owners Group, has continued to work on improving these calculations. Our experience has been that the calculated results - Figure 8-3 - have changed several times in a few months. The causes of such changes have included the following:

- New systems response information, such as downcomer mixing experimental results and models;

- New insights regarding additional sequences to be included, such as the high-pressure extended HPI group;

- Addition of phenomena to the model, such as warm prestressing.

The changes in results have been well within the quoted overall uncertainty band of the calculational methods at their present state of development. However, the estimated risk for a given vessel RT_{NDT} has changed up to a factor of 5; the RT_{NDT} for a given risk level has changed by up to 30°F.

The staff interprets these changes to show the immaturity of the probabilistic calculations at their present state of development.

Completeness aside, the accuracy from both the transient sequences and the vessel calculations, is subject to substantial uncertainties. In particular the probabilistic treatment of fracture mechanics (Section 7, Appendix H) is still under development. Both the methodology and the probability distribution functions used as input information are sources of variation in the results. Detailed study of these variations has not yet been accomplished. Moreover,

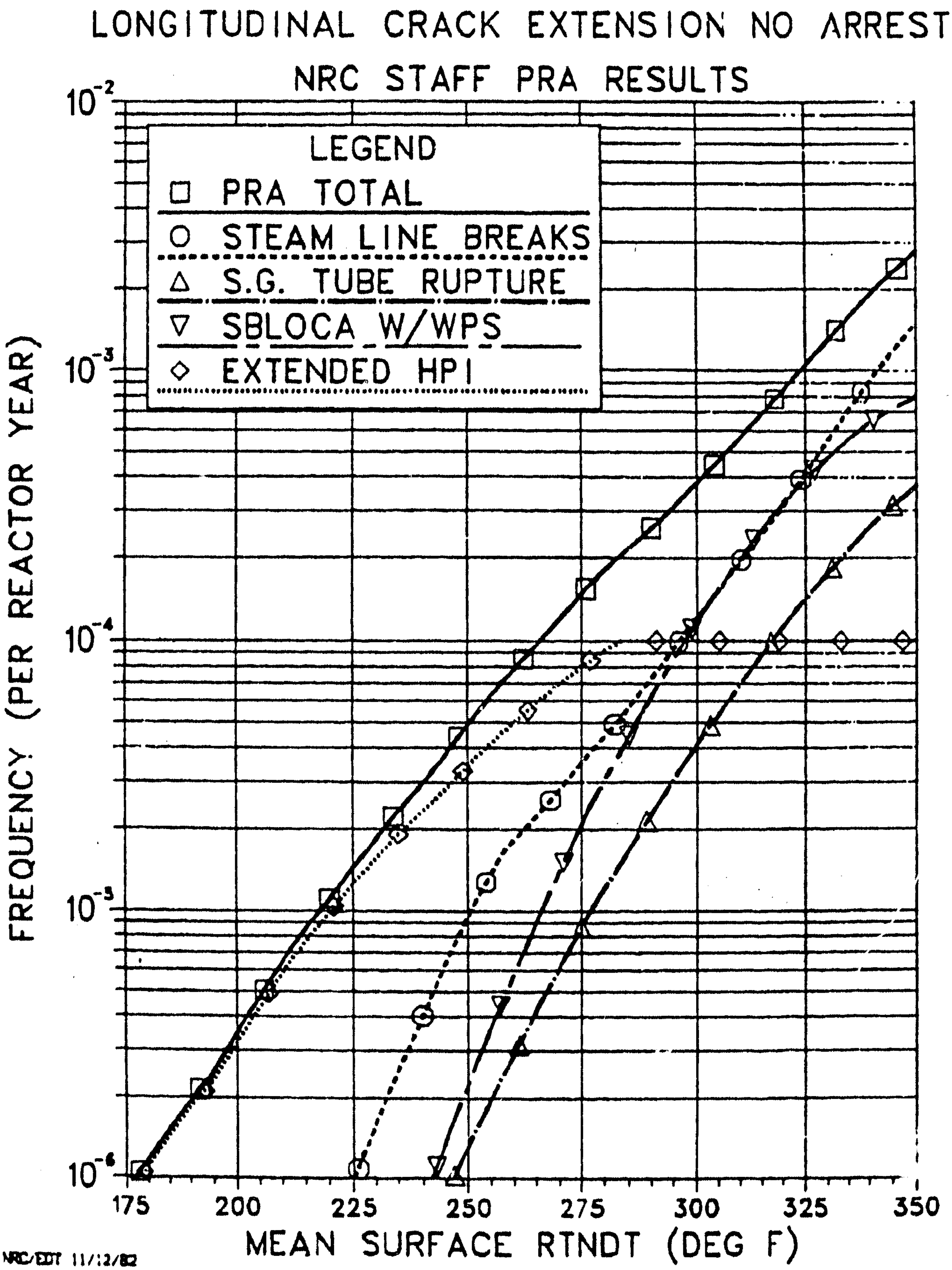

LONGITUDINAL CRACK EXTENSION NO ARREST
NRC STAFF PRA RESULTS
LEGEND
PRA TOTAL
STEAM LINE BREAKS
S.G. TUBE RUPTURE
SBLOCA W/WPS
EXTENDED HPI
FREQUENCY (PER REACTOR YEAR)
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
175
200
225
250
275
300
325
350
MEAN SURFACE RTNDT (DEG F)
NRC/EDT 11/12/82

Figure 8-3

much more extensive sensitivity studies are planned. Therefore, the numerical results given in this Section must be taken for what they are worth. Rather than close estimates of absolute RPV failure probability, the calculated values should be used for insight into trends and sensitivities. The values of the calculated probabilities of failure for a given set of nominal conditions is believed by the researchers (Appendix H) to be uncertain by plus or minus at least two orders of magnitude. Also, the steepness of the curves (Appendix H) shows a high sensitivity of the result (calculated RPV failure probability) to variations in the values of T_f, β and P assigned to the transient group. The calculation of these quantities is approximate, even for a well defined event sequence. The lesson from transients don't look like exponentially decaying temperatures with constant pressures. Thus, another source of uncertainty is introduced by the stylized transients necessarily used in this calculation, at the present state of the art.

These analyses -- both WOG and NRC staff -- are basically for Westinghouse plants. Both probabilistics and severities were calculated for such plants. The use of these analyses to represent B&W and CE plants has not been investigated by the NRC or the industry. The beginnings of a Combustion Engineering Owners Group (CEOG) analysis were described in the June 23, 1982, meeting, but only frequencies have so far been reported, with on evaluation of severities. No comparable study for B&W plants has been reported.

The results of the probabilistic calculations are, therefore, applicable directly only to Westinghouse plants.

The large uncertainties in probabilistic PTS evaluations at the present time have led the staff to use them to estimate the level of safety rather than attempt to derive licensing requirements directly from the probabilistic results.

8.4 Results

With due consideration of the uncertainties discussed just above, we present the results of the probabilistic PTS calculations in Figs. 8-2 and 8-3. The details are given in Appendices G and H.

Figures 8-2 and 8-3 show, as a function of RPV mean value of RT_{NDT}, the expected frequency of RPV failure due to PTS. The abcissas are the reference temperatures, mean value of RT_{NDT}, at the inner surface of a RPV having the mean values of RT_{NDTo}, neutron fluence, and copper content of the probability distribution functions used for these parameters. The ordinate is the failure frequency of the RPV so characterized, per reactor-year, owing to tne PTS transient subclasses (LOCA, SLB, etc.) as labelled, and the total RPV failure frequency due to PTS. New vessels start at the left side of these diagrams, with very low mean value of RT_{NDT} and negligibly small PTS probability. As the vessels are irradiated, their characteristics move to the right, and an increasing number of increasingly probable overcooling transients have increasingly higher probability of inducing RPV failure.

Figure 8-2 gives the results for the WOG distribution of transients; Figure 8-3, the NRC staff distribution.

The steepness of the curves in Figures 8-2 and 8-3 shows a high sensitivity of RPV failure probability to the value of RT_{NDT} (as defined for these curves). A change in RT_{NDT} as small as 20-30°F changes the calculated probability by a factor of 10, on some of these curves. Yet we know neither the actual value of RT_{NDT} for a given RPV, nor the severity of a given transient, to within this order of accuracy. This is another way of restating the substantial uncertainties in the present state-of-the-art of making analyses of this kind. For this reason, the NRC staff recommends that the PTS criteria--screening or otherwise--should not be determined by where these curves cross some acceptable value of risk. Rather, the probabilistic curves are used to estimate the margin of safety for vessel approaching the screening criterion.

8.5 Relationship to Safety Goal

In February 1982, the Commission published for comment a "Proposed Policy Statement on Safety Goals for Nuclear Power Plants" (Ref. 8.2). Although the Safety Goals guidelines have not been adopted (at least not yet), it is instructive to compare the proposed PTS requirements to the guidelines.

Core Melt. - The core melt Safety Goal guideline states, "The likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of reactor operation." This suggests that the core-melt frequency ascribable to one sequence, for example PTS, should not exceed approximately 10^{-5} per reactor-year.

Because of the unusually large uncertainty in the risk estimation for PTS, compared to other sequences, a value of less than 10^{-5} might well be assigned for a safety goal for PTS. We have not done this in the discussion in this section, but have used 10^{-5}. The reader should keep in mind that the risk numbers of PTS given in the following discussion are highly uncertain.

We have no technical analysis of the course and consequences of a PTS sequence that involves RPV failure. Determination of the RPV failure mode (better, estimation of the probabilities of the various failure modes) has not been done and is dependent on the details of the scenario. Moreover, the outcome would likely be dependent also on the plant design details. In particular, ice condenser containments would be expected to have different failure models, with different probabilities, than large dry containments.

The breach in the RPV would be a LOCA, which might or might not prevent ECCS effectiveness. A large through-wall crack peak the midplane level of the core would most probably lead to core melt. Axial cracks and most circumferential cracks would not likely lead to early containment failure; the massive concrete shielding would intercept missiles and the containment could stand the temperature and pressure. (Again, ice condensers have not been evaluated.) Whether a complete circumferential failure (which seems low in probability) would lead to large RPV (and core) motions is not well known.

The result of such approximate and intuitive analysis is that not all PTS failure events lead to core melt, but the fraction that do has not been analyzed quantitatively.

Public Risk. - the Draft Policy Statement includes quantitative guidelines for risk to individual members of the public, and for society at large, from reactor accidents. For analyzing how PTS events contribute significantly to the risk to the public, the following logic applies:

1. PTS event sequences leading to RPV failure have overall frequency F per reactor-year. Figures 8-2 and 8-3 provide a very approximate estimate of F. A plant evaluated (as described in Section 5 or 9 and Appendix E) to be at the 270°F screening criterion is likely to have a true RT_{NDT} of 150-270°F (two sigma 7≅ 60°F). For the mean of 210°F, F≅ 6 x 10^{-6} per reactor-year on the NRC curve (Figure 8-3), and much smaller on the WOG curve (Figure 8-2).

2. A fraction X<1 of RPV failure sequences leads to core melt, giving an expected value of XF core melts per reactor-year.

3. A fraction Y of failures leading to core melt leads to significantly early radioactive releases, so the expected value of the frequency of significant early releases due to PTS is XYF, which is therefore the expected value of the frequency of events involving non-zero early deaths due to PTS.

4. THe risk of early deaths to the average individual within one mile of the site is given by XYFD, where the factor D includes the effects of dispersion and wind direction.

5. To show PTS risk to be lower than 10% of the safety goal guidelines would involve showing

$$XF < 10^{-5} \text{ per reactor-year}$$

and

5x10^{-8}

XYFD$\lesssim$ [redacted] per reactor-year

The limiting value for XYFD is approximate, for an averge site.

We have only approximate values for F, and no quantitative evaluation to give values for X and Y. It is, however, possible to calculate backwards how small these quantities must be to stay within the safety goal guidelines.

D is approximately equal to 5×10^{-2}.

If the core-melt guideline is just barely met, XF equals 10^{-5}. For such a plant, Y would have to be 10^{-1} or less to meet the risk guideline. For a lower value of XF, a proportionately higher value of Y would be low enough to meet the safety goal. For $F = 6 \times 10^{-6}$ (see Item 1, above) and X set arbitrarily at the unrealistically high value of 1, Y would have to be 2×10^{-1} or less to meet the risk guideline.

While we have no analysis of Y for PTS sequences, PTS involves overcooling of the reactor vessel and may well result in a high probability of cooling the containment, thus leading to a low value for Y.

Aside from the containment failure modes given in PRAs for non-PTS events, which apply to PTS events also with some probability, an additional prompt containment failure mode peculiar to PTS arises from a postulated circumferential crack. If the vessel breaks all the way around and separates, the top half will be accelerated upwards. Available analyses are not definitive whether the upward force will break the restraining structures and pipes and allow the loose half to become a jet-propelled missile, or whether such a missile would cause early containment failure. The answer may depend on the details of the plant design. If this mode could dominate the risk, then the plant-specific analyses described in Section 9 should include its evaluation.

ALARA. - The Draft Policy Statement gives a cost-benefit guideline for decis on-making of $1000 per man-rem averted.

For scenarios involving core melt without significant releases, the core-melt guideline will govern and ALARA is not a consideration.

For early containment failure scenarios at an average site, as much as 50×10^{6} man-rem might be involved, at a frequency of XYF. The expected value of the exposure would, therefore, be 50 x 10^6 XYF. For XYF $\lesssim$ 1 x 10^{-6}, the expected value would be less than 50 man-rem, and the ALARA guideline would not be a consideration for these sequences, either. For a highly populated site, the values would be several times higher, but the conclusion would not be changed.

In summary, the results of the approximate probabilistic PTS analysis reported here suggests that the proposed screening criterion is in satisfactory conformance with the Draft Policy Statement on Safety Goals for values of X and Y that seem achievable, even considering th̲ large uncertainties. Further analysis would be appropriate to improve the technical basis of this conclusion.

8.6 Relationship to Licensing Criteria

Several of the Commission's regulations have applicabiltiy to the issue of pressurized thermal shock.

Appendix G of 10 CFR Part 50 sets forth requirements for the fracture toughness of reactor pressure vessels "during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences." In the definitions Section of Appendix A to 10 CFR Part 50, "anticipated operational occurrences" are defined as "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit." Since the usual design life of nuclear plants is forty years, events with an expected frequency of greater than 2.5 x 10^{-2} per reactor-year would be considered "anticipated operational occurrences." On the basis of experienced overcooling events (see Section 4) and low probability event sequences (see Section 6), PTS events with a severity of concern for reactor vessels with a value of RT_{NDT} less than 270°F have an expected frequency of less than 10^{-2} per reactor-year. Thus, the applicability of Appendix G of 10 CFR Part 50 to severe PTS events is questionsable, for pressure vessels that do not exceed the proposed screening criteria. However, the uncertainties are large.

Section IV.A.2.a of Appendix G, 10 CFR Part 50 states that:

> Calculated stress intensity factors shall be lower than the reference stress intensity factors by the margins specified in the ASME Code

> Appendix G, "Protection Against Non-Ductile Failure." The calculational procedures shall comply with the procedures specified in the ASME Code Appendix G, but additional and alternative procedures may be used if the Commission determines that they provide equivalent margins of safety against fracture, making appropriate allowance for all uncertainties in the data and analyses.

The ASME Code Appendix G procedure used for determining "an adequate margin of protection" includes the postulation of a reference semi-elliptical surface flaw having a depth of 1/4 of the section thickness with a length six times its depth. The Appendix G procedure also requires the use of the ASME K_{IR} curve for crack initiation which is approximately 70°F more conservative in the range of interest than the K_{IC} curve used in analyzing PTS. In addition, the stress intensity factor due to pressure is increased by a factor of two. Because pressure stresses dominate for hydro testing and normal startup and shutdown situations, this factor provides additional margin for normal operation, beyond the conservatism provided by the 1/4 T assumed flaw and the use of the K_{IR} curve.

For severe cooldown transients of interest to PTS, however, thermal stresses near the inner vessel surface are dominant. The material toughness is also lower near the surface than deeper into wall because of the lower temperatures near the surface. Hence, consideration must be given to relatively shallow flaws. Thus, procedures different from those of Appendix G are necessary to provide an adequate margin of protection for PTS events. Such procedures will be developed as part of the resolution of the PTS issue. Consideration will need to be given as to whether Appendix G to 10 CFR 50 should be amended or supplemented by separate guidance.

Another potential regulation interface is 10 CFR 50.46 and 10 CFR 50, Appendix K. While the trust of these regulations is to cooling effectiveness, paragraph (b)(5) of 10 CFR 50.46 requires,

> After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

If "successful initial operation" involves a PTS scenario, as can happen for 2 to 6 inch breaks (Sections 2 and 6, Appendix G), then "long-term cooling" can be jeopardized.

This scenario is discussed in Section 6. This sequence was the subject of detailed examination, as discussed in Section 6.2.2. For this sequence, detailed calculations of system response were used, rather than the stylized T_f, β, and P. The WOG calculations, which we accepted, include fluid mixing in the cold leg as predicted from experimental results, heat input from hot piping walls, benefit of warm pre-stressing is assumed, and an assumed temperature of 60°F for the injected coolant. These calculations used in the NRC assumptions for crack arrest, but should be corrected by -10°F to allow for cladding effect (see Section 3). The WOG result is a predicted allowable RT_{NDT} of approximately 270°F, consistent with the proposed screening criterion. Recently obtained information has led the NRC staff to accept a less rapid cooldown for this event. This would indicate a higher, less restricted RT_{NDT} would be acceptable, based on this event.

We conclude that a small break LOCA in a vessel within the proposed screening criterion would cause an acceptably low probability of vessel failure, so 10 CFR 50.46 is not infringed by the proposed requirements.

Another regulation related to PTS is General Design Criterion 31 Appendix A to 10 CFR 50, which states:

> Criterion 31 - Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintainance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

The proposed new requirements relating to PTS provide additional guidance in the implementation of Criterion 31, and it does not appear necessary to modify this regulation.

Section 100.11 of 10 CFR sets forth dose guidelines for assessing the acceptability of Exclusion Area and Low Population Zone boundaries, based upon fission product releases from hypothesized accidents considered credible. In the past, pressure vessel failures have not been considered in the assessments required by 10 CFR 100. The staff believes that the considerations of PTS risk presented above support continuation of this position, but further consideration of the relationship of PTS to 10 CFR 100 is warranted.

8.7 Conservatisms and Non-Conservatism

The calculations summarized in Sections 3, 5, 7 and 8 and described in detail in the appendices and references, contain uncertainties of various sorts. The following paragraphs briefly summarize the most significant sources of uncertainty.

Operating Experience - The three most severe events took place in B&W plants. We have neglected, for lack of sufficient data to do otherwise, plant design differences in evaluating the experience. We have also neglected all the action taken since TMI, Rancho Seco, and Crystal River to improve design and operations and thereby make these transient sequences less likely in the future. These are substantiated conservatisms in the inference from operating experience.

The temperatures used to characterize operating experience were measured in cold legs. The fluid in the downcomer could have been warmer (from mixing) or colder (from stratification) than the measurements.

Operation Actions - The analyses include the probability of the operating staff failing or delaying performing a needed operation, but do not include either successful mitigating actions or wrong actions that could make the events more severe.

Flaws and Cracks - The deterministic calculations assume the presence of a long through-clad flaw of critical depth--a substantial conservatism. The probabilistic calculations use a through-clad crack probability that is highly uncertain and that some people believe is conservative. No account is taken of actual in-service inspection results in these generic calculations.

The crack growth/arrest model used by the staff assumes long initial flaws that grow uniformly over their length. This initial flaw shape is conservative. The growth/arrest shape is discussed in Section 3; we believe that, once a crack initiates, the long crack is a more realistic description than less conservative shapes used in other models.

Stresses - the models include no residual stresses, which is non-conservative. The NRC model includes cladding effect, which is realistic for through-clad cracks. The probabilistic analysis of small-break LOCA used in Section 8 took credit for warm prestressing.

None of the models used for generic events currently includes a warm prestress (WPS), which is a conservatism for transients satisfying the WPS conditions. WPS is included in calculations specifically limited to the small-break LOCA, where it has been demosntrated that WPS is applicable.

Material Properties - The estimation of RT_{NDT} as described in Section 5.4 and 8.5, is a substantial conservatism.

Fracture Mechanics - The use of linear elastic fracture mechanics in the upper-shelf temperature region is appropriate. Assuming vessel failure when the stress intensity factor for a crack reaches the upper-shelf toughness is bel eved by many people to be conservative because considerable ductility exists in tle remaining ligament. Until we have validated applicable elastic-plastic mode s, however, the degree of conservatism cannot be determined.

Uncertainties in Probabilistic Calculations - Substantial uncertainties exis in probabilistic calculations as discussed in Section 8.3. The characterization of event sequences by T_f, β, and P is an oversimplification that may or may not be on the conservative side.

The net result of the above considerations is that the PTS analyses have substantial uncertainty, and are on balance believed by the staff to be substantially conservative. Neither the uncertainty nor the conservatism has been quantified.

Reference 8.3 includes discussions of 38 conservatisms, nonconservatisms, and factors whose effect is not known.

Plants with higher RT_{NDT} would be predicted to have higher PTS risk, so the additional evaluations and requirements of Sections 9 and 10 are proposed.

<u>References</u>

8.1 Letters dated May 28, 1982 (OG-70) and July 15, 1982 (OG-73) from O. D. Kingsley, Chairman, Westinghouse Owners Group to H. R. Denton, NRC; and additional probabilistic PTS results transmitted informally on June 22, 1982.

8.2 "Safety Goals for Nuclear Power Plants: A Discussion Paper," NUREG-0880, February 1982. (The draft policy statement is quoted on pp. vii-xxi.)

8.3 Letter dated November 5, 1982, from L. T. Pedersen, PNL, to Felix Litton, NRC, enclosing draft Supplement 1 to NUREG/CR-2837.

9. PLANT-SPECIFIC ANALYSES AND EVALUATIONS TO BE PROVIDED BEFORE THE SCREENING CRITERION IS EXCEEDED

9.1 Introduction

The study of pressurized thermal shock to determine if there exists a need for interim improvements while the long-term program continues has led the staff to recommend a two-step process. The first step is to establish a screening criterion based on RT_{NDT} to identify reactor vessels where radiation embrittlement has progressed to the point of potential concern. This criterion was selected using simplifications and generic treatment of certain design features, transients, fracture mechanics analyses and plant operating characteristics as described in Sections 2, 3, and 4. The second step, to be taken for plants with vessels with values of RT_{NDT} that exceed or are approaching the screening criterion, involves more detailed plant-specific analyses to determine what, if any, modifications are necessary to the plant design and/or operations to resolve the concern.

The purpose of this section is to outline the analyses and actions to be required of those licensees whose reactor vessels have exceeded the RT_{NDT} screening criterion or will exceed the screening criterion within three calendar years. More detailed requirements must be formulated by the staff in the near future so that it will be clearly understood what methods of analysis are acceptable to the staff and what level of detail is required. Further, and most important, acceptance criteria will be developed and promulgated regarding the required analyses and actions.

9.2 Evaluation of Overcooling Event Sequences

Assessment of pressurized thermal shock concerns on a plant-specific level requires a study of the uinque potential for and consequences of severe overcooling transients at the specific plant. The overcooling transients must be chosen for analysis based on a detailed plant-specific control and safety system design, procedural, and a plant specific human factor study. The study must include a systematic search for, and identification of, potential overcooling event sequences to identify those sequences which are the dominant contributors to the risk of pressure vessel failure. The human factors studies to identify needed procedural changes must be based on an integrated evaluation of all plant procedures at the specific plant for PTS-related and non PTS-related events. The generic studies of potential event sequences done thus far by the staff and by the Westinghouse Owners Group, described in Section 6 of this report, have shown that consideration of only the design basis accident sequences conventionally presented in Safety Analysis Reports does not identify adequately those dominant sequences. The design study must include systems functions pertinent to cooldown transient sequences and must include such systems as the feedwater system, steam generator level control system, steam dump system, steam generator power operated relief valves, charging and letdown system, emergency core cooling system, monitoring instrumentation, and control and safety systems actuation instrumentation. The procedural and human factors study must include operating and emergency procedures, instrumentation available to the operators, operator training, and the ability of the operators to diagnose transients and accidents that occur or do result in a rapid cooldown of the primary system. Any thermal-hydraulic models used in these studies must be verified appropriate for use with PTS scenarios.

The purpose of this study is twofold. First, the results of this study will be used in event-tree analyses which would identify failures that could initiate cooldown transients and quantify the frequency of these events and end states. This information will then be used to select those events that should be subjected to detailed thermal-hydraulic analyses to determine the coldown rates and end states in characteristic pressures and temperatures, which will be used in fracture mechanics analyses whose results will help determine risk.

Second, the results of this study should identify systems, instrumentation, material, and procedural and training program improvements necessary to reduce the probability and consequences of pressurized thermal shock events.

9.3 Vessel Materials Properties (Refer to Appendix E for background and detail)

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

9.3.1 Improve Basis For Initial RT_{NDT}

As noted in Section 5.2, and discussed in more detail in Appendix E, for many older reactor vessels, few data are currently readily available and validated to support the selection of a value for the initial RT_{NDT}. The confidence that can be placed in estimates of the initial RT_{NDT} depends not only on metallurgical tests, but also on the accurate documentation of welding technique, weld wire used, and weld flux used. The credibility of such estimates could be enhanced by performing more tests on archival material, by discovering previously unreported test results on weld specimens from the particular plant, or by evaluating properties of welds considered typical of the plant-specific weld.

9.3.2 Refinement of Chemistry Information for Critical Materials

If it were necessary to assume 0.35 percent copper, because there was no other information, attempts should be made to find archival material suitable for chemical analysis or data on the weld material from other vessels where it may have been used. If the surveillance material matches one of the critical welds, some check for analyses for copper and nickel contents of broken Charpy bars should be considered.

9.3.3 Vessel Fluence (See Appendix F)

Fluence calculations for the critical welds should be rechecked, using modern codes and information from surveillance dosimetry. Location of critical welds relative to the axial and azimuthal flux map should be taken into account, as well as changes in fuel loading during periods when dosimeters were exposed.

9.4 Deterministic Fracture Mechanics Evaluations (See Appendix D)

For the limiting transients as determined in 9.2 and materials properties as determined in 9.3, licensees should provide sufficiently detailed fracture mechanics analyses to permit the NRC staff to interpret the results without its having to redo the calculations. The details should include a listing of the assumptions used, the bases for them and a discussion of the sensitivity of the results to variations in the assumptions. Items to be discussed are:

- Vessel wall thickness and clad thickness; vessel inner radius
- Location and orientation of the assumed initial crack
- Heat transfer coefficient used and material properties, $k, (\frac{E\alpha}{1-\gamma})$ vs. temperature
- Assumed crack shape at initiation and time(s) of initiation
- Crack shape at arrest
- Treatment of cladding-induced stresses
- Upper shelf toughness
- Bases for the determination of limiting RT_{NDT} (at the inner vessel radius)

The results of each transient analyzed should be protrayed as a plot of critical and arrest relative crack depths versus time into the transient. Superpose a line indicating when warm prestressing is deemed to be effective and a curve indicating the depth at which the upper shell toughness is reached. If crack arrest is predicted and accepted at or above the upper shelf, it must be justified.

9.5 <u>Flux Reduction Programs</u> (See Appendix I)

Fuel management programs should be instituted to reduce neutron flux at the reactor vessel wall and at critical weld locations. This would reduce the rate at which the reactor vessel experiences a decrease in ductility and fracture toughness properties. Particular areas of concern in the reactor vessel should be located from an analysis of the material properties of the reactor vessel plate and weld metals. Consideration should be given to replacing fuel assemblies in close proximity to these critical areas. To reduce flux levels, these fuel assemblies could be replaced by spent fuel, zircaloy or stainless steel spacers, or water. Another scheme to be investigated would be an in-out loading pattern where fresh fuel is loaded into the center of the core and moved outward in later cycles. Implementation of revised fuel management techinques have demonstrated a reduction in the neutron flux at the positions of previous maxima by factors of approximately two without derating the reactor coolant power level.

Appendix I contains preliminary estimates of the flux reduction factors (FRP) that can be obtained by reshuffling fuel to place older fuel in the outer positions near the critical welds. It also contains estimates of the FRF that can be obtained from more radical actions such as removal of outer assemblies and replacement with stainless-steel (dummy) elements.

Appendix I also contains results of a study showing what FRF would be necessary to prevent exceeding the screening criteria before end-of-life for each domestic PWR. The factors range from about 10 less than 1 (the latter implying that plant will not reach the screening RT_{NDT} even with no flux reduction).

Finally, Appendix I contains first rough preliminary estimates of what the needed range of FRFs would cost. The costs will be quite dependent on plant specific information (power shape, margin, whether limited by thermal margin or ECCS limits etc.) and are greatly influenced by power derate necessary, if any.

9.6 <u>Inservice Inspection and Nondestructive Evaluation Program</u> (See Appendix L)

Current requirements specified in 10 CFR 50.55a endorse ASME Section XI as defining the examination requirements for reactor vessel welds. The volume of weld to be examined includes the near-surface area; however, currently employed techniques do not provide sufficient basis for assuming that all near-surface cracks can be detected.

The utilizaton of state-of-the-art nondestructive evaluation techniques provides an opportunity to decrease or eliminate a conservatism used in the generic assessment of pressurized thermal shock; that is, small cracks exist at or near the surface of the reactor vessel. Existing inservice inspection programs should be reevaluated to consider incorporation of state-of-the-art examination techniques for inspecting the clad-base metal interface and the near-surface area. This would include plant-unique consideration of the clad surface conditions and may require grinding the clad metal smooth enough to utilize these techniques, and consideration of increased frequency of inspections. Other more detailed recommendations are covered in Appendix L to this report.

9.7 Plant Modifications

To adequately protect reactor vessels from the effects of pressurized thermal shock, the protection needs to be compatible with the plant design and commensurate with the vessel's fracture toughness properties and/or susceptibility to cooldown transients. Modifications to be considered should include the following:

(1) Instrumentation and Controls (See Appendix J)

- (a) reactor vessel downcomer water temperature monitor
- (b) instantaneous and integrated reactor coolant system cooldown rate monitors
- (c) steam dump interlock
- (d) feedwater isolation/flow control logic
- (e) reactor coolant system pressure and temperature monitors
- (f) NDT margin monitor

(2) Automatic Depressurization Logic

(3) Increased Emergency Core Cooling Water and Emergency Feedwater Temperatures (See Appendix K)

Because of design differences and transients response characteristics, plant-specific consideration should be given to any system modification. Further for active system modifications such as an automatic depressurization system, a failure mode and effects analysis should be performed to verify that inadvertent operation of the system would not induce transients more severe than the mitigative capabilities of the plant's safety systems or that otherwise create an unacceptable risk.

9.8 Operating Procedures and Training Program Improvements (See Appendix C)

As a result of generic pressurized thermal shock event tree analysis and actual reactor operating experience it has been shown that operator actions and associated plant response play a key role in the initiation and mitigation of pressurized thermal shock events. The licensees of the seven plants currently being evaluated by the NRC for susceptibility to pressurized thermal shock have reviewed these current operating procedures for information relevant to the pressurized thermal shock issue. Based on the NRC's and the licensee's review of their own procedures, a number of revisions have been incorporated.

These are extensive, industry-NRC ongoing programs to develop a set of integrated procedures that will consider the balance of requirement necessary to ensure core cooling while preventing vessel overcooling. This program is described in Appendix C.

As part of the detailed analyses and actions to be taken when a plant is within 3 calendar years of the screening RT_{NDT}, the licensee should:

(1) Ensure that the actions specified in the procedure guidelines are based on an integrated evaluation of relevant technical considerations, including PTS, core cooling, environmental releases and containment integrity.

(2) Implement upgraded Emergency Operating Procedures which are based on technical guidelines developed or described in (1) above for NUREG-0737, I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents." All licensees will be required to implement these upgraded procedures on a schedule developed in accordance with SECY-82-111B.

The procedures developed and implemented by items (1) and (2) above should address the following types of concerns:

(a) Instructions should include allowance for system response delay times (e.g., loop transport time, thermal transport time).

(b) The need for cooldown rate limits for periods shorter than one hour should be evaluated.

(c) Methods for controlling cooldown rats should be provided.

(d) Guidance should be provided for the operator if cooldown rates or brittle fracture limits are exceeded.

(e) The desired region of operation (e.g., subcooling band) on the pressure-temperature curve should be evaluated to determine if it can be revised to maximize the operator's ability to prevent brittle fracture.

(f) Instructions for controlling pressure following depressurization transients should be provided.

9.9 In-Situ Annealing (See Appendix M)

Annealing of the reactor vessel is a possible, although difficult and expensive, remedial measure for the radiation embrittlement problem. Research sponsored by both the regulated industry and the NRC has provided a basis for selecting the temperatures and duration of the annealing process with some data on reirradiation damage. Research is being funded by the Electric Power Research Institute on the feasibility of annealing. A draft report on annealing proposes the use of electric resistance heating elements supported by a frame that can be lowered into the reactor vessel. The draft report on this study finds no insurmountable difficulties; however, many engineering details remain to be resolved. These include the potential for vessel damage, and protecting the concrete and vessel support structures from the effects of high temperatures. For those plants where proposed remedial actions of the types described in Sections 9.2 through 9.3 above do not result in acceptable risks of vessel failure for the whole design lifetime, a plant-specific engineering evaluation of in-situ annealing should be performed.

The feasibility of conducting a demonstration anneal on an older, irradiated vessel is being investigated and is reported in Appendix M. The investigation has not reached a final conclusion. It is believed that useful information could be obtained regarding both material property recovery and regarding practical difficulties of actually performing the operation, if a suitable vessel could be found. However, the costs would be great and a suitable vessel has not been identified.

9.10 Basis for Continued Operation

Finally, as part of the plant-specific analysis package, the licensee will provide a basis for concluding whether or not continued plant operation is justified, and if it is justified the licensee must demonstrate that interim compensatory measures are in place that will effectively protect the public health and safety and that the licensee is proceeding in good faith in a timely manner to achieve a final resolution.

This basis should include details regardi - frequency of PTS events, description of the dominant risk contributors, and assessment of the total risk from all such events. Vessel and containment failure modes should be discussed, and it should be shown quantitatively how such considerations are factored into the overall risk assessment. The total projected PTS risk for the interim period until acceptance criteria can be met by corrective action should then be compared to the NRC safety goal.

10.0 CONCLUSIONS

10.1 Conclusions

As a result of evaluations performed thus far of the issue of pressurized thermal shock, the NRC staff has reached the following conclusions:

(1) The risk from PTS events for reactor vessels with RT_{NDT} values less than the proposed screening criterion (270°F for axial welds, and 300°F for circumferential welds) is acceptable. On the basis of presently available information, no reactor vessel will exceed the screening criterion for the next few years, therefore there is no need to shutdown or anneal any operating PWR in the next few years.

(2) Most plants can avoid reaching the screening criterion throughout their service life by timely implementation of flux reduction programs. Such flux reduction programs should be implemented on a time schedule that will avoid foreclosure of this option.

(3) Any plant for which the value of RT_{NDT} is projected to reach the screening criterion before the end of service life, using the conservative method of RT_{NDT} determination described in Section 5 and Appendix E, should submit plant-specific evaluations (of the type described in Section 9) to determine what, if any, modifications to equipment, systems and procedures should be required to provide acceptable protection against vessel failure from PTS events for the remainder of plant life. These evaluations should be submitted three years before the vessel is projected to reach the screening criterion.

(4) In the near future, the staff should develop more detailed

guidance for these evaluations and acceptance criteria for determining whether plant modifications are needed based on the evaluations.

(5) Some of the Commission's regulations (Appendix G to 10 CFR Part 50, 10 CFR 50.46, and possibly others) may not appropriately reflect current understanding of the state of reactor vessel embrittlement and the potential for vessel failure as a result of PTS. Timely consideration should be given to the possible need for amendments to the regulations (as discussed in Section 8.6).

10.2 Comment on Overall Approach

It will be evident from this report that the staff is not proposing to resolve PTS issues by requiring a "design-basis pressurized thermal shock event" to be analyzed by a prescribed conservative evaluation model with the results to be compared to specified acceptance criteria. Rather than this traditional approach, the staff has used analyses of overcooling event sequences actually experienced, plus a wide spectrum of possible sequences that have not occurred, together with explicit consideration of the frequencies or probabilities of occurrence of the various events. Moreover, the staff has used analysis models as realisitic as the state of the art permits, with a few explicit conservatisms to provide the needed margin of safety. The overall level of safety thus provided has been estimated (very approximately) using probabilistic analysis.

The staff believes that this approach, a departure from regulatory practice in the past, gives improved coverage of the spectrum of PTS event sequences, improved evaluation of the level of safety, and improved accounting for the various possible events that can mitigate or aggravate an ongoing PTS sequence.

APPENDICES TO

NRC STAFF EVALUATION

OF

PRESSURIZED THERMAL SHOCK

(November 1982)

APPENDIX A

OVERVIEW OF ACTIVITIES CONCERNING THE PRESSURIZED THERMAL SHOCK ISSUE

A.1 INTRODUCTION

The subject of thermal shock to reactor pressure vessels from overcooling transients is not a new concern; both industry and the NRC have held meetings and issued written reports on the subject for several years. The thermal shock concern after a Loss of Coolant Accident (LOCA) has been subject to considerable review in the past. Analyses and experiments indicate that the vessel will still hold water after a large LOCA. Therefore, for large LOCA, thermal shock to the reactor pressure vessel is not a new concern.

The TMI Action Plan (NUREG-0737, Item II.K.2.13 "Thermal Mechanical Report Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss of Coolant Accident with No Auxiliary Feedwater") identified one transient of concern which is characterizd by severe overcooling causing thermal shock to the vessel, concurrent with or followed by repressurization (that is, Pressurized Thermal Shock, PTS). The staff has recognized that there are many other scenarios which could result in PTS. On the basis of events which have occurred at operating PWRs, the staff recognized early in 1981 that some operating reactor pressure vessels of the older plants were approaching material property conditions which made the PTS issue a greater concern. Thus the NRC staff requested a meeting with industry representatives on March 31, 1981, to discuss the PTS problem. This initiated the current effort concerning the PTS issue.

The PTS issue is a concern only for operating PWRs. Boiling water reactors (BWRs) are not a significant PTS concern. BWRs operate with a large portion of water inventory inside the pressure vessel at saturated conditions. Any sudden cooling will condense steam and result in a pressure decrease, so simultaneous creation of high pressure and low temperature is improbable. Also contributing to the lack of PTS concerns for BWRs is the lower fluence at the vessel inner wall and the use of thinner vessel wall which results in a lower stress intensity for a postulated crack.

The attached appendix provides a time table of events concerning the PTS issue.

A.2 Summary of Industry Meetings with the Staff

On March 31, 1981,[1] the NRC staff met with the PWR Owners Groups and representatives of NSSS vendors to discuss the effects of potential thermal shock to reactor pressure vessels by overcooling transients and the potential consequences of subsequent repressurization at relatively low temperatures. The staff requested the industry to make a plant-by-plant assessment of the problem and to scope and bound the problem. As a result of this meeting the industry representatives committed to a report by May 15, 1981, providing an account of what immediate problems exist. Subsequently, by letter dated April 20, 1981,[2] the NRC requested the Owners of PWR operating plants to respond by May 22, 1981, identifying the specific action which the plant Owners propose to take.

Meetings were held with Babcock & Wilcox (B&W), Westinghouse (W) and Combustion Engineering (CE) Owners Groups (OG) on July 28, 29, and 30, 1981,[3] respectively, at the request of the staff in order to present the staff's analysis of the problem and the actions the staff intends to take and to hear from the PWR Owners the results of their analyses and their proposed actions concerning the problem. The staff concluded at this meeting that Owners of plants of each NSSS type which have the highest RT_{NDT} values would be requested to take action to resolve the problem for their plants. Subsequently, the NRC requested under 10 CFR 50.54f, by letters dated August 21, 1981,[17] the licensees of Oconee 1, TMI-1, Robinson 2, Turkey Point 4, San Onofre 1, Calvert Cliffs 1, Fort Calhoun and Maine Yankee (1) a 60-day response for information related to RT_{NDT} and operator action to prevent PTS and ensure vessel integrity and (2) a 150-day response for information which would define actions and schedules for resolution of the PTS issue and analyses to support continued operation.

Follow-up meetings were held with the W, B&W, and CE OGs on September 18 and 22, and October 7, 1981,[4,5,6] respectively, to review the progress and to discuss the technical issues concerning the systems analyses, operator responses, and the materials and fracture mechanics aspects of the PTS issue.

The WOG indicated their report for the TMI Action Plan Item II.K.2.13 due at the end of the year would address (1) the Small Break Loss of Coolant Accident with Loss of Feedwater (SBLOCA + LOFW) (TMI II.K.2.13) and other scenarios including steam line breaks, (2) fracture mechanics calculations for each operating plant, (3) the date and RT_{NDT} for each plant when acceptable conditions will not be met, and (4) evaluation of remedial actions. WOG indicated that the most limiting plant has at least 3 EFPY remaining before there is a concern.

The B&WOG indicated that their work would be concentrated on the Oconee 1 150-day response to the August 21, 1981 letter and plant-specific analyses thereafter.

The CEOG indicated that the report due at the end of the year would address the TMI Action Plan Item II.K.2.13 and other scenarios including the main steam line break event. The CEOG indicated that the most limiting plant has at least 5 EFPY remaining before there is a concern assuming the no-crack initiation criteria.

Meetings were held with the WOG and CEOG including the Owners of the six selected plants who received the August 21, 1981 letter on February 24 and March 3, 1982,[7,8] at the request of the staff. A meeting was held with Duke Power Company on March 24, 1982.[9] These meetings were to discuss the respective Owners groups' reports and the "150 day" responses concerning San Onofre 1, Robinson 2, Turkey Point 4, Fort Calhoun, Calvert Cliffs 1, Maine Yankee, and Oconee 1. TMI-1 was not included in these discussions since GPU elected to delay their "150 day" submittal until June 1982. These meetings were designed to respond to specific staff concerns which were identified with the published meeting notices and later were, in part, transmitted to the Owners of the selected plants.[34-41]

Meetings were held with the Omaha Public Power District and the WOG on May 6, and 10, 1982,[10,11] respectively at their request to update the staff on the progress of the respective programs and the responses to the staff's concerns identified in the previous meetings.

The WOG provided the results of a study involving a methodology leading to a probabilistic risk assessment (PRA) related to PTS. The conclusion of this study was that the likelihood of a cooldown transient can challenge the reactor vessel is less than 10^{-4} to 10^{-3} per reactor year for that lead plant at 5 EFPY from today. WOG maintains that the total risk to public health is in the area of 10^{-9}.

The CEOG provided responses to the staff concerns identified in the meeting of March 3, 1982. In particular the CEOG provided the results of their review of operating experience of CE operating plants and the results of a probability analysis related to the PTS issue. The review representing 49 reactor years of operating experience identified 16 events which met a screening criterion. Of those only two met the selection criteria. These actual overcooling events were much less severe than the event analyzed in the "150 day" response and there was no uncontrolled repressurization in either event. The probability study concluded that the main steam line break (MSLB) is the most severe event and ranges between a probability of 10^{-6} to 10^{-4}.

A meeting was held on June [illegible], 1982,[12] with General Public Utilities (GPU) at their request to provide the staff a status report on the PTS program for TMI-1 and to present a summary of the "150 day" response for TMI-1. Significant in this study was the use of the COMMIX Code in the mixing analysis. The COMMIX Code shows warmer temperatures for the SBLOCA events than the BAW 1648 or Oconee 1 mixing models. The SBLOCA and turbine bypass valve failure were the only events analyzed. GPU determined that based on EOL RT_{NDT} of 335°F for the most critical weld, operation would be acceptable for 32 EFPY.

A meeting was held on June 9, 1982,[13] with the PWR industry representatives at the request of the staff for the purpose of discussing the current NRC staff considerations of possible recommendations for PTS requirements. The staff was considering a limit of T_f RT_{NDT} of 230°F for longitudinal welds and 255°F for circumferential welds based on a transient which resulted in a final temperature/pressure of 250°F/2500 psi (β = 0.15) which would initiate a crack. The industry representatives did not agree with the conservatism of the staff considerations. They objected to the crack initiation criteria. They believed the final temperature was too low and the pressure was not possible. They objected to the data base which was used for the probabilistic determinations. The staff provided the industry two weeks to submit comments in order for staff to consider the industry news in the determination of the staff's position.

Meetings were held on June 22 and 23, 1982,[14,15] with the WOG and CEOG respectively at their request to respond to the staff's request for comments to the staff's proposed recommendations on PTS requirements. WOG proposed a screening criteria of a RT_{NDT} of 310°F and 335°F at longitudinal and circumferential welds respectively. This criteria was based on T_f = 290° at the surface of the reactor vessel weld. The WOG PRA and the NRC probabilistic fracture mechanics was coupled with the W probabilistic transient evaluation to yield safety goals somewhat lower than those reported by the staff. The WOG analysis indicated that the PTS issue would be of no concern to operating plants for the transient for the next five years of plant operation.

The CEOG recommended the use of CEN-189 best-estimate initial RT_{NDT} values. They recommended the current Regulatory Guide 1.99 but used to predict the upper bound shift for high-copper, high-nickel material at fluence greater than 10^{19} nvt and that Guthrie (HEDL) correlation be used to predict the upper bound shift for medium-low copper, high-low nickel material at fluence less than 10^{19} nvt. The CEOG believes arrest will occur. The probabilistic analysis indicated that the MSLB bounds the PTS events.

A meeting was held with the WOG on July 30, 1982,[100] at the request of the staff to discuss the apparent discrepancies between the WOG and the staff concerning the limiting transients which produce the greatest overcooling, the frequencies of such transients, and the fracture mechanics analysis associated with the transients. In particular, the meeting discussed the small break

LOCAs (SBLOCAs) which result in stagnation flow and the factors in the fracture mechanics analysis which account for the differences between the WOG and the staff. WOG indicates that SBLOCAs in the area of 2" to 3" were the sizes of concern which result in stagnation flow and the frequencies of such events were conservatively estimated to be 6 x 10^{-4}. Factors which account for the differences in the fracture mechanics analyses were heat transfer coefficient used, crack length assumptions, and effects of the clad. WOG assumed the heat transfer coefficient was not 300° continuous. It varies as explained in WCAP 10019. WOG assumed an elliptic crack versus the staff's assumptions of an infinite long crack. WOG assumed the clad has no effect.

A follow-up meeting[101] was held with the WOG on August 11, 1982. WOG indicated that a lower limit for the SBLOCA of concern was 5 x 10^{-5} (a medium value). More realistic mixing assumptions concerning other factors such as metal heat resulted in approximately 60° increase to prior results of analysis of the SBLOCA.

For the longitudinal flaw the calculational differences between the staff and the WOG amount to 45° RT_{NDT}.

The staff proposed a screening criteria as follows:

T_F = 260°F at 10^{-2} frequency
RT_{NDT} = 270° for longitudinal welds
RT_{NDT} = 300°F for circumferential welds.

The above is based on operating references.

A.3 Summary of Industry Responses to Staff Requests

At the meeting of March 31, 1981, with the PWR industry representatives, the PWR Owners Groups agreed to provide individual owners groups reports by May 15, 1981, which would provide an accounting of what immediate problems exist. By letters dated April 20, 1981,[2] the NRC requested the Owners of operating PWR plants to provide responses by May 22, 1981 relating to their participation in the Owners groups programs and specific action which they intend to take. By letters dated August 21, 1981,[17] the NRC requested 30-, 60- and 150-day responses from each of eight selected utilities owning plants which represented the three different vendor NSSS and reactor vessels with the highest irradiation damage of each group.

The NRC responded[18-41] to each of the utilities responses to the August 21, 1981 letter. As a result of the 60- and 150-day responses the staff requested additional information from each utility which received the August 21, 1981 letter.

Each of the Owners groups provided responses by May 25, 1981[42-43] with an accounting of the immediate concern and their plans for resolving the issue. Owners of all operating PWR plants indicated their participation in the Owners groups programs by letters in response to the NRC letter dated April 20, 1981.

Tables A-1, A-2 and A-3 provide the summaries of the 30-, 60- and 150-day responses, respectively[45-66] of the selected utilities which received the August 21, 1981 letter. Table A-4 provides a summary of the W and CE generic reports[67-68] concerning the PTS issue.

A.3.1 Responses Relating to Westinghouse Plants

The WOG response dated May 14, 1981 from Mr. Robert W. Jurgensen[44] indicated that all Westinghouse operating plants could sustain severe thermal shock transient, including repressurization to beyond January 1983. The WOG program would be completed by December, 1981. Each utility of a Westinghouse plan would provide additional information including a schedule for remedial action if requested on completion of the WOG program.

Tables A-2, A-3 and A-4 provide summaries of the "60 and 150 day" responses and the generic reports. These responses were supplemented by additional information received from the WOG and each of the three selected Westinghouse operating plants in May 1982.[69-72]

The Westinghouse WOG report concludes that a number of reactor vessels will require more plant-specific evaluations and may require that remedial actions be implemented at some point in the vessel life to demonstrate vessel integrity to end-of-life. The licensees of the three Westinghouse reactors all concluded that vessel integrity will be maintained to or beyond end-of-life.

The supplemental information provided by the WOG at the meeting of May 10, 1982,[11] concludes that the probability of a transient of PTS concern for the "lead" plant at 5 EFPY from today is between 10^{-4} to 10^{-3}.

By letter dated May 28, 1982,[69] the WOG provided supplemental information on Reactor Vessel Integrity in the form of a report "Summary of Evaluation Related to Reactor Vessel Integrity." This report supported the conclusions provided by the WOG at the meeting of May 10, 1982.

By letter dated June 16 1982,[73] the WOG provided a discussion of benefits and penalties of fuel management schemes to reduce fluence in the form of a report "Fuel Management To Reduce Neutron Flux." This report provides methods of reducing the flux to the pressure vessel with no power derating or economic penalty.

By letter dated June 22, 1982,[14] the WOG provided the "Review of the Emergency Response Guidelines Related to Pressurized Thermal Shock." This report explicitly identified those steps in the Emergency Response Guidelines (ERG) that have been written to provide operator direction to prevent a mitigated PTS event. The report also determined those areas of the ERGs that should be modified to more clearly identify appropriate operator responses to prevent or mitigate potential PTS events.

By letter dated July 15, 1982 [103], the WOG provided the material which was presented at the meeting of June 22, 1982, with the staff on the PTS issue. The letter provided discussions on the calculational differences between WOG and the NRC staff and the use of the technology relating to PTS in future regulatory actions. The WOG utilizes a transient with a final temperature of 290°F to obtain a screening RT_{NDT} value of 310°F to obtain a screening RT_{NDT}

value of 310°F. The WOG recommends this value for a screening criteria for prioritizing plants for plant specific programs for the resolution of the PTS issue.

By letter dated September 2, 1982 [104] supplemented by letter dated September 16, 1982 [105] the WOG provided material which was presented at the August 11, 1982, meeting with the NRC staff. The WOG indicated substantial agreement with the staff's methods, techniques, assumptions and the screening values for RT_{NDT} (270° longitudinal welds and approximatley 300°F for circumferential).

A.3.2 Responses Related to Combustion Engineering Plants

The CEOG response dated May 15, 1982, from Mr. K. P. Baskin[43] indicated that the steam line break transient produces the largest magnitude and rate of heat removal for the CE-NSSS design. With this transient, approximately 5 EFPY of operation would have to elapse before vessel integrity would theoretically become a concern.

The CEOG response indicates that a program is planned to address all aspects of the PTS issue and a generic response to TMI Action Plan Item II.K.2.13 would be provided by January 1, 1982.

The Combustion Engineering CEOG Generic Report[68] concludes that all CE plants can withstand the postulated small break LOCA (SBLOCA) with extended Loss of Feedwater (LOFW) scenarios for the assumed life of the plant. The 150-day responses[60-62] from the three licensees of operating CE plants all indicate that vessel integrity will be maintained for the lifetime of the plant.

The supplemental information provided by each of the three selected CE operating plant owners[75-77] indicated that the main steam line break event is the most limiting event and ranges between a probability of 10^{-6} to 10^{-4}. They also provided an identification of overcooling events from the operating history of CE plants. In addition the responses discussed the sensitivity of controlling overcooling transients to operator action.

By letter dated June 14, 1982, the CEOG provided a response to the NRC staff proposed position that was presented at the June 9, 1982 meeting. This letter reiterated the CEOG conclusion that the MSLB event is the most limiting and probable concerning the PTS issue. The CEOG concludes that an RT_{NDT} value of 320°F is more appropriate for crack initiation criteria for the NRC proposed transient (T_F = 250°, P = 2500 psi, β =0.50m^{-1}).

The NRC staff proposed crack initiation criteria was considered unnecessarily conservative. The ability of the CE-NSSS to cool down as rapidly as the NRC proposed temperature transient while maintaining pressure at 2500 psi is considered physically impossible. The CEOG contends that the NRC calculated probability of the NRC proposed temperature transient is much too high. The CEOG disagrees with the approach taken by the NRC to resolve the PTS issue.

Omaha Public Power District provided comments concerning the staff's proposed position by letter[79] dated June 26, 1982. OPPD suggested that some type of screening criteria would be appropriate to focus on plants which might develop a potential PTS concern. The screening criteria should reflect the assessment of plant operating histories and major design differences. The use of best-estimate RT_{NDT} value is most appropriate.

A.3.3 Responses Related to B&W Plants

The B&WOG response dated May 12, 1981 from Mr. John J. Mattimoe[42] indicated that the SBLOCA with no repressurization is the bounding accident. This assumes operator action would mitigate repressurization (by throttling HPI and utilizing atmospheric dump or turbine bypass valves). B&W contended that the analysis is conservative and there is no concern for thermal shock through 1982. The B&W Owners submitted, in December 1980, BAW 1648, which addressed TMI Action Plan Item II.K.2.13 "Thermal Mechanical Report - Effect of HPI on Vessel Integrity for SBLOCA with Additional Loss of Feedwater." The B&WOG plans with respect to PTS to submit plant-specific analyses to address the conservatisms in the generic analysis. No generic report was planned for B&W plants.

Oconee 1 and TMI-1 were the B&W selected plants for the August 21, 1981 letter.

The 150-day response from Duke Power Company[66] concerning the Oconee 1 vessel concludes that no changes to the plant or additional fuel management, or reactor vessel annealing is necessary to assure safe operation of Oconee 1 through the design life of the plant. The Oconee 1 report indicated that severe PTS events were in the probability range of 10^{-6} to 10^{-4}. The Duke Power Company letter dated April 30, 1982[80] provided additional information concerning operator responses and sensitivity of transient analysis to operator action times.

By letter dated March 17, 1982,[81] GPU Nuclear informed the staff that the "150 day" response concerning TMI-1 would be submitted as soon as the revised mixing analysis could be inputted into the B&WOG plant-specific analyses (estimated completion June 1982).

By letter dated June 1, 1982,[82] GPU provided a response to the NRC letters of August 21, 1981, and December 18, 1981. This letter provides a summary of an analysis which GPU proposed to provide at the end of June, 1982. The summary concludes that the TMI-1 reactor pressure vessel integrity will not be compromised due to PTS events during the lifetime of the plant. Also the rate of embrittlement of the TMI-1 vessel may be reduced further if the plant switches to low leakage fuel scheme in the near term reloads. GPU indicated because of the concerns by the PTS issue, operator response will be significantly improved through increased awareness and additional training.

By letter dated July 7, 1982,[102] GPUN provided the "150 day" response to the NRC August 21, 1981 letter.

By letter dated June 22, 1982,[83] the B&WOG provided a response to the NRC staff's request at the meeting of June 9, 1982 concerning the NRC staff proposed position on the PTS issue. The B&WOG indicated that the generic position is unsound, unrealistic, and inappropriate unless used solely as a screening basis. Also the NRC proposed crack initiation criteria was considered to be highly conservative and the proposed generic transient does not realistically represent an actual B&W plant response. The B&WOG recommended the use of

RT_{NDT} as a means of "flagging" plants with potential concerns. Each plant should be analyzed for a realistic probable transient.

Letters[84-88] were received from Duke Power Company, Arkansas Power & Light Company, Florida Power Corporation, GPU Nuclear and SMUD in response to the staff's request concerning the proposed staff position concerning the PTS issue. Duke Power Company indicated that the staff proposed approach can be utilized as a screening method of identifying plants for detailed analyses with respect to the PTS issue. However, the staff's analysis of the frequency of the transient events is not applicable to any real plant. The Duke Power Company letter expressed the concern that the staff has failed to provide a feedback loop such that plant improvements made are directly included in the analyses. Also the screening criterion may need to be established on a group of plants or even on an individual plant basis rather than a generic PWR basis.

Arkansas Power & Light Company's comments[85] on the staff proposed position follows:

(1) Indexing the operation to actual fracture toughness rather than on RT_{NDT} should be pursued.
(2) RT_{NDT} could be used as a screening criteria.
(3) The staff's generic approach does not consider basic design differences.
(4) The transient selected by the staff is unrealistic.
(5) The crack initiation criteria is not consistent with the ASME Code.
(6) AP&L disagrees with some of the basic assumptions of the staff's proposal.

Florida Power Corporation[86] included the following comments on the staff's proposal:

(1) System pressure does not remain constant as proposed by the staff.
(2) Emphasis should be focused on actual fracture toughness rather than RT_{NDT}.

GPU[87] offered the following comments to the staff's proposal:

(1) The staff's proposed failure criteria is too conservative.
(2) The proposed governing transient is too severe and overly conservative.
(3) Emphasis should be placed on the actual toughness of the vessel material rather than RT_{NDT}.

SMUD indicated that the PTS issue cannot be realistically evaluated by focusing on a single parameter such as RT_{NDT}.

A.4 NRC Staff Audit of Operating Procedures, Operator Qualifications and Training With Respect to the PTS Issue

On March 16, 1982, a NRC short-term task force on PTS was organized to make a detailed review and prepare a report on the efforts on PTS at the H. B. Robinson Nuclear Plant. Specifically, the task force was to provide a report characterizing the problems, methodology of resolution, bases for conclusions and recommendations regarding the adequacy of in-place training programs and operating procedures.

The report[89] "NRC Staff Audit of Robinson and Procedures and Training for Pressurized Thermal Shock" dated April 15, 1982, recommended that prior to restart the Robinson 2 operators and STAs should be retrained in areas related to the PTS issue and that SI termination criteria and procedures be changed to accommodate the PTS issue.

The task force also recommended similar audits be performed at the other seven plants which were identified with the August 21, 1981, letter.

By letter dated April 26, 1982,[90-95] the seven other utilities of plants of concern were requested to cooperate in this effort.

A.4.1 Robinson 2 Audit

A visit to the Robinson 2 site took place on April 5-7, 1982, to evaluate procedures and training. By letter dated April 20, 1982,[96] the staff confirmed the understanding that of general acceptance of the recommendations of the task force report. This was confirmed in writing by CP&L by letter dated May 4, 1982.[97]

A.4.2 Oconee 1 Audit

A review of Oconee's procedures and training for PTS was conducted May 11-13, 1982. In general, the review team found the operators adequately knowledgeable of the PTS issue, except that knowledge of past PTS events at other facilities was weak. The procedures provided mitigative actions to prevent PTS, but needed to be strengthened to provide actions if an unacceptable pressure/temperature condition was reached. The audit team felt that a means should be provided for plotting cooldown rate and subcooling margin with the plant computer out of operation.

A.4.3 Fort Calhoun Audit

A review of the procedures and training for PTS at Omaha Public Power developments Fort Calhoun plant was completed on June 8-10, 1982 by PNL. General recommendations[99] regarding procedures and control instrumentation made by PNL included:

(1) The values of parameters of interest in procedures should be consistent as appropriate.
(2) Emergency procedures should note both minimum and maximum subcooling temperatures.
(3) Emergency procedures should identify only one form of saturation curve.
(4) The current NDT curve should be in every procedure which references it.
(5) The subcooling margin indications should be available for all ranges of RCS temperatures.

A.4.4 San Onofre 1 Audit

An onsite audit was conducted of the San Onofre Unit 1 procedures and training for PTS June 3-4, 1982. Preliminary findings from the audit indicate that the procedures are based on plant-specific analyses of transients and that the operations personnel were familiar with PTS even though their training was not completed at the time of the audit. The findings indicate that the remainder of the training program should include instruction on past cooldown events. Findings on the San Onofre 1 procedures are included in the audit report. The procedures were generally found adequate for PTS considerations, and were based on Westinghouse analysis. The findings indicate that a method for plotting cooldown rate should be provided to the operators.

A.4.5 Maine Yankee Audit

A review of Maine Yankee's procedures and training for PTS was conducted on May 25-27, 1982. The review team found the plant operations personnel and STAs adequately knowledgeable of the PTS issue, and the procedures provided adequate guidance for preventing PTS. One significant operating philosophy already in place at Maine Yankee is the throttling of HPI flow to maintain as close to 50°F subcooling as possible during potential cooldown events. It was noted by the review team that no written exam was conducted after the lectures on PTS. Rather, a seminar method was used to determine the level of comprehension. Questions regarding PTS have been included in the written requalification examinations. The review team concluded that the operators were sufficiently knowledgeable of PTS. No changes to the operating procedures or training program were recommended to meet the objectives of the audit.

A.4.6 Calvert Cliffs 1 Audit

An onsite audit was conducted of the Calvert Cliffs procedures and training for PTS on July 6-8, 1982. Procedure recommendations were identified in the instructions for preferred methods of accident mitigation, priorities for mitigative actions, and procedure cross-reference. Training recommendations included additional training on accident mitigation methods.

A.4.7 Turkey Point 4 Audit

An onsite audit was conducted of the Turkey Point 4 procedures and training for PTS on July 13-15, 1982. No substantive recommendations were identified.

Table A-1 Summary of responses to NRC letters dated August 21, 1981 concerning thermal shock issue

October 7, 1981

Plant Name (Date of licensee's 30 day response to ltr of 8/21/01)	Licensee's 30 day response concerning 5 items to be submitted in 60 days				
	(1) Request for present RT_{NDT}@ plates and welds	(2) Rate of RT_{NDT} increase	(3) RT_{NDT} limit for continued operation	(4) Basis for RT_{NDT} limit	(5) Questions concerning operator actions
Calvert Cliffs 1 (September 24, 1981)	Will be answered within 60 days	Will be answered within 60 days	Do not think a a simple RT_{NDT} value is an appropriate limit.	Will provide a more reasonable criteria for a limit for continued operation in 150 day response.	Will be answered in 60 days
Fort Calhoun (September 22, 1981)	Will provide value for plate material within 60 days. Will provide value for weld material in 150 day response.	Will provide value for plate material within 60 days. Will provide value for weld material in 150 day response.	Do not think a simple RT_{NDT} value is an appropriate limit.	Will provide a response within 60 days. Response will provide basis for response to (3).	Will be answered in 60 days.
Turkey Point 4 (September 23, 1981)	Will provide responses within 60 days.	Will provide responses within 60 days.	Response will be delayed until March 1, 1982.	Information has been provided in 5/14/81 ltr. Will provide additional information with response to (3)	Will be answered in 60 days.
Robinson 2 (September 21, 1981)	Will provide responses within 60 days.	Will provide responses within 60 days.	Response will be deferred until 150 day response.	Information has been provided, however additional informamation will be provided in 150 day response.	Will be answered in in 60 days.

TABLE A-1 (Continued)

Plant Name (Date of licensee's 30 day response to ltr of 8/21/01)	Licensee's 30 day response concerning 5 items to be submitted in 60 days				
	(1) Request for present RT_{NDT} plates and welds	(2) Rate of RT_{NDT} increase	(3) RT_{NDT} limit for continued operation	(4) Basis for RT_{NDT} limit	(5) Questions concerning operator actions
San Onofre 1 (October 5, 1981)	Will be provided by November 4, 1981 (15 day delay).	Will be provided by November 4, 1981 (15 day delay).	Will be provided as soon as made available. Owners Group work will not be complete until end of year.	Will be provided by November 4, 1981 (15 day delay).	Will be provided by November 4, 1981.
Oconee 1 (No 30 day response expected - telephone conversation indicated no conflicts seen).	Will provide 60 day response.	Will provide 60 response.	Will provide 60 day response.	Will provide 60 day response.	Will provide 60 day response.
TMI-1 (October 1, 1981)	Will provide response within 60 days.	Will be answered within 60 days.	Will be answered within 60 days.	Will be answered within 60 days.	Will be answered within 60 days.
Maine Yankee) (September 29, 1981)	Could answer within 60 days, however, would prefer to respond more fully within a short time after 60 day response is due.	Could answer within 60 days, however, would prefer to respond more fully within a short time after 60 day is due.	Could answer within 60 days, however, would prefer to respond more fully within a short time after 60 day resp. is due.	Could answer within 60 days, however, would prefer to respond more fully within a short time after 60 day resp. is due.	Could answer within 60 days, however, 'ould prefer to respond more fully within a short time after 60 day resp. is due.

Table A-1 (Continued)

Licensee's 30 day response concerning the 150 day response	Remarks
Will provide a plan within 150 days. Item 7 of request for additional information will not be complete. Will provide a schedule for complete response.	NRC letter dated 10/22/81 indicated staff would accept what licensee will provide. Staff will continue effort which may result in specifying conservatisms where more definitive info. is not available.
Will provide the 150 day response.	NRC ltr. dtd. 10/27/81 acknowledged receipt of ltr. dt. 10/20/81 and requested licensee to submit weld material data as soon possible as a supplement to ltr. dtd. 10/20/81.
Will provide response Jan. 1, 1982. Will provide schedules & additional analyses for remedial actions by 3/1/82.	NRC letter dated 10/22/81 indicated staff would accept what licensee will provide. Staff will continue effort which may result in specifying conservatisms where more definitive info. is not available.
Will provide 150 day response.	NRC letter dated 10/22/81 indicated staff would accept what licensee will provide.

Licensee's 30 day response concerning the 150 day response.	Remarks
	Staff will continue effort which may result in specifying conservatisms where more definitive info. is not available.
No conflict seen.	NRC letter dated 10/22/81 indicated staff would accept what licensee will provide. Staff will continue effort which may result in specifying conservatisms where more definitive info. is not available.
Will provide 150 day response.	NRC ltr. dtd. 10/22/81 confirms that licensee anticipates no conflicts in providing 60 day and 150 day response.
Intend to submit response following plant-specific analysis established by B&W OG (Oconee 1-Dec. 31, 1981, Rancho Seco-Mar. 1, 1982. Others-later).	NRC ltr. dtd. 10/22/81 indicated that the staff will use Oconee 1 data as it is applicable to TMI-1. Staff encourages licensee to submit info. as it becomes available. Remainder of ltr. similar to to response for Calvert Cliffs 1.
Will respond to II.K.2.13 by mid June 1982. More detailed fracture mechanics data available in 1982.	NRC ltr. dtd. 10/23/81 indicated staff would favor the response according to the licensee's licensee's proposed schedule. The lack of timely information on requested information could result in conservatisms in the staff conclusions.

Table A-2 Summary of 60 day responses to NRC letters dated August 21, 1981, concerning thermal shock to RPV
November 10, 1981

Plant Name (Date of 60 day response)	(1) Initial & Present RT_{NDT} for plates & welds	(2) Rate of RT_{NDT} increase for plates & welds	(3) RT_{NDT} limit for continued oper.	(4) Basis for RT_{NDT} limit	(5) Response concerning operator actions	Remarks
Calvert Cliffs I (10/20/81)	Limiting RT_{NDT} Values Initial Values Plate 20° Welds ° circ. -20° long. 10° @ 4.77 EFPY (12/31/81) Plate 92° Welds circ. 146° long. 178° Peak ID Fluence 7.05x1018 n/cm2	Licensee provided RT_{NDT} values for 7.97 EFPY (12/31/85) in resp. Plate 115° Welds circ. 194° long. 235° Peak ID Fluence 1.18x1019 n/cm2	Licensee does not consider it appropriate to define an upper limit RT_{NDT} value.	Adoption of RT_{NDT} limit would not permit consideration of warm prestressing or other factors.	Operators can control feed rate and terminate HPSI to prevent overpressurization. Generic program will review procedures after detailed analyses of transients.	Licensee RT_{NDT} values are well within the staff's estimate.

Table A-2 (Continued)

Plant Name (Date of 60 day response)	(1) Initial & Present RT_{NDT} for plates & welds	(2) Rate of RT_{NDT} increase for plate & welds	(3) RT_{NDT} limit for continued oper.	(4) Basis for RT_{NDT} limit	(5) Response concerning operator	Remarks actions
Fort Calhoun (10/20/81)	Limiting RT_{NDT} Values Initial Values Plate 10° Welds circ. -20° long. -20° @ 5.36 EFPY (12/31/81) Plate 112° Welds circ. 245° long. 255° Peak ID Fluence - 7.04x1018 n/cm2 RT_{NDT} values are based on generic material properties. Properties for archive material will be provided in 150 day response.	Licensee provided RT_{NDT} values for 8.58 EFPY (12/31/85) in response. Plate 142° Welds circ. 270° long. 268° Peak ID fluence - 1.12x1019 n/cm2	Licensee does not consider it appropriate to define an upper limit RT_{NDT} value.	Adoption of RT_{NDT} limit would not permit considerations of warm prestressing.	HPSI throttling and termination and feedwater throttling criteria are provided in emergency transient procedures to prevent repressurization. CE will review procedures & where warranted procedure revisions will be proposed and evaluated. Following this, procedures will be changed and operating staff retrained as necessary.	Licensees RT_{NDT} values are within the estimates. By letter dtd. 10/23/81 the staff requested properties for archive material as soon as it is available as a supplement to to the 60 day resp.

Table A-2 (Continued)

Plant Name (Date of 60 day response)	(1) Initial & Present RT_{NDT} for plates & welds	(2) Rate of RT_{NDT} increase for plates & welds	(3) RT_{NDT} limit for continued oper.	(4) Basis for RT_{NDT} limit	(5) Response concerning operator	Remarks actions
Maine Yankee (11/2/81)	Limiting RT_{NDT} Values Initial Values Plate -10° Weld -30° As of 9/30/81 - (43 x 106 MWH electric RT_{NDT} @ ID = 180° Peak ID Fluence 5.4x1018 n/cm2	Licensee provided RT_{NDT} values for 26 more calendar years & end of life (35 total calendar years) for welds. 26 more cal. yrs 300° End of life 295°	Licensee does not consider it appropriate to define an upper limit RT_{NDT} value.	The program the licensee is working on considers the many variables involved in the vessel capabilities.	Operators are instructed with procedures to limit repressurization that results frm HPSI operation and removing RC pump operation during transients. Licensee maintains these actions contribute to problem.	Licensees RT_{NDT} values are well within the staff's estimates.
Turkey Point 4 (10/21/81)	Limiting RT_{NDT} Values Initial Values Forging 50° Circ. Welds 3° Current Values 5.61 EFPY (9/30/81 Forgings 85° Circ. Welds 193° Peak ID Fluence - 1.1x1019 n/cm2 Fluence @ 1/4 T 6.6x1018 n/cm2	70/EFPY for next 10 yrs 5°/EFPY for remaining life. For forgings, this represents 30° inc. for remaining design life of vessel.	Licensee stated in letter dtd. 9/23/81 that response will be delayed until 3/1/82.	Licensee stated in letter dtd. 9/23/81 information has been provided in 5/14/81 ltr. Will provide additional information with response to (3).	Licensee indicated no operator action is required for LOCA. Operator action is required within 10 min. for large MSLB. This includes criteria in procedures for HPSI termination and throttling AFW. LOCA procedures have similar procedures. Operators are trained in procedures and on simulator.	Licensees RT_{NDT} values are well within the staff's estimates.

Table A-2 (Continued)

Plant Name (Date of 60 day response)	(1) Initial & Present RT_{NDT} for plates & welds	(2) Rate of RT_{NDT} increase for plate & weld	(3) RT_{NDT} limit for continued oper.	(4) Basis for RT_{NDT} limit	(5) Response concerning operator	Remarks actions
Robinson 2 (10/26/81)	Limiting RT_{NDT} Values Initial Values Plate 46° @ 1/4 T Welds 0° Current Values ID Plate 124° 1/4 T Plate 113° ID Weld 242° 1/4 T Weld 210° Fluence @ ID Plate 1.42x1019 n/cm2 Fluence @ ID Weld 1.30x1019 n/cm2	7°/EFPY for next 10 yrs 5°/EFPY for remaining life. 45° total for plate.	Licensee stated in ltr. dtd. 9/21/81 that responses will be deferred until 150 day response.	Licensee stated in ltr. dtd. 9/21/81 that information has been provided; however, additional information will be provided in 150 day repsonse.	Operators are provided in procedures HPSI termination criteria and FW throttling criteria. HPSI pumps have 1500 psi shutoff heads. Training programs are established.	Licensees RT_{NDT} values are within the estimates.
San Onofre (11/4/81)	Limiting RT_{NDT} Values Initial Values Plate 60° Weld (long.) 0° Current Values @ 8.93 EFPY (10/31/81) Plate 222° Weld (long.) 229°	For Plate 4°/EFPY For Welds 3°/EFPY	Licensee stated that response will be provided upon completion of W Owners Group work.	RT_{NDT} should not be used as sole parameter to determine vessel integrity. Such a limit should be qualified to the specific method of calculation. Refers to Owners Group report of 5/14/81.	Existing procedures provide HPSI termination criteria for LOCA and SLB. Provides no provisions for throttling HPSI. Provided instruction to throttle feedwater. For SLB operator action not required before 10 min. Training programs are provided. HPSI shutoff head is 1160 psi.	Licensees RT_{NDT} values are well within the staff's estimates.

Table A-2 (Continued)

Plant Name (Date of 60 day response)	(1) Initial & Present RT_{NDT} for plates & welds	(2) Rate of RT_{NDT} increase for plates & welds	(3) RT_{NDT} limit for continued oper.	(4) Basis for RT_{NDT} limit	(5) Response concerning operator actions	Remarks
Oconee 1 (10/20/81)	Limiting RT_{NDT} Values Initial Values Plate @ Nozzle 60° Weld circ. 20° long. 20° Current Values 5.13 EFPY (10/1/81) Plate 89° Weld circ. 145° long. 160° Plate Fluence 1.94x1018 n/cm2 Weld Fluence 2.27x1018 n/cm2	Licensee provided fluence rate of increase for peak fluence and for critical weld location. Peak Fluence Rate - 0.37x1018 n/cm2/EFPY Weld Fluence Rate - 0.33x1018 n/cm2/EFPY	Licensee does not consider it appropriate to establish an upper limit RT_{NDT} value.	A RT_{NDT} limit would not provide confidence to predict toughness of materials and assurance that material with the greatest index is the controlling material for a given analysis.	Emergency procedures require operator action for controlling steam line break (overcooling) and LOCA. These include throttling and termination criteria for HPSI. The operator can take manual control of feedwater systems to limit plant cooldown.	The licensee's RT_{NDT} values are within the staff's estimates.

Table A-2 (Continued)

Plant Name (Date of 60 day response)	(1) Initial & Present RT_{NDT} for plates & welds	(2) Rate of RT_{NDT} increase for plates & welds	(3) RT_{NDT} limit for continued oper.	(4) Basis for RT_{NDT} limit	(5) Response concerning operator actions	Remarks
Three Mile Island 1 (10/23/81)	Limiting RT_{NDT} Values Initial Values Plate @ 1/4 T 40° Welds @ 1/4 T 20° Current Values Plate 83° Weld circ. 177° long. 170° Fluence for Plate 2.3x1018 n/cm2 Fluence for circ. welds 2.1x1018 n/cm2 Fluence for long. welds 1.7x1018 n/cm2	For Plate 6.2° RT_{NDT}/EFPY For Circ Welds 22.8° RT_{NDT}/EFPY For Long. Welds 19.9 RT_{NDT}/EFPY These are the current rates. As plant life increases da/dt decreases.	Use of RT_{NDT} as a limiting parameter for continued operation is not considered appropriate by licensee.	The owners group will establish a set of parameters that are expected to be other than RT_{NDT}.	B&W Report BAW 1648 Guidelines have been incorporated in TMI-1 Emergency Procedures. For SBLOCA procedures provide for HPSI throttling (termination) criteria and feedwater control criteria. Training on these procedures is a part of operator training and retraining program.	The licensee's RT_{NDT} values for longitudinal welds are slightly higher than the staff's estimates (10°).

Table A-3 Summary of "150 day" responses concerning PTS

Plant (NSSS vendors)	Response to letters and gen. contents	Conclusions	Limiting transients	Criteria of accept.	Warm prestressing considered	Operator actions considered actions	Remedial Actions
Robinson 2 1/25/82 (W) Referenced WCAP 10019	1. Irradiation data 2. Weld material info. 3. Basis for continued operation 4. Operator actions 5. Remedial actions	31 cal. yrs. of vessel life remaining for all transients considered.	Rpt. provided a table transients considered which include following: 1. Large LOCA 2. SBLOCA 3. LSLB 4. SSLB 5. Rancho Seco	Min. flaw depth for crack initiation is greater than 1.0 in. Crack arrest occurs within 75% of vessel	Yes, for all transients considered.	Refers to WCAP 10019. Credit is taken for LSLB. AFW terminated HPSI terminated in 10 min.	1. Will have low leakage core 2. Will keep abreast on annealing developments. 3. Studying benefits of heating RWST. 4. Verification analysis by EPRI.
Turkey Pt. 4 1/21/82 (W) Referenced WCAP 10019	1. Irradiation information 2. Weld material info 3. Transient fracture analysis showing basis for continued operation.	Reactor vessel integrity will be maintained throughout design life.	Rpt. provided a table of transients considered which include following: 1. Large LOCA 2. SBLOCA 3. LSLB 4. SSLB 5. Rancho Seco	Min. flaw depth for crack initiation is greater than 1.0". Crack arrest occurs within 75% of vessel wall thick.	Yes, all transients except SSLB	Cannot determine but WCAP 10019 provides following: Control AFW	Since integrity has been demonstrated, no need for action plan.

Table A-3 (Continued)

Plant (NSSS vendors)	Response to letters and gen. contents	Conclusions	Limiting transients	Criteria of accept.	Warm prestressing considered	Operator actions considered actions	Remedial Actions
San Onofre 1/25/82 (W) Referenced WCAP 10019	1. Irradiation effects 2. Material property info 3. Basis for continued operation 4. Operation actions 5. Remedial Actions	Reactor vessel integrity will be maintained beyond design lifetime.	Rpt. provided a table of transients considered which include following: 1. Large LOCA 2. SBLOCA 3. LSLB 4. SSLB 5. Rancho Seco	Min. flaw depth for crack initiation is greater than 1.0". Crack arrest occurs within 75% of vessel wall thick.	Yes, for large and small LOCAs only.	For LSLB Terminate HPSI Terminate AFW to faulted SG. For SSLB Isolate break (PORV) Terminate HPSI	Plan for remedial actions not warranted. Low leakage core is place.
Ft. Calhoun 1/18/82 (CE)	1. Thermal-Hydro Eval. (a) SLB (b) Overcooling (antipated occurences) 2. Fracture Mech Analy. for SLB 3. Response to Dec. 18 ltr. 4. Fluence data	Integrity will be maintained for lifetime of plant.	MSLB most limiting. Overcooling AOO-stuck open dump valve. (SBLOCA + LOFW analyzed in CEN 189)	For MSLB (low probability) - crack arrest. For AOO + Single failure - crack arrest For AOO - no crack initiation	Benefit from W.P. not considered, however, it was not needed. It would have been credited if needed and criteria met.	Yes For MSLB - 30 min. to reduce HPSI flow. For MSLB - trip RC pumps in 30 seconds. For AOO trip RCP in 10 min. Reduce HPSI in 90 min.	1. Will implement reduced radial leakage fuel scheme in Cycle B. 2. Will study other fuel arrangement schemes 3. Do not plan increase in ECC water temp. 4. Evaluating annealing. 5. Program plan will evaluate control systems, procedures & potential design mods.

Table A-3 (Continued)

Plant (NSSS vendors)	Response to letters and gen. contents	Conclusions	Limiting transients	Criteria of accept.	Warm prestressing considered	Operator actions considered actions	Remedial Actions
Maine Yankee 1/21/82 (CE)	APP A - response to 4-150 day questions APP B - response to RFI of 8/21/81 APP C - response to 12/18/81 ltr. May do further RETRAN analyses	Vessel will retain integrity throughout design life.	MSLB most limiting (cooldown below 300°)	No crack initiation. Response references CEN 189 Report. Prob. of MSLB is very low.	Benefit from W.P. not considered, however, it was not needed. It would have credited if needed and criteria met.	Yes For MSLB Trip RCP @ 30 sec. Terminate HPSI @ 30 min. For AOO Trip RCP @ 10 min. Terminate HPSI @ 90 min.	1. Low leakage fuel mgmt. for Cycle 7. 2. Will operate RWS to maintain higher temp. not to exceed 80° 3. Will keep informed on annnealing. 4. Will evaluate control strategy after plant-specific evaluation is in place.
Calvert Cliffs 1/28/82 (Resp. to 12/18/81 ltr 1/21/82) (CE)	1. Was responsive 2. Fluence cal. 3. Systems Analysis 4. Fracture mechanics	No crack initiation for assumed plant life for SBLOCA + LOFW. Same for stuck open dump valve (AOO) if needed For MSLB, satisfactory performance for 21 add'l EFPY	MSLB most limiting, AOO + single failure. SBLOCA + LOFW analysis in CE 189	No crack initiation for AOO Crack arrest for MSLB	Benefit from W.P. not considered, however, it was not needed. It would have been credited and criteria met.	Yes For MSLB Trip RCP @ 30 sec. Reduce HPSI flow @ 30 min. For AOO Trip RCP @ 10 min. Terminate AFW @ 10 min. Reduce HPSI @ 90 min.	1. Scoping studies on fuel mgmt. 2. Do not plan to increase RWST temperature. 3. No discussion on annealing. 4. Control system changes may be considered.

Table A-3 (Continued)

Plant (NSSS vendors)	Response to letters and gen. contents	Conclusions	Limiting transients	Criteria of accept.	Warm prestressing considered	Operator actions considered actions	Remedial Actions
Oconee 1 1/15/82 (B&W)	1. Overcooling transient analysis** 2. SBLOCA analysis 3. Mixing analysis 4. Vessel wall thermal analysis 5. Material properties 6. Fluence determination 7. Fracture mechanics analysis 8. Frequency determination 9. SLB analysis **Turbine bypass system failures, overfill transients	Vessel failure is not calculated to result from postulated transient. With minimal downcomer mixing, no credit for mixing in hot leg, no credit for W.P.-16 EFPY. With credit for W.P., for SBLOCA - 32 EFPY. For overcooling transient 25 EFPY (Design life - approx. 27 EFPY)	SBLOCA + LOFW Overcooling transient	Crack initiation with arrest within 1/4 T	Yes, for SBLOCA. No for overcooling transient	SBLOCA + LOFW Trip RCP. Throttle HPIS @ 93 min. Overcooling Transient Trip RCP. Isolate EFWS @ 20 min. Except MSLB - isolate all feedwater within 5 min. Only assumed above actions where necessary to mitigate consequences and achieve acceptable EFPY.	1. 18 month fuel cycle provides decrease in leakage flux. 2. Current water temperature sufficient. 3. In-place annealing not required. 4. No control system changes are necessary.

Table A-3 (Continued)

Plant	Responses and General Contents	Conclusions	Limiting Transients	Criteria of Acceptance	Warm Prestressing Considered
TMI-1 7/7/82 (B&W)	1. Transient Analysis a) Experience b) Overcooling Transient c) SBLOCA 2. Mixing analysis 3. Vessel Wall Thermal Analysis 4. Material Properties 5. Fluence Determination 6. Fracture Mechanics Analysis	Vessel integrity for SBLOCA & stuck open TBV is maintained at 32 EFPY and longer. Assumed 1/4 T flaw-no defect growth.	1. SBLOCA + LOFW 2. 6 Stuck Open	1. No crack initiation during 2. Crack initiation with arrest	No

Operator Action Considered	SBLOCA & LOFW	Overcooling Transient	Remedial Actions
	1. Trip RCP within 2 min. of ESFAS 2. Throttle HPI @ 180 min.	1. Trip RCP within 30 sec. of HPI 2. Throttle HPI and isolate TBV within 12 min.	1. Currently investigating 18 month low leakage fuel cycl 2. Initiating a program to reinforce operator awareness o PTS through review of procedures.

Table A-4 Summary of generic reports concerning PTS

Owners Group	General Contents	Conclusions	Limiting Transients*	Criteria of Acceptance	Warm Prestressing Considered	Operator Action Consider	Potential Remedial Action
Westinghouse WCAP 10019 December 1981 "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants"	1. Limited transient development 2. Fluence Calc. 3. Stress & Fracture Mechanics for Transients 4. Vessel Integrity Evaluations 5. Potential Remedial Actions 6. Conclusions (for each operating plant) 7. Don't address idenfication of events causing highest PTS risk	All plants can continue operation a number of yrs (3 for the least) before acceptance criteria is violated. A table provides no. of yrs. for each plant. Eight plants are 5 yrs or less	**1. Small Steam Line Break 2. Rancho Seco 3. Large Steam Line Break 4. Small LOCA 5. Large LOCA *In order of severity. **Most limiting.	1. No initiation of flaws less than 1 inch deep. (Flaws J 1 in. deep not assumed to exist) or 2. Crack Arrest occurs within 75% of wall thickness.	Benefit of W.P. considered for SBLOCA and some large LOCA and large SL breaks. Benefit was not considered for other transients.	Yes-Control AFW Trip RCPs as examples - Rept. is not very definitive.	1. Heating RWST to 80° - provide of 3 to 30 EFPY operation. 2. Limit AFW 3. Control Systems to mitigate transients a. RC Press. Relief System b. Safety Injection Control c. AFW Control 4. Core Modifications a. Low leakage loading 5. Annealing Vessel a. Is feasible

Table A-4 (Continued)

Owners Group	General Contents	Conclusions	Limiting Transient	Criteria of Acceptance	Warm Prestressing Considered	Operator Action Consider	Potential Remedial Action
Combustion Engineering CEN-189 "Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCAs with Loss of Feedwater for CE NSSS"* December 1981 *(This is the Post-TMI "feed & bleed" rept. It is not a Generic PTS report.)	1. Only addresses SBLOCA with loss of all FW transient 2. Thermal Hydro-analysis 3. Discussions on mixing Additional studies are expected to permit removal of certain conservatisms 4. Scoping studies indicate range of HPSI flows must be considered 5. Fluence Calculations 6. Material Properties 7. Vessel Integrity Evaluation 8. Plant-Specific Analysis	Each plant's vessel can safely withstand SBLOCA + LOFW for design life without crack initiation.	1. Only considers SBLOCA + LOFW **(Note that MSL break is most limiting but was only considered in the 150 day responses)	1. No initiation of flaws of credible size, or if it does initiate 2. Arrest after limited extension.	Benefit of WP was considered	Yes: 1. PORVs opened in 10 min. 2. AFW reinstated after 30 min.	None considered.
No B&W Report -- promised plant-specific analyses. No generic report promised.							

REFERENCES

1. NRC Memorandum dated April 7, 1981, from T. J. Walker to S. S. Pawlicki - Minutes of PWR Owners Groups Meeting with NRC on March 31, 1981.

2. NRC Letters dated April 20, 1981 from D. G. Eisenhut to Licensees of Operating PWR Nuclear Power Plants - Thermal Shock to Reactor Pressure Vessels (Generic Letter 81-19).

3. Summary of Meetings with the Babcock & Wilcox, Westinghouse, and Combustion Engineering Owners Groups on July 28, 29 and 30, 1981, Respectively, Concerning Pressurized Thermal Shock to Reactor Pressure Vessels - dated August 18, 1981.

4. Summary of Meeting with the Westinghouse Owners Group of September 18, 1981 Concerning Pressurized Thermal Shock to Reactor Pessure Vessels - dated October 1, 1981.

5. Summary of Meeting with the B&W Owners Group on September 22, 1981 Concerning Pressurized Thermal Shock to Reactor Pressure Vessels - dated October 1, 1981.

6. Summary of Meeting with the Combustion Engineering Owners Group on October 7, 1981 Concerning Pressurized Thermal Shock to Reactor Pressure Vessels - Dated October 21, 1981.

7. Summary of Meeting with the Westinghouse Owners Group and the Three Selected Owners on February 24, 1982 Concerning the Pressurized Thermal Shock Issue - dated March 8, 1982.

8. Summary of Meeting with Combustion Engineering Owners Group and the Three Selected Owners of CE Designed Plants on March 3, 1982 Concerning the Pressurized Thermal Shock Issue - dated March 12, 1982.

9. Summary of Meeting with Duke Power Company on March 24, 1982 Concerning the Pressurized Thermal Shock Issue for Oconee Unit No. 1 - dated October 31, 1981.

10. Summary of Meeting with Omaha Public Power District on May 6, 1982 Concerning the Pressurized Thermal Shock Issue - dated May 13, 1982.

11. Summary of Meeting with the Westinghouse Owners Group on May 10, 1982 Concerning the Pressurized Thermal Shock Issue - dated May 21, 1982.

12. Summary of Meeting with GPU Nuclear on June 2, 1982 Concerning the PTS issue for TMI-1 - dated June 16, 1982.

13. Summary of Meeting with PWR Industry Representatives on June 9, 1982 Concerning the PTS issue.

14. Summary of Meeting with WOG on Reactor Vessel Integrity on June 22, 1982 Concerning the PTS issue - dated June 30, 1982.

15. Summary of Meeting with CEOG on June 23, 1982 Concerning the PTS issue - dated July 8, 1982.

16. Summary of Meeting with the WOG on July 30, 1982 Concerning the PTS issued.

17. NRC Letters dated August 21, 1981, from Darrell G. Eisenhut to eight selected utilities (Florida Power & Light Company, Carolina Power & Light Company, Southern California Edison Company, Baltimore Gas & Electric Company, Omaha Public Power District, Maine Yankee Atomic Power Company, Duke Power Company and GPU Nuclear Corporation) Concerning Pressurized Thermal Shock to Reactor Pressure Vessels.

18. NRC Letter dated October 2, 1981 from Mr. T. M. Novak, NRC, to Mr. A. E. Lundvall, Jr., Baltimore Gas & Electric Company, Concerning Responses to the NRC August 21, 1981 Letter.

19. NRC Letter dated October 26, 1982, from Mr. T. M. Novak, NRC, to Mr. W. C. Jones, Omaha Public Power District, Concerning Responses to NRC Letter dated August 21, 1981.

20. NRC Letter dated October 23, 1981 from Mr. T. M. Novak, NRC, to Mr. Robert H. Groce, Maine Yankee Atomic Power Company, Concerning Responses to NRC Letter dated August 21, 1981.

21. NRC Letter dated October 26, 1981 from Mr. T. M. Novak, NRC, to Mr. J. A. Jones, Carolina Power & Light Company, Concerning Responses to NRC Letter dated August 21, 1981.

22. NRC letter dated October 26, 1981 from Mr. T. M. Novak, NRC, to Dr. Robert E. Uhrig, Florida Power & Light Company, Concerning Responses to NRC Letter dated August 21, 1981.

23. NRC Letter dated October 23, 1981 from Mr. Gus C. Lainas, NRC, to Mr. R. Dietch, Southern California Edison Company, Concerning Responses to NRC Letter dated August 21, 1981.

24. NRC Letter dated October 23, 1982 from Mr. T. M. Novak, NRC, to Mr. William O. Parker, Jr., Duke Power Company, Concerning Responses to NRC Letteer dated August 21, 1981.

25. NRC Letter dated October 23, 1981 from Mr. T. M. Novak, NRC, to Mr. Henry D. Hukill, Metropolitan Edison Company, Concerning Responses to the NRC Letter dated August 21, 1981.

26.-
33. NRC Letters dated December 18, 1981, to Maine Yankee Atomic Power Company, Baltimore Gas & Electric Company, Omaha Public Power District, Carolina Power & Light Company, Florida Power & Light Company, Southern California Edison Company, Duke Power Company, and Metropolitan Edison Company -

Provided Evaluations of the "60 day" Responses to the NRC Letter dated August 21, 1981, and Requested Additional Information to be Provided in the "150 day" Responses.

34. NRC Letter dated March 15, 1982 to Southern California Edison Company Concerning Request for Information Related to their "150 day" Response.

35. NRC Letter dated March 16, 1982, to Carolina Power & Light Company Concerning Request for Information Related to their "150 day" Response.

36. NRC Letter dated March 16, 1982, to Florida Power & Light Company Concerning Request for Information Related to their "150 day" Response.

37. NRC Letter dated March 16, 1982 to Mr. Oliver Kinglsey, Chairman of WOG, Concerning Request for Information Related to W Generic Program on PTS.

38.-
40. NRC Letters dated March 18, 1982 to Maine Yankee Atomic Power Company, Omaha Public Power District and Baltimore Gas & Electric Company Concerning Requests for Information Related to their "150 day" Responses.

41. NRC Letter dated April 4, 1982 to Duke Power Company Concerning Request for Information Related to their "150 day" Responses.

42. B&WOG Letter dated May 12, 1981 from John J. Mattimoe, Chairman B&WOG, to Harold Denton, NRC, Concerning Report on Reactor Vessel Brittle Fracture Concerns in B&W Operating Plants.

43. CEOG Letter dated May 15, 1981 from Mr. K. P. Baskin, Chairman CEOG, to Mr. Darrell G. Eisenhut, NRC, Concerning Reactor Vessel Pressurized Thermal Shock.

44. WOG Letter dated May 14, 1981 from Mr. Robert W. Jurgensen, Chairman WOG, to Mr. D. G. Eisenhut, NRC, Concerning Thermal Shock to Reactor Pressure Vessel.

45. Baltimore Gas & Electric Company Letter dated September 24, 1981 to NRC Concerning 30 day Response for Calvert Cliffs 1 to August 21, 1981 Letter.

46. Omaha Public Power District Letter dated September 22, 1981 to NRC Concerning 30 day Response to August 21, 1981 Letter for Fort Calhoun.

47. Maine Yankee Atomic Power Company Letter dated September 29, 1981 to NRC Concerning 30 day Response to August 21, 1981 Letter for Maine Yankee.

48. Florida Power & Light Company Letter dated September 23, 1981 to NRC Concerning 30 day Response to August 21, 1981 Letter for Turkey Point 4.

49. Carolina Power & Light Company Letter dated September 21, 1981 to NRC Concerning 30 day Response to August 21, 1981 Letter for Robinson 2.

50. Southern California Edison Company Letter dated October 5, 1981 to NRC Concerning 30 day Response to August 21, 1981 Letter for San Onofre 1.

51. Metropolitan Edison Letter dated October 1, 1981 to NRC Concerning 30 day Response to August 21, 1981 Letter for TMI-1.

52. Baltimore Gas & Electric Company Letter dated October 20, 1981 to NRC Concerning 60 day Response to August 21, 1981 Letter for Calvert Cliffs 1.

53. Omaha Public Power District Letters dated October 20 and November 12 and 13, 1981, to NRC Concerning 60 day Response to August 21, 1981 Letter for Fort Calhoun.

54. Maine Yankee Atomic Power Company Letter dated November 2, 1981 to NRC Concerning 60 day Response to August 21, 1981 Letter for Maine Yankee.

55. Florida Power & Light Company Letter dated October 21, 1981 to NRC Concerning 60 day Response to August 21, 1981 Letter for Turkey Point 4.

56. Carolina Power & Light Company Letter dated October 26, 1981 to NRC Concerning 60 day Response to August 21, 1981 Letter.

57. Southern California Edison Company Letter dated November 4, 1981 to NRC Concerning 60 day Response to August 21, 1981 Letter for San Onofre 1.

58. Duke Power Company Letter dated October 20, 1981 to NRC Concerning 60 day Response to August 21, 1981 Letter for Oconee 1.

59. Metropolitan Edison Company Letter dated October 23, 1981, to NRC Concerning 60 day Response to August 21, 1981 Letter for TMI-1.

60. Baltimore Gas & Electric Company Letters dated January 21 and 28, 1982 to NRC Concerning Request for Information dated December 18, 1981 and 150 day Response to August 21, 1981 Letter, Respectively, for Calvert Cliffs 1.

61. Omaha Public Power District Letter dated January 18, 1982 to NRC Concerning 150 day Response to August 21, 1981 Letter for Fort Calhoun.

62. Maine Yankee Atomic Power Company Letter dated January 21, 1982 to NRC Concerning 150 day Response to August 21, 1981 Letter for Maine Yankee.

63. Carolina Power & Light Company Letter dated January 25, 1982 to NRC Concerning 150 day Response to August 21, 1981 Letter for Robinson 2.

64. Florida Power & Light Company Letter dated January 21, 1982 to NRC Concerning 150 day Response to August 21, 1981 Letter for Turkey Point 4.

65. Southern California Edison Company Letter dated January 25, 1982 to NRC Concerning 150 day Response to August 21, 1981 Letter for San Onofre 1.

66. Duke Power Company Letter dated January 15, 1982 to NRC Concerning 150 day Response to August 21, 1981 Letter for Oconee 1.

67. WOG Letter dated December 30, 1981 to NRC from O. D. Kinglsey, Chairman WOG, Concerning WCAP-10019 "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants."

68. CEOG Letter dated December 32, 1981 to NRC from K. P. Baskin, Chairman CEOG, Concerning CEN-189 "Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCAs with Loss of Feedwater for Combustion Engineering NSSS."

69. WOG Letter dated May 28, 1982 to NRC from Mr. O. D. Kinglsey, Chairman WOG, Concerning Response to Request of NRC Letter dated March 16, 1982.

70. Carolina Power & Light Company Letter dated May 4, 1982 Concerning the NRC Requests for Information dated March 16 and April 20, 1982.

71. Florida Power & Light Company Letter dated May 3, 1982 to NRC Concerning Response to NRC Request for Information dated March 16, 1982.

72. Southern California Edison Company Letter dated May 25, 1982 to NRC Concerning Response to NRC Request for Information dated March 16, 1982.

73. WOG LEtter dated June 16, 1982 from O. D. Kingsley, Chairman WOG, to H. R. Denton, NRC, Concerning Report "Fuel Management to Reduce Neutron Flux."

74. WOG Letter dated June 22, 1982, from O. D. Kinglsey, Chairman WOG, to H. R. Denton, NRC, transmitting report "Review of Emergency Response Guidelines Relative to PTS."

75. Baltimore Gas & Electric Company Letter dated May 4, 1982 to NRC in Response to NRC Request for Information dated March 18, 1982.

76. Omaha Public Power District Letter dated April 30, 1982 to NRC in Response to NRC Request for Information dated March 18, 1982.

77. Maine Yankee Atomic Power Company Letter dated May 11, 1982 to NRC in Response to NRC Request for Information dated March 18, 1982.

78. CEOG Letter dated June 14, 1982 from Ken Baskin, Chairman CEOG, to H. R. Denton, NRC, Concerning the Proposed NRC recommendations on the PTS issue.

79. Omaha Public Power District Letter dated June 28, 1982 to NRC as a result of the meeting of June 23, 1982 with the staff.

80. Duke Power Company Letter dated April 30, 1982 in Response to NRC Request for Information dated April 5, 1982.

81. GPU Nuclear Corporation Letter dated March 17, 1982 to NRC Concerning 150 day Response to August 21, 1981 Letter for TMI-1.

82. GPU Nuclear Letter dated June 1, 1982 to the NRC Concerning a summary of the "150 day" response to the August 21, 1981 letter and request for information letter dated December 18, 1981.

83. B&WOG Letter dated June 22, 1982 from A. P. Rochino, Chairman of B&WOG, to H. R. Denton, NRC, Concerning the staff proposed recommendations on the PTS issue.

84. Duke Power Company Letter dated June 21, 1982, Concerning the staff's proposed recommendations on the PTS issue.

85. Arkansas Power & Light Company Letter dated June 21, 1982 Concerning the staff's proposed recommendations on the PTS issue.

86. Florida Power Corporation Letter dated June 28, 1982 Concerning the staff's proposed recommendations on the PTS issue.

87. GPU Nuclear Letter dated July 7, 1982 Concerning the staff's proposed recommendations on the PTS issue.

88. SMUD Letter dated June 21, 1982 Concerning the staff's proposed recommendations on the PTS issue.

89. NRC Internal Memorandum from G. R. Maetis, Chairman of Robinson PTS Task Force, to H. L. Thompson, Acting Director of DHFS, dated April 15, 1982 concerning staff audit of Robinson 2 procedures and training for PTS.

90.-
95. NRC Letters dated April 26, 22, 22, 22, 22 and 23 to Florida Power & Light Company, Maine Yankee Atomic Power Company, Baltimore Gas & Electric Company, Omaha Public Power District, Southern California Edison Company, and Duke Power Company requesting cooperation in the audit and evaluation effort on plant procedures and operator training related to PTS.

96. NRC Letter dated April 20, 1982 confirming the Carolina Power & Light Company's commitment to the recommendations of the NRC Robinson 2 Task Force on PTS.

97. Carolina Power & Light Company Letter dated May 4, 1982 confirming the utility's commitment to the recommendations of the NRC Robinson 2 Task Force on PTS and providing additional information which was requested in the NRC March 16, 1982 letter.

98. Pacific Northwest Laboratory's Letter dated June 4, 1982 Concerning the review of the procedures and training for PTS at Oconee 1.

99. NRC Summary of June 8-10, 1982 meeting with OPPD regarding the procedures and operator training relative to the PTS issue - dated June 16, 1982

100. NRC summary of July 30, 1982, meeting with WOG regarding SBLOCA which result in stagnation flow, frequencies of such events and differences between staff and WOG concerning the fracture mechanics analyses dated August 9, 1982.

101. NRC Summary of August 11, 1982 meeting with WOG to resolve differences between the staff and WOG concerning the screening criteria dated August 20, 1982.

102. GPU Nuclear letter dated July 7, 1982, to NRC transmitting the TMI-1 report concerning PTS (the "150 day" response to the NRC August 21, 1981, letter).

103. WOG letter dated July 15, 1982, from O.A. Kingsley, Chairman of WOG to H.R. Denton, NRC, concerning the June 22, 1982 meeting with the staff on the PTS issue.

104. WOG letter dated September 2, 1982, from O.P. Kingsley, Chairman WOG to H.R. Denton, NRC concerning the July 30 are August 11, 1982, meetings with the staff on the PTS issue.

105. WOG letter dated September 16, 1982, from O.D. Kingsley, Chairman WOG to H.R. Denton as a follow-up to the letter dated September 2, 1982.

APPENDIX B

CHRONOLOGY OF EVENTS CONCERNING PRESSURIZED THERMAL SHOCK

December 31, 1980 B&W licensees submitted Thermal Mechanical Report (BAW-1648).

March 31, 1981 - NRC meeting with PWR Owners Groups concerning thermal shock with repressurization issue. Owners Groups committed to a report by May 15, 1981 to put thermal shock issue into perspective.

April 20, 1981 letter to all operating PWR licensees requesting Owners Groups Reports by May 15, 1981 and licensee's responses by May 22, 1981.

May 4, 1981 - Commission Information Paper (SECY-286 (2B)).

May 12, 1981 - Board Notification.

May 15, 1981 - Received May 15 reports from Owners Groups.

May 19, 1981 - ACRS subcommittee meeting to discuss Owners Groups responses.

May 28-June 4 - Received responses from all operating PWR licensees.

June 5, 1981 - ACRS Briefing.

June 11, 1981 - Commission Briefing.

July 28, 29, 30, 1981 - Meetings with Babcock & Wilcox, Westinghouse and Combustion Engineering Owners Groups.

September 15, 1981 - Commission Briefing.

August 21, 1981 - NRC letter to eight licensees of eight operating PWR plants (Fort Calhoun, Robinson 2, San Onofre, Maine Yankee, Turkey Point 4, Calvert Cliffs 1, TMI-1 and Oconee 1) requesting 60-day response and 150-day response concerning Thermal Shock.

September 21 through October 5, 1981 - Received letters from 7 licensees in response to the August 21, 1981 letter identifying conflicts with the request.

September 18, 1981 - Meeting with Westinghouse Owners Group.

September 22, 1981 - Meeting with Babcock & Wilcox Owners Group.

October 7, 1981 - Meeting with Combustion Engineering Owners Group.

October 23-28, 1981 - Letters to eight licensees regarding their exceptions to the August 21, 1981 letter.

October 20 through November 13, 1981 - "60 day" responses from the eight licensees who received the August 21, 1981 letter.

December 8, 1981 - Commission Paper SECY 81-687 dated December 8, 1981, Subject: Designation of PTS as an Unresolved Safety Issue.

December 18, 1981 - NRC evaluations and request for information concerning "60 day" response.

December 30, 1981 - Westinghouse Owners Group Report Concerning Pressure Vessel Integrity.

December 31, 1981 - Combustion Engineering Owners Group Report concerning TMI Action Item II.K.2.13.

January 15 through January 25, 1982 - "150 day" responses to August 21, 1981 letter from seven utilities. GPU did not submit a "150 day" response for TMI-1.

February 24, 1982 - Meeting with WOG to discuss the WOG generic report and the "150 day" responses of three W plants of PTS concern.

March 3, 1982 - Meeting with CEOG to discuss the CEOG generic report and the "150 day" responses of the three CE plants of PTS concern.

March 5, 1982 - Commission Information Paper (SECY 82-97) Subject: Commission Briefing on PTS.

March 9, 1982 - Commission Briefing, Status Report on PTS.

March 16, 1982 - Appointment of Special Task Groups to (1) investigate the reducing of irradiation damage to vessels, and (2) audit the operator training and procedures for the PTS concern at Robinson 2.

March 15, 16, 18 and 24, April 5, 1982 - Letters to seven of the eight licensees of the PTS concerned plants and the WOG requesting additional information related to the "150 day" responses and the generic reports.

March 24, 1982 - Meeting with Duke Power Company to discuss the Oconee 1 "150 day" response.

March 26, 1982 - Transmittal of Task Action Plan for USI A-49, "Pressurized Thermal Shock" (PTS).

March 24, 1982 - Meeting with Duke Power Company concerning the PTS issue for Oconee 1.

April 15, 1982 - Report of special task force on PTS for Robinson 2.

May 20, 1982 - Preliminary Assessment of Techniques for Fluence Rate Reduction for PWR Pressure Vessels.

April 30-May 4, 1982 - Received responses from licensees of special plants concerning PTS to NRC request for information during March 1982.

May 6, 1982 - Meeting with OPPD concerning PTS issue. Discussed OPPD's response of April 30, 1982.

May 10, 1982 - Meeting with WOG concerning the PTS issue. Discussed response of WOG due at end of May.

April 26, 1982 - Licensees of other six special plants of PTS concern requested to cooperate in audits of operating procedures and training.

May 28, 1982 - Received WOG Supplemental Information on Reactor Vessel Integrity.

June 2, 1982 - Meeting with GPU Nuclear concerning PTS, Summary of "150 day" response.

June 3, 1982 - ACRS Meeting - Discussed staff's consideration of possible recommendations for PTS requirements.

June 9, 1982 - Meeting with PWR industry representatives concerning the staff's considerations of possible recommendations for PTS requirements.

June 22, 1982 - Meeting with WOG concerning PTS issue - Followup to June 9, 1982 meeting.

June 23, 1982 - Meeting with CEOG concerning PTS issue - Followup to June 9, 1982 meeting.

June 16, 1982 - WOG report on Fuel Management to Reduce Neutron Flux.

June 22, 1982 - WOG report on PTS review of ERGs.

June 21-July 7, 1982 - Responses from CEOG, OPPD, B&WOG, and licensees of all operating B&W plants concerning staff's consideration of proposed recommendations on PTS requirements.

July 9, 1982 - Meeting with OPPD concerning PTS issue.

July 30, 1982 - Meeting with WOG concerning PTS issue - discussed staff's position on PTS issue.

August 11, 1981 - Meeting with WOG concerning the PTS issue - Discussed the differences between the staff and WOG in the evaluation of the critical RT_{NDT} for the limiting transient and provided a discussion of the staff's proposed position regarding a screening criteria.

APPENDIX C

PROCEDURES AND TRAINING

C.1 Human Factors Considerations

It was recognized by the task force early in the review of Pressurized Thermal Shock (PTS) that plant operators played a key role in the evaluation and mitigation of PTS events. There are some key considerations that must be evaluated in determining the acceptability of operator action as a mitigative action. The sum of these considerations is the need to provide a well-balanced, integrated approach to accident mitigation that is based on technical analysis and considers PTS in the context of other related concerns, such as core cooling, environmental releases, and containment integrity.

The first consideration is that reactor vessels have been designed to withstand the worst design-basis accident. The consequences of a vessel failure are so significant that we have always required vessel and system design adequate to prevent it. The second consideration is a concern for the ability of the operators to "prevent" PTS from breaking a vessel. Operators in general are excellent throughout the industry. But any human can make errors, both cognitive and operative. The likelihood of error increases with an increase in stress, poor control room design, fatigue, instructions inadequate to deal with the particular sequence in progress, and other similar factors. Because of possible human errors and the potential severe consequences of PTS, the NRC does not consider operator action an acceptable long-term "solution" to the PTS issue.

However, the NRC staff recognizes that there is a genuine need to provide clear, concise, and integrated procedures and training to the operators, to ensure they know the technical issues involved not only for this issue, but for other vital considerations they must be concerned with in plant operations. After the TMI-2 accident, NRC-directed 'enhancements' to HPI termination criteria were developed by the industry. The results of these changes is, as perceived by the NRC staff, a 'mindset' to maintain HPI flow after an accident at all costs. Current analysis of accidents with continuous HPI flow shows that the challenge to vessel integrity is more severe than previously considered. In subsequent evaluations, the staff and the industry have learned that real events, with multiple failures, have led to transient cooldowns more severe than previously analyzed. This led the staff to recognize that a balance of considerations must be used to control the operation of HPI and other safety-related plant equipment. The industry should have a clear understanding of those considerations, including understanding of PTS, in determining the best method of operating plant equipment.

C.1.1 Westinghouse Plants

The utilities of the three Westinghouse-designed plants being evaluated for PTS provided a list of procedural steps dealing with HPI termination and control of feedwater. These steps were provided in the 60-day responses to D. G. Eisenhut's August 21, 1981 letter to the eight plants being evaluated for PTS. In response to D. G. Eisenhut's December 18, 1981 letter to the three utilities with Westinghouse-designed plants, additional procedures information was provided at the same time the 150-day response to our August 21, 1981 letter was provided.

At a meeting in February 1982, in Bethesda, Md., Westinghouse presented to the NRC staff an evaluation of the PTS mitigative actions contained in the Westinghouse guidelines, which were developed in response to NUREG-0660 Item I.C.1. The guideline for steam line breaks includes modified HPI termination criteria, to account for vessel integrity considerations, as described in the following: The HPI termination criteria require a level in the steam generator, a level in the pressurizer, adequate subcooling margin, and a minimum pressure. For vessel integrity considerations, the minimum pressure for HPI termination has been lowered in the steam line break guideline from 2000 psig to 700 psig when primary loop temperature is below 350°F.

A letter from S. Varga to the three Westinghouse licensees dated March 16, 1982 requested evaluations regarding the need and effectiveness of upgrading current procedures, and requesting a formal commitment to upgrade operator understanding of PTS. Responses from Carolina Power and Light (H. B. Robinson), Florida Power and Light (Turkey Point 4) and Southern California Edison (San Onofre 1) dated May 4, 1982, May 3, 1982, and May 20, 1982, respectively, were received.

C.1.1.1 H. B. Robinson

C.1.1.1.1 Present Procedures

H. B. Robinson's emergency operating procedures were based on the Westinghouse guidelines developed in response to NUREG-0660 Item I.C.1. They include the modified HPI termination criteria for steam line breaks. CP&L stated in their 150-day response that they believe procedures governing operator action and programs governing operator training should provide a balanced approach to handling transients and accidents. Their heatup and cooldown curves are used to define acceptable operation to prevent PTS events. An additional training program on the recent PTS concerns was completed March 31, 1982. H. B. Robinson continues to tie their efforts into the Westinghouse procedures development effort. Modifications to the Robinson procedures are being made, as outlined in Section C.1.1.1.3.

C.1.1.1.2 Present Operator Training

As stated in the previous section, H. B. Robinson believes in a balanced approach to operator training. As described in their 150-day response, CP&L has committed to assuring that each of their operators has a complete understanding of the PTS issue. CP&L stated in a June 25, 1982 letter that operator training has been upgraded as outlined in the staff audit report.

C.1.1.1.3 Plant Audit

On April 5-7, 1982, the procedures and training related to PTS were audited at the site. Some specific changes were recommended to the operating procedures to lower the required minimum pressure for HPI termination and to provide explicit instruction for pressure control during cooldown. More specific training was recommended, to include instruction on previous overcooling events, walk-throughs of procedures as a shift team, and CP&L evaluation of the shift's ability to cope with a PTS event. In a letter from E. Eury to T. Novak dated May 4, 1982, CP&L committed to address the staff's recommendations and identified other procedure modifications required as the result of Westinghouse's review of the guidelines on which the procedures are based.

C.1.1.2 Turkey Point 3

C.1.1.2.1 Present Procedures

Turkey Point Units 3 and 4 emergency operating procedures were developed based on the Westinghouse guidelines developed in response to NUREG-0660 Item I.C.1. As stated in the FP&L 150-day response, they include the modified HPI termination criteria for steam line breaks, and specific direction to terminate HPI when termination criteria are met. Operating pressure-temperature limit curves are included. Emergency operating procedures provide instructions to (1) minimize RCS cooldown rate and (2) prevent repressurization following overcooling. In a letter from R. Uhrig to S. Varga dated May 3, 1982, FP&L committed to modify their procedures based on the information provided in the staff's H. B. Robinson audit report.

Additionally, FP&L stated that other NRC concerns with existing procedures will be resolved in the guidelines (and subsequent procedures) developed in response to NUREG-0737 Item I.C.1. These concerns include both technical and human factors considerations.

C.1.1.2.2 Present Operator Training

As stated in the FP&L 150-day response, pressure-temperature limit curves are presented and discussed in the Licensed Operator Training Program. Simulator training includes handling overcooling transients. In a letter from R. Uhrig to S. Varga dated May 3, 1982, FP&L stated they will be augmenting operator training based on the staff's recommendations in the H. B. Robinson audit report.

C.1.1.2.3 Plant Audit

On July 13-15, 1982, the procedures and training related to PTS were audited at the site.

C.1.1.3 San Onofre 1

C.1.1.3.1 Present Procedures

San Onofre 1's emergency operating procedures were developed based on the Westinghouse guidelines developed in response to NUREG-0660 Item I.C.1. As stated in Southern California Edison's 60-day response, they include modified HPI termination criteria for steam line breaks, but do not provide specific direction to terminate HPI when termination criteria are met. In a recent procedure modification made for the Systematic Evaluation Program evaluation of steam line breaks, the operators were specifically directed to terminate HPI. Information obtained from the staff's H. B. Robinson audit report is also incorporated into the recently revised procedures.

C.1.1.3.2 Present Operator Training

As stated in their 150-day response, SCE provided formal operator training for PTS during the operator requalification training program conducted in February 1982. Recent format changes to procedures, modified HPI termination pressures, and upgraded knowledge of steam generator tube ruptures have recently been incorporated into the San Onofre 1 emergency operating procedures. These procedures changes will require additional training of the San Onofre 1 operators, prior to startup from their current outage.

C.1.1.3.3 Plant Audit

An onsite audit of the San Onofre Unit 1 procedures and training for PTS was conducted on June 2-4, 1982. Preliminary findings from the audit indicated that the procedures are based on plant-specific analyses of transients and that the operations personnel were familiar with PTS even though their training was not completed at the time of the audit. It was noted that the remainder of the training program should include instruction on post-cooldown actions. Recommendations regarding the San Onofre 1 procedures are included in the audit report. The procedures were generally found adequate for PTS considerations, and were based on Westinghouse analysis. It was recommended that a method for plotting cooldown rate should be provided to the operators.

C.1.2 Combustion Engineering Plants

The licensees of the three Combustion Engineering (CE)-designed plants being evaluated for PTS provided a description of the procedural actions for dealing with HPI termination and control of feedwater.

These steps were provided in the 60-day response to D. G. Eisenhut's December 18, 1981 letter to the three licensees of CE-designed plants. Additional procedures information was provided at the same time the 150-day responses to the August 21, 1981 letter were provided. At a March 1982 meeting in Bethesda, Md., licensee representatives of CE plants presented to the NRC staff an evaluation of the mitigative actions in the specific plant procedures. A letter from R. Clark to A. Lundvall, Jr., dated March 18, 1982, requested additional information regarding the basis and sensitivity of operator action assumed in the analyses performed for the 150-day responses. Responses from Omaha Public Power District (Fort Calhoun), Baltimore Gas and Electric (Calvert Cliffs) and Maine Yankee Atomic Power Company (Maine Yankee) were received by letters dated April 30, 1982, May 4, 1982, and May 11, 1982, respectively.

C.1.2.1 Fort Calhoun

C.1.2.1.1 Present Procedures

Fort Calhoun's emergency operating procedures include design-basis events to cover the requirements for "10 CFR Part 50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." In OPPD's 60-day response, a detailed list of procedural actions was provided, including explanations of their applicability to PTS. In their 150-day response, Omaha Public Power District, OPPD stated that based on their evaluation of their procedures, and on the analysis performed for the PTS issue, changes to the

Fort Calhoun procedures should be made. These changes included the need to provide specific criteria for HPI and charging termination, and improved cautions to assure operator compliance with cooldown curves. These changes, and any other additional modifications based on the plant's analysis, were to be completed by June 1, 1982. These changes were in place at the time of the audit of the Fort Calhoun procedures and training. Fort Calhoun's procedures also require HPI operation for at least 20 minutes, based on previous NRC requirements.

C.1.2.1.2 Present Operator Training

As described in OPPD's 60-day response, the operator training program is part of a two-phase effort. The first phase is performing analysis for PTS. The second phase is determining modifications, if any, that may be necessary, including procedure changes. Retraining of operators will be conducted on the procedure revisions. Instruction on the PTS issue have been conducted for the operators and for all levels of OPPD's management.

C.1.2.1.3 Plant Audit

An onsite audit was conducted at Fort Calhoun to determine the level of operator understanding of PTS concerns. This audit was conducted June 7, 1982. Preliminary findings from the audit indicate that the procedures were generally adequate for PTS, and are based on a plant-specific analysis performed by CE. The operators were generally knowledgeable of the PTS issue. Recommendations from the audit team included the upgrading of pressure-temperature curves, and the consolidation of NDT, saturation and subcooling curves onto one plot for more effective utilization of the curves by the operators.

C.1.2.2 Calvert Cliffs

C.1.2.2.1 Present Procedures

Development of Calvert Cliffs' procedures is part of a two-phase program to address PTS. The first phase is the development of analyses for PTS. The second phase involves changing plant operating procedures, if necessary. In

BG&E's 60-day response to D. G. Eisenhut's August 21, 1981 request for information, specific procedural actions were provided, related to operation and termination of HPI and charging flow, and control of feedwater. An evaluation of the Calvert Cliffs' procedures has been conducted by plant personnel who feel that the procedures adequately address PTS, considering the risk involved. BG&E considers an integrated, analyzed approach to plant operations, of which PTS is one concern, to be the only reasonable approach to responsible plant operations. As stated in a letter dated May 4, 1982, from R. Bryant to D. Eisenhut, BG&E agrees with the staff that vessel integrity concerns should be properly addressed. Changes to Calvert Cliffs' procedures have been made to remind the operators to observe the vessel integrity-related pressure temperature limits. BG&E stated that they will continue to be involved in the CE Owners' Group efforts for emergency operating procedures upgrades for NUREG-0737 Item I.C.1.

C.1.2.2.2 Present Operator Training

As described in BG&E's 60-day response, operator training will be conducted on any operating changes resulting from the plant's analysis. Operator training based on the changes identified was completed by June 30, 1982.

C.1.2.2.3 Plant Audit

An onsite audit was conducted of the Calvert Cliffs' procedures and training for PTS on July 6-8, 1982. The following changes to the Calvert Cliffs procedures were recommended: (1) provide clearer instructions for preferred methods of accident mitigation, (2) predetermine priorities of mitigative actions, and (3) upgrade procedure cross-references. Recommendations for training improvements included the need for additional training on accident mitigation methods, to include pressure-temperature control in various abnormal conditions (e.g., with and without vessel upper-head bubbles, and with and without forced circulation).

C.1.2.3 Maine Yankee

C.1.2.3.1 Present Procedures

Maine Yankee Atomic Power Company (MYAPC) provided a summary of their operator actions in their 60-day response to D. G. Eisenhut's August 21, 1981 letter to the eight plants being evaluated for PTS. These actions include criteria for HPI and charging termination and feedwater operation. The report further stated that a maximum subcooling limit was already in the plant's procedures. The subcooling limit is 200°F, and was based on pressurizer overstress concerns.

C.1.2.3.2 Present Operator Training

The training program described in the 60-day response included a discussion of operator training on emergency operating procedures, emphasizing maintenance of 50°F subcooling. A more detailed training outline was provided in the 150-day response, and included technical as well as operational information. A schedule, included in the 150-day response, showed that training for operating crews was completed by June 1982, and that RO and SRO trainees would receive training in this area.

C.1.2.3.3 Plant Audit

A review of Maine Yankee's procedures and training for PTS was conducted on May 25-27, 1982. The review team found the plant operations personnel and Shift Technical Advisors adequately knowledgeable of the PTS issue, and the procedures provided adequate guidance for preventing PTS. One significant operating philosophy already in place at Maine Yankee is the throttling of HPI flow to maintain as close to 50°F subcooling as possible during potential cooldown events. It was noted by the review team that no written exam was conducted after the lectures on PTS. Rather, a seminar method was used to determine the level of comprehension. Questions regarding PTS have been included in the written requalification examinations. The review team concluded that the operators were sufficiently knowledgeable of PTS. No changes to the operating procedures or training program were recommended.

C.1.3 Babcock & Wilcox Plants

Oconee Unit 1 is the B&W-designed plant being evaluated for PTS. All analyses performed by B&W are specific to the Oconee plant.

C.1.3.1 Oconee

C.1.3.1.1 Present Procedures

Oconee 1's current emergency operating procedures include design-basis events to cover the requirements of 10 CFR Part 50 Appendix B and were developed based on design analysis. In the Duke Power Company 60-day response to D. G. Eisenhut's August 21, 1981 letter to the eight licensees whose plants were being evaluated for PTS, a discussion of the procedural actions related to PTS were provided. The actions discussed included feedwater operation, HPI operation, and instrumentation. Also included was a graph of pressure vs. temperature, with allowable operating regions indicated for conditions with and without reactor coolant pumps operating. In the graph's notes, the operators are instructed to maintain a 50°F to 100°F subcooling band with RCPs off. Duke Power Company stated in their 150-day response that based on the analysis presented in their letter, no major changes in existing plant procedures were considered necessary. The letter also stated that when implemented, the Abnormal Transient Operating Guidelines will include appropriate operator instructions for mitigation of overcooling transients.

C.1.3.1.2 Present Operator Training

As stated in Duke Power Company's 60-day response, the Oconee operators receive instruction on HPI termination and feedwater control during requalification training. Training on plant response and emergency operating procedures is also conducted on the B&W simulator. The 150-day response further stated that Duke Power recognizes the importance of ensuring operators have sufficient training and the procedures are adequate to prevent the occurrence of severe thermal shock events. Additional training to augment operator understanding of PTS is to be conducted, but Duke considers the current knowledge of in-place plant procedures to be acceptable for the short term. Duke also stated that their operators have been made aware of the PTS concern, although no formal training has been conducted.

C.1.3.1.3 Plant Audit

A review of Oconee's procedures and training for PTS was conducted May 11-13, 1982. In general, the review team found the operators adequately knowledgeable of the PTS issue, except that knowledge of past PTS events at other facilities was weak. The procedures provided mitigative actions to prevent PTS, but needed to be strengthened to provide actions if an unacceptable pressure-temperature condition was reached. The audit team felt that a means should be provided for plotting cooldown rate and subcooling margin with the plant computer out of service.

C.2.0 Conclusions

Technical guidelines for Emergency Operating Procedures (EOPs) are being developed generically by the NSSS vendor owners' groups in response to TMI Action Plan Item I.C.1, "Short-Term Accident Analysis and Procedures Revision." These guidelines and their supporting analyses will address the actions required for mitigating a wide range of accidents and transients including multiple failures and operator errors. These guidelines will include the operator actions necessary to prevent or mitigate pressurized thermal shock. Incorporation of PTS concerns in the guidelines is beneficial and more effective than current procedures in that the analyses supporting the guidelines will verify that the mitigating actions for PTS do not result in inadequate core cooling or other problems. The Westinghouse Owners' Group has reviewed its existing Emergency Procedures Guidelines and is considering the PTS issue in developing the remainder of these guidelines. This effort was completed in July 1982. The B&W Owners' Group has incorporated desired operating regions in the Anticipated Transient Operating Guideline (ATOG) for Oconee, which take PTS concerns into account. The B&W approach is considered acceptable until the long-term PTS program has been implemented. The CE Owners' Group has submitted draft Emergency Procedure Guidelines which provide a desired operating range for pressure and temperature. The CE guidelines are presently being reviewed by the staff. Another CE Owners' Group activity deals with verifying the "correctness" of the actions specified in the guidelines with respect to the PTS issue. The revision of the guidelines to be submitted in August 1982 will include the results of this activity.

The NRC staff recognizes that the owners' groups' efforts on the emergency procedure guidelines would not be completed until late 1982 and staff review will not be completed until early 1983. The staff considers this schedule acceptable considering the low probability of occurrence of PTS events, the past operating history of PTS precursor events, and upgrades in instrumentation reliability resulting from the Rancho Seco and Crystal River events. Nevertheless, the staff has undertaken a program to audit the procedures and training covering pressurized thermal shock at the following plants: H. B. Robinson, Oconee 1, San Onofre 1, Maine Yankee, Fort Calhoun, Turkey Point 3, and Calvert Cliffs 1. The purpose of these audits is to assess the adequacy of current procedural steps and operator training necessary to mitigate PTS events and to determine if corrective actions are required before the longer term PTS program provides acceptance criteria and generic resolution of the issue. Additional plant-specific recommendations or generic recommendations may result from these audits.

Based on the audits conducted to date, the staff concludes that industry operators are generally knowledgeable of the PTS issue and of the mitigative actions for PTS included in their procedures. Further, the procedures reviewed, with some specific exceptions delineated in the reports, provide a scheme for mitigation of PTS events. The procedures are usable for PTS, and can be understood by the operators.

C.2.1 Audit Results

The general conclusion based on review of the seven plant's procedures and training programs was that operators are adequately knowledgeable of the technical issues involved in PTS, and were aware of procedures guidance currently existing for dealing with PTS.

C.2.2 Procedures

The seven plants currently being evaluated for PTS have reviewed their current emergency operating procedures for instructions relevant to the PTS issue.

H. B. Robinson's procedures, based on generic Westinghouse guidelines, included in its HPI termination criteria a minimum required pressure of 2000 psig, with HPI shut-off head at approximately 1500 psig. This could have resulted in extended HPI operation when not desired. Based on an NRC audit and the licensee's evaluation, H. B. Robinson has lowered the minimum pressure for HPI termination to 1560 psig, changed the temperature monitoring for operation from the RCS hot leg to the RCS cold leg, strengthened the emphasis on terminating HPI when its termination criteria are met, and provided more detailed instructions on RCS pressure and temperature control. The staff finds the H. B. Robinson procedural guidance adequate for the immediate PTS concern.

Turkey Point 3 procedures, based on generic Westinghouse guidelines, contain specific direction for HPI termination when the criteria are met. Personnel at Turkey Point 3 have reviewed the staff's H. B. Robinson audit report and made changes to Turkey Point's procedures based on the findings from the staff's H. B. Robinson audit report. Based on these commitments the staff finds the Turkey Point 3 procedural guidance adequate for the immediate PTS concern. Further verification was conducted during the onsite audit.

San Onofre 1 procedures, based on generic Westinghouse guidelines and modified for the SEP evaluation, contain directive actions for termination of HPI and the information learned from the H. B. Robinson evaluation. The staff finds the San Onofre 1 procedural guidance adequate for the immediate PTS concern. Findings from the onsite audit are included in the audit report.

Fort Calhoun procedures provide some specific guidance to the operators for operation of HPI, charging, and feedwater. Omaha Public Power District (OPPD) has identified changes necessary to provide criteria for HPI termination to reflect the PTS concern and improved precautions to assure operator compliance with cooldown-based pressure-temperature curves. The staff concurs with the need for these changes. A reevaluation of the requirement for running HPI for at least 20 minutes after initiation should be made by OPPD and the staff. We strongly recommend removing any minimum running time requirements for HPI. A more detailed evaluation of Fort Calhoun procedures was conducted during the onsite audit.

Calvert Cliffs' procedures provide some specific guidance to the operators for operation of HPI and feedwater. Cautions and a Technical Specification are intended to provide assurance that HPI or feedwater flow will be terminated prior to vessel challenge. The instructions, by themselves, do not provide the specific guidance the staff feels is necessary. They should include a directive action step for control of HPI. A determination was made during the plant audit that some procedures modifications were necessary for the operators to effectively deal with PTS.

Maine Yankee's procedures provide some specific guidance to the operators for operation of HPI, charging and feedwater, including a subcooling band (50°F minimum, 200°F maximum). Maine Yankee has requested, in their discussions, that the staff reevaluate its position on requiring HPI flow for a minimum of 20 minutes, and on requiring immediate RCP trip after a safety injection. We concur that this needs to be done before PTS can be completely addressed in any plant's procedure. The staff finds the Maine Yankee procedural guidance adequate for the immediate PTS concern.

Oconee Unit 1's procedures provide some specific guidance to the operators for operation of HPI and feedwater. When below 500°F, the operators are instructed to maintain a subcooling band (50°F minimum, 100°F maximum). The operator is specifically directed to throttle HPI when 50°F subcooling is reached. The staff finds the Oconee procedural guidance adequate for the immediate PTS concern.

C.2.3 Training

The seven plants currently being evaluated for PTS have all stated that they are augmenting their operator training for PTS. The staff conclusions regarding individual plants are included in each audit report.

C.2.3.1 Improvements in Emergency Operating Procedures

Westinghouse performed an evaluation of procedural actions for PTS by reviewing, step by step, guidelines that have a realistic technical basis. In reviewing the technical basis for each step, a determination could be made of its applicability to the PTS concern. This program shows the importance and viability of an integrated approach to accident mitigation, where new technical problems can be evaluated in a manner that includes incorporation of concerns of other technical issues. Combustion Engineering and Babcock & Wilcox have done a significant amount of work on developing their own approach to generic guidelines. All three owners' groups are developing guidelines to be function-oriented, in accordance with NUREG-0737, Item I.C.1. This approach to accident mitigation will provide a means to significantly reduce operator error by providing mitigative actions that are not dependent on diagnosis of specific transients or accidents. This approach will increase the accuracy of operator response by reducing complex diagnostic problems to a prioritized, simplified, function-level response that will be used even if an event is incorrectly diagnosed.

The staff concurs, and strongly encourages, the approach stated by the seven plants being evaluated to ensure that the guidelines and subsequent plant procedures developed in accordance with NUREG-0737 Item I.C.1 address PTS, as well as coordinate the PTS actions with actions to mitigate other serious transients or accidents. We believe this is the best method to provide an integrated set of emergency operating procedures to deal with a wide range of transients and accidents, and will provide the analytic base for evaluation of future technical problems.

In reviewing industry responses to comply with NUREG-0737 Item I.C.1, the staff will review the technical guidelines for emergency operating procedures (EOPs) and will review a description of how EOPs are developed from the guidelines for each operating plant. This will provide assurance that procedures at each plant will be based on analysis of PTS and other events. This review will be performed for all operating reactors and operating license applicants.

C.3.0 Recommendations

In view of the above described ongoing programs to develop and implement a set of integrated procedures, and considering the low risk due to PTS events from plants below the screening RT_{NDT} (which is where all plants are, currently), we conclude that current procedures and training are acceptable, for the present. However, since PTS risk increases with further irradiation, we make the following recommendations for the future:

(1) The industry and the NRC staff will ensure that actions to mitigate PTS are included in the technical guidelines developed for NUREG-0737, Item I.C.1. The NRC staff will, in their review of the analyses that form the basis for the technical guidelines, ensure that the actions specified in the guidelines are based on an integrated evaluation of relevant technical considerations, including PTS, core cooling, environmental releases, and containment integrity.

(2) When plants are determined to be within 3 years of exceeding the staff criteria for RT_{NDT}, those licensees should be required to implement upgraded Emergency Operating Procedures which are based on technical guidelines developed for NUREG-0737, I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents." All licensees will be required to implement these upgraded procedures on a schedule developed in accordance with SECY-82-111B.

The procedures developed and implemented by items (1) and (2) above should address the following types of concerns:

(a) Instructions should include allowance for system response delay times (e.g., loop transport time, thermal transport time).

(b) The need for cooldown rate limits for periods shorter than one hour should be evaluated.

(c) Methods for controlling cooldown rates should be provided.

(d) Guidance should be provided for the operator if cooldown rates or brittle fracture limits are exceeded.

(e) The desired region of operation (e.g., subcooling band) on the pressure-temperature curve should be evaluated to determine if it can be revised to maximize the operator's ability to prevent brittle fracture.

(f) Instructions for controlling pressure following depressurization transients should be provided.

This item should be completed in the same time frame stated in item (1).

(3) The staff recommends that the initial training on the procedures developed from the guidelines discussed in recommendation 1 and 2 above include a specific section on the technical concerns of PTS, and the specific manner in which the procedures provide the mitigative actions. This training should be integrated into each plant's overall training program.

(4) The staff recommends that training programs for periodic operator requalification include the recommendations of item (3) above.

This item should be implemented at the first requalification training cycle following implementation of the upgraded procedures.

The staff feels that these recommendations provide balanced approach to ensure the adequacy of operator response to PTS events. This is accomplished by determining the adequacy of operator understanding at the plants of most concern, then providing for all plants the best available means to ensure the procedures used for plant operation cover a wide range of transients and accidents, while covering a wide variety of multiple failures.

APPENDIX D

REACTOR VESSEL FRACTURE MECHANICS ANALYSIS

The vessel integrity analyses, the results of which are reported in this document, include a determination of the temperature distribution across the vessel wall versus time, the thermal stresses as a consequence of this temperature distribution, as well as fracture mechanics results. The analyses were performed either by the NRC staff using its in-house program or by ORNL using the OCA program. These programs are described in the following sections. Illustrations of typical temperature, stress and stress intensity factor distributions across the vessel wall at a certain time in the transient are shown in Figures D-1 (a), (b) and (c) respectively. It should be noted in Figure D-1 (c) that the stress intensity factor, K_I, for long axial cracks is higher than for long circumferential cracks, especially for cracks that extend relatively deep into the vessel wall. K_I is due to contributions from thermal stresses, pressure stress and other stresses that may be present. Superposed in Figure D-1 (c) are K_{Ic}, the vessel toughness that determines crack initiation, and K_{Ia}, the toughness at crack arrest. When K_I exceeds K_{Ic}, crack initiation is expected (for axial cracks having depths between points C and C^1 in the diagram), if warm prestressing is not effective (warm prestressing is discussed in D.3). The crack would then grow to a depth where K_I intercepts the arrest curve, K_{Ia} (point A in the diagram). Similar results would occur for a circumferentially oriented crack except that arrest will generally occur at the shallower depths.

Equivalent calculations are made at other times into the transient and the results cross-plotted on a critical crack depth diagram as shown on Figure D-2 (b). Also shown in Figure D-2 (b) is the depth at which the upper shelf toughness of the metal is reached (nominally 200 ksi $\sqrt{in}$). If the arrest point falls above the upper shelf, arrest is assumed not to occur.

Figure D-2 (a) illustrates the trend of K_I for a particular crack depth versus time for a hypothetical PTS transient. If pressure remains constant or decreases with time, K_I will increase to a maximum and then decrease as the thermal stresses die out. The time at which K_I reaches its maximum determines the time of warm prestressing (WPS). When the entire initiation curve falls to the right of the WPS time as shwon in Figure D-2 (b), crack initiation is not expected to occur; however, care must be exercised in reaching this judgment because of analytical and material uncertainties. The dotted line in Figure D-2 (b) indicates that crack initiation might occur because of uncertainties and might reinitiate later in time because of an increase in K_I at that time. If this were to occur, arrest is not expected because then the arrest curve is above the upper shelf toughness.

D.1 NRC Analytical Procedures

The NRC procedure to evaluate the effect of cooldown transients and postulated accident scenarios on the integrity of reactor vessels was developed in 1978 and subsequently updated to include technological data as they become available. It is designed primarily for investigations of thermal shock to the beltline region of vessels with a vessel radius to wall thickness ratio of about ten.

Heat transfer algorithms are based on classical closed form solutions which provide temperature distributions across a vessel wall versus time into a cooldown transient. These temperature profiles are used to calculate thermal stresses versus time and depth into the wall. The calculation of fracture mechanics stress intensity factors is based on the linear-elastic boundary integral equations method for cylinders and the superposition of stresses due to all causes particularly those due to temperature differences, pressure and residual stresses in welds. Although certain simplifying assumptions are usei in the procedure, its results have been compared with those from more sophisticated analyses and found to be in good agreement.

D.1.1. Assumptions

Geometry

For heat transfer and thermal stress analyses, slab geometry is assumed. For typical reactor vessels with a vessel radius to wall thickness ratio of ten or more, the error introduced by this assumption is negligible compared to other uncertainties inherent in the analyses. This assumption permits a more simple calculational procedure that is adaptable to programmable calculators or computers. Cylindrical geometry is used, however, in the fracture mechanics analyses.

Heat Flow

In a cooldown or heatup transient, heat flow is assumed to occur only in the wall thickness direction. Thus, the procedures are one-dimensional.

Heat Transfer Coefficient

The heat transfer coefficient, h, during a typical transient can vary over a considerable range depending on the hydraulic and thermal conditions. Its magnitude may even be difficult to determine versus time as the transient progresses because of hydraulic and other uncertainties. However, for values of heat transfer coefficients in the range of interest for most thermal transients (approximately 300 Btu/hr ft^2 °F), short perturbations to higher values do not cause significant increases in thermal stresses. Therefore, for typical transients of interest, metal temperature and stress distributions are obtained by utilizing a constant heat transfer coefficient. The value used is conservatively selected on the basis of experience and judgment. For maximum conservatism, a value of infinity can be used. The heat transfer coefficient is also assumed to be the same at all water cooled portions of the vessel wall.

Temperature Dependence of Metal Properties

The physical properties of materials are temperature dependent. When thermal transients result in a significant temperature range and difference through the vessel wall, accurate results require consideration of this phenomenon. Data for materials of interest are taken from recent ASME publications.

Analytical Model

Prior to the thermal transient, the water temperature is assumed to have remained constant for a sufficiently long time so that the vessel wall is at a uniform temperature equal to the water temperature. Prior to and during the transient, heat flow at the outer insulated surface of the vessel is assumed to be zero and the vessel cooled or heated only at the inner surface with no sources of heat within the metal. For typical transients of interest, these assumptions introduce minimum uncertainties in the end results.

Finite Number of Series Terms

Solutions for metal temperature distributions at various times during a transient are in the form of an infinite series. Because of obvious practical considerations, it is necessary to truncate the series to a finite number of terms. The error introduced by limiting the number of series terms is significant only at or very shortly after the start of the transient (time = zero) where an infinite number of terms is required to obtain correct temperatures. Shortly thereafter, however, higher terms in the series decay rapidly to insignificant values. Because, for transients of interest, the maximum thermal stresses generally occur relatively late in the transient, little or no error is introduced by utilizing a finite number of terms. Six series terms are used for deterministic analyses; however, the last two terms contribute very little. Therefore, for probabilistic analyses only four terms are used.

Effect of Cladding

Because the material and physical properties of the stainless steel cladding differ from those of the carbon steel wall, the cladding effect must be accounted for in reactor vessel integrity analyses. The presence of cladding affects the heat transfer and stress calculations as well as the fracture mechanics analyses. The heat transfer coefficient is readily adjusted to account for the higher thermal resistance of the stainless steel clad (Figure D-3). The stress effect of the clad, however, depends on the stress relief and operational history of the vessel. Once this is established, this effect is accommodated by superposition of cladding induced stresses with those from other causes including those due to temperature variations across the wall of the vessel (Figure D-4). Fracture mechanics effects of cladding depend on the assumed shape and location of postulated cracks. Procedures for treating long through-clad axial cracks are used. The treatment of elliptical and circumferential cracks needs further development. In general, thermal stresses and stress intensity factors due to temperature differentials across the wall are calculated assuming only the thermal resistance of the clad, then calculating the clad induced stresses and stress intensity factors assuming a constant metal temperature and superposing the results. Thus, the effect of cladding is accounted for in the heat transfer, thermal stress and fracture mechanics analyses when long axial through-clad cracks are assumed.

The NRC model for determining the clad effect for postulated long through-clad cracks is as follows:

- Assume that the clad is stress-free at reactor operating temperature. As the vessel wall cools down, tensile stresses in the cladding and lesser compressive stresses in the base metal develop and reach a clad stress of about 30 ksi at room temperature.

- The average clad temperature is assumed to be the cooled surface temperature during a transient; however, to determine the incremental effect due to the clad, the entire wall temperature is assumed to be constant (the effect of the actual temperature variation across the wall during a transient is superposed later). The lower thermal conductivity of the cladding is included in the determination of the surface temperature by a reduction in the heat transfer coefficient.

- Knowing the clad incremental stress, the stress intensity incremental effect due to the clad is then calculated via the influence function technique described briefly in Section D.1.4.

The results of an example calculation of the clad effect on the stress intensity factor as determined independently by the staff and ORNL are shown in Figure D-5.

D.1.2 Stress Algorithms

The total peak stresses (thermal plus pressure plus residual plus any other stresses) are assumed to be less than, or at least not significantly larger than, the material yield strength so that components of stress can be added and that linear-elastic fracture mechanics procedures can be utilized. For rapid thermal transients, high stresses usually occur locally at the inner vessel wall and acceptable stress distributions (total stress below yield) over the remaining section can still be obtained if the overstressed region is relatively thin.

D.1.3 Postulated Initial Cracks

Long through-clad cracks, either axial or circumferential, are assumed to exist in the welds of limiting (highest) RT_{NDT}. In this case, the cladding effect is conservatively applied in that the stresses due to the different expansion coefficients of the clad and base metal are added to the nominal thermal stresses. For short through-clad cracks or underclad cracks it is conceivable that the cladding can have a beneficial effect if the cladding is sufficiently tough, that is, it is less affected by irradiation damage than the base material. In this case, it could deter crack elongation or could even prevent crack initiation depending on the specific transient. At present, there are differences of opinion as to clad toughness after irradiation, and further research is needed as to the behavior of short or underclad cracks in an overcooling event. Also, analyses to date omit consideration of weld residual stresses and in the case of circumferential cracks, the effect of dead weight stresses. An uneven temperature distribution in the azimuthal direction increases K_I value for circumferential cracks. Therefore, the NRC concludes that the more conservative assumption of long through-clad cracks should be used at least for scoping calculations, until further information s developed to permit a relaxation of this assumption.

D.1.4 Fracture Mechanics Algorithms

Fracture mechanics analyses utilize the linear-elastic boundary integral methods of Heliot, Labbens and Pellissier-Tanon (References D.1 and D.2).

At each time step, the thermal and other stresses are expressed as polynomial functions of the relative depth intc the wall of the vessel:

$$\sigma\left(\frac{X}{L}, t\right) = \sum_{j=o}^{4} \sigma_j \left(\frac{X}{L}\right)^j$$

where σ_j's are constants determined by curve fitting. The stress intensity factor for this stress distribution is then;

$$K_I = \sqrt{\pi a} \sum_{j=o}^{4} \sigma_j \left(\frac{a}{L}\right)^j i_j$$

In the NRC procedure, the i_j's are expressed as polynomial functions of the relative crack depth. Different expressions for the i_j's are used for different crack geometries and directions.

The stress distribution due just to the cladding, however, cannot be expressed by a polynomial equation without resorting to a large number of terms. For this application, the staff used the basic equations in the references and adapted them to obtain an expression for long axial cracks in a cylinder (expressions for other crack geometries and directions need further development):

$$K_I = \sqrt{\pi a} \left(\frac{2}{\pi - 2}\right) \left\{ \left(i_o - \frac{2}{\pi}\right) \int_o^{\frac{\pi}{2}} \sigma(\omega)\, d\omega - (i_o - 1) \int_o^{\frac{\pi}{2}} \sin \omega\, \sigma(\omega) d\omega \right\}$$

where:

$$\sin \omega = \frac{X}{a}, \quad o \leq x \leq a$$

and i_o is the influence function for a uniform stress.

D.2 ORNL Analytical Procedures, OCA-I, OCA-II

In addition to performing its own PTS analyses, the NRC staff also utilized the services of ORNL. The ORNL analytical code differs from that of the NRC, yet compatible results are obtained. The ORNL program is described in Reference D.3 from which some of the following is taken.

The OCA-I code is a computer program that performs a two-dimensional linear-elastic fracture mechanics analysis for long axial inner-surface flaws in a cylinder subjected to time-dependent thermal and pressure loadings. Six basic calculations are performed: (1) a one-dimensional thermal analysis to obtain temperature distributions through the wall of the cylinder as a function of time; (2) stress analysis, neglecting presence of flaw, using thermal and pressure loadings; (3) calculation of stress intensity factor (K_I) as a function of flaw depth and time; (4) calculation of static initiation and arrest toughness values (K_{Ic} and K_{Ia}) at tip of flaw as a function of flaw depth and time; (5) calculation of K_I/K_{Ic} and K_I/K_{Ia} as a function of flaw depth and time; and (6) construction of the critical-crack-depth curves, which indicate the behavior of the flaw at all times during the transient.

Input to the thermal analysis includes the coolant temperature vs. time, the fluid-film heat transfer coefficient, and the initial temperature of the cylinder. All necessary material properties, with the exception of the reference temperature (RT_{NDT^o}) and the concentrations of specific impurities (copper and phosphorous), are included in OCA-I, but different values may be inputted. The calculation of K_{Ic} and K_{Ia} considers the temperature and fast-neutron-fluence distributions through the wall, RT_{NDT^o} and the copper and phosphorous concentrations, which influence the radiation damage effect.

The K_I calculation is based on a superposition technique that uses the uncracked-cylinder stresses and a set of unit-load K_I values (K^*) that correspond to cylinder dimensions typical of a 1000-MW(e) pressurized-water reactor pressure vessel (4.37-m ID x 4.80-m OD). The K^* values were calculated using finite-element techniques and are included in OCA-I.

The development of OCA-I was prompted by a growing interest in the behavior of surface flaws in reactor pressure vessels during overcooling accidents. The OCA-I code was designed specifically for these accidents in an effort to minimize time and expense associated with the analysis. To this end, special provisions were made for parametric-type analyses. OCA-II, which was used for later studies, includes plotting refinements plus the incorporation of the cladding effect in the stress intensity factor.

The OCA-II code (Reference D.4) which was developed by Oak Ridge National Laboratory utilizes:

(a) the latest NRC calculation method for determining neutron fluence attenuation with depth into the vessel wall, which is described in Section D.4 of this report,

(b) the latest NRC calculation method for determining shift in RT_{NDT} with neutron fluence, which is described in Section D.4 of this report,

(c) a finite element, one-dimensional code with a constant heat transfer coefficient, h, in thermal analyses,

(d) thermal, pressure and clad stresses and infinitely long axial through-clad crack in the finite element linear-elastic fracture mechanics (LEFM) analyses,

(e) any prescribed water temperature during the transient.

The OCA-II code and NRC LEFM analyses performed for this study do not include plate-to-plate weld residual tensile stresses. We believe that OCA-II and NRC stress, thermal and fracture mechanics analyses are sufficiently conservative to permit a parametric study of vessel fracture without including these stresses.

D.3 Warm Prestressing

Although warm prestressing (WPS) can theoretically prevent crack initiation during a pressurized thermal shock transient, the staff believes that the fluctuations of pressure and temperature during these transients are possible; therefore, our scoping calculations did not rely upon WPS to prevent crack initiation. The NRC staff believes that it would not be prudent for operators to rely primarily on warm prestressing to assure reactor vessel integrity during pressurized thermal shock transients. The staff is aware of and accepts the theoretical basis for warm prestressing. One explanation for the WPS effect is:

> "During temperature reduction, initiation of crack propagation from an arrested crack in the reactor vessel cannot occur while the K value is constant or decreasing." (Reference D.5)

Another explanation rests on a physical picture of blunting at the crack tip and development of favorable residual stresses caused by the warm prestress.

The theoretical basis for warm prestressing assumed that K_I is decreasing with time in monotonic fashion after it reaches its maximum value. In a real transient, the pressure component of K_I may rise and fall in an unpredictable fashion as the system is being brought to a stable condition. Some variation in the thermal component of K_I may also occur. Of particular concern to the staff is that emergency operating procedures at some facilities permit repressurization after a thermal transient to as high as 2000 psig. Thus, the potential benefit effects of WPS may be deliberately negated.

Experiments have shown that when there is an increase in K_I after cooldown to a temperature at which K_I exceeds K_{Ic}, there is an ever-increasing probability of fracture as K_I increases such that the probability is very nearly one for $K_I = K_I$ maximum. The probability of fracture decreases to acceptably low values for $K_I - K_{Ic} \leq 25$ percent of (K_I max. - K_{Ic}). (Reference D.6) The experimental information also shows clearly that the beneficial effects of warm prestress are nearly eliminated if K_I drops to a low value after reaching K_I-maximum and then increases, for example, repressurization late in transient after the vessel has cooled down.

During a typical transient scenario, the reactor coolant temperature and pressure both decrease initially from their normal operating values. Thereafter, the trend of both temperature and pressure depends on the nature of the even and the actions taken by operators and/or automatic systems. Because of the relatively rapid decrease of the reactor coolant temperature, thermal stresses are developed in the vessel wall which are superposed on the pressure stresses.

The net result is an increase of the total stress intensity factor, K_I versus time. The thermal component of K_I reaches a maximum and then decreases and the wall temperature tends toward a uniform value. Typically, the total K_I also has a maximum during this initial period. Thereafter, the change in K_I versus time depends on the assumed actions taken by operators and by automatic systems.

There are, of course, many possible variations in the cooldown scenario that will produce different degrees of departure from the ideal monotonic decrease of K_I after reaching K_I-maximum. Our knowledge is insufficient to draw the line between acceptable versus unacceptable transients with regard to the acceptance of warm prestressing, other than to say that we ought not to rely on it at this time. The exceptions are transients such as certain LOCAs where pressure is limited as described below.

Following a severe cooldown transient, the NRC staff believes that facility operators should limit reactor pressure by manual and/or automatic means to the extent practicable. Preferably, reactor pressure should be decreased monotonically consistent with 50°F subcooling and the pressurizer water level increased only to its normal operating range. In particular, water-solid conditions should be avoided especially if the reactor coolant reaches low temperatures. Repressurization should not be permitted until the transient has been evaluated, and for severe transients, the vessel should be inspected to assure its integrity.

Even if these procedures are followed, it still is conceivable that a small crack may initiate and grow deeper. However, in the absence of pressure, it will not penetrate the wall. With pressure stresses also present, it is possible that a crack would create an opening in the vessel, especially when the wall material has cooled down.

In conclusion, the staff believes that it would not be prudent to rely on warm prestressing to assure reactor vessel integrity during a pressurized thermal shock transient. The basis for this position rests on uncertainties regarding system considerations and on insufficient experimental information to confirm the benefits of warm prestressing under these circumstances at this time.

While the odds in favor of warm prestressing being a viable phenomenon to prevent initiation or reinitiation of a crack during a particular transient scenario may be relatively high, facility operators should also consider the relative risk.

For small break LOCAs of sufficient size such that the pressure is limited to some low value during the critical period of the transient or is monotonically decreasing because of the inability of the ECC and charging systems to maintain high values, then conditions are attained where warm prestressing can be effective and credit can be considered for it.

D.4 Determination and Utilization of Material Toughness

To make the fracture analyses of pressurized thermal shock, it is necessary to have values for the fracture toughness of the material at the tip of the postulated cracks in the reactor vessel wall. Toughness must be known as a function of time in the transient, and temperature and fluence must be known as a function of position in the wall.

D.4.1 ASME Code Section XI Curves

The fracture analyses performed by utilities, vendors and the NRC have all utilized the values of K_{Ic} and K_{Ia} given in Section XI of the ASME Code and reproduced in Figure D-6. The toughness values are given as a function of the temperature, T minus RT_{NDT}, the reference temperature, nil-ductility transition. The quantity, RT_{NDT} is the sum of two quantities; the initial RT_{NDT} and the ΔRT_{NDT} caused by irradiation. Appendix E of this report describes the bases for estimating initial RT_{NDT} and ΔRT_{NDT} for the individual plants. Estimates are given for the inside surface of the vessel wall (at the clad-base metal interface) for the critical locations, which are almost always the welds, either a longitudinal weld or a circumferential weld in the beltline. The second step is to determine the attenuation of ΔRT_{NDT} through the vessel wall.

D.4.2 Attenuation of Fluence and RT_{NDT} through the Vessel Wall

Some recent changes have been made in the way the attenuation calculations are performed. These are illustrated in Figure D-7. In the past, the attenuation of fluence has been calculated by an exponential equation fitted to the results of calculations given in surveillance reports, as follows.

$f = f_o e^{-.33x}$

f = fluence at any point, n/cm^2 (E> 1 MeV)

f_o = fluence at inside wall

x = distance from inside wall, inches

However, changes in the neutron energy spectrum within the wall cause the use of the above formula to be unconservative. Therefore, the NRC has chosen to use displacements per atom (dpa) as the damage function, following a report received from HEDL (Reference D.7). They provided six plots of the ratio, dpa/fluence ($E > 1$ MeV), versus depth in the vessel wall. At 8.0 inches, the ratio averaged 2.06. To achieve this reduction in the attenuation at 8.0 inches, the equation for fluence attenuation becomes:

$$f = f_o e^{-.24x}$$

Thus, we use a "dpa equivalent" attenuation equation, while retaining the description of fluence in terms of n/cm^2 ($E > 1$ MeV).

As illustrated in the lower part of Figure D.7, the combination of the dpa-equivalent equation for attenuation of fluence and the Guthrie trend cur e formula gives an expression for the attenuation of RT_{NDT} that is much less steep than that previously used. We believe that the new expression is realistic and have incorporated it into our program and the OCA-II code described in Section D.2.

The staff recognizes that the technical community is not in total agreement that dpa is the best physical model for the correlation of neutron damage as a function of fluence where the neutron energy spectrum varies significantly.

However, it does seem important to account for spectral changes and dpa is the most generally accepted method.

D.5 Stress/Fracture Mechanics Procedures Summary

The analytical methods used by NRC, ORNL and vendors differ somewhat but yield essentially the same results if all input assumptions are the same. Differing conclusions result primarily from assumptions as to crack shape, clad effects, effect of warm prestressing, etc.

When materials properties and the transient are known, these procedures can predict crack behavior quite well as demonstrated by results from the ORNL thermal shock experiments.

For generic studies, the NRC uses an exponential decay of water temperature to envelope a variety of transients. The staff has also used the Rancho Seco event as an analytical model. Our objectives are:

Avoid crack initiation, if possible.
Avoid vessel failure, in any event.

The staff has studied the PTS issue both deterministically (conservative assumptions) and probabilistically (mean values of parameters) to assess risk to a vessel.

D.6 Discussion of Results

The NRC has performed both deterministic and probabilistic fracture mechanics analyses to generate a basis for judgment regarding the safety margins against PTS transients especially for the more highly irradiated vessels. Although recognizing that the transients that occurred at Rancho Seco in 1978 and Ginna in 1982 are unique, and are very unlikely to happen in the same way again, the staff concludes that they provide measures of the severity of a PTS event. The NRC and ORNL have arbitrarily utilized an idealized Rancho Seco pressure-temperature transient as a benchmark model for other vessels.

For generic investigations, however, a postulated exponential decay of water temperature has proved to be more appropriate in that it can be characterized by two parameters, β (min.$^{-1}$) which is the reciprocal time constant and T_f (°F) the final postulated equilibrium temperature. The initial temperature, T_o, is the normal operating temperature. Thus, $T_w = T_f + (T_o - T_f)\, e^{-\beta t}$ Information obtained from transients that have actually occurred at nuclear facilities indicates that the above formulation adequately describes the water temperature at least for the initial critical portion of the transient. After T_f is approached, operator and/or systems action can, of course, affect the longer term variation of water temperature. Typical values of β have been found to be in the range of 0.05 to 0.15 min.$^{-1}$ as a consequence of the physical limitations of a real facility. Higher values of β of about 0.5 min.$^{-1}$ or more have been estimated for hypothetical, low probability design basis transients. Typical values of T_f for the worst cooldown transients to date (including those at BWR facilities) are in the range of 250 to 300°F. The Rancho Seco event, for instance, resulted in β of about 0.05 min.$^{-1}$ and a T_f of about 290°.

For the more likely transients, the times of crack initiation have been calculated to be 20 to 30 minutes or more after the onset of cooldown, the actual time varying up to one hour depending on the pressure and RT_{NDT}. Thus, operators have time to gain control of the event if properly instructed and trained.

The OCA II Code was utilized to determine the lowest RT_{NDT} for crack initiation as a function of constant pressure, final water temperature (T_f) and the reciprocal time constant (β). From these data, Figures D-8 and D-9 were plotted which indicate the effect of T_f, β and pressure on crack initiation.

The principal objective of the NRC (and the industry) is to prevent crack initiation, and for more probable PTS events, this may be possible. However for the less likely events such as a postulated small break LOCA, crack initiation is likely in vessels with a relatively high RT_{NDT}. For these cases, the objectives must be to prevent any crack from propagating through the wall. Early in the transient, a pre-existing crack can initiate and propagate to the order of half the wall thickness or somewhat less, and then arrest because the metal at this location is still much warmer than at the cooled surface and because the metal at this depth has experienced less irradiation damage. As previously mentioned, linear elastic fracture mechanics (LEFM) methods are used for analysis of PTS transients. Typical values of K_I at the first crack initiation range from 60 to 100 ksi $\sqrt{in}$.

LEFM is not valid for tough materials such as are encountered on or above the upper shelf. Other techniques are necessary. These techniques have been developed for analysis of piping flaws and for pressure vessels under pressure loads only (Task Action Plan A-11). To date, they have not been adequately developed for treatment of more complex stress patterns such as occur in a PTS event.

Therefore, the NRC conservatively assumes that, if a crack is calculated to propagate above the upper shelf of the material (200 ksi $\sqrt{in}$ is assumed), it is assumed to continue propagating through the wall. It is recognized that subsequent elastic-plastic or fully plastic analyses may show that this may not be the case. On the other hand, it must be recognized that if the pressure is high enough, crack propagation through the wall is possible, even in tough material, because the remaining ligament may not be sufficient to sustain the pressure and residual thermal stress loads. Pending further research in this area, the NRC concludes that a conservative approach must be taken.

D.6.1 OCA-II Parametric Study

The OCA-II code was used to make a parametric study of the effects of pressure P, final water temperature, T_f, and the reciprocal time constant, β, on the critical values of RT_{NDT} at the inside wall for crack initiation and crack penetration through the wall (no arrest). (Strictly speaking, initial RT_{NDT} should be mentioned as a variable, because it is only ΔRT_{NDT} that attenuates through the vessell wall, but the difference in critical values of RT_{NDT} for different initial RT_{NDT} values is negligible.)

Some of the results of the parametric study, plotted in Figure D-8, show that the T_f - RT_{NDT} is a fairly reasonable normalizing parameter, although the curves for different T_f values are separated by as much as 10-20 degrees at low pressure. Figure D-8 indicates that crack initiation will occur at lower material RT_{NDT} as pressure increases or final water temperature decreases. Figure D-9 indicates that crack initiation will occur at lower material RT_{NDT} as β increases, but, the effect is slight for values of β greater than 0.15. The "dogleg" in the curves of Figures D-8 and D-9 occurs because the critical crack size changes. At low pressure, K_I - thermal predominates in the fracture analysis and the critical crack sizes are a fraction of an inch, whereas at high pressure the critical size is near the arbitrary limit of 1.25 inches.

A cross plot of these Figures, shown in Figure D-10, illustrates the effect of pressure and T_f on the critical value of RT_{NDT}, for a given value of β (0.15 min.$^{-1}$). To use Figure D-9, the plant condition as characterized by RT_{NDT} is related to the transient severity as characterized by P, T_f and β to determine if the vessel is safe from crack initiation. This is, of course, a deterministic calculation, which assumes that the critical flaw depth given by the analysis is indeed present in the critical weld. Stated in another way, if the value of RT_{NDT} used is the true value, the probability of crack initiation is the probability that the critical flaw is indeed present.

Also shown in Figure D-10 is a set of "no arrest" lines, which merge with the solid lines for crack initiation at about 600 psig. This means that at very low pressure, cracks will arrest if T_f is between the solid line and the dashed line. At higher than 600 psig, the analysis shows that a crack, once it has initiated, will penetrate the vessel wall. The assumptions on which this analysis is based are thought to be conservative--they assume that the material will behave as indicated by linear-elastic fracture mechanics.

Finally, in Figure D-10 there is a steeply slanting dashed line marked "Circumferential cracks." It was drawn on the basis that at low pressure K_I - thermal is the same for cracks of any orientation (which is nearly true for shallow cracks) and on the basis that K_I - pressure for circumferential cracks is approximately one-half of that for axial cracks.

The fluid film heat transfer coefficient, "h", is another variable (in addition to T_f, P and β) that is part of the characterization of a transient. The parametric study described above was made using an "h" of 1000 Btu/hr ft^2 °F, which is characteristic of a "pumps on" condition. To check the effect of a change in "h" to 300, for a "pumps off" condition, eight cases were repeated, using OCA-II. The results, shown in the following table, are the differences in critical RT_{NDT} (in degrees F) for a calculation using h = 300 minus the result for h = 1000. As expected, a higher value of RT_{NDT} can be tolerated when "h" is lower, but the difference is only about 10°F or less at high pressures. The difference is seen to be greatest at low pressure, where K_I - thermal is the predominant part of K_I - total, and for a severe cooldown. This means that the near vertical lines of Figure D-10 would move to the left 5°F at P = 2500 psig and about 29°F at 500 psig. Figure D-11 is a repeat of Figure D-10 for h = 300 instead of 1000.

	T_f = 150°F		T_f = 300°F	
	β = 0.015	β = 0.15	β = 0.015	β = 0.15
P = 500 psig	9	29	0*	25
P = 2500 psig	11	5	7	5

*Both calculations stopped at RT_{NDT} = 400°F

D.6.2 Fracture Mechanics Analysis for Several PWR Recorded Transients

In the past, a number of events have occurred that can be categorized as PTS transients. Some of these have previously been analyzed by fitting the actual temporal temperature and pressure variations with smoothed and/or bounding curves in order to facilitate the analysis. These transients have recently been reanalyzed using the recorded temperature and pressure traces with all their respective fluctuations. The results are presented in an ORNL report, Appendix O, and as is discussed elsewhere in this document, were used as part of the basis in arriving at RT_{NDT} screening criteria.

D.6.3 Fracture Mechanics Example Analyses

In addition to the many uncertainties regarding PTS scenarios such as the temperature and pressure profiles versus time, the degree of mixing of cold water with warm water, etc. there exists parametric uncertainties in the stress and fracture mechanics analyses. The treatment of these uncertainties becomes significant when the cooldown temperature is to approximately RT_{NDT} because small changes in assumptions can influence whether or not a crack will initiate.

Assuming infinitely long cracks, h = 330 Btu/hr ft² °F (including clad effect) and using, for example, an assumed downcomer water temperature transient of

$$T_w = 250 + 300\ e^{-0.15t}\ {}^{o}F, \quad t = \text{minutes}$$

which is only slightly more conservative than transients that have actually occurred and RT_{NDT} at the cooled surface of 294°F which is only slightly greater than that which exists in some facilities, the NRC staff found that to prevent crack initiation, the pressure versus time would have to be less than as shown in Figure D-12. That is, the pressure should be reduced to near saturation conditions by about 30 minutes if warm prestressing (WPS) is assumed to be ineffective. If the pressure had been reduced approximately monotonically, then WPS, which occurs at about 18 minutes for this assumed transient, could also preclude crack initiation. From the results of this transient provided by ORNL, which were calculated using somewhat more conservative assumptions regarding input parameters, crack initiation was predicted at about 24 minutes even for zero pressure if WPS is not effective. The main contributor to this difference in conclusions is believed to be the effective heat transfer coefficient used in the respective analyses. Thus, for cases where the final temperature is in the range of RT_{NDT}, the sensitivity of results to the various input parameters needs to be investigated before final conclusions can be reached as to limiting pressures.

A factor for consideration regarding these transients is that, in general, larger pre-existing cracks are necessary before crack initiation would occur for the cases of higher RT_{NDT}'s. This factor is not illustrated in the figures in this appendix.

This same temperature transient was also analyzed for different values of RT_{NDT} at the vessel inner radius and for a circumferential crack. The results are shown in Figure D-13. Note that the effect of the clad is approximately

$8^{\circ}F$ and that a circumferential crack will tolerate about a $30^{\circ}F$ higher RT_{NDT} (considering crack initiation only) for this transient. Similar variations would be expected for other transients. This example illustrates the benefits to be attained by monotonically decreasing pressure in the event of a moderately severe thermal transient in that it is possible to avoid crack initiation.

For much more severe thermal transients, crack initiation may occur due to high thermal stresses. In this case it is necessary to consider the potential for crack arrest. Figure D-14 is a schematic representation of a critical crack depth diagram to illustrate the analytical model used by the staff for determining acceptable arrest criteria. An upper shelf toughness of 200 ksi $\sqrt{in.}$ is assumed; however, higher or lower values may be more appropriate for a specific reactor vessel. When the thermal stress intensity factor is known at the time of warm prestressing (WPS), the maximum pressure is determined such that arrest will occur at or before the time of WPS and for crack depths below the upper shelf curve. The limiting case is shown as point "A" in the figure. The thermal transient selected for this example is:

$$T_w = 60 + 480\ e^{-\beta t}$$

Figure D-15 illustrates the effect of the cooldown rate with a water to metal heat transfer coefficient of 300 Btu/hr ft^2 $^{\circ}F$. Figure D-16 shows the equivalent results for a lower coefficient. Note that the sensitivity to the heat transfer coefficient is greatest for the more rapid cooldown. Figure D-17 shows the effect of various assumptions regarding the attenuation of RT_{NDT} in the metal as discussed in Section 3 and Appendix E of this document. The above figures are for long axial cracks. Figure D-18 shows the effect of assuming long circumferential instead of axial cracks. In terms of RT_{NDT}, instead of being about $30^{\circ}F$ for crack initiation, the difference now is about $100^{\circ}F$ for crack arrest, depending on the specific pressure. Also shown in Figure D-18 is the effect of crack shape at arrest. (An a/c value of 0.1 represents a crack which is 20 times as long as it is deep.)

Figure D-19 is for another transient. It illustrates the uncertainty in RT_{NDT} that can occur due to the selection of the time of warm prestressing because of the relative flatness of K_I versus time near its peak value. Again, the difference between axial and circumferential cracks is shown when warm prestressing and arrest are considered.

As stated earlier, the NRC staff assumes an infinite flaw length in its analyses; that is, an ellipse with an aspect ratio of zero. For circumferential cracks that arrest at some depth, this assumption is believed to be reasonable if the vessel wall is uniformly cooled in that direction. On the other hand, growing axial cracks could be limited in length by reaching the ends of a critical weld and intercepting tougher plate material. Also, they could extend into less irradiated regions of the vessel wall and hence into tougher materials even within the weld. Thus, the assumption of an infinitely long axial crack is conservative.

Figure D-20 shows the effect of the assumed crack shape at arrest. If an aspect ratio a/c = 0.1 is assumed instead of zero, there is a gain in RT_{NDT} of about 60° for the case illustrated. This appears to be reasonable in that, if the crack arrested at the ends of an axial weld, it would be approximately half-wall thickness in depth. An assumed aspect ratio of 1/3 would lead to higher tolerable RT_{NDT}'s; however, analyses and experiments related to growing cracks during a severe thermal transient indicate that cracks arresting with this shape are very unlikely. Also wall penetrations might occur before the ends of the crack reached tough materials. Therefore, the staff does not accept this assumption. If other than infinitely large arresting cracks, say those with an aspect ratio up to 0.1, are to be accepted, then reasonable assumptions have to be made regarding all stresses especially weld residual stresses that can be present in addition to those due to pressure and temperature distribution.

These illustrations are intended to demonstrate the importance of limiting the reactor system pressure in the event of a severe cooldown transient as well as the necessity to allow for uncertainties both in analyses of transients and in material properties.

Based on the examples illustrated in Section D.6.3 and on the analyses of other organizations for similar PTS scenarios, it is seen that variations in input assumptions can lead to differences of limiting RT_{NDT}'s for axial crack initiation. Specific differences will, of course, depend on specific scenarios. The following are typical results:

	Assumption	Effect on RT_{NDT}, °F
(a)	Clad stress vs. no clad stress	10°
(b)	Continuous flaw for initiation vs. elliptical flaw (a/c = 1/3)	20°
(c)	h = 300 Btu/hr ft^2 °F vs. Westinghouse free convection correlation	15°

The above assumption differences account for a total RT_{NDT} variation of about 45° between staff analyses and those of Westinghouse. The Westinghouse model for fluence attenuation into the wall is equivalent to the dpa model of the staff. Other vendors, however, may still be using other models. The attenuation effect on limiting RT_{NDT}'s for crack initiation is not expected to be great but for crack arrest situations, the difference can be significant as illustrated in Figure D-17.

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D.6. J. J. McGowan, "An Assessment of the Beneficial Effects of Warm Prestressing Under Severe Thermal Shock," WCAP 9178, March 1975.

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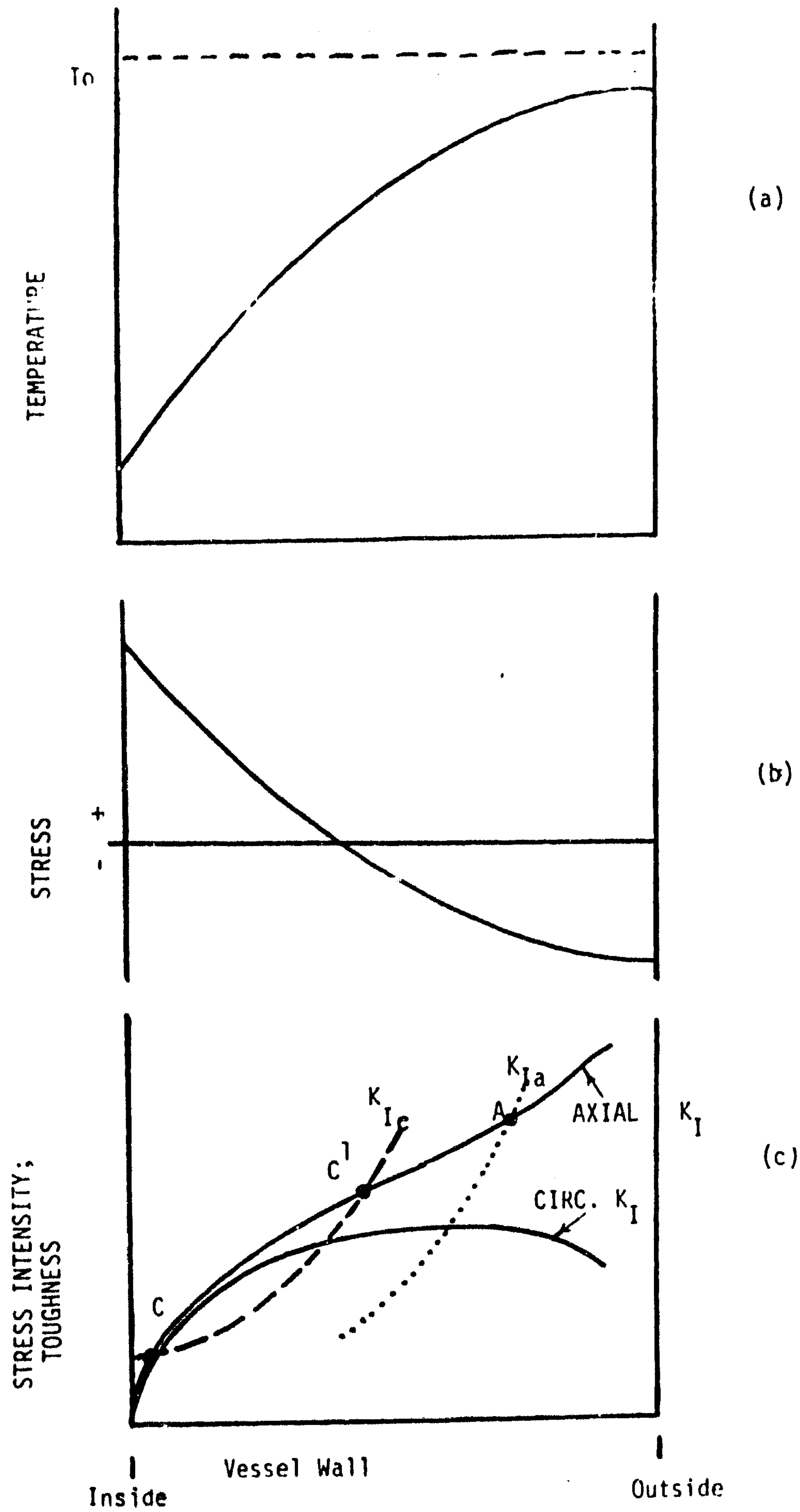


FIGURE D-1 RESULT OF PTS ON VESSEL WALL

HYPOTHETICAL PTS TRANSIENT

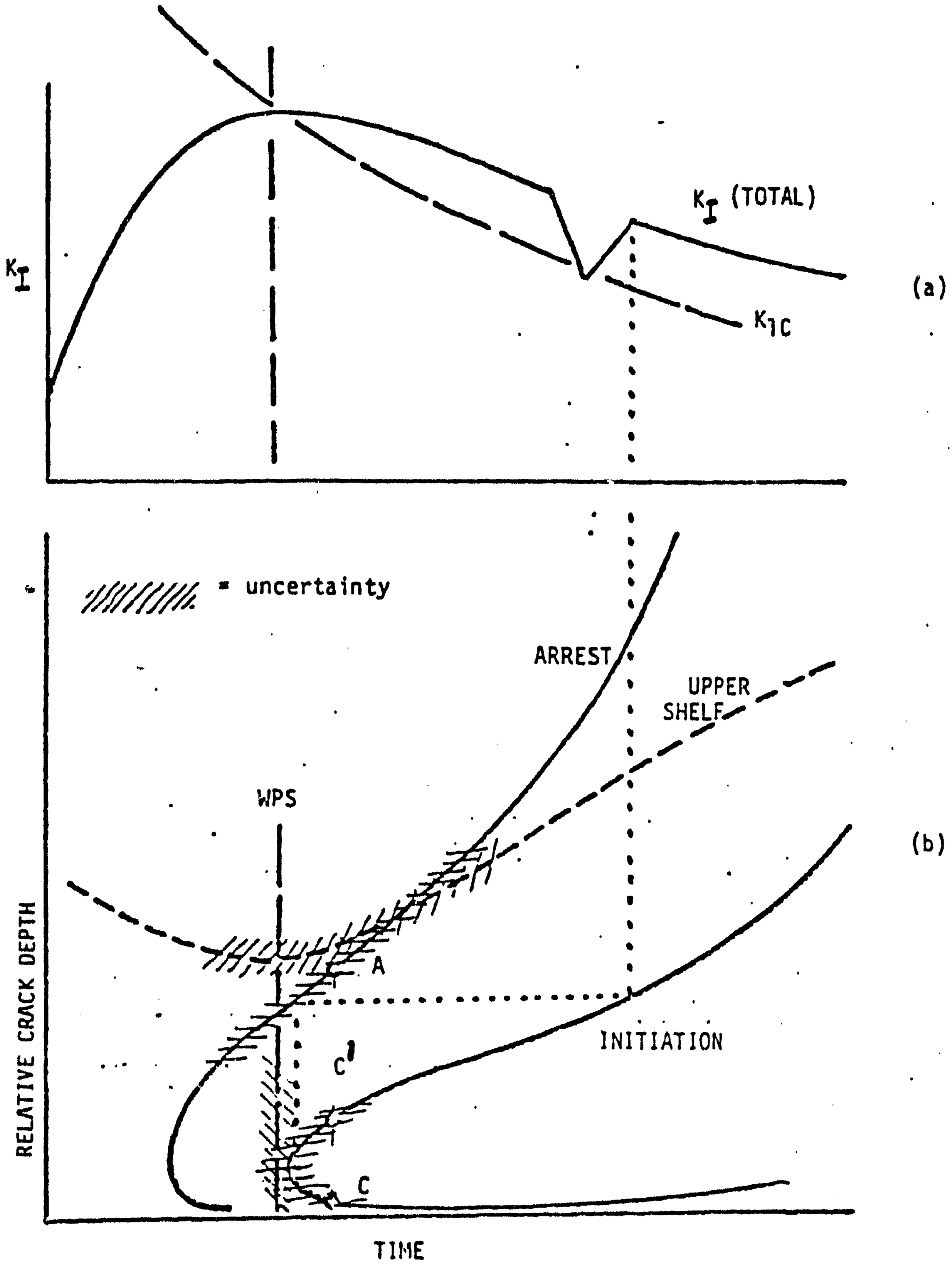


FIGURE D-2

CLAD EFFECTS

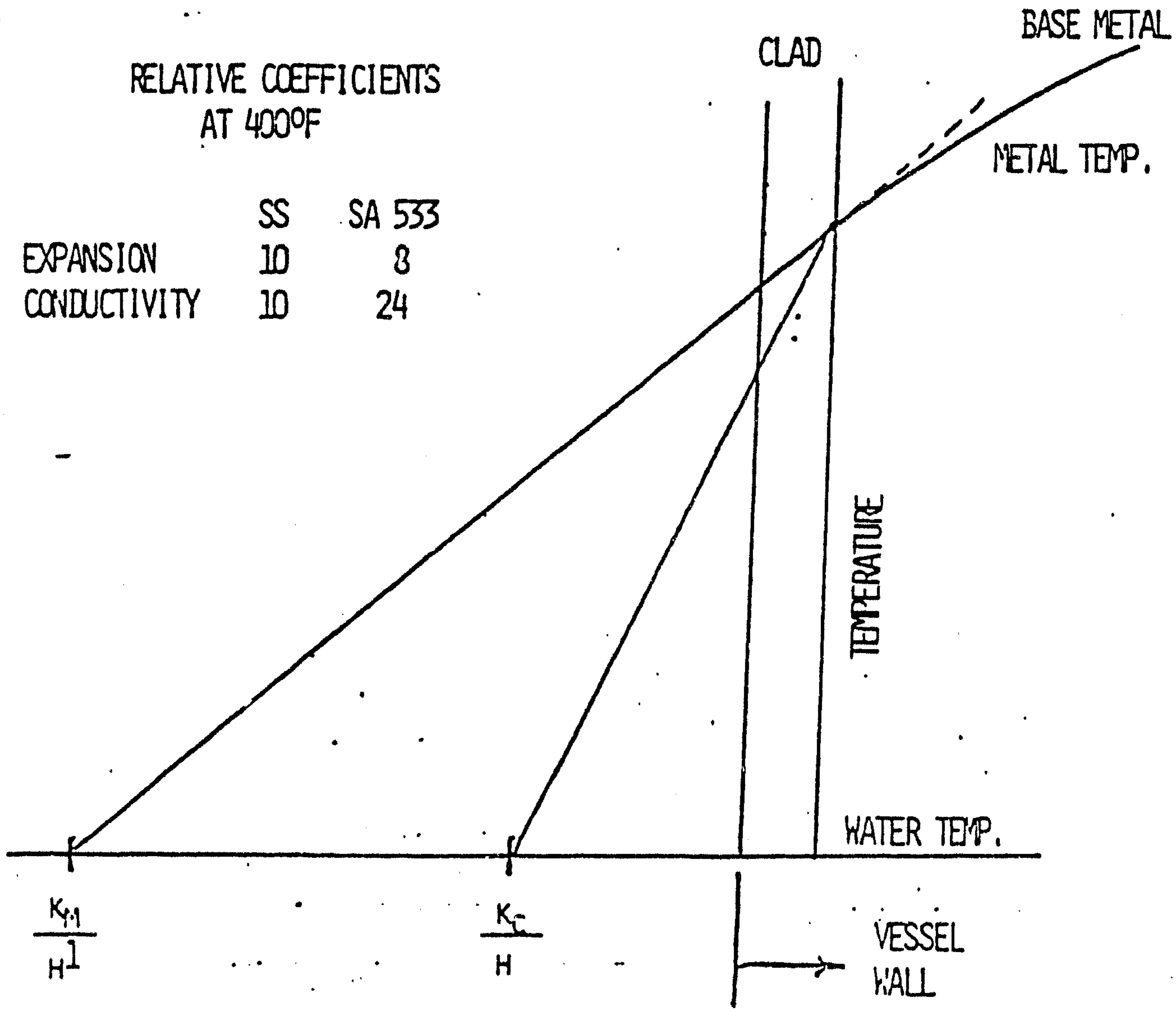
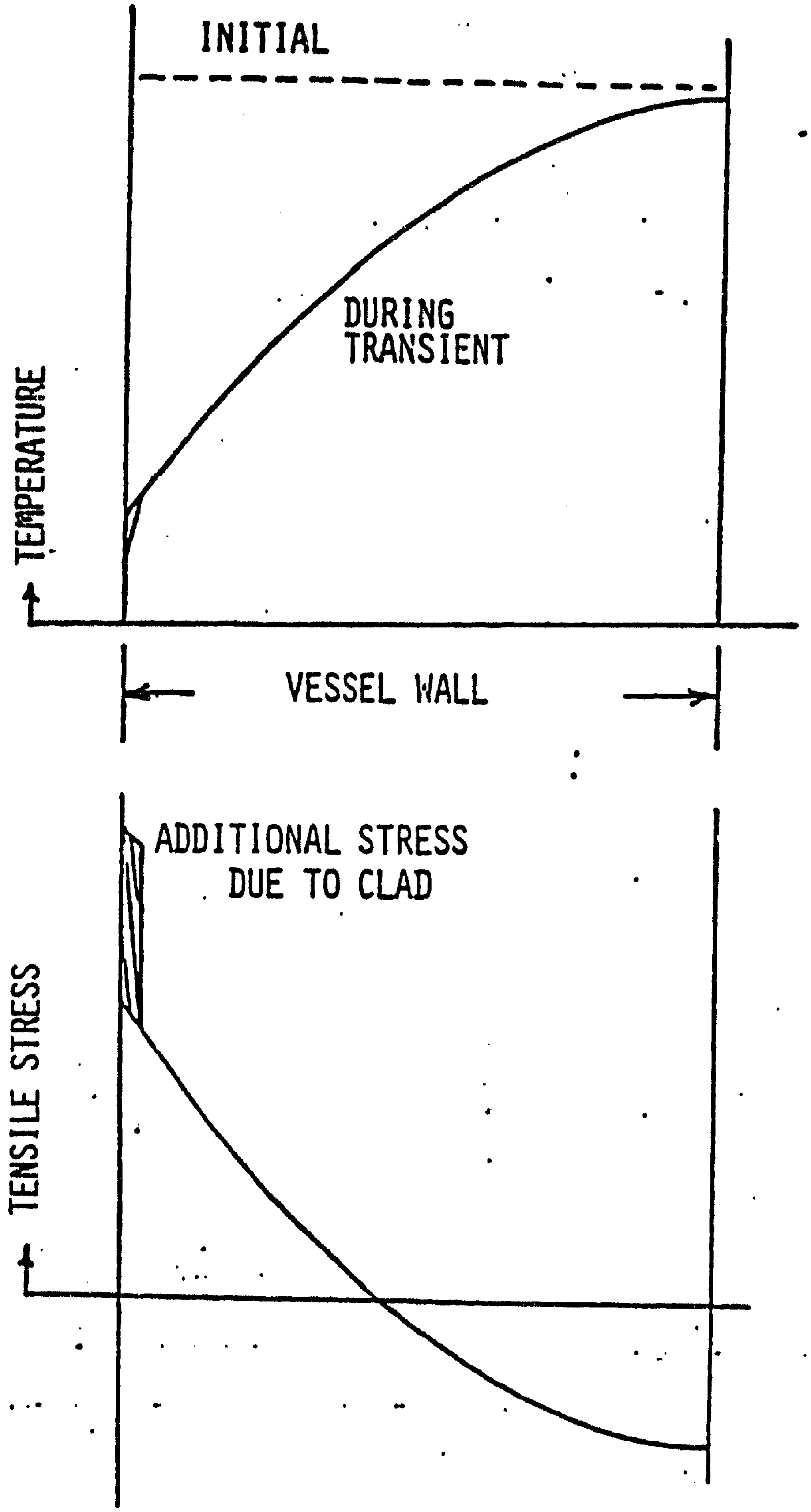


FIGURE D-3



TEMPERATURE AND THERMAL STRESS DISTRIBUTION
DURING A
COOLDOWN TRANSIENT

FIGURE D-4

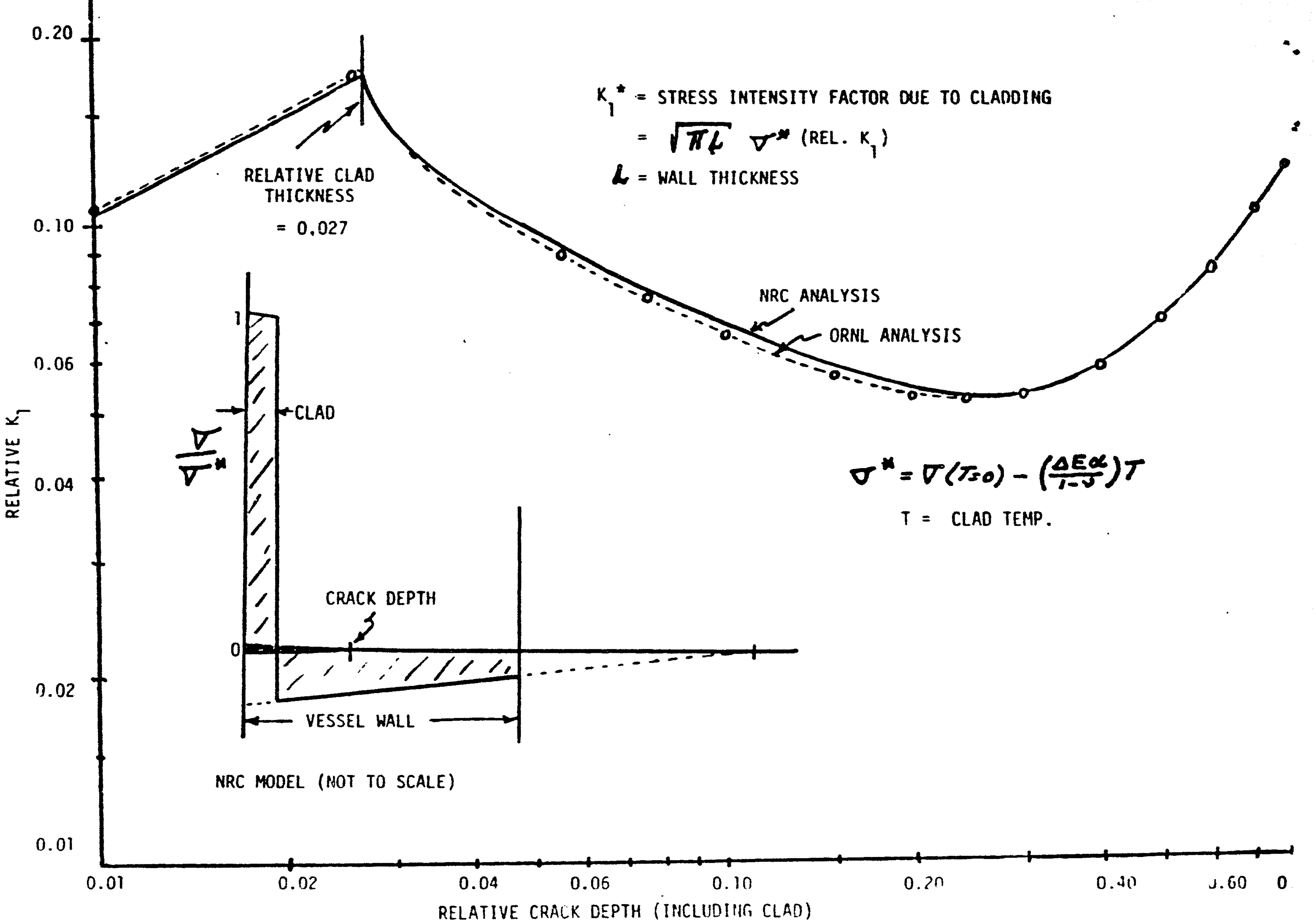

K_1^* = STRESS INTENSITY FACTOR DUE TO CLADDING
$= \sqrt{\pi L}\ \sigma^*$ (REL. K_1)
L = WALL THICKNESS
RELATIVE CLAD THICKNESS = 0.027
NRC ANALYSIS
ORNL ANALYSIS
$\sigma^* = \sigma(T=0) - \left(\frac{\Delta E \alpha}{1-\nu}\right) T$
T = CLAD TEMP.
$\frac{\sigma}{\sigma^*}$
1
0
CLAD
CRACK DEPTH
VESSEL WALL
NRC MODEL (NOT TO SCALE)
RELATIVE K_1
0.20
0.10
0.06
0.04
0.02
0.01
0.01
0.02
0.04
0.06
0.10
0.20
0.40
0.60
RELATIVE CRACK DEPTH (INCLUDING CLAD)

FIGURE D-5

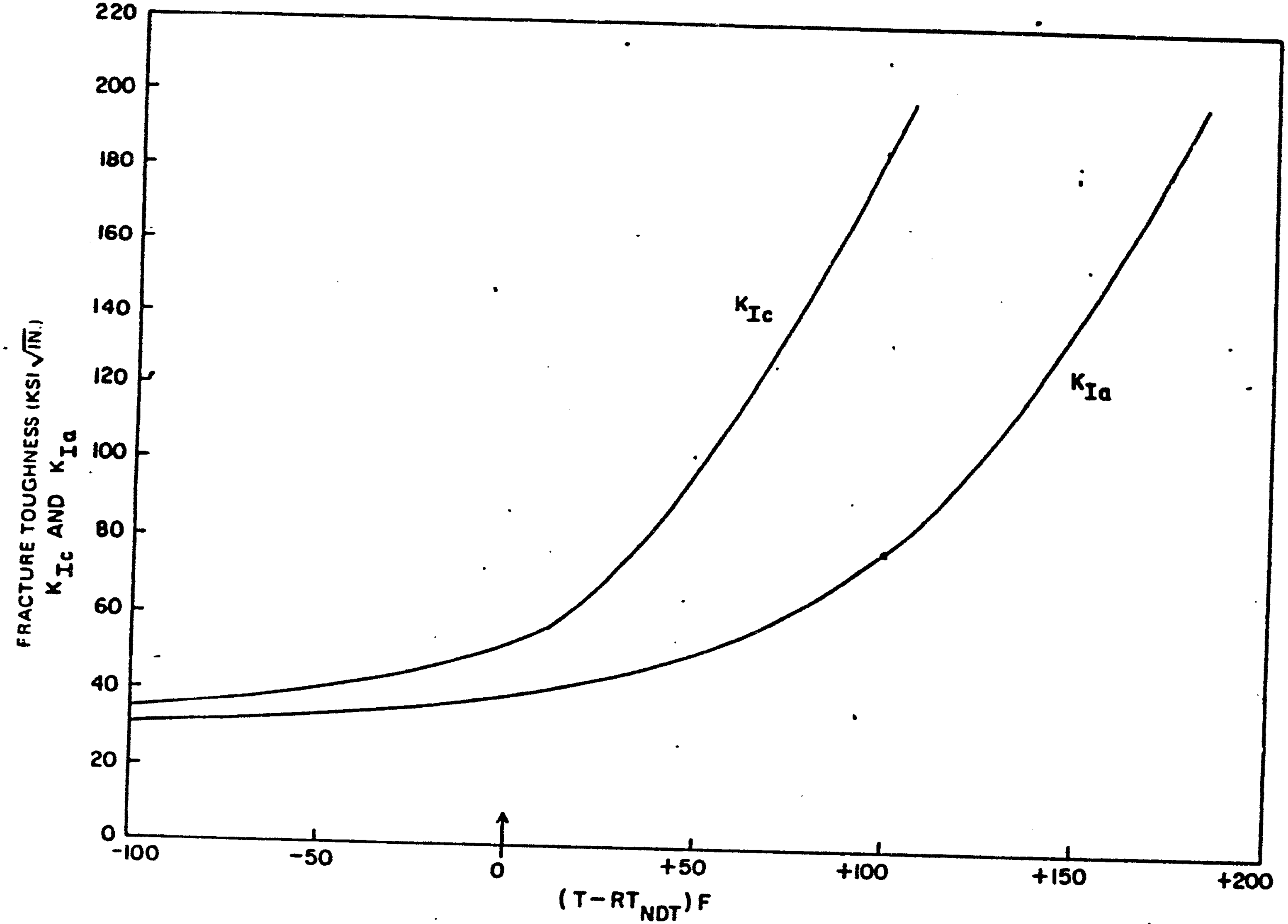


FIG. A-4200-1 LOWER BOUND K_{IA} and K_{IC} TEST DATA FOR SA-533 GRADE B CLASS 1, SA-508 CLASS 2, AND SA-508 CLASS 3 STEELS

FIGURE D-6, ASME CODE SECTION XI CURVES FOR RELATING FRACTURE TOUGHNESS TO $T-RT_{NDT}$

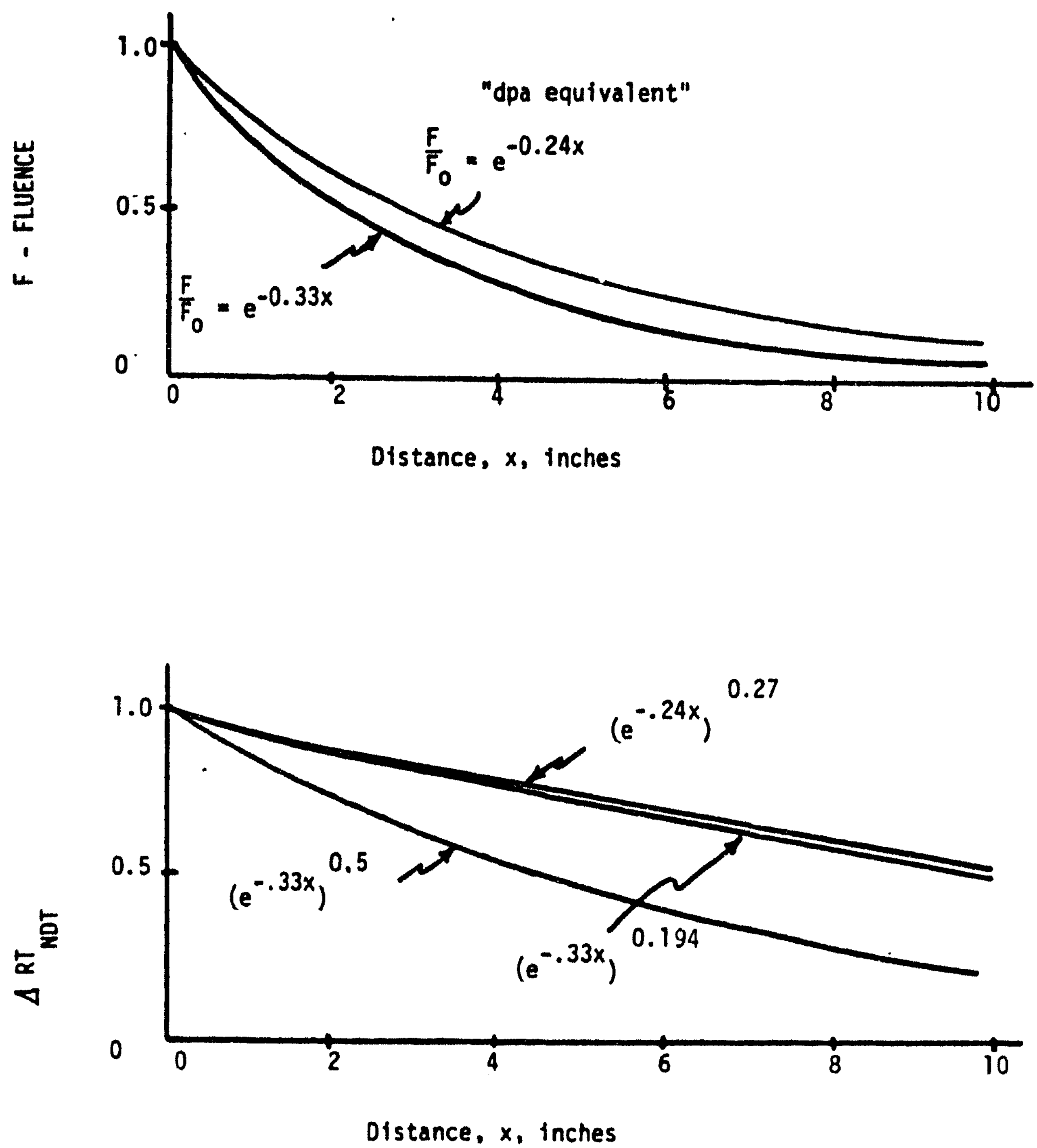
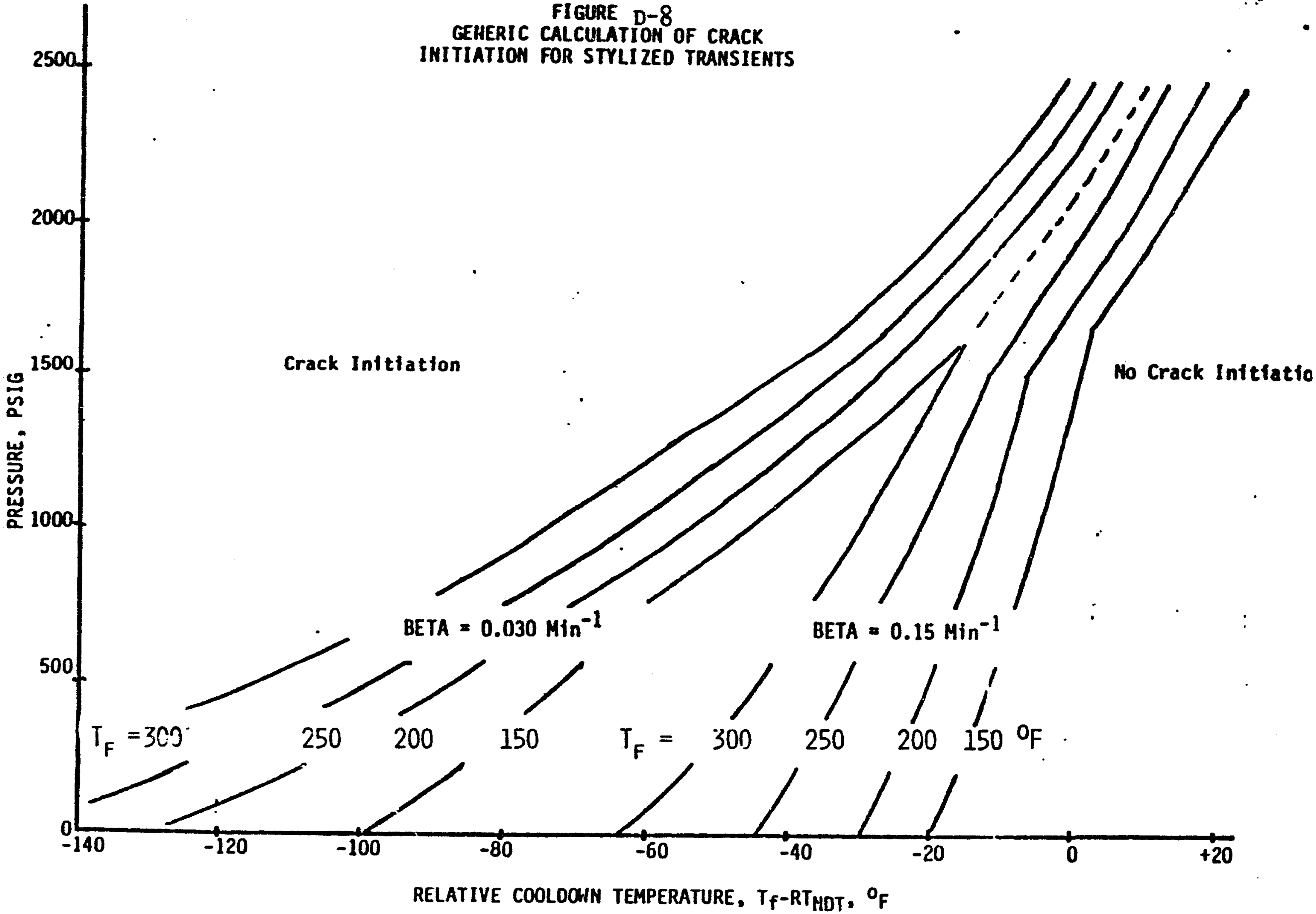


FIGURE D-7, ATTENUATION OF FLUENCE AND ΔRT_{NDT} THROUGH THE VESSEL WALL

FIGURE D-8
GENERIC CALCULATION OF CRACK
INITIATION FOR STYLIZED TRANSIENTS



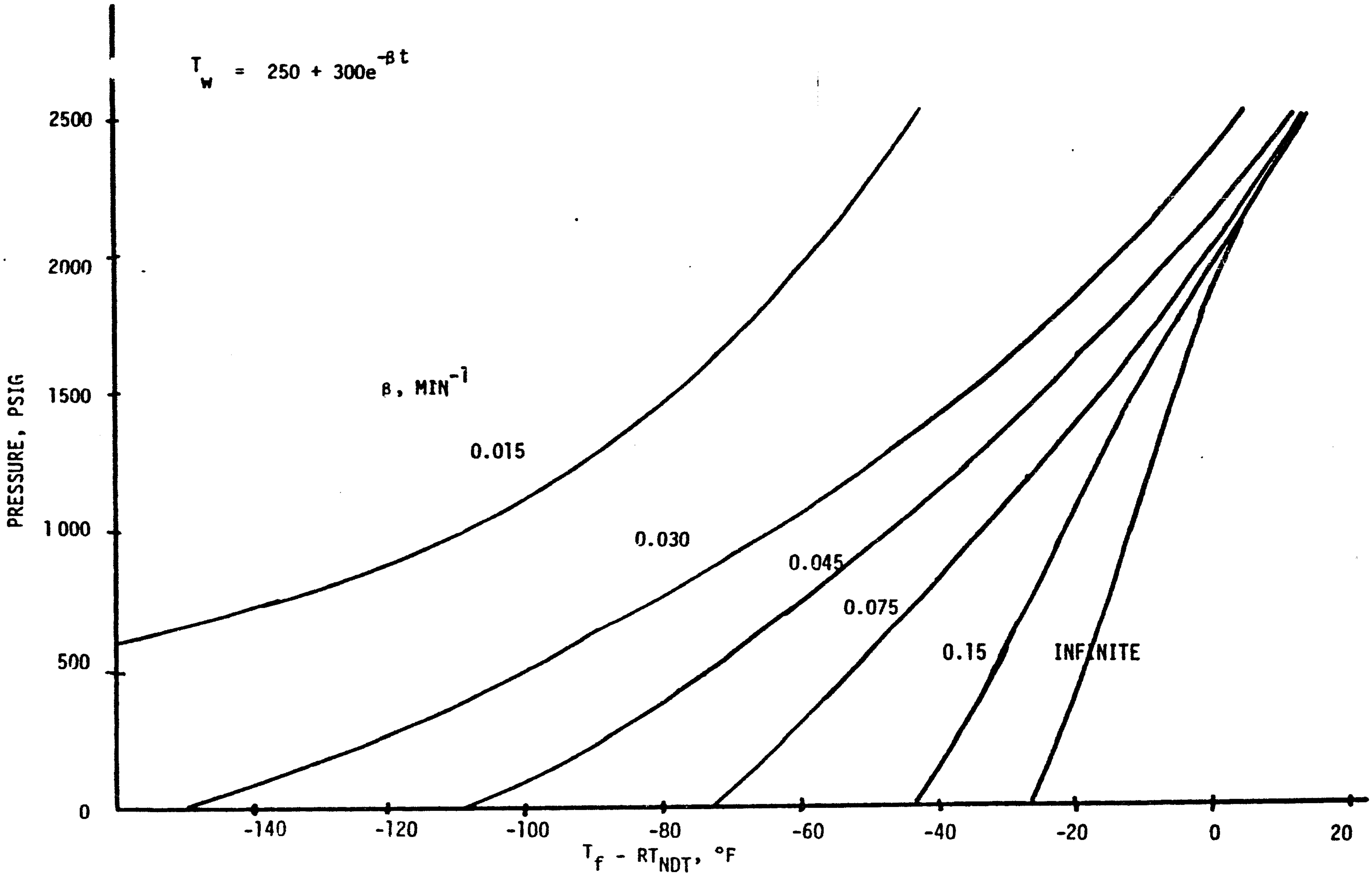


FIGURE D-9 EFFECT OF THE RECIPROCAL TIME CONSTANT, β, ON THE CRITICAL VALUES OF PRESSURE AND $T_f - RT_{NDT}$ FOR CRACK INITIATION

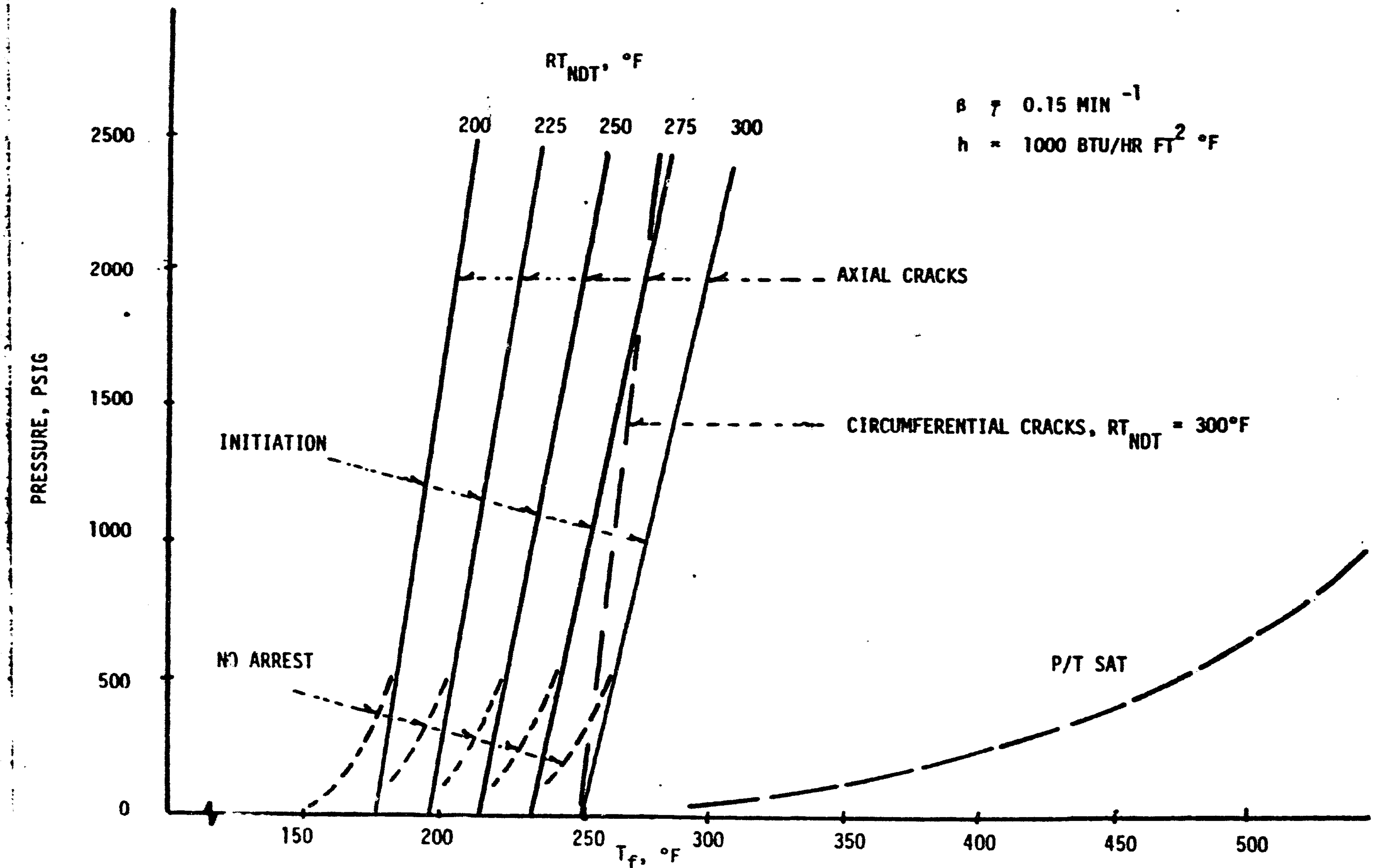


FIGURE D-10 EFFECT OF T_f AND RT_{NDT} ON THE CRITICAL PRESSURE

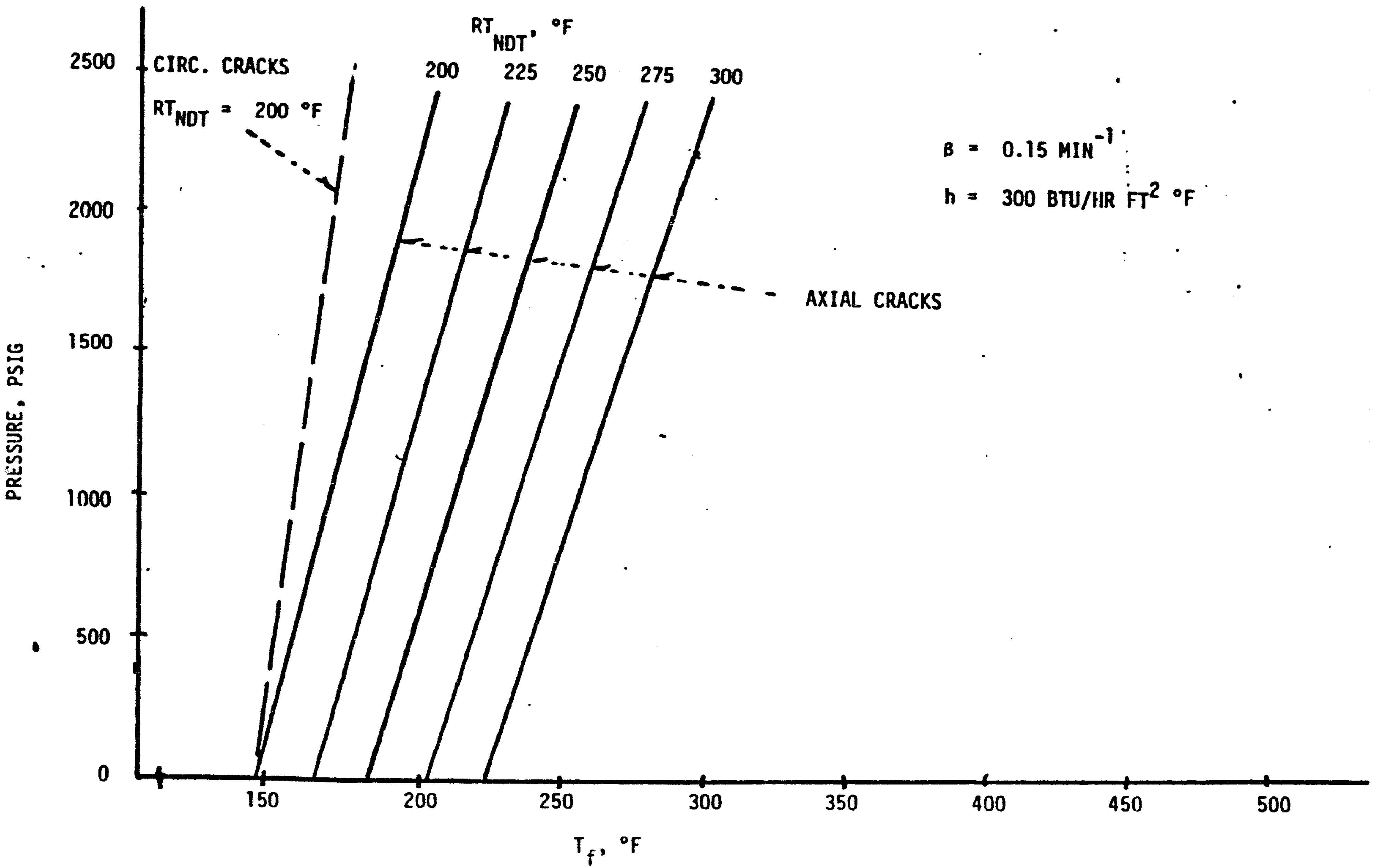
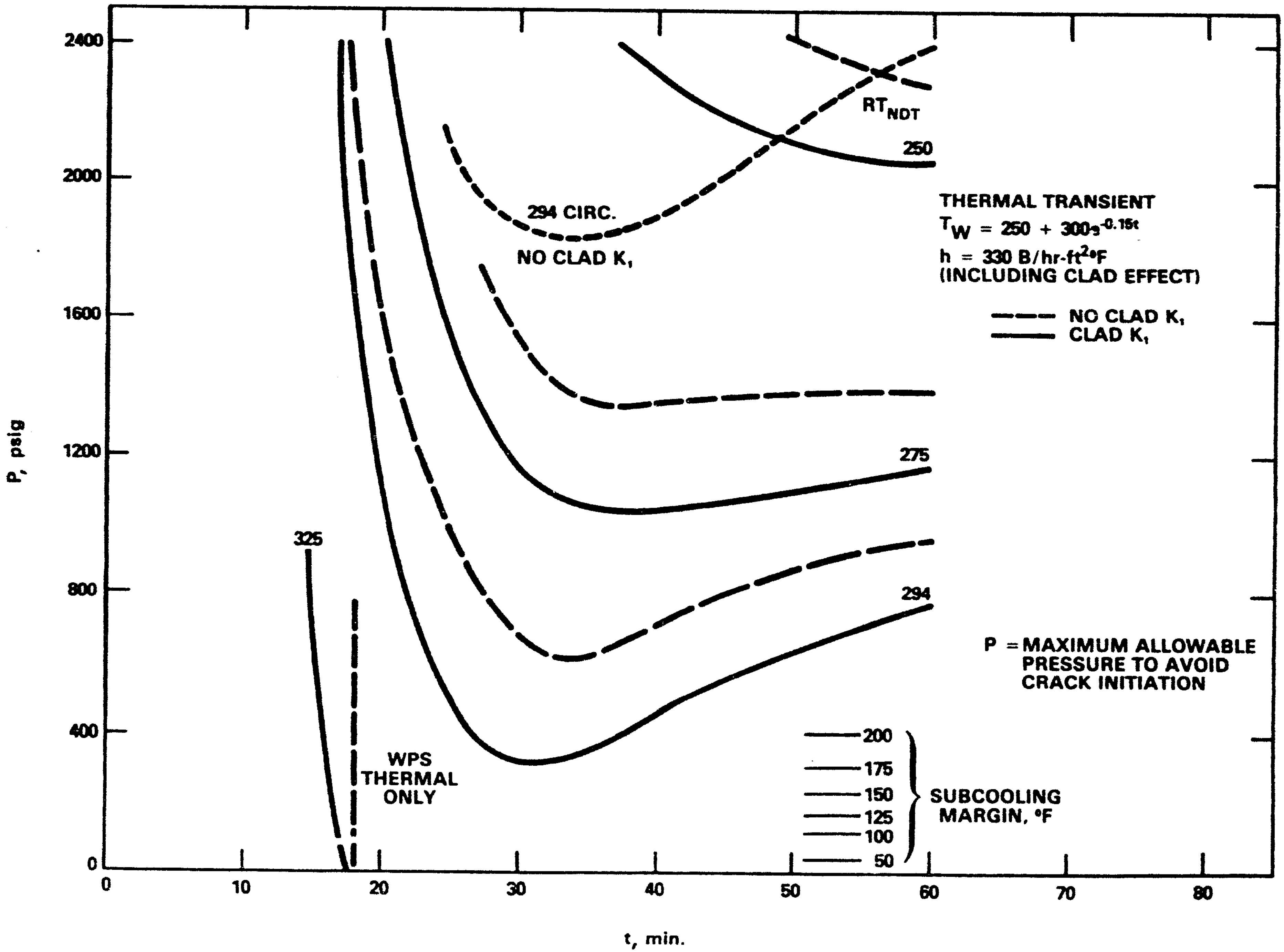


FIGURE D-11 EFFECT OF T_f AND RT_{NDT} ON THE CRITICAL PRESSURE FOR CRACK INITIATION



2400
2000
1600
1200
800
400
0
P, psig
0
10
20
30
40
50
60
70
80
t, min.
RT$_{NDT}$
250
294 CIRC.
NO CLAD K$_1$
THERMAL TRANSIENT
$T_W = 250 + 300e^{-0.15t}$
$h = 330$ B/hr-ft^2°F
(INCLUDING CLAD EFFECT)
NO CLAD K$_1$
CLAD K$_1$
275
325
294
P = MAXIMUM ALLOWABLE PRESSURE TO AVOID CRACK INITIATION
WPS THERMAL ONLY
200
175
150
125
100
50
SUBCOOLING MARGIN, °F

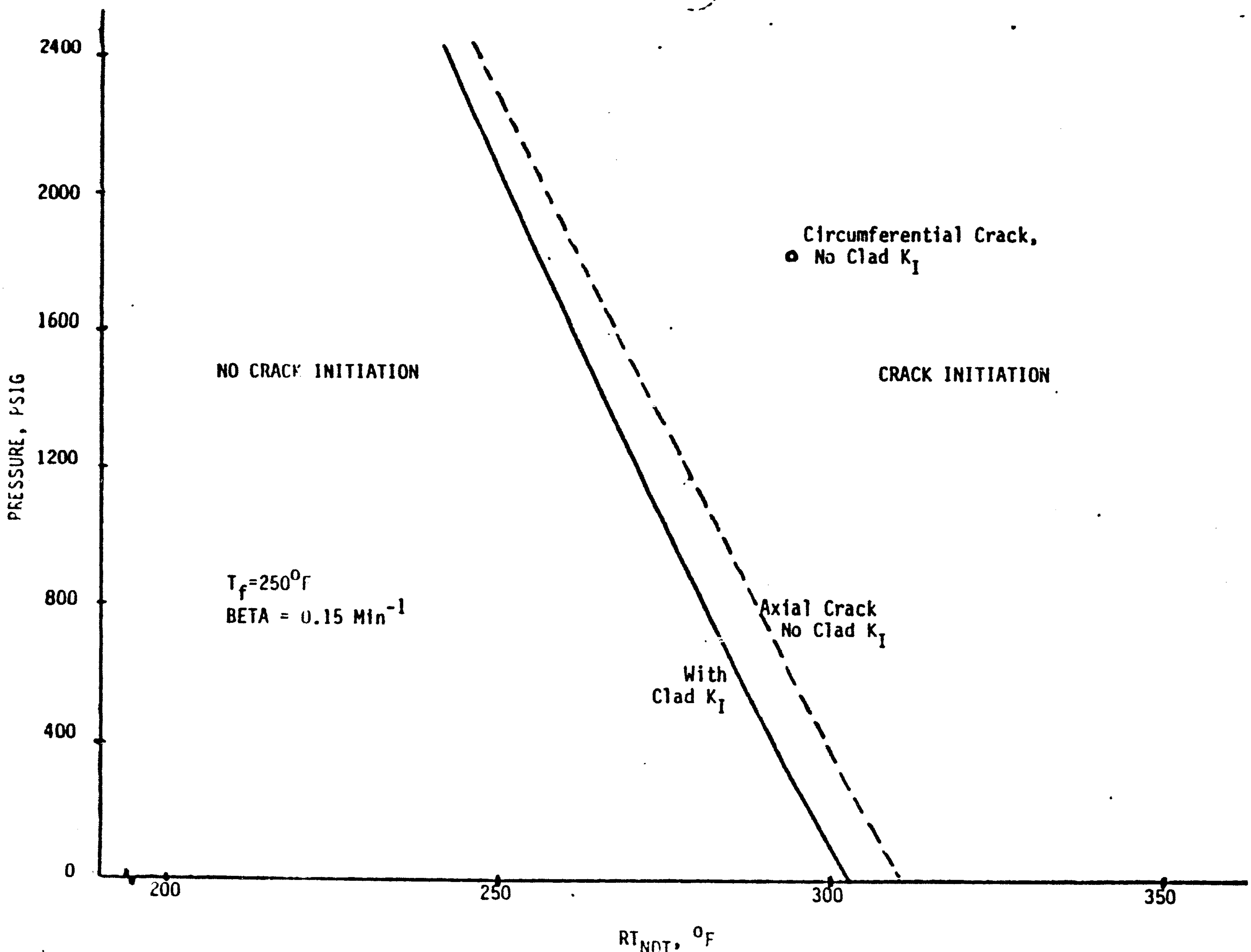


FIGURE D-13 SENSITIVITY STUDIES

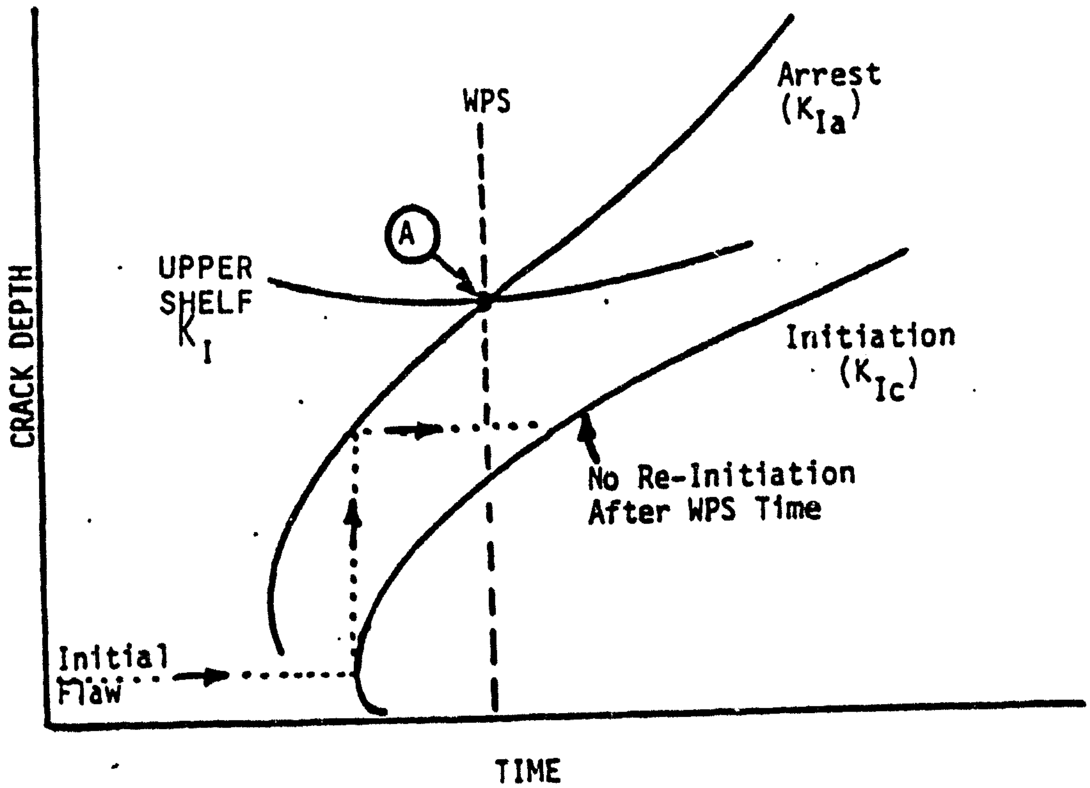
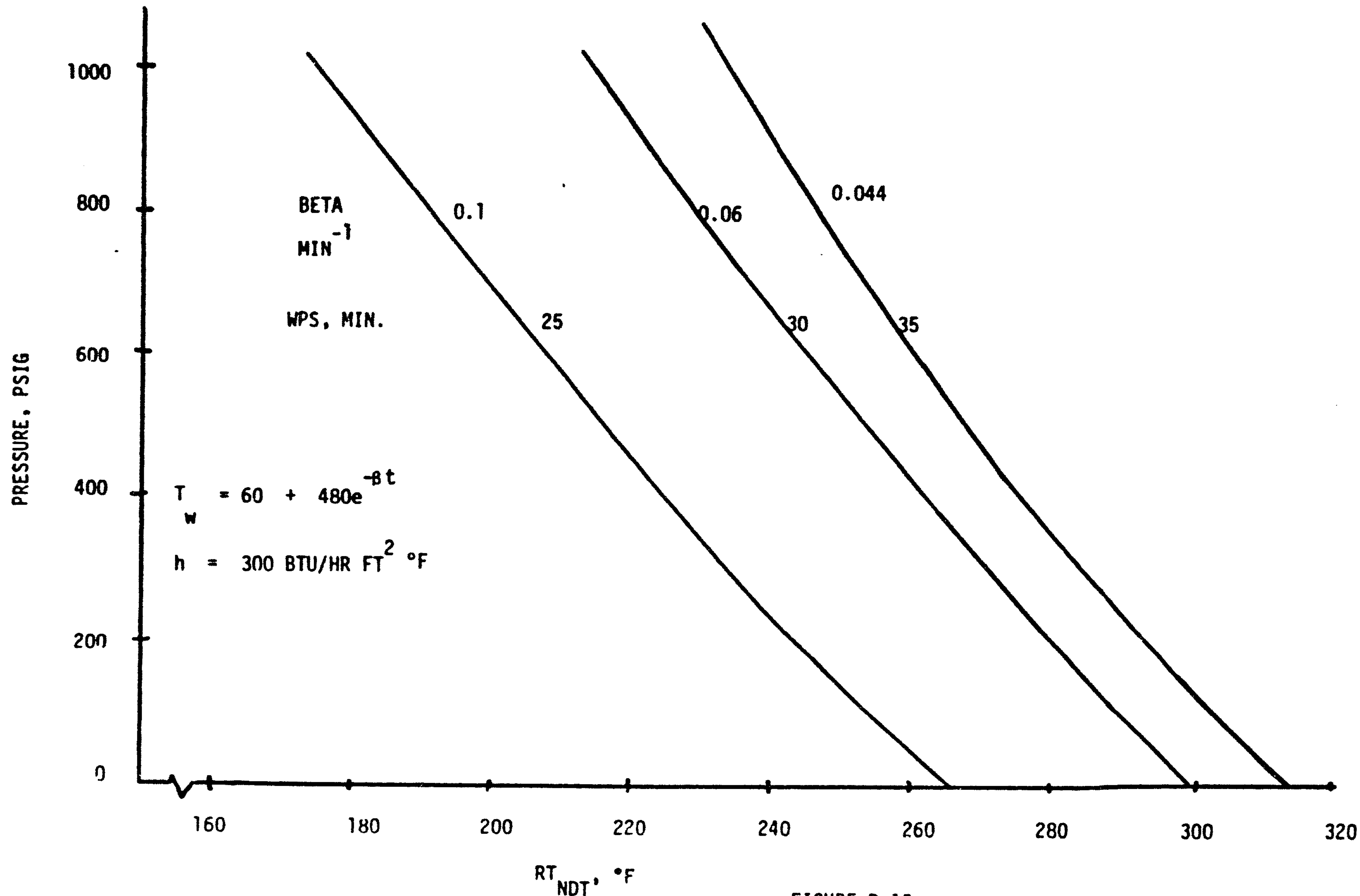
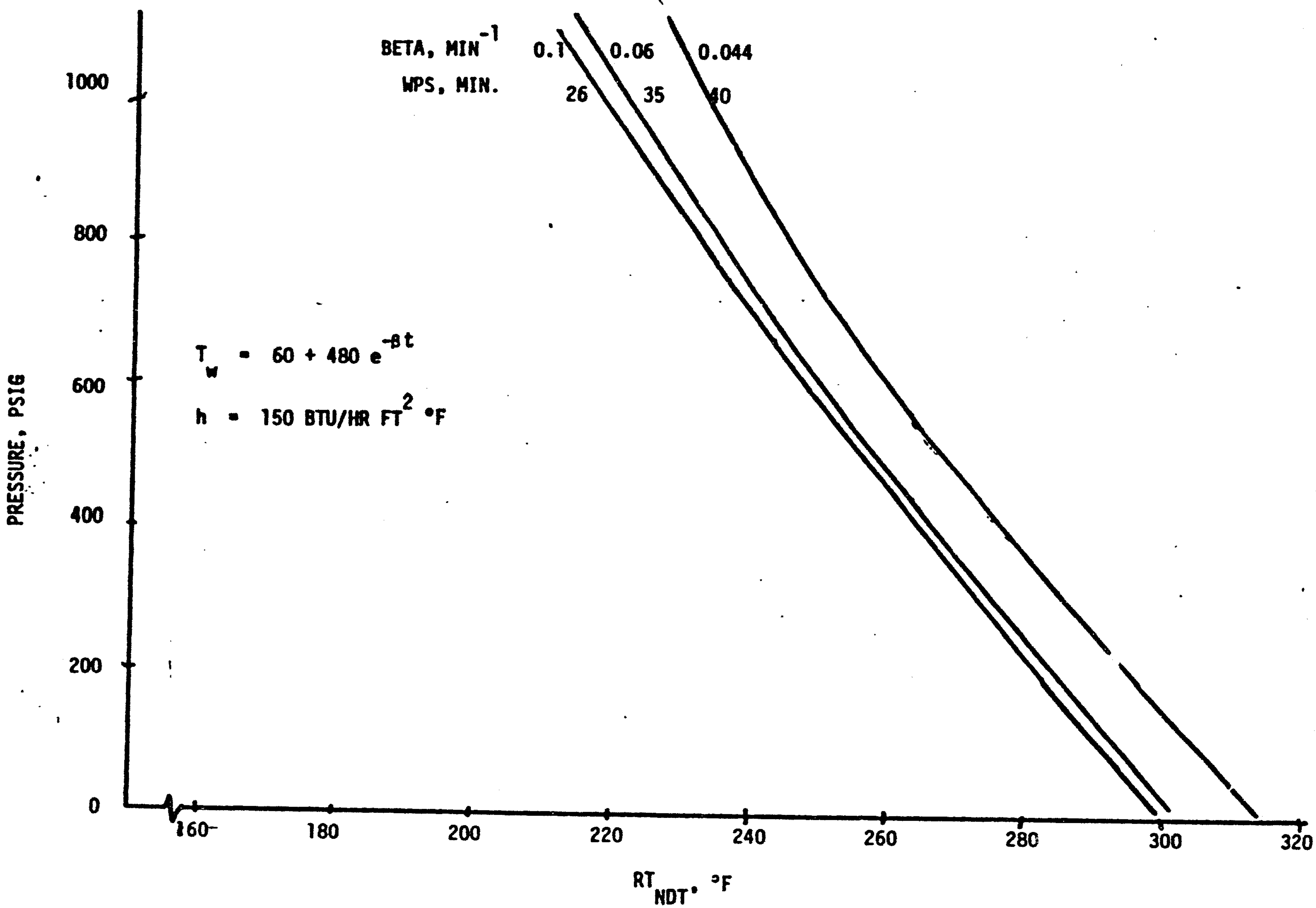


FIGURE D-14 NRC CRACK
INITIATION AND ARREST
MODEL



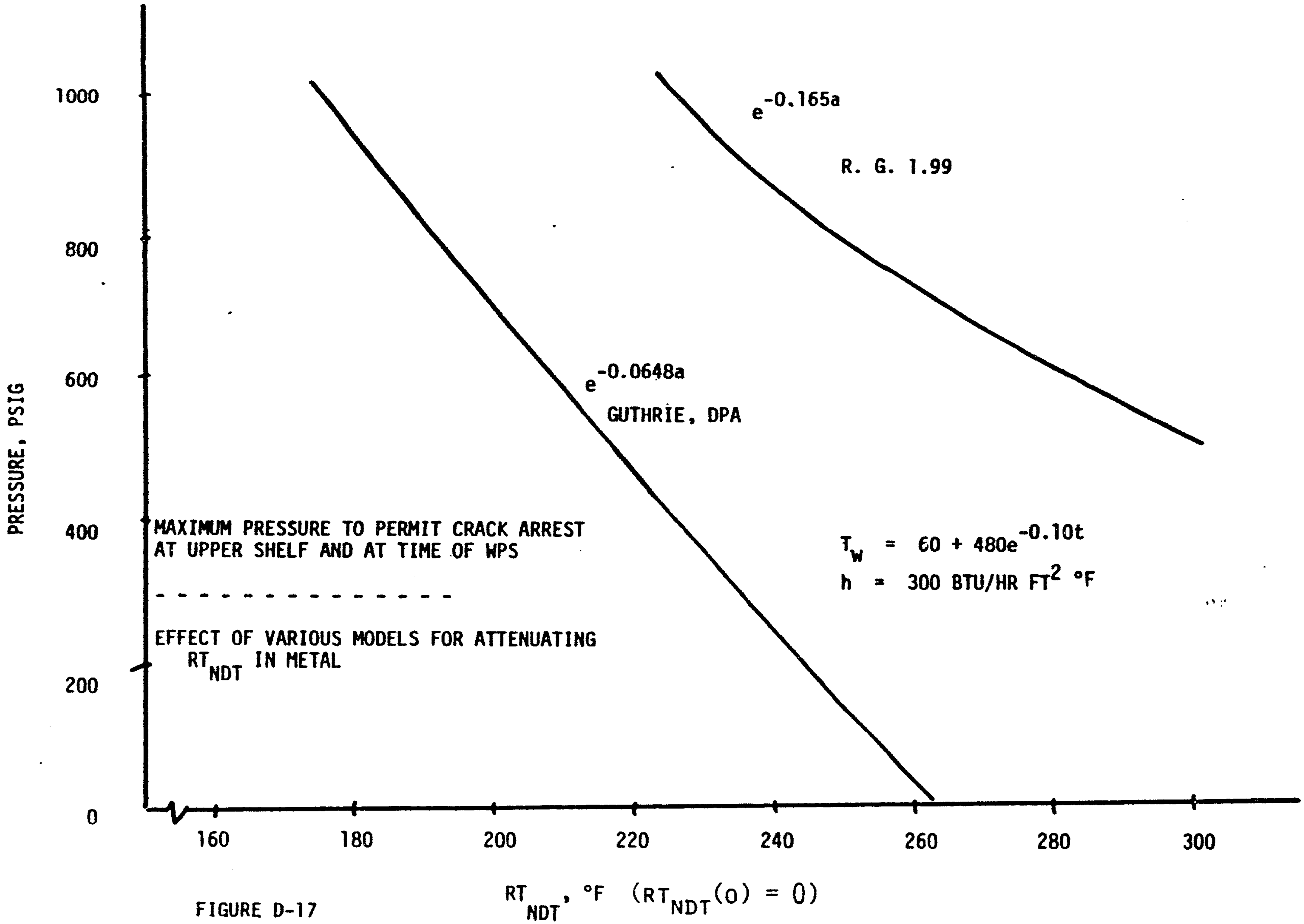
PRESSURE, PSIG
1000
800
600
400
200
0
160
180
200
220
240
260
280
300
320
RT_{NDT}, °F
BETA
MIN^{-1}
WPS, MIN.
0.1
25
0.06
30
0.044
35
$T_w = 60 + 480e^{-\beta t}$
$h = 300$ BTU/HR FT^2 °F

FIGURE D-15



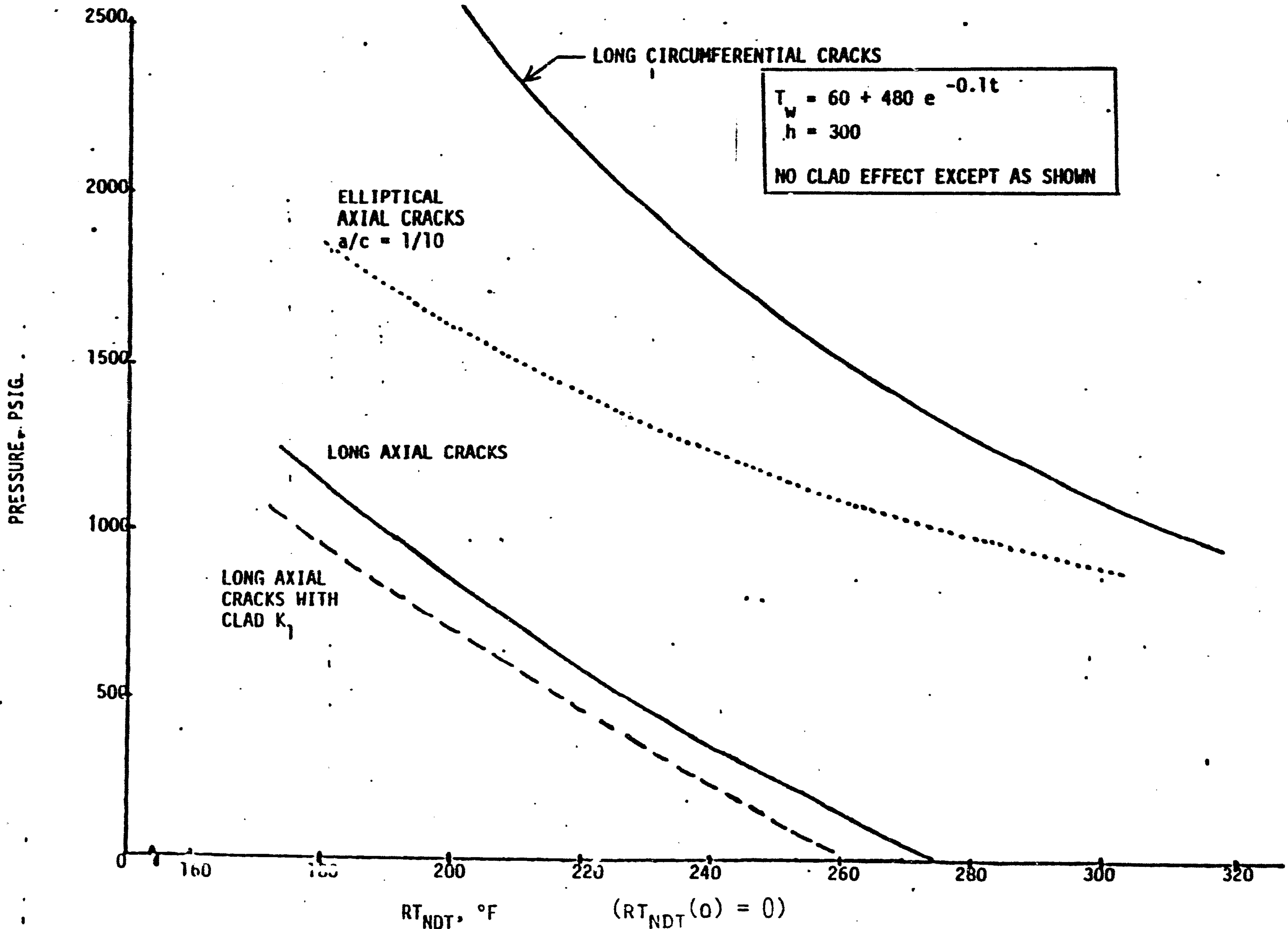
BETA, MIN^{-1} 0.1 0.06 0.044
WPS, MIN. 26 35 40
$T_w = 60 + 480\,e^{-\beta t}$
h = 150 BTU/HR FT^2 °F
PRESSURE, PSIG
1000
800
600
400
200
0
160
180
200
220
240
260
280
300
320
RT_{NDT}, °F

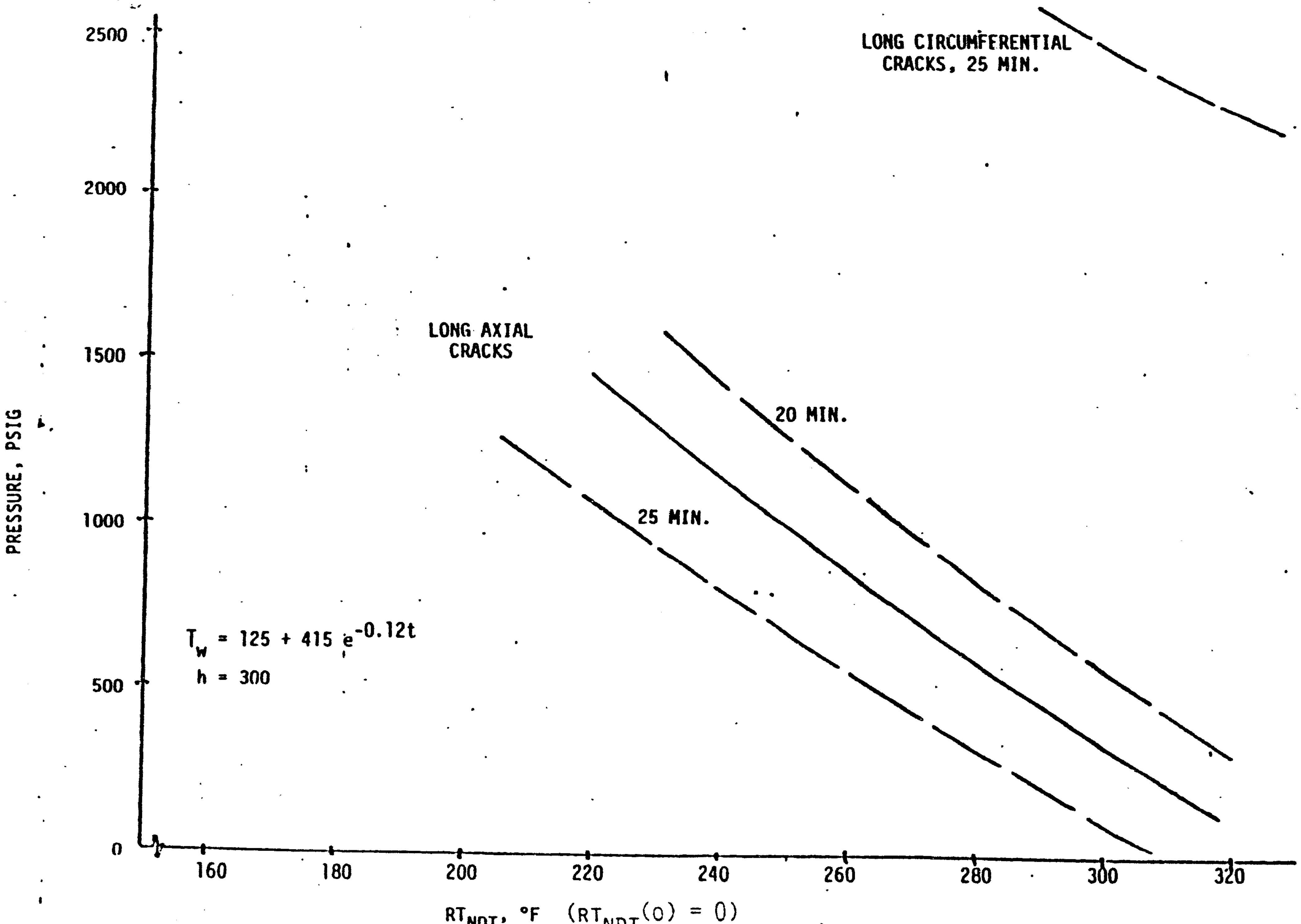
FIGURE D-16



PRESSURE, PSIG
1000
800
600
400
200
0
160
180
200
220
240
260
280
300
e^-0.165a
R. G. 1.99
e^-0.0648a
GUTHRIE, DPA
MAXIMUM PRESSURE TO PERMIT CRACK ARREST
AT UPPER SHELF AND AT TIME OF WPS
EFFECT OF VARIOUS MODELS FOR ATTENUATING
RT_NDT IN METAL
T_W = 60 + 480e^-0.10t
h = 300 BTU/HR FT^2 °F
RT_NDT, °F (RT_NDT(o) = 0)

FIGURE D-17


LONG CIRCUMFERENTIAL CRACKS
$T_w = 60 + 480\ e^{-0.1t}$
$h = 300$
NO CLAD EFFECT EXCEPT AS SHOWN
ELLIPTICAL AXIAL CRACKS a/c = 1/10
LONG AXIAL CRACKS
LONG AXIAL CRACKS WITH CLAD K_I
PRESSURE, PSIG.
2500
2000
1500
1000
500
0
160
200
220
240
260
280
300
320
RT_{NDT}, °F
($RT_{NDT}(o) = 0$)
FIGURE D-18



LONG CIRCUMFERENTIAL CRACKS, 25 MIN.
LONG AXIAL CRACKS
20 MIN.
25 MIN.
$T_w = 125 + 415\ e^{-0.12t}$
h = 300
PRESSURE, PSIG
2500
2000
1500
1000
500
0
160
180
200
220
240
260
280
300
320
RT_{NDT}, °F ($RT_{NDT}(o) = 0$)

FIGURE D-19

EFFECT OF ASPECT RATIO

AT CRACK ARREST

EXAMPLE

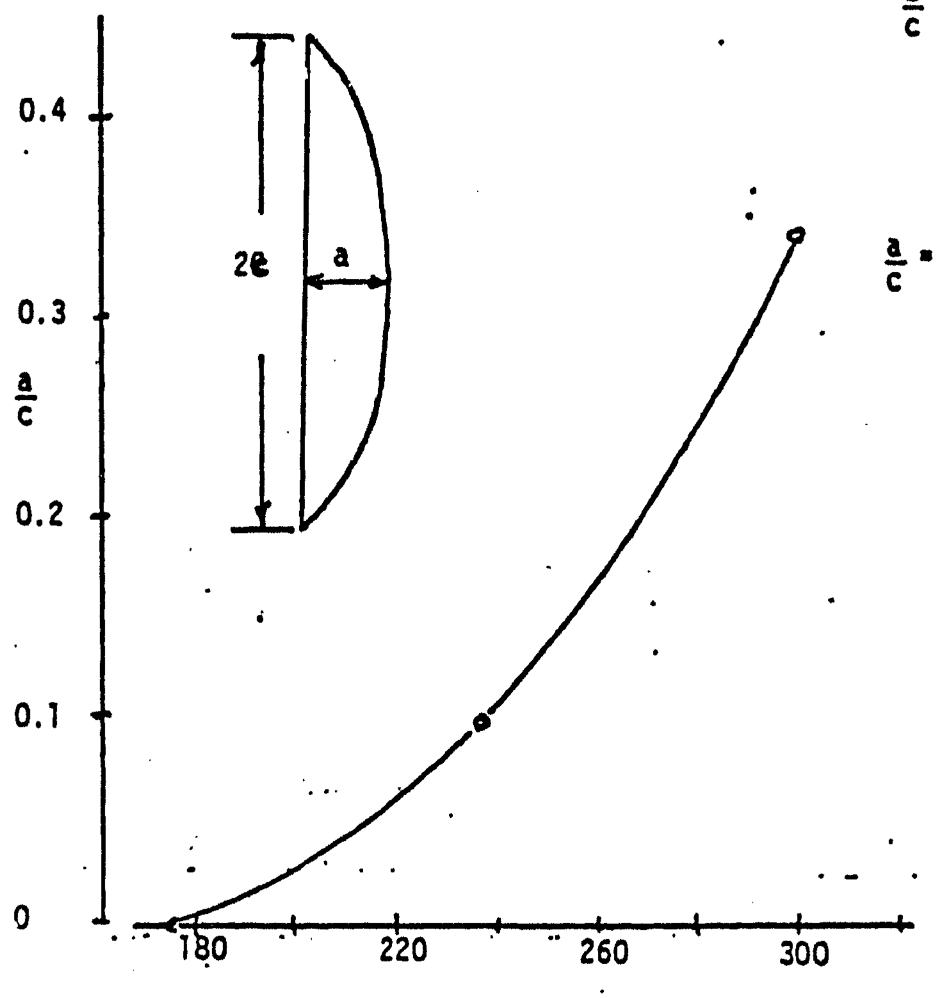
$P = 1250$ psig

$T_W = 60 + 480\ e^{-0.1t}$

$h = 300$

no clad effect

$t = 25$ min. (WPS)



RT_{NDT}, °F ($RT_{NDT}(0) = 0$)

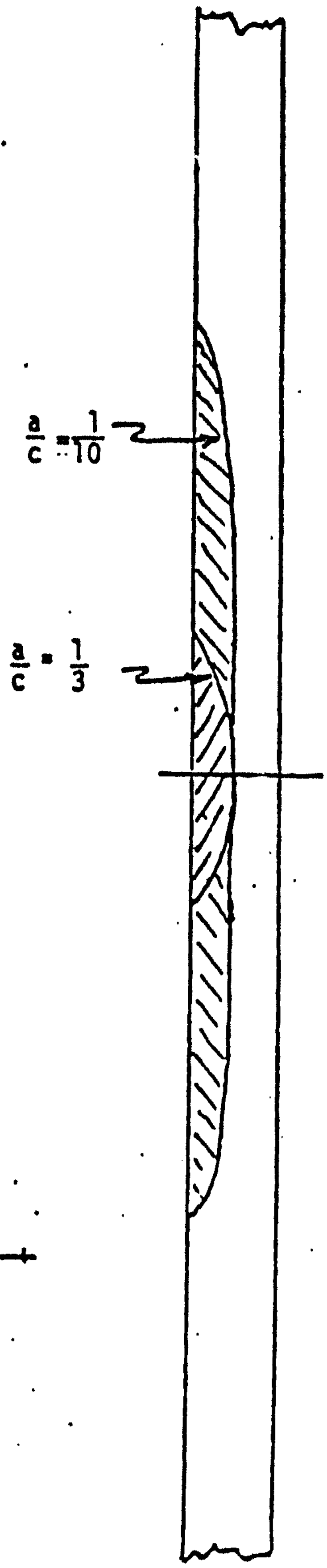


FIGURE D-20

APPENDIX E

DETERMINATION OF RT_{NDT} FOR PLANTS FOR COMPARISON WITH SCREENING CRITERIA

E.1 Introduction

In Appendix D, RT_{NDT} was shown to be an important quantity in the fracture analysis of PTS, because the toughness values, K_{Ic} and K_{Ia} are given in the ASME Code, Section XI as a function of $T - RT_{NDT}$. (In such analyses, the metal temperature, T, and the adjusted reference temperature, RT_{NDT}, are the values at the tip of the postulated crack.) Moreover, the results of the parametric studies described in Sections 3 and 7 and Appendixes D and H show that $T_f - RT_{NDT}$ is an important factor in the characterization of cooldown transient severity for a given plant. In this case, T_f is the asymptotic cooldown temperature of the water in the downcomer, and RT_{NDT} is estimated at the inside surface of the vessel. This finding led to consideration of RT_{NDT} as a screening criterion. Obviously, RT_{NDT} for a given plant is not related to the severity or probability of occurrence of a PTS in that plant and is therefore not necessarily the overall criterion for rating plants. Nevertheless, the value of RT_{NDT} at the inside surface of the vessel is a good screening criterion for the tendency of a reactor vessel to suffer damage from PTS.

RT_{NDT} is the sum of two quantities: the initial RT_{NDT} from tests made at the time the vessel was fabricated and the ΔRT_{NDT} estimated from tests designed to measure the effects of neutron radiation. The purpose of this discussion is to describe the bases for estimated initial RT_{NDT} and ΔRT_{NDT} for the individual plants. Estimates will be given for the inside surface of the vessel wall (at the clad-base metal interface) for the critical locations, either a longitudinal weld or a circumferential weld in the beltline or occasionally a beltline plant or forging.

As described below, there are a number of uncertainties in the estimation of initial RT_{NDT} and ΔRT_{NDT}, and thus there is the difficult question of establishing a proper, known, degree of conservatism in the estimate of RT_{NDT}. To resolve this question, a Working Group on RT_{NDT} was assembled for a two-day meeting (June 17 and 18, 1982) to review the NRC methods and recommend a method for use in the report. The work of that group is described in Referenced E.10. The method described below follows the recommendations of the Working Group.

E.2 Initial RT_{NDT}

E.2.1 Code Definition

The Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code contained the first requirements for measurement of RT_{NDT} for the plates, forgings, and welds that make up the reactor vessel, measurements to be made at the time of fabrication. Two types of tests are required--drop weight tests and Charpy tests. However, most of the vessels in question were fabricated in the 1960's when only Charpy tests were required.

E.2.2 Absence of Actual Measurements of RT_{NDT}

Typically, the data available to the NRC staff comprise 3 Charpy tests at 10°F for each plate, forging and weld, complete Charpy curves for the surveillance weld and base materials, and in cases where the base material was controlling, some drop weight data on archive or surveillance material. In the past, the NRC has used the guidelines of Branch Position MTEB 5-2 to obtain an estimate of initial RT_{NDT}. In summary, those guidelines were to use the Charpy 30 ft. lb. level, but not lower than 0°F. The Charpy curves from the surveillance tests were used to guide any extrapolation needed to get the 30 ft. lb. temperature from the 3 tests results at +10°F.

In summary, values of initial RT_{NDT} measured according to ASME Code rules are not generally available for the welds in question. Estimates based on the 3 Charpy test results and MTEB 5-2 are not very satisfactory, because they are overconservative for some cases.

E.2.3 Generic Data

From compilations of data obtained subsequent to the time the vessels in question were made, it is possible to divide the welds into two groups according to the weld flux used, and to develop a mean value and a standard deviation for the generic data. One must then decide if it is purdent to use the mean generic value as the best estimate for the vessel welds in question. Except for some archive material, the welds that are represented in the data base were made at a later time than the vessel welds. There may have been some differences in weld chemistry or welding practice. Furthermore, even if there were actual RT_{NDT} values for the vessel weld in question, the samples would come from weld metal qualification welds, not from actual vessel weld prolongations and not from full thickness test pieces. Thus, a mean plus 2 sigma value appears to be the best engineering estimate the initial RT_{NDT} for use in the screening criterion.

In the Combustion Engineering Report, CEN-189 (Ref. E.2) there is a table of values of initial RT_{NDT} which contains 49 values for Linde 0091, 20 values for Linde 124, and 13 values for unidentified weld fluxes, some of which we have identified as Linde 1092. By inspection, the three groups appear to be in the same population, and the total has been treated as such to yield a mean value of -56°F and a standard deviation of 17°F. It was pointed out by PNL (Ref. E.3) that these data are not normally distributed, but were skewed to the high side. However, the resulting error is swamped by the uncertainty in the application of these data to the actual vessels. An earlier weld flux, ARCOS B-5, used on one or two vessels, was deemed to be in the same population based on comparison of available Charpy energy values.

For Linde 80 weld flux, a set of 10 values provided by Babcock and Wilcox (Ref. E.4) had a mean value of 0°F and the range was from -40° to +20°F. Because the sample size for the Linde 80 welds was small, the standard deviation was taken to be the same as for Linde 0091 welds, 17°F.

E.2.4 Comparison with Vendor's Values

Westinghouse (WCAP 10019) (Ref. E.1) used MTEB 5-2 to estimate RT_{NDT} values. Combustion Engineering (CE) (Ref. E.2) proposed two bases: (1) 60°F below the Charpy 50 ft. lb. level, and (2) an upper-2-sigma value from generic data for the weld fluxes in their vessels, Linde 0091, 1092, and 124. Their utilities used method 1, but the CE report made for each plant used method 2. Babcock and Wilcox (B&W) used upper bound values from generic data for Linde 80 weld flux, which was used in their vessels. An exception is Three Mile Island 1, for which a lower value of initial RT_{NDT} was used, basis not specified.

The following table compares vendors' values with NRC values. The latter are mean plus two sigma values. As described as paragraph E.4, in combining initial and ΔRT_{NDT}, the full two sigma value is reduced about 10 degrees by the use of the quantity $2\sqrt{\sigma_0 2 + \sigma_\Delta{}^2}$.

	Linde 80 flux	Linde 0091 etc. flux
NRC	0°F mean plus 34 = 34°F	-56°F mean plus 34 = -22°F
W	0 to +10°F	0 to +10°F
CD		-20°
CE Utilities		-50°F
B&W	+20F	

E.3 Adjustment of RT_{NDT} Due to Radiation (ΔRT_{NDT})

E.3.1 Trend Curves versus Surveillance

Most of the plants in question in the thermal shock issue have withdrawn at least one surveillance capsule and tested the irradiated specimens therein. The fluence is generally not exactly the value of interest, but the result can be extrapolated to the fluence of interest by using one of the trend curves to be described.

However, there are problems associated with using individual surveillance results as the sole source of information about a plant. First, the surveillance weld often does not match the critical vessel weld exactly, i.e., the weld wire heat numbers are different. A broader problem is that caused by scatter in the ΔRT_{NDT} data. This results in part from the fact that ΔRT_{NDT} is the difference between the curves for irradiated and unirradiated material, both of which were fitted to data that typically shows considerable scatter.

Thus, there is a preference for the use of trend curves instead of individual surveillance data. To use any of the trend curves, the chemistry of the material must be known, in particular, the copper content. This is obtained from analysis of the weld metal qualification weld for the weld wire heat number and weld flux number that were used for the critical weld. If not available, data were sought for that weld wire heat number as used in other vessels. Failing that, best estimates were made from the surveillance weld (even through the heat numbers did not match) and from generic data for welds made in that time period. As a last resort, a value of 0.35% copper was used, that being the value which gave the upper limit or bounding line for all data in Regulatory Guide 1.99 Revision 1 (Ref. E.5) as described below.

E.3.2 Regulatory Guide 1.99 Revision 1, Bounding Curves

Regulatory Guide 1.99 Rev. 1 published in April 1977, contains the procedure recommended at that time by the NRC to obtain ΔRT_{NDT}, the "adjustment of reference temperature" as a function of chemistry and neutron fluence. Copper was the dominant residual element in the chemistry term (the other was phosphorus) as can be seen at the top of Figure E-1. The exponent on the fluence term is 0.5, but there is a cut-off or upper limit line for which the exponent is 0.194 for high copper content and fluence exceeding 6×10^{18} n/cm^2 (E>1 MeV).

Criticism leveled at Regulatory Guide 1.99 became more insistent when the PTS issue made it necessary to look hard at all sources of conservatism. It was said that (a) the curves were too conservative at high fluences, especially for low-nickel materials, and (b) the phosphorus term was not supported by recent studies such as the MPC report (Ref. E.6) described below and an EPRI report (Ref. E.11), and should be dropped. Nevertheless, Regulatory Guide 1.99 was used for high-nickel materials by all 3 vendors in the reports that were concurrent with the utilities' 150 day reports. The high-nickel materials are ASTM A 533 plates, A 508 forgings, and welds of comparable chemistry, for which the nickel content is generally between 0.5 and 1.0 percent. The low-nickel materials are ASTM A 302 plates and welds of comparable chemistry, which generally have less than 0.25 percent nickel as a residual element. A relatively small number of older vessels have low-nickel material.

E.3.3 Gutherie Trend Curves

Evidence has been accumulating for several years that the low-nickel materials are less sensitive to neutron radiation. When the PWR surveillance data base was analyzed by the NRC in October 1981, the difference between high and low nickel content material was apparent. Westinghouse and CE reported similar findings and presented empirical equations for the low-nickel material. (B&W have no plants with low-nickel materials in the reactor vessel.) The PWR surveillance data have now been fitted by a multiple regression analysis technique. The work was done at HEDL by George Guthrie, whose name is attached to the new trend curves (Ref. E.7). The Guthrie mean curve is as follows:

$$\Delta RT_{NDT} = [-10 + 470\ Cu + 350\ Cu\ Ni]\ [f/10^{19}]^{0.27}$$

ΔRT_{NDT} = adjustment of reference temperature, degrees F
Cu = weight percent copper
Ni = weight percent nickel
f = fluence, n/cm^2 (E > 1 MeV)

The use of a copper-nickel product term reflects the advice of J. R. Hawthorne (Ref. E.8) of the Naval Research Laboratory to the effect that nickel seems to enhance the effect of copper, but nickel does not cause increased embrittlement in the absence of copper. The product term is also consistent with work reported by Varsik and Byrne (Ref. E.9) in which their "chemistry factor" was the product of copper and a quantity, nickel plus other elements.

Figures E-2, E-3, and E-4 show how the Guthrie formula fits the PWR surveillance data. The residual value (predicted minus measured) for each line of data is plotted against fluence, copper content, and nickel content to give a graphical check on the effectiveness of the multiple regression analysis.

E.3.4 Guthrie Upper Bound Trend Curves

The standard deviation for the data analysis described in paragraph E.3.3 was 24 degrees F. From inspection of Figure E-2, it appears that a constant 2-sigma upper bound is satisfactory over the fluence range of interest.

E.3.5 Comparison with MPC Curves

As further support for the Guthrie mean curve, Figure E-5 gives a comparison of the Guthrie mean curve for representative copper and nickel contents with a mean curve developed by the Metal Properties Council for ASTM Committee E-10 on Nuclear Technology and Applications (Ref. E.6). The latter is being balloted as an ASTM Standard. The MPC data base contains all of the test reactor and surveillance data that fit the criteria for material form and irradiation temperature that were available in November, 1977. There is reasonably good agreement between the MPC trend curves and the Guthrie curves, considering that the MPC curves were for a range of nickel content, but were without a nickel term in the equation.

The MPC trend curve did not contain a phosphorus term, because in the regression analysis the addition of a phosphorus term did not produce any significant decrease in the residual variance. In a further study of this finding, the MPC Task Group found a statistically significant relationship of phosphorus content to copper content, i.e., high phosphorus was found with high copper. Thus, their combined effects were represented in the trend curve formulation by a copper term alone.

E.3.6 Comparison with Vendor's Curves

Westinghouse and Combustion Engineering drew bounding curves for low-nickel material. Figure E-6 gives a comparison of the Gutherie mean plus 2-sigma curves for 0.15% nickel material with the low-nickel trend curves presented by Westinghouse and CE. The latter lie below the Guthrie curves over most of the range of fluence.

E.4 Screening Value of RT_{NDT}

The Working Group on RT_{NDT} (Ref. E.10) agreed that the value of RT_{NDT} to be used in screening plants should be calculated as the sum of 3 quantities: the mean value of initial RT_{NDT} (RT_{NDTo}), plus the mean value of ΔRT_{NDT} at the inside surface of the vessel, plus twice the square root of the sum of the squares of the standard deviation on each, i.e., $2\sqrt{\sigma_o^2 + \sigma_\Delta^2}$.

E.4.1 Uncertainties

Uncertainties in the screening value of RT_{NDT} arise from several sources. Those associated with the estimate of initial RT_{NDT} were discussed in paragraph E.2. For ΔRT_{NDT}, there is the scatter about the trend curve (shown in Figures E-2, E-3 and E-4) which is made up of the uncertainty in response of material to radiation, plus errors in the copper and fluence values in the data base and errors in the Charpy shift measurement itself. In addition, there is uncertainty in the copper content of the critical weld in the vessel. Because copper was introduced as a plating on the weld wire, and plating thickness was not controlled, variation in copper content through the vessel wall and along the length of the weld is expected to be considerable From a number of measurements for certain weld wire heat numbers, one standard deviation is expected to be about 0.03 percent copper, typically. This is equivalent to 15 degrees F in the plants with higher fluences.

Nevertheless, the copper contents used in calculating RT_{NDT} for plants wer best-estimate values. They were not mean plus 2 sigma values. This is one reason why the Working Group on RT_{NDT} felt that the screening values should have the 2 sigma measure of error added to the mean.

E.4.2 Alternative Calculation of RT_{NDT}

For high values of copper and nickel contents, the method described above gives values higher than those predicted by that part of the Upper Limit of R.G. 1.99, given by the equation:

$$\Delta RT_{NDT} = 283\ (f/10^{19})^{0.194}$$

Experience has shown that the latter bounds the available data. Therefore, the screening value of RT_{NDT} is taken to be the lower of two quantities:

$$RT_{NDT} = RT_{NDT\ o} + \text{Guthrie Mean } \Delta RT_{NDT} + 2\sqrt{\sigma_o^2 + \sigma_\Delta^2}$$

or

$$RT_{NDT} = RT_{NDTo} + 283\ (f/10^{19})^{0.194} + 2\ \sigma_o$$

as illustrated schematically in Figure E-7.

The 2-sigma term in the second equation does not include the error in ΔRT_{NDT} because the term for ΔRT_{NDT} is an upper-bound equation.

The Upper Limit line of R.G. 1.99 actually consists of two branches, the one described above, for fluences above 6×10^{18}, and a lower branch that has an exponent of 0.5. The latter was not used, because it does not bound all of the observed data in that fluence range. Thus, for the purpose of this screening criterion, the alterative equation,

$$\Delta RT_{NDT} = 283\ (f/10^{19})^{0.194}$$

is used at fluences below 6×10^{18} as well as for higher fluences.

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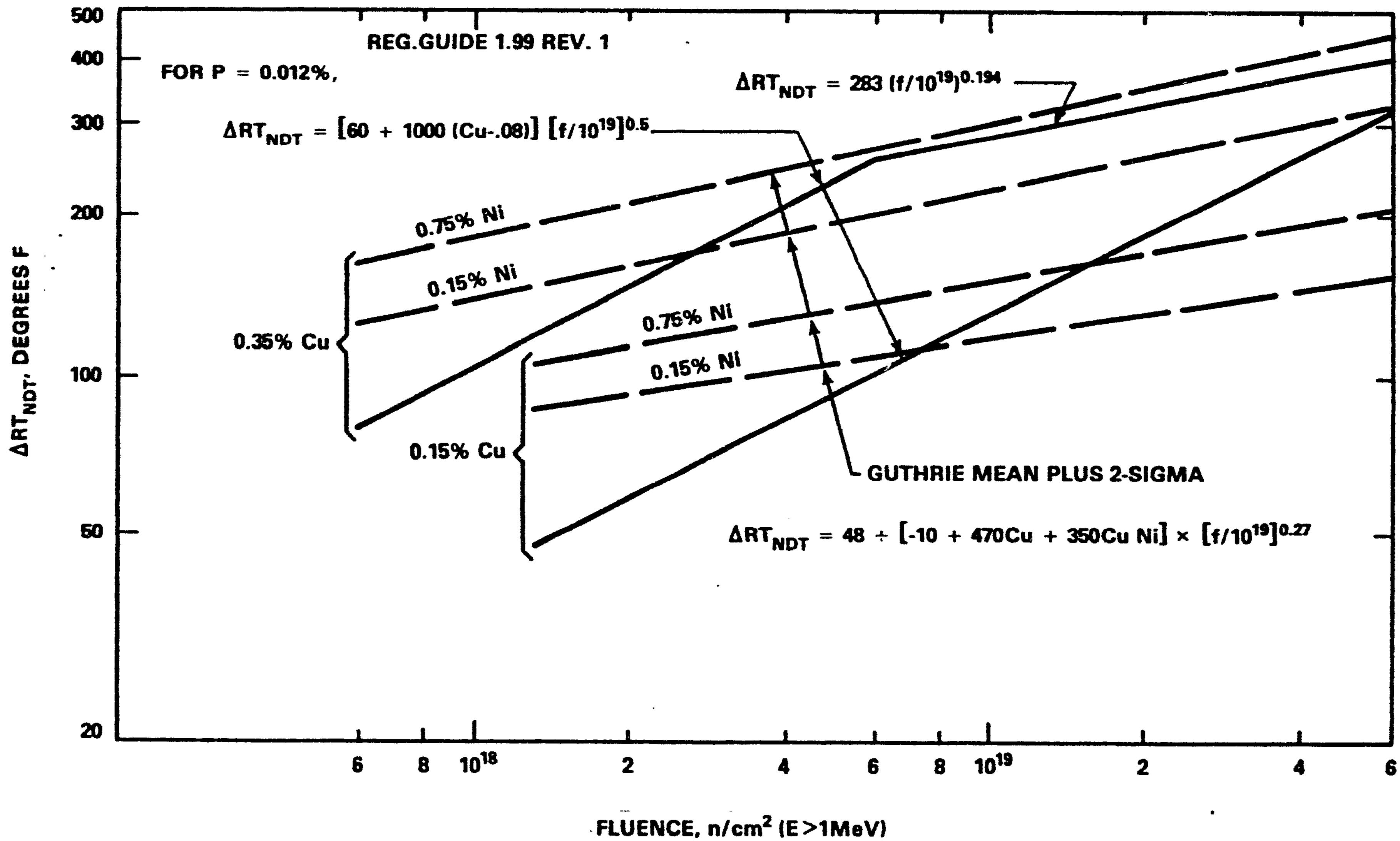

FIGURE E-1: COMPARISON OF REG. GUIDE 1.99 REV. 1 AND GUTHRIE MEAN PLUS 2-SIGMA TREND CURVES

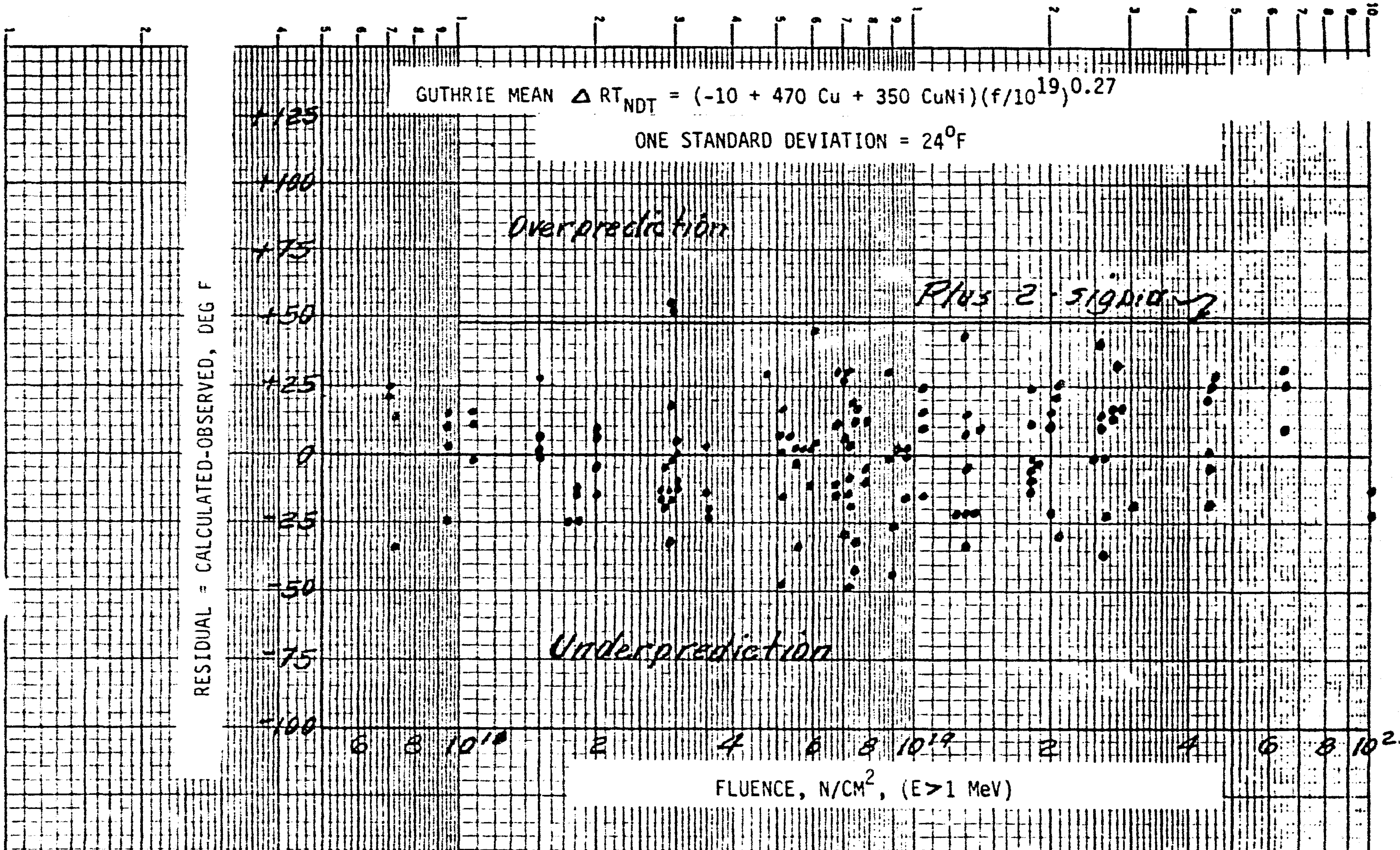

FIGURE E-2 ANALYSIS OF THE FIT OF THE GUTHRIE FORMULA TO THE PWR SURVEILLANCE DATA BASE. EFFECT OF FLUENCE

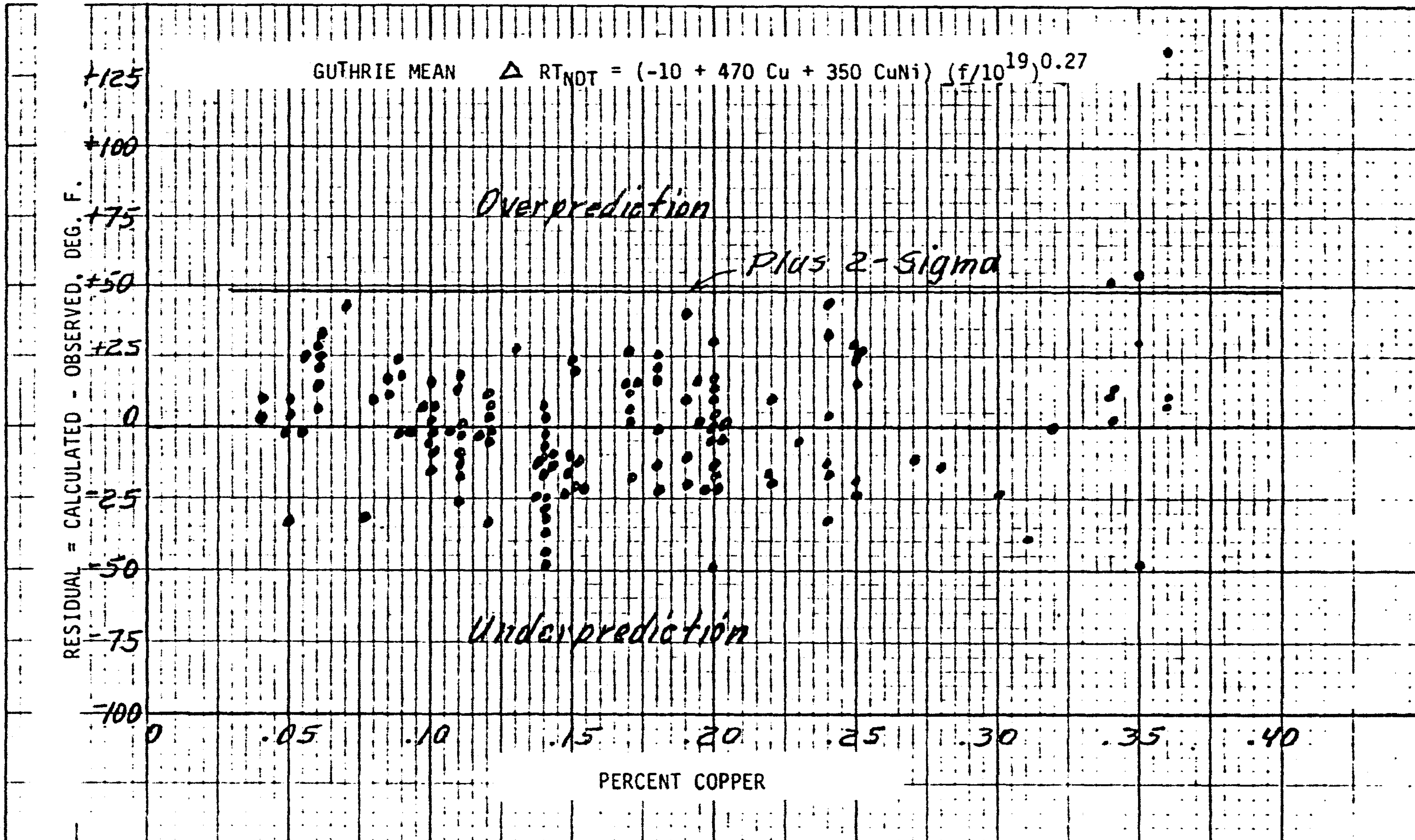

FIGURE E-3 ANALYSIS OF THE FIT OF THE GUTHRIE FORMULA TO THE PWR SURVEILLANCE DATA BASE EFFECT OF COPPER CONTENT

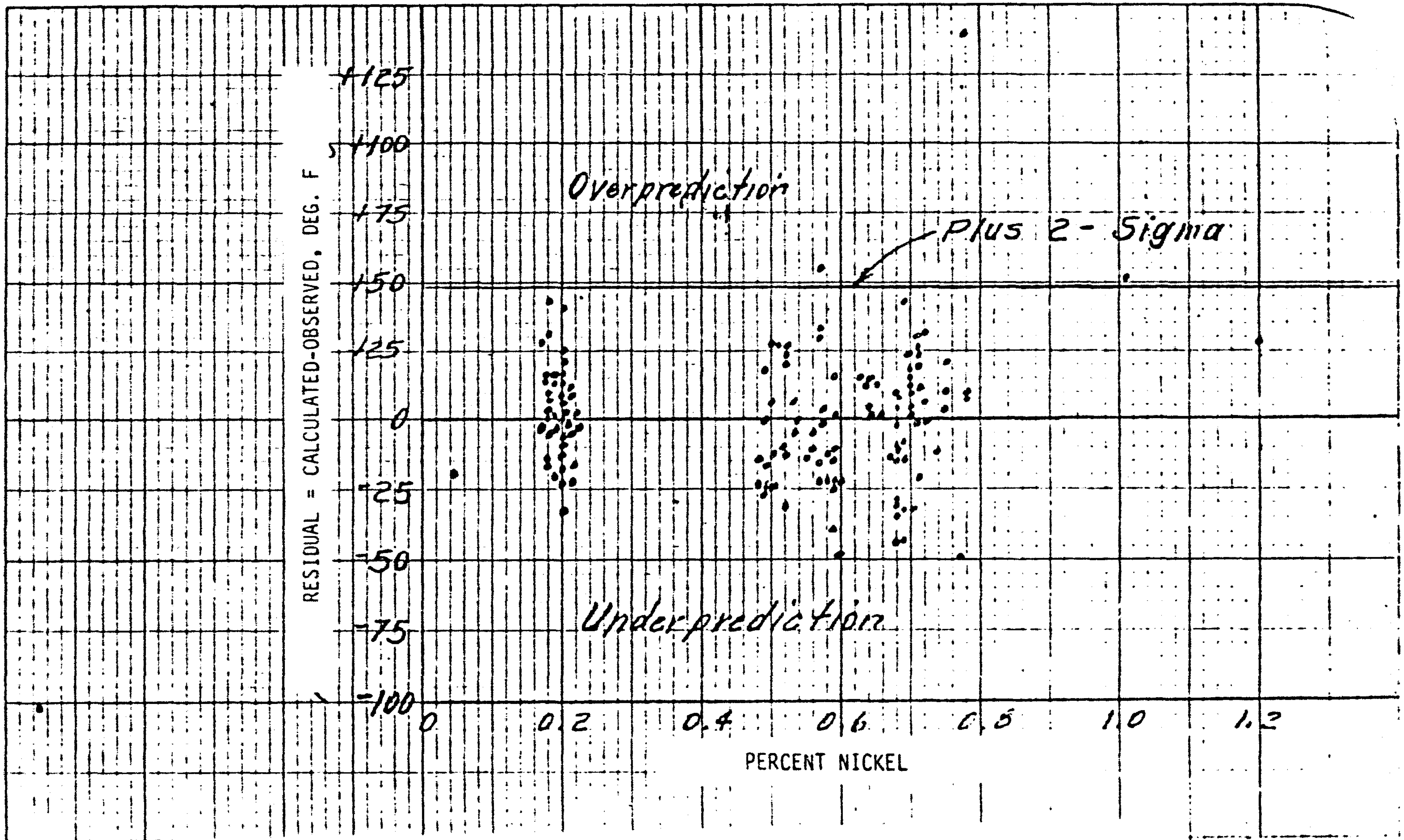

FIGURE E-4 ANALYSIS OF THE FIT OF THE GUTHRIE FORMULA
TO THE PWR SURVEILLANCE DATA BASE
EFFECT OF NICKEL CONTENT

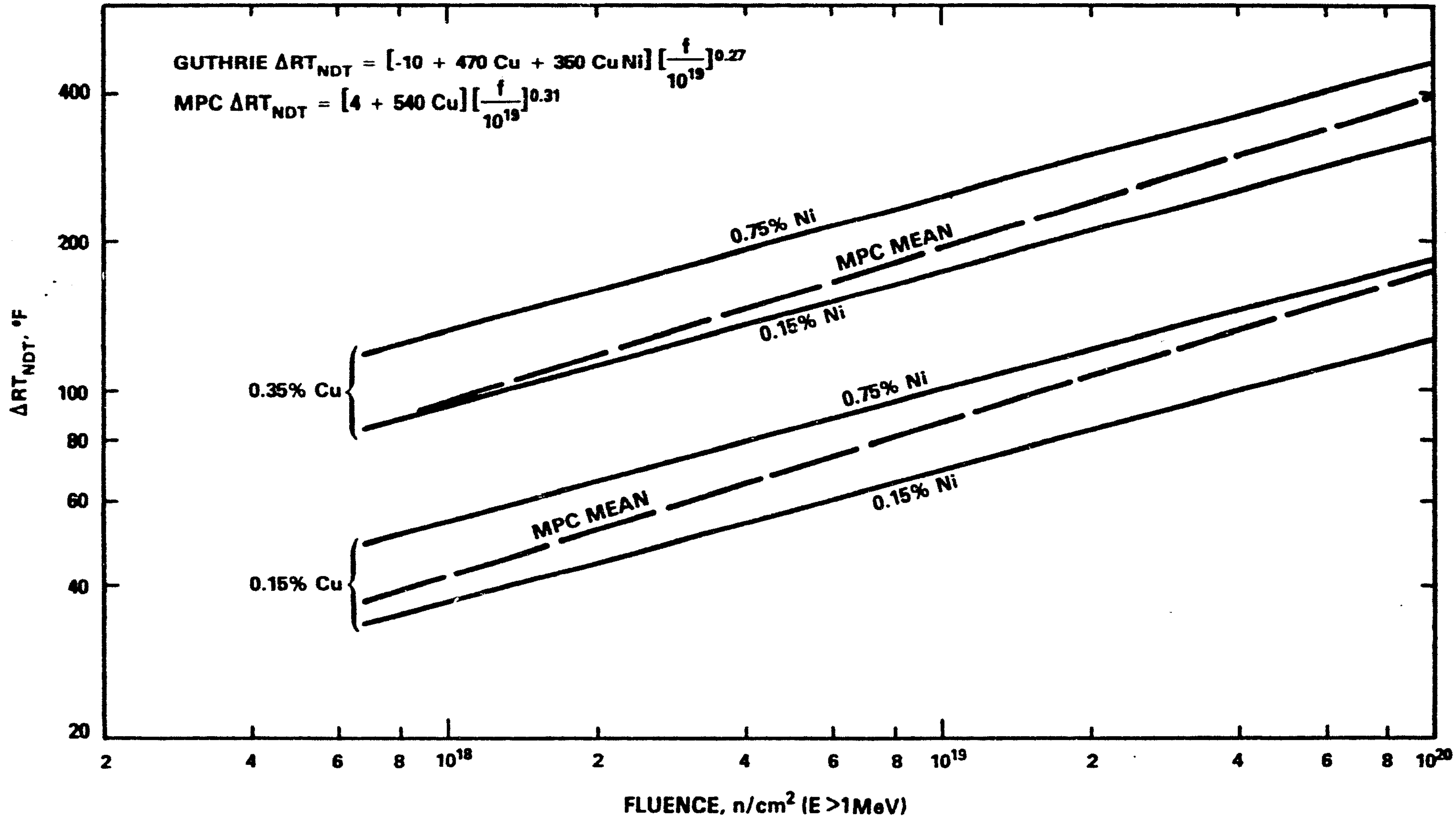

FIGURE E-5: COMPARISON OF GUTHRIE AND MPC FORMULAS FOR THE MEAN VALUES OF ΔRT_{NDT} FOR REPRESENTATIVE COPPER AND NICKEL CONTENTS

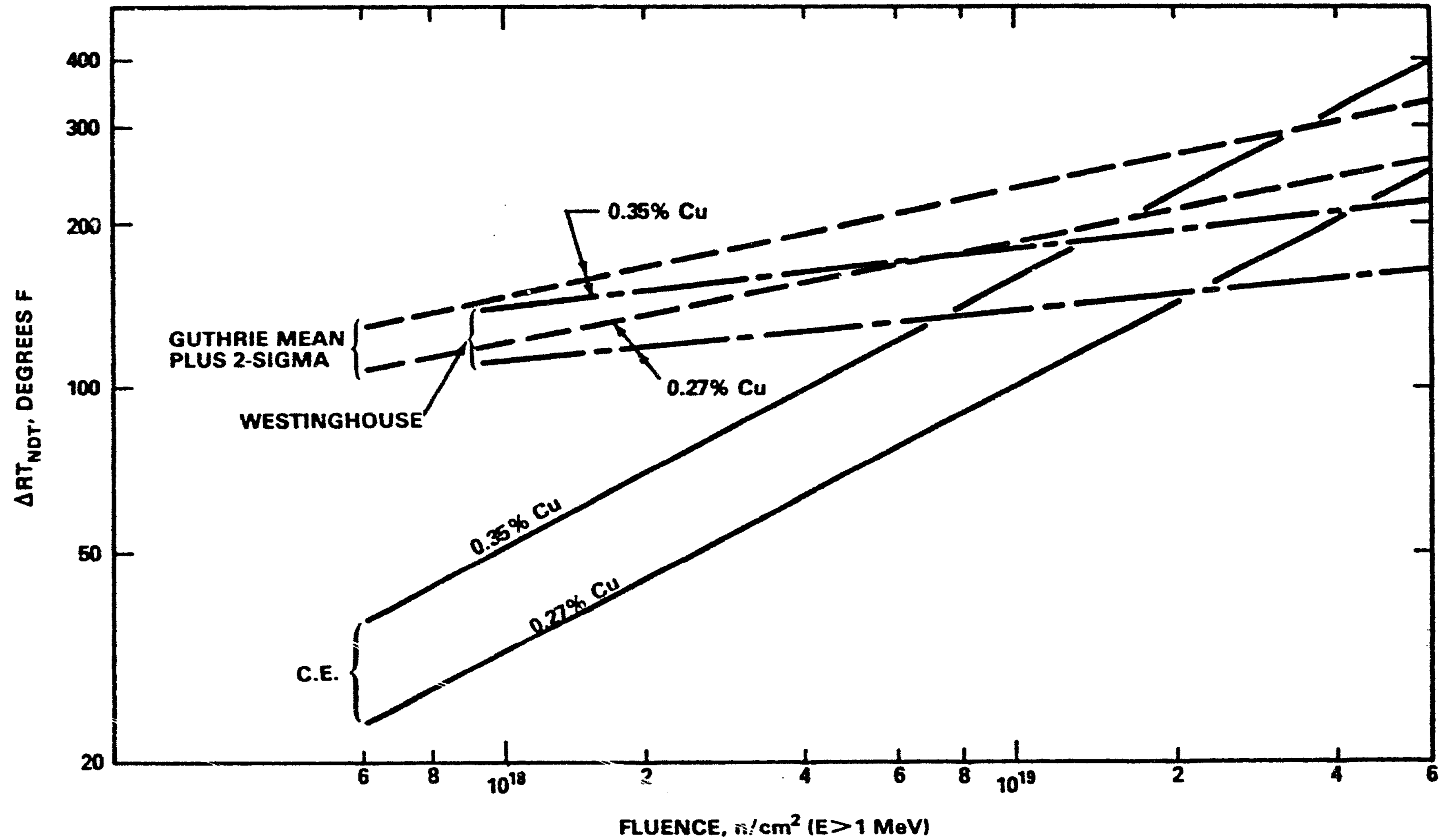

FIGURE E-6: COMPARISON OF C.E., WESTINGHOUSE AND GUTHRIE TREND CURVES FOR LOW-NICKEL MATERIAL

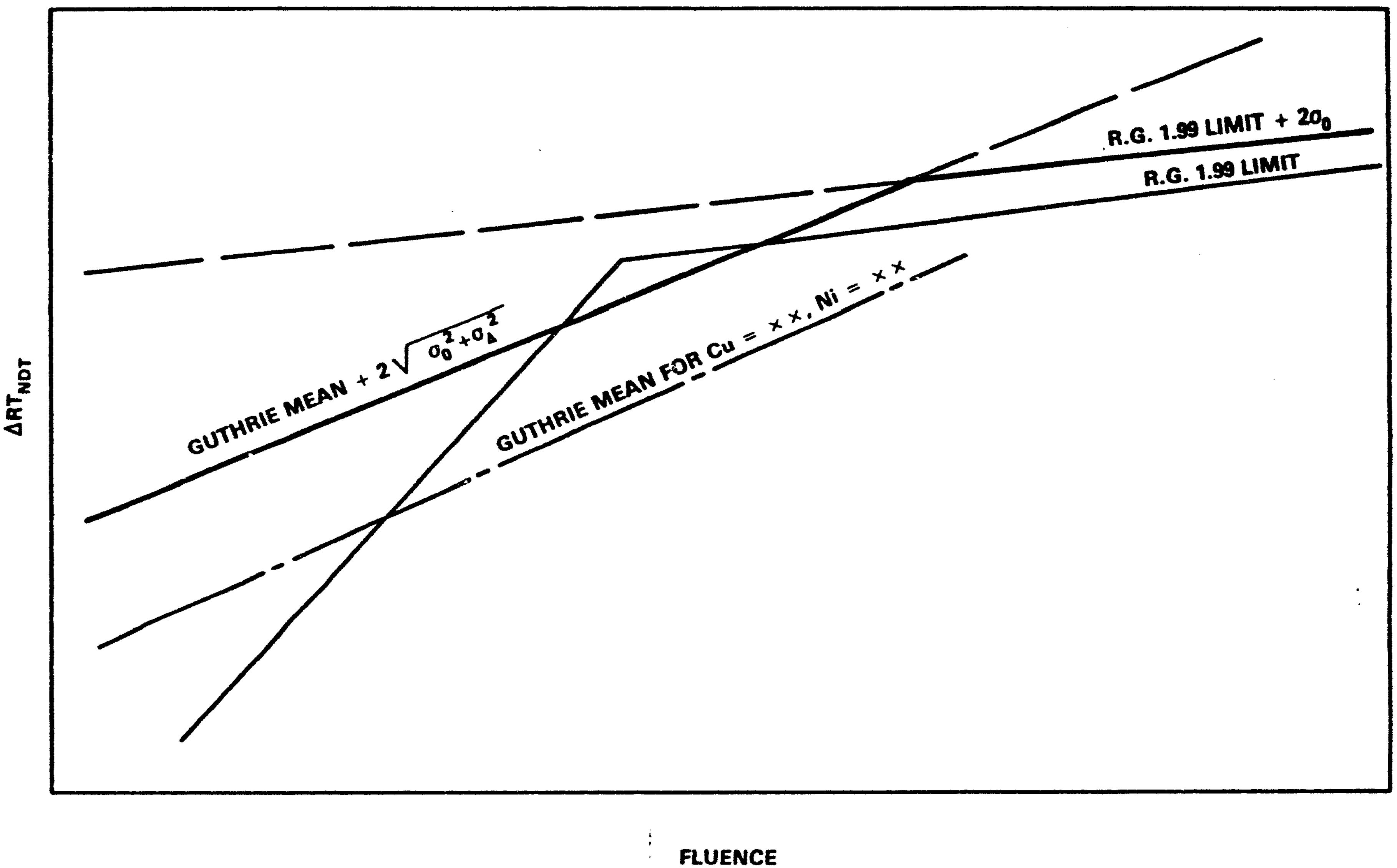

FIGURE E-7: SCHEMATIC DIAGRAM OF TREND CURVES

APPENDIX F

PRESSURE VESSEL FAST NEUTRON FLUENCE UNCERTAINTY

F.1 Introduction

The following discussion deals with the components of the staff's fast fluence (>1 MeV) predictive calculational uncertainty.

There are two major sources of uncertainty in fast fluence computations, i.e., (a) the uncertainty which results from the measured values of the fluence used in benchmarking the computer codes, and (b) computational, which originates from uncertainties of input quantities to the code.

F.2 Benchmarking Uncertainty

The prediction of the calculation is benchmarked to measured values of carefully performed experiments. The benchmarking process has been instrumental in recent improvement of the uncertainty as shown in Fugure F-1. It can be seen that in the early years of commercial nuclear power the predictive uncertainty was very large. Figure F-1 represents the FSAR predicted values of the fluence and their comparison to a posterior measured value with the surveillance capsule. Measured values from the surveilance capsules and the Pool Critical Assembly improved the predictive capability in the 1970s and is shown in 1980-81 when surveillance capsules were removed. The staff has a techniccal assistance program to BNL to benchmark the neutron transport code DOT 3.5 and verify the fluence values in the eight pressure vessels which have been thought to have marginal toughness. At this time the benchmarking is nearly complete.

The benchmarking includes data from the following:

(i) The Pool Critical Assembly pressure vessel dosimetry benchmark experiment (Ref. F.2).

(ii) The ANO-1 surveillance capsule and reactor cavity flux measurements (Refs. F.3, F.4, F.5, and F.6).

EPRI-sponsored measurements in the reactor cavity provide flux values to an estimated accuracy of ±15 percent (undesignated distribution). Surveillance capsule measurements are being used to adjust the fluence calculated on the inside of the pressure vessel.

(iii) Fort Calhoun surveillance capsule.

(iv) Main Yankee surveillance capsule.

Figure F-2 shows a typical configuration of a surveillance capsule. The overall length corresponds to that of a fuel assembly and contains an upper, middle, and lower tensile monitor compartments. Tensile specimens are housed in this section along with radiation monitors (Figure F-4). Charpy impact specimens are housed in separate compartments (Figure F-3). Typical locations of surveillance capsules are shown in Figure F-5.

The causes of uncertainty in dosimetry measurements are related to reaction rate cross-sections, the photofission correction, counting calibration, flux-time history, etc. The overall benchmarking uncertainty is ±15 percent (1σ).

F.3 Computational Uncertainty

Computational uncertainty results from uncertainties in cross-section data (inelastic scattering of iron is a particular source of error), modeling, numerical methods, source representation, geometry, etc., which are inputs to the DOT 3.5 code. The DOT series of codes are two-dimensional neutron transport codes based on finite differencing with anisotropic scattering in (x,y), (r,Θ), or (R,z) geometries. The DOT 3.5 version is operational at BNL (Ref. F.7)

In order to evaluate calculational uncertainties and provide an additional independent assessment of the uncertainty, a direct parametric analysis is being performed. In this analysis major uncertainty components (e.g., source representation, geometry, cross-section, etc.) have been identified and are being quantified. DOT sensitivity calculations are being performed to propagate these uncertainties and determine their effects on vessel fluence and ΔRT_{NDT} (Ref. F.8). The expected uncertainty is ±15 percent (1σ).

We estimate the overall predictive uncertainty to be ±20 percent (1σ) comparable to ±15 to ±20 percent recently claimed by the vendors (Refs. F.9 and F.10).

The above is illustrated in diagrammatic form in Figure F-6 which illustrates the overall uncertainty, its components, and the sources of the experimental uncertainty.

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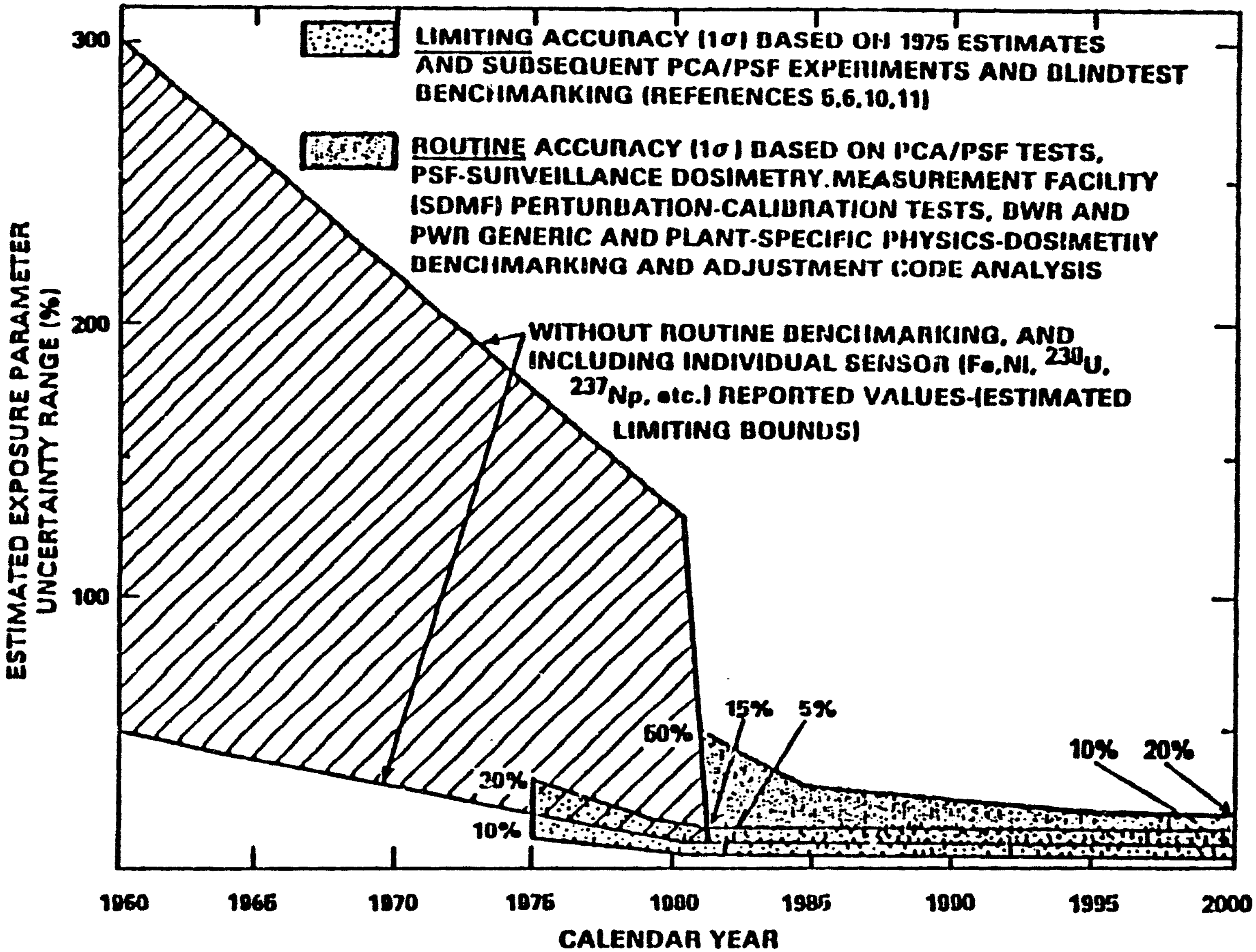
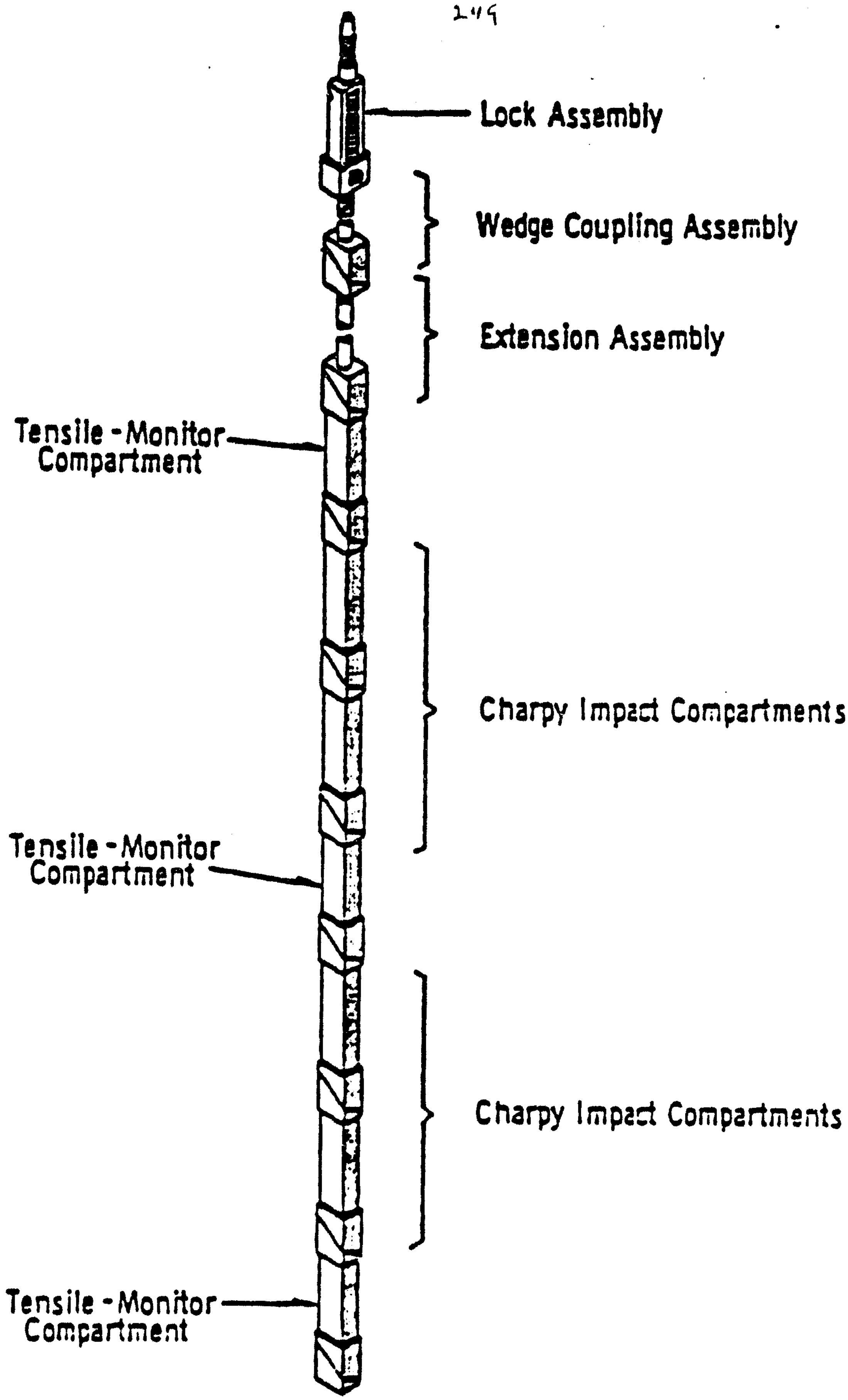
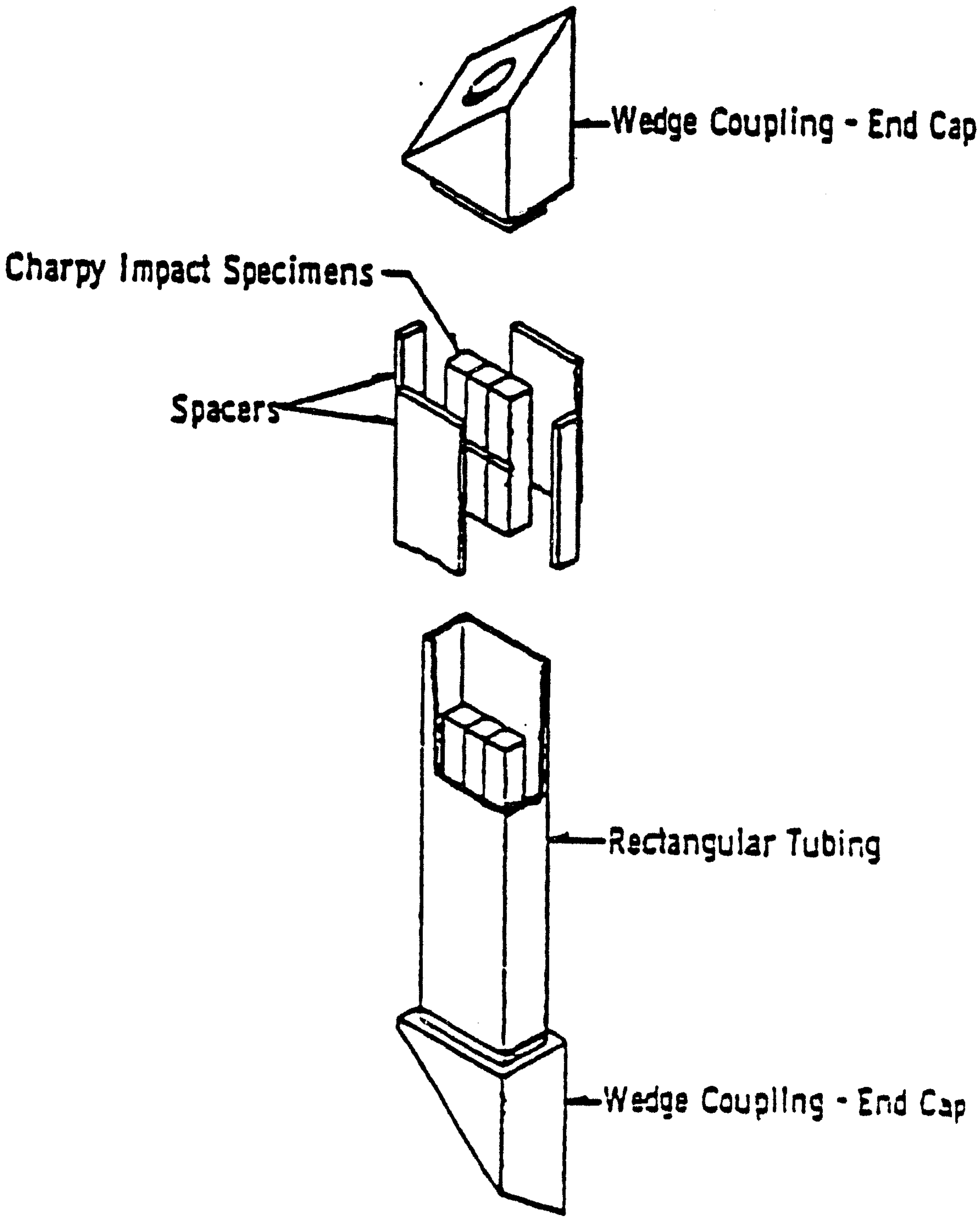


FIGURE F-1 Estimated Exposure Parameter Uncertainties Obtained from FSAR and Surveillance Capsule Reports.



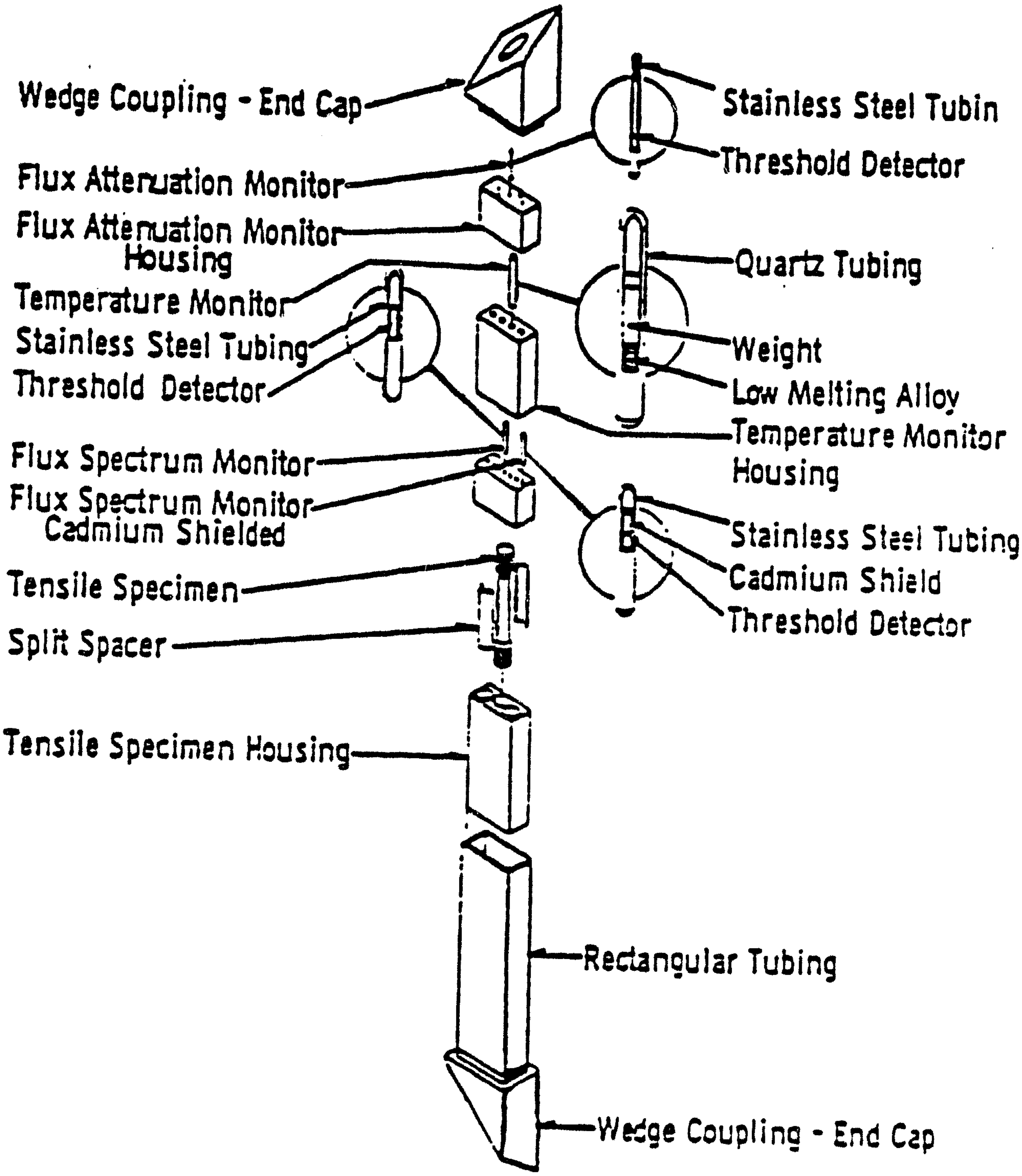
OMAHA PUBLIC POWER DISTRICT Fort Calhoun Station Unit No. 1	SURVEILLANCE CAPSULE ASSEMBLY

FIGURE F-2



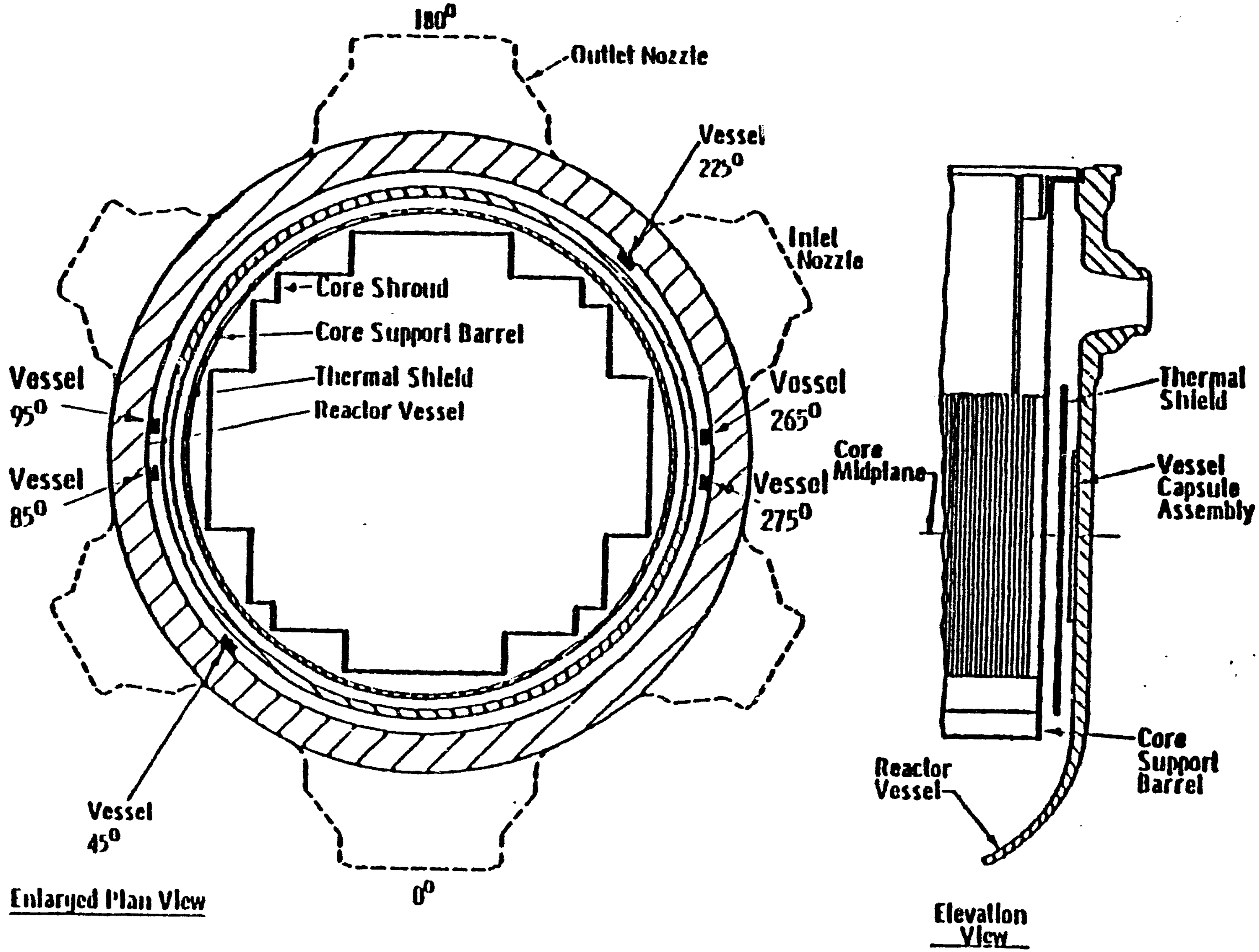
OMAHA PUBLIC POWER DISTRICT Fort Calhoun Station Unit No. 1	CHARPY IMPACT COMPARTMENT ASSEMBLY

FIGURE F-3



OMAHA POWER DISTRICT Fort Calhoun Station Unit No. 1	TENSILE-MONITOR COMPARTMENT ASSEMBLY

FIGURE F-4



OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Station
Unit No. 1

LOCATION OF SURVEILLANCE CAPSULE ASSEMBLIES

FIGURE F-5

FIGURE F-6

FAST FLUENCE UNCERTAINTY

± 20% (1σ) Predictive Uncertainty

Benchmarking (experimental)
± 15% (1σ)

Calculational Uncertainty
± 10-15% (1σ)

Benchmarking

Pool Critical Assembly

ANO-1: Surveillance Capsure
Reactor Cavity

Ft. Calhoun: Surveillance
Capsule

Maine Yankee: Surveillance
Capsule

Calculational

Sensitivity analysis to be
performed for major
uncertainty components

APPENDIX G

OVERCOOLING EVENT SEQUENCES LEADING TO POTENTIAL PTS CONDITONS

G.1 Critical Parameters

G.1.1 Temperature

The temperature of concern is the fluid temperature at the reactor pressure vessel (RPV) inner surface (at the point of interest, typically a weld). For PTS analyses, the initial steady-state temperature is taken to be 550°F. Final temperatures of 350°F and less are expected to be potential PTS initiating events.

G.1.2 Rate of Cooldown

The rate of cooldown is also important. Events which can be controlled within the limits of Appendix G heatup/cooldown limits are not considered to be PTS initiating events. As a guide, transients which do not exceed 100°F per hour cooldown may be excluded from PTS consideration.

Present fracture mechanics analysis methods input the temporal coolant temperature behavior in a stylized manner approximated with an equation of the form:

$$T = To - (To - T_f)\,(1 - \exp(-\beta t))$$

Where To - initial temperature, °F

T_f - final temperature, °F

β - cooldown parameter, per minute

t - time, minutes

Although using this formulation to characterize the coolant temperature temporal behavior is advantageous from a fracture mechanics standpoint, it does not consider variations in the cooldown rate which can occur as a result of operator action, and control or protection systems operations. Also, the predicted final temperature is uncertain, as stored heat in the primary system metal and the decay heat output from the reactor core will increase the coolant temperature once action is taken to terminate the cooldown.

When approximating the coolant temperature temporal behavior as a result of specific initiating event with this form of stylized equation, the selection of T_f and β (the final temperature and the cooldown parameter) which best approximates the actual transient behavior, requires some engineering judgment. In general, the final temperature is selected as the lowest calculated value and the cooldown parameter is either the natural (for example best fit) cooldown parameter or an adjusted value for cases where the temperature increases following termination of the cooldown. The cooldown parameter used for these analyses has been adjusted to account for this temperature increase and is based on the Westinghouse approach. This approach considers the fracture mechanics response to the actual temperature transient and the fracture mechanics response to the stylized formulation with the adjusted β value, designated β*. The designated β* value is obtained from:

$$\beta^* = 2/t^*$$

where t* is the time of lowest temperature, minutes and β* is never less than the natural cooldown rate.

G.1.3 Pressure

The predominant stress associated with PTS is the thermal stress associated with the rapid cooldown of the RPV wall. It is noted that the thermal stress alone is not sufficient to cause loss of RPV integrity. The other important stress is due to pressure. Maintaining system pressure relatively high or repressurizing following a cooldown is of primary concern. Control of the primary system pressure is achieved by operator action to terminate charging and HPI flow to the primary. Other systems, such as letdown, pressurizer heater and pressurizer sprays are not considered here for pressure control.

A review of current Westinghouse guidelines for HPI termination indicates that primary system pressures of 2250 psig could be obtained prior to HPI termination for certain coldown events, such as steam line breaks. This pressure can also be reached as a result of delayed operator action in terminating charging or safety injection flow. Therefore, 2250 psig was considered appropriate for this evaluation unless a lower pressure can be justified when a specific transient is considered.

G.1.4 Frequency

The expected frequency of occurrence of an event which could result in PTS conditions is an important parameter because it defines the relative probability of a PTS initiator. When combined with other probabilities and consequences associated with specific events in a sequence, the risk due to PTS from that sequence can be estimated. Theoretically, by adding the risk from all significant event sequences, the total risk can be obtained. The practical difficulty is obtaining reasonable assurance that all significant event sequences are in fact included.

G.1.5 Uncertainty in T_f, β, and P Values for Characterization

It is not possible, at this time, to determine the uncertainty in either T_f or β as used to characterize a PTS transient. The computer programs and models used to determine the thermal-hydraulic characteristics of the PTS transients are still under review by the staff. The input data and assumptions used in these computer programs and models range from "best estimate" to "conservative for PTS," and in some cases inconsistent data are used to enhance the cooldown or the cooldown rate. While the staff has endorsed the use of best estimate analyses to characterize the PTS transient behavior, we relied very heavily on the analyses presented by Westinghouse to support our evaluation. (Reference G.1)

G.2 Events to be Considered

The selection of events to be considered is based on the event sequences identified by Westinghouse. The staff has reviewed the sequences, the associated frequencies of occurrences, the β's and T_f's. Based on a number of technical meetings with Westinghouse, we have established a set of events of significance to the PTS issue which have been identified as Main Steam Line Break (MSLB), Small Steam Line Break (SSLB), Small Break Loss-of-Coolant Accidents (SBLOCA), and Steam Generator Tube Rupture (SGTR).

In order to characterize individual event sequences within each of these groups, certain additional parameters have been identified which determine the significance of these sequences as a PTS challenge.

The level of decay heat present during an initiating event is an important parameter in the cooldown from a given transient. The level of decay heat is related principally to the operational status (full power operation, hot zero power, other) immediately preceding the transient. The frequency of challenging event sequences are thus differentiated by the operational status of the plant.

The time allowed prior to initiation of proper operator action is another parameter that is important in some sequences. This variable has been used as a parameter in the results which characterize certain sequences that are presented below.

G.2.1 Main Steam Line Break (MSLB)

The MSLB, with a break area larger than 6 inches equivalent diameter, results in a rapid cooldown of the primary system. The final downcomer coolant temperatures can be on the order of 200°F, dependent on the decay heat level and operator action time to terminate auxiliary feedwater. The system will repressurize as a result of continued safety injection flow, and may repressurize in excess of 2000 psig dependent on the time operator action is taken to terminate safety injection. Stored metal heat and decay heat also aid in the repressurization.

Frequency of Main Steam Line Break

There have been three large breaks in steam-bearing lines in commercial nuclear plants. Two (at H. B. Robinson in 1970 and Turkey Point in 1971) occurred during hot functional tests prior to commercial operation and caused failures in branch lines or headers associated with the steam generator safety valves. These two failures were attributed to underdesign of the piping at the failed locations. A recent failure occurred at Oconee in a line off of the high pressure turbine exhaust. The preliminary evaluation attributed the failure to wall thinning by steam erosion.

The estimated median frequency of a pipe break (>6 inches) in the primary coolant system was 10^{-4}/RY in WASH-1400 based on evaluations of several data sources including nonnuclear experience. The steam lines including bypass lines (>6 inches) between the steam generator and the main steam isolation valves in 3-loop Westinghouse plants are about 1.7 times longer than comparable primary system piping which would yield an equivalent steamline break frequency of 1.7×10^{-4}/RY. Median estimates of the frequency of steamline breaks (up to MSIV) in the Zion and Indian Point Probabilistic studies were 4×10^{-4}/RY and 10^{-4}/RY, respectively, based on no major failures in these lines during commercial nuclear operations. There are about 600 reactor-years of commercial operation at nuclear power plants, so the recent Oconee failure represents a point estimate of 1.6×10^{-3}/RY for steamlines downstream of the MSIV which have a lower level of quality control than the steamlines upstream of the MSIV. The length of steam piping (>6 inches) downstream of the MSIV may vary from plant to plant but we assumed it to be three times the length of piping upstream of the MSIV, so that an equivalent failure frequency estimate for the piping upstream of the MSIV would be about 5×10^{-4}/RY.

The two events at Robinson and Turkey Point occurred prior to commercial operation during shakedown tests expected to identify plant design and operating deficiencies so that these data should not be given equal weight with the Oconee event. Bush (Reference G.2) estimated that about 70% of the piping failures occur in the first two years of operation. Thus if one assumes that the Robinson and Turkey Point events are part of the first two years' operation at all plants (140 RY), the effective steamline pipe break frequency for mature plants would be estimated to be about 4×10^{-3}/RY.

Based on an evaluation of several sources of data, Bush (Reference G.2) estimated the frequency of large disrupture failures to be between 10^{-5} and 10^{-6}/RY and also pointed out that "virtually any type of piping damage may be considered to result from some kind of human error or lack of knowledge in design, fabrication, or operation."

A probabilistic fracture mechanics analysis of pipe fracture in the primary coolant loop of a PWR plant was presented in Reference G.3. This study estimated very low pipe break frequencies $<10^{-6}$/RY based on cyclic stresses and seismic events imposed in service. These results are limited by the scope of the analysis. Thus, these results are given little weight compared to experience.

The estimates of steamline break frequency appear to cluster around 10^{-4}/RY. A frequency of 1.7×10^{-4}/RY is used for evaluating steamline breaks for screening purposes. It is estimated that the upside uncertainty is a factor of 10 and the downside a factor of 100 based on the preceding discussion. The Westinghouse Owners Group contends that the frequency of major steamline breaks (>6 inches) is much smaller based on Reference G.3.

We considered other failures in addition to the initiating event that might result in extended HPSI operation such as failure of reactor coolant pump seals, unisolated letdown lines, and a stuck-open PORV. We have assumed that the RCP seal cooling will be maintained during this event; however, this assumption should be confirmed for the Westinghouse plants. Therefore, the conditional probability of RCP seal failures is considered to be low. Similarly, we have assumed that the letdown line will be automatically isolated and will not automatically open on reseting SIS or containment isolation signals. This assumption should be confirmed for the Westinghouse plants. In some plants with high shut-off head ECCS pumps, the PORV may be challenged. The conditional probability of a PORV sticking open is estimated to be 10^{-2}/D in WASH-1400. The conditional probability of an operator failing to isolate the PORV in ten minutes is about 2×10^{-2}/D from Reference G.4. The combined event frequency is not significant (10^{-6}/RY). Thus, only simple large steamline breaks without any additional failures will be considered for this sequence category.

Operator Actions

The operator is trained to isolate the faulted steam generator in a steamline break event based on low pressure in one steam generator compared to the others. Operator action time to isolate the feedwater to the faulted steam generator is presented in Figure II-4 of Reference G.5 based on simulator experience. Using human error estimates from Reference G.6 for high stress situations and assuming three operators in the control room, human error probabilities (HEPs) for failing to recognize that the AFW has to be isolated are estimated to be 0.7 at 5 minutes and 0.04 at 10 minutes compared to 0.3 and 0.9 from Figure II-4 of Reference G.5. We will assume HEPs of 0.5 at 5 minutes and 0.1 at 10 minutes, 3×10^{-3} at 30 minutes (based on WASH-1400 estimate for switchover from injection ro recirculation) and 10^{-3} at 60 minutes for purposes of this screening analysis. These human error split fractions are presented in an event tree format in Figure G.1. The conditional probabilities for successful termination of auxiliary feedwater in specific time intervals are indicated in Figure G.1. The estimated uncertainty in the probability of successful operator action is a factor of 2 to 10.

The operator could potentially control the primary system pressure by terminating high pressure injection and venting through the PORV. This is a highly dynamic situation with the pressure recovering rapidly after water has reentered the pressurizer. The generic procedure guidelines have an allowable HPSI termination pressure of 2000 psi. For these screening analyses we have assumed that the pressure will be at 2250 psi.

Operational Status of the Plant at the Time of the Initiating Event

The operational status or power split data developed by Westinghouse considers the decay heat level following each outage and startup (the assumed point of the PTS initiating event) based on operating data. The staff has reviewed the operating history over a period of four years for one Westinghouse plant. The data indicates that the time at power (greater than 50% rated) is on the order of 90% to 95%, as shown in Figure G.2.

For the purpose of this evaluation, the power split selected is 0.9 for hot full power (HFP) and 0.1 for hot zero power (HZP). HFP is equivalent to decay heat levels of 1% or greater, HZP is equivalent to decay heat levels of 0.25% or less. It is noted that Westinghouse uses three levels: 0.75 for HFP, 0.1 for HZP, and 0.15 for intermediate power levels (0.25% to 1% decay heat), based on Westinghouse operating experience. HFP cases, unless otherwise stated, are cases run by Westinghouse using the conservative assumption of being at hot shutdown or hot zero power with decay heat levels of 1% or greater. The larger secondary side mass inventory enhances the cooldown, resulting in a conservatively low temperature calculation.

NRC Staff Characterization of the MSLB

Two cases are used to characterize the MSLB for this evaluation. The first case is for initiation from hot full power and nominal operating conditions. The data was obtained from the Westinghouse Owners Group (WOG) Emergency Response Guidelines E-2, Loss of Secondary Coolant (September 1, 1981). The case selected is for a 0.6 sq ft break. The data have been extrapolated for times greater than 10 minutes. This event is summarized in Table G.1 as a function of operator time to terminate auxiliary feedwater to the faulted steam generator. Also presented are the parameters for initiation from hot zero power. These data are taken from the Westinghouse data for the 0.77 sq ft break case.

WOG Characterization of the MSLB

The data used in the Westinghouse evaluation for the HFP case is conservatively based on the use of initial operating conditions at hot zero power. The data are presented in Table G.2.

G.2.2 Small Steam Line Break (SSLB) or Stuck Open Steam Generator Safety/ Relief Valve (SOSGRV)

The SSLB, or stuck open SG safety/relief valve, can result in an overcooling transient similar to the MSLB but of a longer duration due to the smaller break size. This event is expected to occur at a much higher frequency than the MSLB.

Frequency of Small Steamline Break

The steamline contains several pressure relief devices to accommodate rapid pressure increases caused by plant trips or other perturbations. There are turbine bypass lines that are connected to the steamline downstream of the MSIV and discharge into the condenser. Valves in these lines are designed to open at pressures above operating conditions and provide normal decay heat removal following plant trip. Since these lines can be isolated by the MSIV, we believe that the frequency of a secondary depressurization caused by malfunctioning of these valves is bounded by potential relief valve failures upstream of the MSIVs.

For very rapid steam pressure increases associated with turbine trip or inadequate MSIV closure, the atmospheric dump valves or safety valves upstream of MSIV are likely to be challenged and have a finite probability of sticking open. In Reference G.1, Westinghouse estimated the frequency of a single stuck-open relief valve upstream of the MSIV to be 3.4×10^{-3}/RY and CE estimated the frequency to be 1.5×10^{-2}/RY in Reference G.7. A survey of events relevant to PTS has indicated only two events in about 350 RYs having a small secondary depressurization that resulted in relatively high cooldowns. For purposes of this screening analysis, we will assume a frequency of 10^{-2}/RY for small steamline breaks (equivalent to a stuck-open steam generator safety/ relief valve) with an uncertainty spread of a factor of 3.

We considered additional failures besides the initiating event that might result in extended HPSI flow. Specifically, we looked at the same failures explored for large steamline breaks. The potential impact of RCP seal failures and an unisolated letdown line are judged to be small as discussed under large steamline breaks. The conditional probability of an unisolable open PORV is less than 5×10^{-4} which would yield a combined small steamline break-small LOCA of less than 5×10^{-6}/RY. We do not have a thermal-hydraulic analysis of this event for extended time periods and, therefore, cannot evaluate its contribution. This type of event should be explored as part of the long-term program. Its estimated frequency is sufficiently low that it does not represent an immediate concern.

Operation Actions

The severity of this event can be reduced by terminating auxiliary feedwater to the faulted system generator. For this analysis we will assume the same operator action probabilities as were used in the section addressing main steamline breaks for similar actions.

NRC Staff Characterization of the SSLB

Two cases are used to characterize the small steamline break. The data are taken from the Westinghouse conservative bounding analysis performed at HZP for a 0.22 sq ft break. A decay heat level of 1% is used to represent HFP and a decay heat level of 0% for HZP. The actual cases analyzed were from HZP conditions, therefore, the HFP case is a conservative representation. The transient summary is presented in Table G.3 as a function of operator time to terminate auxiliary feedwater to the faulted steam generator for the initial conditions of HFP and HZP.

WOG Characterization of the SSLB

The data used in the Westinghouse evaluation for these cases appear to be based on a break size greater than 0.22 sq ft. The data are presented in Table G.4.

G.2.3 Small Break Loss-of-Coolant Accident (SBLOCA)

The SBLOCA was considered in two categories. The first category is for break sizes less than 2 inches equivalent diameter. For this category, the safety injection (or makeup) flow exceeds the break flow. For breaks in this range, the break cannot remove all the decay heat generated in the core, and natural circulation, using the steam generators to remove the decay heat not removed by the break, will occur, maintaining loop flow. Within this first category are RCP seals, primary PORV or safety valve leakage or failure as well as actual primary piping leaks. The cooldown rates for these SBLOCAs are not expected to violate Appendix G limits (less than 100°F per hour cooldown). For these reasons this category of small breaks was excluded as not being significant for this evaluation.

Typically a SBLOCA in this range will depressurize the primary system to the pressure corresponding to the saturation pressure of the secondary (steam side). The primary coolant temperature remains just above the secondary coolant temperature in order to maintain a decay heat removal path until sensible heat addition to the incoming SI water is capable of removing this energy. At that time RCS pressure and temperature should decrease slowly. If SI flow at this equilibrium pressure meets or exceeds the break flow, the RCS should remain in natural circulation. Operator actions which reduce the secondary pressure may have a favorable impact on natural circulation, but would increase the cooldown rate.

The second category includes the break size range from 2 to 6 inches equivalent diameter. For these breaks, the safety injection flow is less than the break flow, resulting in a net mass loss of primary coolant. In addition, for breaks greater than 2 inches, the break can remove all of the core decay heat. The combined effect of mass loss and energy removal by the break will result in an extended loss of natural circulation flow in the coolant loops. The continued injection of cold safety injection water into the stagnant loops has the potential to allow relatively unmixed safety injection water to rapidly reach the downcomer and contact the vessel wall. The exact break size where loss of flow occurs is dependent on the safety injection flow rate (and makeup flowrate), the break location, the decay heat level, and the SG (heat sink) performance.

The factors which affect the rate of cooldown following a SBLOCA are:

Break Location
Break Size
SI Flow
Decay Heat
Secondary Pressure (steam dump and feedwater)

Frequency of Small SBLOCA 2 to 6 inch range

The staff has reviewed the Westinghouse thermal hydraulic calculations and has determined that small LOCAs which have equivalent break sizes of 2 to 6 inches will result in stagnated loop conditions and primary system pressures less than 1000 psi. We have examined events that might result in stagnated loop conditions such as LOCAs in the 2 to 6-inch range. There are several small diameter pipes (in the range of 2 to 4 inches) connected to the main primary system piping. These include charging and letdown lines, RTD bypass lines, pressurizer spray and scoopline, PORV line, and safety injection lines. We have also considered the following events.

(a) pipe breaks less than 2 inches with loss of one train of HPSI,
(b) unisolated open PORVs with the loss of one train of HPSI,
(c) failure of all RCP seals due to loss of cooling.

A median frequency of 3×10^{-4}/RY for small LOCAs in the range of 2 to 6 inches is based on WASH-1400, and the uncertainty band is ± a factor of 10. Assuming a lognormal distribution, the estimated mean frequency would be 1.5×10^{-3}/RY. The Westinghouse Owners Group estimated the mean frequency of LOCAs in this range to be 6×10^{-4}/RY.

In evaluating small LOCAs, we have focused on events which stagnate the flow in the pressure vessel and involve a monotonic decrease in primary system pressure. Other stagnated loop flows type events may result in repressurizing the primary system following an extended period of HPSI operation with stagnated flow and low pressure. There are several events that have been considered in this category. These include:

(a) Stuck open safety valve that subsequently closes after the vessel is chilled.

(b) Considerable RCS inventory is lost during the initial surge in an ATWS event. If the reactor coolant pumps are tripped, no primary system leaks occur, and the HPSI operates for an extended period of time, low vessel temperatures may be obtained with high system pressures.

(c) A small-small LOCA initially with only one train of ECCS operating may result in stagnated loop conditions. If the other ECCS trains are subsequently recovered, high pressures may be achieved after the vessel is chilled.

(d) A small-small LOCA with no cooling in one steam generator may lead to high pressure low temperature conditions.

(e) A small steamline break with a subsequently stuck open PORV may lead to extended HPSI flow with a stagnated loop (caused by isolating the faulted steam generator).

(f) Loss of all feedwater that results in losing primary system inventory with a subsequent recovery of feedwater and HPSI operation without loop flow may also lead to these conditions.

Engineering analyses are not available for all of the events postulated above. The most significant is thought to be the stuck open safety valve that subsequently recloses. In Ref. G.4, it is indicated that the Westinghouse Owners Group estimated the frequency of a stuck open safety valve to be $\sim 10^{-5}$/RY assuming the PORV's are blocked 90% of the time. This estimate is based on a conditional failure to close probability of 10^{-3}/D. The review presented in Ref. G.4 reestimated the conditional failure probability to be $\sim 10^{-2}$/D so that the frequency of a stuck open safety valve in Westinghouse plants would be about 10^{-4}/RY. This frequency estimate may be somewhat high; however, considering the other potential events noted above, it is prudent for screening purposes to use a frequency of 10^{-4}/RY for extended HPSI operation with loop stagnation and high pressure.

NRC Staff Characterization of SBLOCA

Three breaks in the hot leg for a typical 3 loop PWR were reviewed. These breaks were of equivalent diameters of 1, 2 and 3 inches. For the 2-inch break, loop natural circulation is lost at about 12 minutes, and for the 3-inch break at about 7 minutes. In addition, three cold leg breaks, 2, 3, and 4 inches were also reviewed. The hot leg breaks are bounding for those cases where circulation is lost due to the higher energy removal rates. Loss of circulation occurs later for an equivalent cold leg break. The loss of circulation is dependent on the break location, the break size, the decay heat level, and operation of the RC pumps (pumps off). For breaks 2 inches and larger, circulation is assumed to be lost. For breaks less than 2 inches, the cooldown rates are less than 100°F per hour and therefore were not included in this evaluation since such slow cooldown does not present a PTS concern.

The most limiting break size for PTS concerns is the 2-inch break. The primary system pressure remains relatively high (1000 psig) for an extended period of time (30 minutes) following loss of loop natural circulation flow. For larger breaks, the pressure will be lower as a result of the faster time to uncover the break and depressurize the system.

The transient characterization based on recent work performed by SLI (Reference G.8) was used to determine the appropriate mixing control volume. This volume is the cold leg piping from the loop seal to the vessel plus one-half of the vessel downcomer volume. The wall heat from the piping, the vessel, and the core support barrel was included in the analysis. The results of this analysis show a leveling of the coolant temperature at about 125°F (assuming a safety injection temperature of 60°F) for the time range of interest, which is 2000 to 3000 seconds after loss of flow is assumed.

On August 11, 1982, the staff solicited advice from leading thermal-hydraulic experts to determine how mixing during no-loop flow conditions could best be treated for this evaluation. In particular, we were interested in determining what was the most realistic acceptable method for predicting mixing under no-flow conditions. There was a consensus at the meeting that the method presented by Leviy in Reference G.8 was appropriate; however, stored energy in the vessel metal, the lower plenum volume, and the pipe volume upstream of the safety injection location to the pump should also be accounted for. In addition these volumes should be included in the mixing volume.

Westinghouse presented its evaluation at this meeting. There was general agreement on the methodology used by Westinghouse. The Office of Nuclear Regulatory Research (RES) requested three of their experts to perform independent analyses of this event, to confirm the Westinghouse results.

The RES analyses have been reviewed by the staff and the results of these analyses (Reference G.9 and G.10) have been used to characterize the SBLOCA event. The models used have been compared to recent Creare experiments, for a 1/5 scale geometry (Reference G.11). These comparisons indicate that the models are a reasonable representation of the mixing phenomena under no-loop flow condition. The characterization used for this evaluation for this event sequence is presented in Table G.5.

WOG Characterization of SBLOCA

For breaks in the cold leg, the Westinghouse evaluation uses a 1-inch break, which bounds the RC pump seal failure case and assumes no operator action. For decay heat levels of 1%, the T_f is 315°F and β is 0.007 per minute. At 0% decay heat, T_f is 105°F and β is 0.007 per minute. For breaks in the hot leg, Westinghouse uses a 0.75-inch break, slightly less than the area of a typical PORV (1.4 inches). For decay heat levels of 1%, T_f is 528°F and β is 0.004 per minute. At 0% decay heat, T_f is 166°F, and β is 0.004 per minute. Westinghouse excluded larger breaks from the PRA evaluation. For these breaks, crack extension is accepted. However, as a result of the decreasing pressure, the Westinghouse fracture mechanics analyses, which includes warm prestress, has shown that crack arrest occurs for these cases.

G.2.4 Steam Generator Tube Rupture (SGTR)

The SGTR transient is governed by operator actions prescribed in current plant emergency procedures. The tripping of the RC pumps, the use of the PORV during a SGTR event, and the subsequent cooldown and depressurization to isolate the break all impact on the transient. When the RC pumps are tripped, natural circulation in the faulted loop can be lost for a short period of time, since the recommended action is to isolate the secondary side of the faulted steam generator, thus eliminating its heat removal capability, prior to the recovery procedures. Delayed safety injection termination can enhance the cooldown and increase the consequences of the event. In general, the SGTR event will not result in loss of vessel integrity when current plant procedures are observed, as the downcomer temperature is expected to remain greater than 325°F. The SGTR event does become a potential initiator for PTS if additional failures are included in the event sequence. Two of these are of definite interest. They are failure by the operator to terminate safety injection in accordance with the prescribed procedures and a stuck-open steam generator safety valve.

Frequency of SGTR

The estimated frequency of a steam generator tube rupture event is 2×10^{-2}/RY (4 events in about 250 RY for Westinghouse plants) which by itself is not a particular PTS concern if the HPSI is terminated in a timely fashion (~30 minutes). The Ginna-type event (1982), which resulted in extended HPSI operation, may lead to PTS conditions of concern. Since this is one event out of four, the estimated frequency of SGTR events that lead to extended HPSI operation (1 hour) is 5×10^{-3}/RY unless subsequent guidelines are effective in reducing the frequency of this event.

We have considered other failures in addition to the SGTR which may result in extended HPSI operation. In the Ginna event, the operator isolated the atmospheric dump valve on the faulted steam generator and challenged the steam generator safety valve. Assuming a conditional probability of 10^{-2}/D for failure of the safety valve to close (WASH-1400), there is a potential event of an SGTR with a stuck-open SG safety with an estimated frequency of

2×10^{-4}/RY. The frequency of this scenario can be reduced if the HPSI is terminated in a timely fashion (~30 minutes) and/or the atmospheric dump valves are not isolated.

Another potential compound event is extended HPSI operation which results in flooding the steamline and it subsequently fails. The staff has estimated the probability of failure of water-filled steamlines to be negligible. If we assume a conditional failure probability of 10^{-3} for water-filled steamline failure, then the estimated frequency for a combined event of SGTR followed by a steamline break is 10^{-5}/RY (there have been two SGTR events with water in the steamlines).

The PORV may be used to depressurize the primary system. The conditional probability of having an unisolated open PORV is 5×10^{-4} as discussed previously. The frequency of this combined event is estimated to be 10^{-5}/RY. We do not have thermal-hydraulic calculations to describe this event for an extended time period. We would judge that it may be bounded by a small LOCA event which has an estimated frequency of 3×10^{-4}/RY.

We also considered the potential for developing excessive steam generator tube leaks following steamline breaks. We had insufficient information to evaluate this event in the short term; however, they should be pursued in the longer term study of pressurized thermal shock.

In summary, we will use the following frequencies for SGTR events for this screening process.

(a) simple SGTR with HPSI termination in 1 hour	5×10^{-3}/RY
(b) SGTR with stuck-open secondary safety valve	2×10^{-4}/RY
(c) SGTR with subsequent steamline break	1×10^{-5}/RY

There are considerable uncertainties in these event frequencies which may be as much as a factor of 10.

NRC Staff Characterization of SGTR

The base case SGTR is similar to the one used by Westinghouse T_f = 325°F, β = 0.12 per minute, at 1000 psig. This case assumed operator action as prescribed in current plant procedures. Delayed SI termination will result in lower final temperatures. Based on the Westinghouse data, the characterization is as follows, as a function of operation time to terminate safety injection flow.

The data for the SGTR with a stuck-open safety valve is taken from the Westinghouse study case for the SGTR with a small break inside containment (SLB). The data for the case with a steamline break is taken from the Westinghouse study case for an SGTR with a small break outside containment (SOSV). The transient characteristics are presented in Table G.6.

WOG Characterization of SGTR

The Westinghouse data is used with the exception of the base case β value used by Westinghouse of 0.09 per minute. Based on a review of operating experiences, a value of 0.12 per minute appears to be more appropriate and is used for this evaluation.

G.2.5 Excess Feedwater

The contribution to PTS from an excess feedwater transient is considered to be negligible due to the low frequency of occurrence for Westinghouse and Combustion Engineering plants and is not considered in this evaluation. Because of the difference in steam generator water level and inventory and main feedwater design for Babcock and Wilcox plants, the excess feedwater transient should be considered as a potential PTS initiating event for B and W plants.

G.2.6 Other Events

There are numerous other events, or variations (multiple failures) of the above described events, which may also warrant consideration in a PTS evaluation of this nature. However, due to the scoping nature of this evaluation, combined with the time constraints to produce this preliminary evaluation, all of these additional events could not be systematically identified and evaluated. These other events include situations where loop flow is lost, as a result of loss of the heat sink (SG). For example, Westinghouse has indicated that secondary side depressurization (MSLB and SSLB) might result in loss of natural circulation some time after 30 minutes, dependent upon the transient specifics such as break size, delayed operator action, and decay heat level. Once loop flow is lost, the cooldown will be accelerated due to the nonmixing of the safety injection flow. In addition to cases where loop flow can be lost, there are many other events which should be considered as part of future, detailed evaluations. These are generally scenarios which include multiple failures (for example, MSLB and subsequent SGTR; multiple RC pump seal failure due to loss of component cooling water; and feed and bleed operations). A detailed PTS event tree analysis would be required to determine those events which could result in PTS conditions. Such an evaluation would have to consider plant-specific information, such as safety injection pump shut-off head, safety injection and auxiliary feedwater flow rates and temperatures, and specific control and protection system set-points. This evaluation would have to be performed as part of a long-term PTS effort.

The above evaluations gave no credit for the operator reducing primary system pressure to minimize the potential for PTS and also assumed that the operator would not deliberately put the plant into a PTS condition after it has been stabilized following some event.

There have been events such as the TMI-2 accident in which the plant was repressurized even though low fluid temperatures were present. We cannot adequately evaluate the potential for the operator repressurizing the primary system because of some unforseen situation or dilemma. However, the potential impact of such events can be considered by combining operational experience with postulated events.

G.2.7 Limitations

The selection of events noted above to be analyzed for establishing a screening criterion were based on the dominant sequences identified by the Westinghouse Owners Groups in the June 22, 1982 meeting. In addition, the staff has considered some limited variations of these events in an attempt to identify additional failure that may lead to significant PTS conditions. Transients with significant reliance on ad-hoc operator action will be based on operating experience. These events were based on a generic 3-loop Westinghouse plant, so plant-to-plant variations are not considered. The set of sequences considered above is not complete, but was meant to provide some short-term perspective for a screening criterion for Westinghouse plants in anticipation of more detailed long-term studies that are being pursued by NRC and the industry.

In developing the sequence frequencies, we have assumed that the reactor coolant pump seals are cooled following all the above events, the letdown line is isolated, the feedwater system has three isolation devices in series that are activated on the different signals, the allowable primary to secondary leakage and the operator will depressurize for small LOCAs (<2"). These assumptions should be confirmed as part of the longer term program. In addition, we have omitted some compound sequences because of insufficient information to perform an adequate evaluation in the short-term. This evaluation does not include low temperature overpressurization for which pressure relief devices are available.

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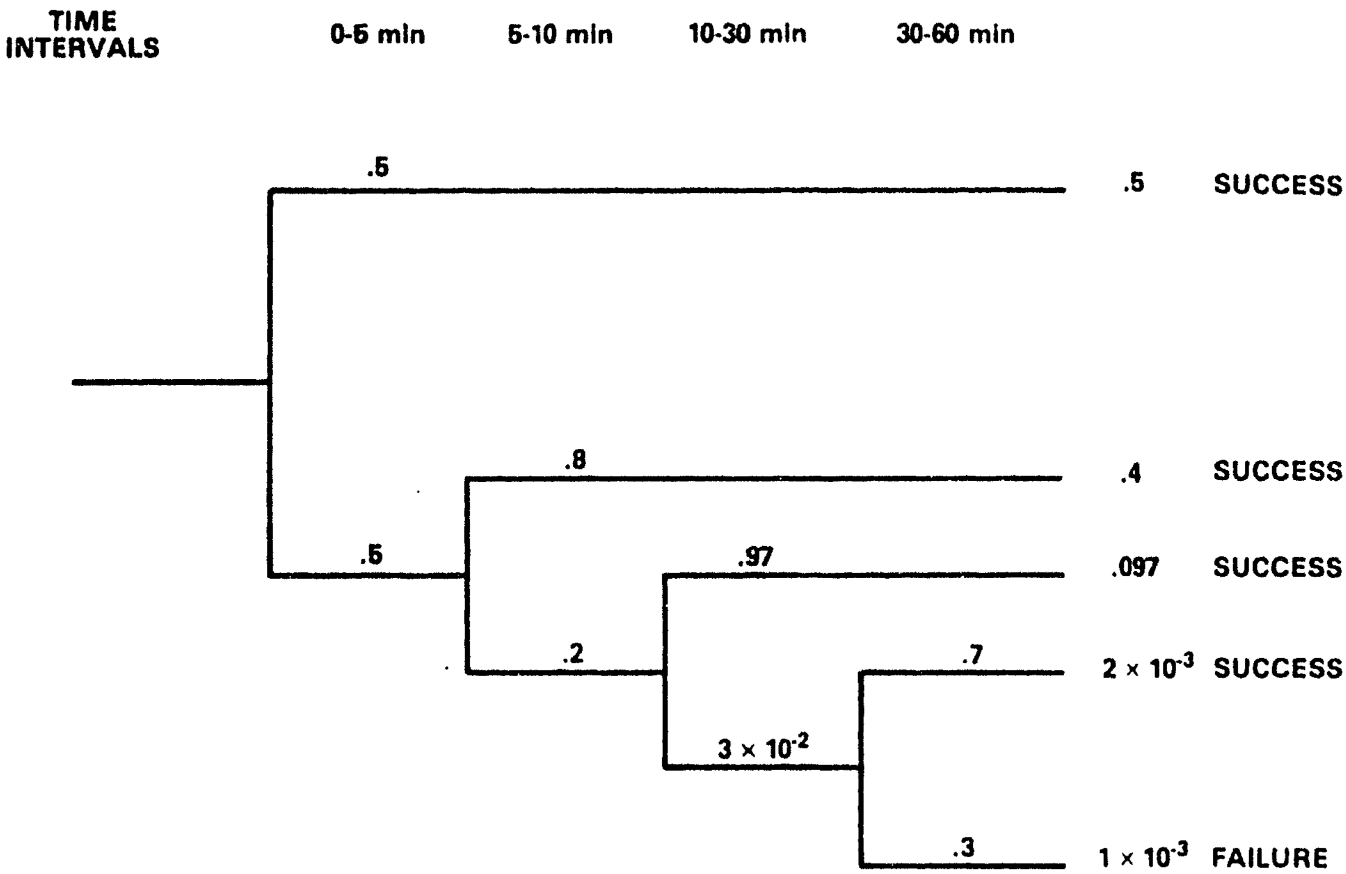


FIGURE G-1: HUMAN ERROR SPLIT FRACTIONS FOR AUXILIARY FEEDWATER TERMINATION

FIGURE G-2

PLANT POWER HISTORY EVALUATION

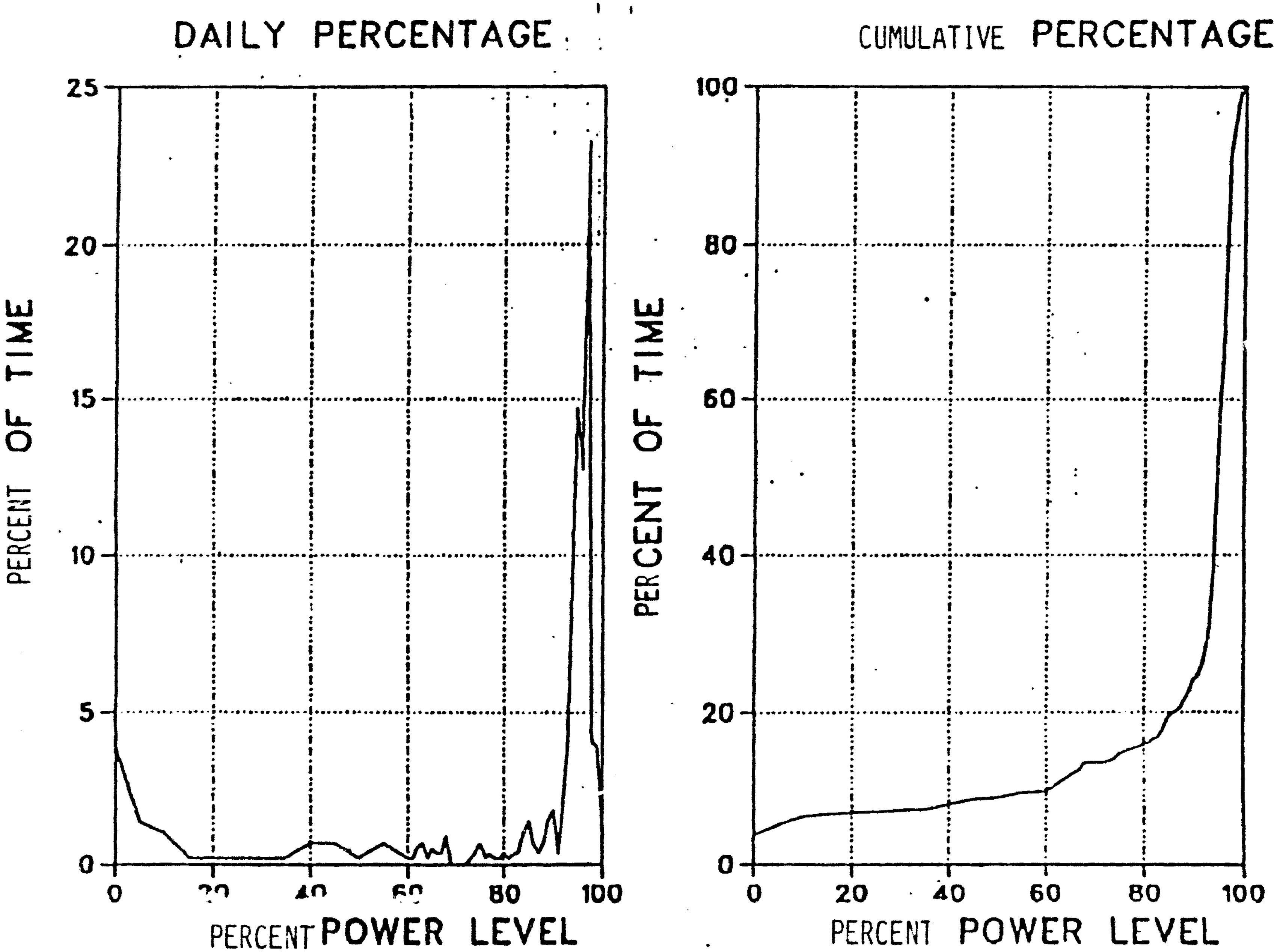


Table G.1 NRC staff event parameters for the main steam line break (MSLB)

Plant Status	Hot Full Power					Hot Zero Power				
Operator Isolates Auxiliary Feedwater min.	5	10	20	30	60	5	10	20	30	60
Event Frequency per Reactor-Year	8×10^{-5}	6×10^{-5}		1.5×10^{-5}	$3. \times 10^{-7}$	8.5×10^{-6}	6.8×10^{-6}		1.6×10^{-8}	$3. \times 10^{-8}$
β min^{-1}	0.4	0.2	0.1	0.09	0.09	0.4	0.2	0.2	0.2	0.2
Final Reactor Coolant System Temperature at Vessel Wall, °F	450	390	350	300	250	212	212	212	210	190
Final Pressure (psig)	2250	2250	2250	2250	2250	2250	2250	2250	2250	2250

Table G.2 Westinghouse Owners Group (WOG) event parameters for the main steam line break (MSLB)

Plant Status	Hot Zero Power				
Operator Isolates Auxiliary Feedwater min.	5	10	20	30	60
Event Frequency per Reactor-Year	2×10^{-8}	7×10^{-8}		7×10^{-10}	$4.\times10^{-11}$
β min^{-1}	0.4	0.2	0.2	0.2	0.2
Final Reactor Coolant System Temperature at Vessel Wall, °F	220	220	212	210	200

Table G.3 NRC staff event parameters for the small steam line break (SSLB)

Plant Status	Hot Full Power					Hot Zero Power				
Operator Isolates Auxiliary Feed-water, Min.	5	10	20	30	60	5	10	20	30	60
Event Frequency per Reactor-Year	4.5×10^{-3}	3.6×10^{-3}	8×10^{-4}	6.3×10^{-5}	1.8×10^{-5}	4.5×10^{-4}	3.6×10^{-4}	8×10^{-5}	6.3×10^{-6}	1.8×10^{-6}
β min^{-1} 0.4	0.2	0.1	0.06	0.06	0.4	0.2	0.1	0.06	0.06	
Final Reactor Coolant System Temperature at Vessel Wall, °F	385	320	250	220	200	375	310	235	200	175

Table G.4 Westinghouse Owner's Group (WOG) event parameters for the small steam line break (SSLB)

Plant Status	Hot Full Power					Hot Zero Power				
Operator Isolates Auxiliary Feed-water, Min.	5	10	20	30	60	5	10	20	30	60
Event Frequency per Reactor-Year	$2.\times10^{-3}$	$9.\times10^{-4}$	9×10^{-5}		4×10^{-6}	5×10^{-5}	2×10^{-5}	2×10^{-6}	2×10^{-6}	1×10^{-7}
β min^{-1} 0.4	0.2	0.1	0.06	0.06	0.4	0.2	0.1	0.06	0.06	
Final Reactor Coolant System Temperature at Vessel Wall, °F	320	290	260	250	220	230	230	230	220	200

Table G.5 NRC staff event parameters for the small-break loss-of-coolant accident (SBLOCA), and extended HPSI operation with a stagnant loop

	SBLOCA	Extended HPSI Operation
Event Frequency Per Reactor-Year	1.5×10^{-3}	10^{-4}
β, min^{-1}	0.05	0.05
Final Reactor Coolant System Temperature at Vessel Wall, °F	125	125
Final Pressure, psig	1000	2250

Table G.6 NRC staff event parameters for steam generator tube rupture

With delayed safety injection termination by the operator delay in terminating safety injection, minutes	20	30	60
Event Frequency per Reactor-Year	5×10^{-3} (total for all delays)		
β, min^{-1}	0.1	0.07	0.04
Final Reactor Coolant System Temperature At Vessel Wall, °F	290	260	200
Final Pressure, psig	1000	1000	1000

With Steam Line Break or Stuck-Open SRV	Stuck open SRV Outside Containment	SLB Inside Containment
Event Frequency Per Reactor-Year	2×10^{-4}	1×10^{-5}
β $min.^{-1}$	0.04	0.04
Final Reactor Coolant System Temperature at Vessel Wall, °F	170	170

APPENDIX H

REACTOR PRESSURE VESSEL FAILURE PROBABILITY STUDY

Reactor pressure vessels (RPV) in nuclear power plants have traditionally been considered extremely reliable structural components. Indeed, studies completed in the United States and Europe have concluded that the disruptive failure rate (loss of the pressure retaining boundary) for nuclear pressure vessels is less than 10^{-6} at a 99% confidence level for RPVs designed, fabricated, inspected, and operated in accordance with the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers. However, recent results from surveillance and research programs and operating experience suggest that the issue of RPV failure probability should be reassessed. The renewed interest in RPV failure probability is due to the observation that thermal hydraulic transients occurring in commercially operating nuclear power plants are subjecting RPVs to unanticipated loadings which could contribute significantly to the failure probability of RPVs. In addition, operating experience and research programs over the past few years have provided additional information that more clearly defines both material property variations in RPVs and the effect of neutron irradiation on the material's resistance to fracture. The objective of this study is to assess the contribution to RPV failure probability of recently observed thermal hyraulic transients using recent material property data. (Note: The material properties formula and the model for ΔRT_{NDT} attenuation in the vessel wall differ from those specified in Appendix E. Future work will use consistent models.)

Generally, RPV reliability studies have used either one of two methods to calculate the probability of RPV failure. These methods are (1) the analysis of statistical data from observed non-nuclear pressure vessel failures to infer failure rates for both nuclear and non-nuclear pressure vessels and (2) the use of mathematical models that predict failure rates by analytically generating pressure vessel failures. Mathematical models used in the later technique have been primarily closed form analyses. In this effort, Monte Carlo simulation techniques have been used because of the ability to consider a greater number of significant random variables and to perform a wide range of sensitivity studies. The results of extensive sensitivity studies which have been conducted are extremely important because they quantify the effect of uncertainties in the input parameters, thereby providing an estimate of the accuracy of the calculated failure probabilities, and they identify the significant variables and variable interactions. The results are best applied in a relative sense for use in decision making, and extreme caution must be exercised in applying the results in an absolute sense.

Section H.1 of this report describes the reactor pressure vessel considered in this study, Section H.2 describes the fracture mechanics techniques and simulation model used to calculate RPV failure probabilities; Section H.3 presents results of a reference case and sensitivity analysis performed using the simulation code; and Section H.4 presents a discussion and conclusions of the study.

H.1 Reactor Pressure Vessel Description

The reactor vessel geometry in this study has a 9-inch wall thickness and a 90-inch mean radius. Figure H-1 presents a schematic of how the RPV is fabricated. The failure probability is calculated for one vertical weld in the two beltline shell courses, which have lengths of approximately 72 inches. These dimensions are typical of most operating PWR vessels. Only the welds are considered because they have the greatest propensity for flaws, are most sensitive to radiation damage, and hence, should dominate the failure probability. The reactor vessel is fabricated of carbon steel with stainless steel cladding on the internal surfaces that are in contact with the primary coolant.

H.2 Probabilistic Model

H.2.1 Fracture Mechanics Algorithms

Pressurized thermal shock transients can subject the reactor pressure vessel to an unusual combination of high thermal and pressure stresses that create the potential for fracture of the reactor pressure vessel. Given well defined pressure and temperature-time histories for a pressurized thermal shock transient, heat transfer and stress analyses can be conducted using either closed form or numerical analysis techniques.

In this study closed form solutions have been utilized for the heat transfer and stress analyses. The closed form solutions allow the primary coolant temperature time history to be expressed as either a fourth order polynomial or an exponential function of the form:

$$T = T_f + (T_o - T_f)e^{-\beta t} \qquad \text{(H-1)}$$

where T is the temperature of the primary coolant as a function of time; T_o and T_f are the initial and final primary coolant temperatures, respectively; β is the decay constant that determines the rate of cooldown; and t is time. The pressure time history is represented by a fourth order polynomial. The heat transfer analysis is performed using an effective heat transfer coefficient which takes into account the fluid film heat transfer coefficient and the thermal resistance of the stainless steel cladding. However, the stresses due to the difference in thermal expansion between the stainless steel cladding and the base metal have not yet been included in the probabilistic code. A sensitivity study in Section H.3.2.9 provides an indication of how these stresses might affect the calculated failure probabilities.

The temperature and stress intensity values calculated using the above techniques were found to be in excellent agreement with the temperatures and stress intensity values calculated by the OCA-I code developed at ORNL.

Once the transient temperature and stress states have been calculated for the pressurized thermal shock event, linear-elastic fracture mechanics analysis is used to evaluate RPV integrity. Linear-elastic fracture mechanics (LEFM) is used to determine if a pre-existing flaw will propagate unstably through a material under certain loading and material conditions. The LEFM criteria for unstable fracture is:

$$K_I > K_{Ic} \qquad \text{(H-2)}$$

where K_I is the applied stress intensity factor and K_{Ic} is the critical stress intensity factor. Warm prestressing which can effectively inhibit crack extension even when K_I exceeds K_{Ic} (see Section D.3) was not considered in the analyses with the exception of the sensitivity studies presented in Section H.3.5. Although for many of the transients analyzed, warm prestressing would be effective, these transients were only assumed for convenience in conducting parametric studies. System considerations and operator actions do not ensure that warm prestressing will be effective in every case.

The applied stress intensity factor, K_I, is a function of the stress state; crack depth, a; and flaw and component geometry. The stress state at any time in a pressurized thermal shock transient is defined by the pressure and temperature-time histories. The component geometry of interest in this study is the RPV beltline with an assumed longitudinally oriented flaw. The assumed longitudinal orientation is that expected in longitudinally oriented welds and is the flaw orientation that experiences the maximum stress and K_I in the reactor vessel beltline. Deterministic analyses assume that a flaw of a specific depth exists with certainty. In the probabilistic model developed in this study, the crack depth is treated as a random variable.

The critical stress intensity factor, K_{Ic}, is the material's resistance to unstable fracture. K_{Ic} is a function of the temperature at the crack tip; the material's initial nil-ductility reference temperature, RT_{NDTo}; and the shift in RT_{NDT}, ΔRT_{NDT}. The temperature at any depth in the vessel wall is defined by the heat transfer analysis of the pressurized thermal shock transient.

RT_{NDTo} is a material property determined by a destructive material testing procedure and is a measure of the temperature at which the material begins a transition from a "brittle" to ductile fracture mode. Determination of RT_{NDTo} is subject to material variability and measurement errors. Furthermore, estimates of the RT_{NDTo} for a specific plant often must be made from a generic data base not totally representative of the specific material of interest. Therefore RT_{NDTo} is treated as a random variable in the probabilistic model.

The shift in RT_{NDT} is a result of neutron irradiation. As the vessel beltline fluence increases, the RT_{NDT} of the material becomes higher. This means that in order to exhibit the same resistance to fracture, K_{Ic}, the material must be at a higher temperature. The attenuation of fluence through the RPV wall for the results presented in this study was represented by the following relation

$$F(a) = F_{ID}\, e^{-.33a} \qquad \text{(H-3)}$$

where a is the depth in inches into the vessel wall and F_{ID} is the fluence (> 1 MEV) in neutrons/cm^2 at the surface of the RPV wall. More recent studies based on the concept of displacement per atom, dpa, consider a wider spectrum of neutron energies and suggest that the exponential decay constant should be smaller to more accurately predict radiation damage through the RPV wall. Fluence on the inside surface of the RPV wall varies with location in the RPV beltline due to the core design and power profile. In addition, there are relatively large uncertainties in calculating fluences. Thus, fluence has been considered a random variable in this study.

In the probabilistic analyses, the mean shift in RT_{NDT} has been represented by the following function:

$$\Delta RT_{NDT} = [-4.83 + 476 \cdot Cu + 267 \cdot Cu \cdot Ni]\,[F/10^{19}]^{0.218} \qquad \text{(H-4)}$$

where ΔRT_{NDT} is the mean shift in RT_{NDT}, Cu is the copper content in weight percent, Ni is the nickel content in weight percent, and F is the fluence in neutrons (> 1 MEV)/cm^2. This equation was developed at HEDL through regression analysis of surveillance and research program results. Copper and nickel contents vary throughout the RPV material, and uncertainties exist with the values specified for plant specific welds. Hence copper and nickel contents should be treated as random variables. Copper content was treated as a random variable in this study. However, the effect of nickel has just recently been recognized; and hence, nickel was not considered as a random variable in the original development of the code. Future versions of the code will include nickel as a random variable. The results presented here were generated assuming a constant nickel content of 0.65%. The equation used in Appendix E and in Section 5 of the main report for calculating the mean shift in RT_{NDT} is based on a more recent regression analysis performed by HEDL. In order to maintain consistency between sensitivity studies, the earlier form of the equation (equ. H-4) was used throughout the probabilistic analyses. This equation predicts a lower shift in RT_{NDT} than the equation used in Section 5.

The surveillance and research program data on ΔRT_{NDT} as a function of fluence exhibit significant variability as illustrated in Figure H-2. However, it is believed that much of the variability is due to variability and uncertainty in the measured fluences and copper contents in the data base. Therefore, it seems inappropriate to consider this variability twice and for the results presented in this study the mean trendline for ΔRT_{NDT} versus fluence specified in equation (H-4) was used. A proposed sensitivity study to be conducted in the future is to compare the results of this study with results generated by using mean copper contents and fluences and treating ΔRT_{NDT} as a random variable. However, for this study it was desirable to be able to conduct sensitivity studies on copper content and fluence; hence, these parameters were treated as random variables.

Once the initial RT_{NDT} and shift in RT_{NDT} have been specified either deterministically or probabilistically, the critical stress intensity factor, K_{Ic}, can be calculated. Figure H-3 shows a plot of K_{Ic} data versus $T-RT_{NDT}$, where T is the temperature of the material and RT_{NDT} is the sum of the initial RT_{NDT} and the shift in RT_{NDT}. Because K_{Ic} is a material property, it exhibits some variability and is treated as a random variable. A mean curve for K_{Ic} versus $T-RT_{NDT}$ was developed through regression analysis. The equation for this mean curve is:

$$\bar{K}_{Ic} = 36.2 + 49.4\exp(0.0104(T-RT_{NDT})) \quad \text{for } T-RT_{NDT} \leq -50°F \qquad \text{(H.5a)}$$

$$\bar{K}_{Ic} = 55.1 + 28.0\exp(0.0214(T-RT_{NDT})) \quad \text{for } T-RT_{NDT} > -50°F \qquad \text{(H-5b)}$$

If crack initiation is predicted, the crack may arrest as it runs deeper into the wall encountering hotter, less irradiated, and hence, tougher material. Arrest of the crack is predicted if

$$K_I < K_{Ia} \qquad \text{(H-6)}$$

where K_{Ia} is the stress intensity factor for crack arrest. Figure 5-4 shows the data for K_{Ia} versus $T\text{-}RT_{NDT}$ and a mean curve fit using regression analysis. The equation for the mean curve is:

$$K_{Ia} = 19.9 + 43.9\exp(.00993(T\text{-}RT_{NDT}) \text{ for } T\text{-}RT_{NDT} < 50°F \quad \text{(H-7a)}$$

$$K_{Ia} = 70.1 = 6.5\exp\ (.0196(T\text{-}RT_{NDT}) \text{ for } T\text{-}RT_{NDT} > 50°F. \quad \text{(H-7b)}$$

Both the mean crack initiation and crack interest toughness were truncated at an upper shelf value of 200 KSI $\sqrt{in}$. Thus if crack arrest is not predicted before K_I reaches a value of 200 KSI $\sqrt{in}$. vessel failure is predicted.

H.2.2 Simulation Model

Figure H-5 illustrates the simulation model developed for RPV failure probability. The left hand column in the figure is the deterministic analysis which includes the heat transfer, thermal and pressure stress, and applied stress intensity value calculations for a range of crack depths at ten time steps in the transient. Matrices of temperature and K_I values are stored for use later in the simulation analysis.

The variables designated "simulate" in the diagram are treated as random variables, and their values are sampled from a statistical distribution defined by input parameters. As discussed in the previous section, crack depth, a; fluence; RT_{NDTo}; copper content; K_{Ic}; and K_{Ia} were treated as random variables in this study. A value for each of these random variables is sampled from the appropriate statistical distribution. Once the flaw size is simulated, the corresponding K_I value is retrieved from the K_I matrix developed earlier in the code. The mean K_{Ic} value is calculated according to the equation (H-5) using the temperature corresponding to the time step and simulated crack depth and an RT_{NDT} based on the values of copper content, fluence, and RT_{NDTo} sampled from their corresponding statistical distributions. Since the K_{Ic} data exhibits significant variability, the K_{Ic} value is simulated by sampling from a distribution about the mean K_{Ic} value.

If crack initiation is predicted, the crack is allowed to advance through the RPV wall in discrete steps of 0.25 inches, and a check for crack arrest is made at each crack advance. K_{Ia} is treated in a similar fashion to K_{Ic} as mentioned above. If crack arrest is predicted, the code continues to analyze successive time steps in the transient using the arrested crack depth. Since the applied K values and material temperature at the crack tip are a function of time in the transient, reinitiation of the crack may occur.

Each pass through the simulation loop depicted in Figure H-5 represents a single computer experiment conducted to determine if RPV failure will occur. Up to a million passes through this loop can be made. The code keeps track of the number of crack initiations and RPV failures and the probabilities of crack initiation and RPV failure are estimated by dividing these values by the total number of trials. Thus the code actually performs millions of deterministic calculations with each set of calculations based on a different set of values selected from the appropriate statistical distributions for the significant variables. This is equivalent to subjecting a population of up to a million operating reactor pressure vessels to the pressurized thermal shock transient of interest and then inferring the failure probability based on the number of observed failures.

H.2.3 Statistical Distributions of Random Variables

The simulation model described above suffers from the same problem as all analytic models, its output is only as good as its input. Unfortunately, very little information exists in the literature regarding the required statistical inputs, and the time frame of this initial study was not sufficient to allow the necessary research and analysis to develop rigorous statistical inputs. Therefore, m ny of the statistical distributions associated with the random variables in th model are based on expert opinion and have somewhat ill-defined "levels of on-fidence." It is appropriate to interject at this point that, because of th uncertainties associated with the input parameters, the best use of the res lts of this study is in a relative sense to assist in the decision-making proce s.

The number and size of cracks in the weld material of the RPV is probably e random variable with the greatest uncertainty. Several crack size distributions exist in the literature. These distributions are based on the experience of RPV fabricators and nondestructive examinations. The flaw distribution is of course difficult to quantify since the flaws of interest are not the flaws that have been detected, but those of unknown size and number that remain in the RPV because they were not detected. Figure H-6 shows the probability of having a flaw of depth a in a reactor pressure vessel longitudinal beltline weld as estimated in the OCTAVIA computer code. The weld volume associated with the OCTAVIA flaw distribution was defined as the volume of longitudinal weld material in the beltline region of a PWR. To obtain the flaw distribution, for a single beltline weld, as considered in this study, the OCTAVIA flaw distribution was adjusted assuming that the flaws were equally distributed among six longitudinal beltline welds. For illustration, the crack depth, a, in Figure H-6 is represented as a continuous random variable. However, in this study, the crack depth was used as a discrete random variable. For the curve in Figure H-6, approximately nine distinct crack depths ranging from 0.125 to 3.5 in. were used and the probabilities indicated at these crack depths were reduced by a factor of 1/6 to represent the probability of a flaw in one weld and were used to construct a stepwise cumulative probability distribution. The Monte Carlo simulation in the computer code used the stepwise cumulative distributions to generate a crack depth for each simulation cycle.

The distribution of RT_{NDTo} is dependent on the variability in the material and measurement error. In discussions with the metallurgists at materials testing laboratories, they indicated that they believed their accuracy in determining RT_{NDTo} was $\pm 20°F$. No data exist from which to infer the shape of the distribution. Therefore, for a reference case, a normal distribution with a standard deviation of 15°F was assumed. Sensitivity studies were conducted assuming that the standard deviation was 30°F.

The variance in fluence is due to the power distribution in the reactor core and inaccuracies in calculation. Experts at Hanford Engineering and Development Laboratory in Richland, Washington, have estimated the uncertainty in fluence estimates to be on the order of $\pm$ 30% (1σ) using common practice techniques. For the reference case, a normal distribution with a standard deviation of 30% was assumed. Sensitivity studies were conducted assuming standard deviations of 50% and 15%.

A study was conducted to evaluate the sensitivity of the calculated failure probabilities to the tails associated with the normal distributions assumed for RT_{NDTo} and fluence. In this study the distributions were truncated at the mean plus and minus three standard deviations. The results indicated no appreciable difference, and it was concluded that the tails of the assumed normal distributions do not dominate in the calculations.

Copper was introduced into the welds of the RPV from welding rods that were copper coated to improve the welding process. Chemical composition analyses of welds from RPV prolongations have recently provided extensive data for welds representative of those in operating plants. Rigorous statistical analysis of these data is not yet complete. However, the distribution does appear to be symmetrical with a standard deviation in the range of .02% to 0.5%. For the reference case, a normal distribution with a standard deviation of 0.025% was assumed. In the sensitivity studies a 0.07% standard deviation was considered. In all analyses the range of simulated copper content values was limited to 0.08% to 0.40% copper.

As described in Section H.2.1, K_{Ic} and K_{Ia} and also treated as random variables with a normal distribution and a 10% standard deviation about their respective means curves. Due to lack of sufficient data, the distribution of K_{Ic} and K_{Ia} about their mean is difficult to rigorously determine. However, several papers have suggested using a normal distribution about the mean with a standard deviation of 10%, and this distribution was assumed in generating the results presented here. The normal distribution about the mean was applied to both the transition and upper shelf toughness regions. Sensitivity studies were conducted to evaluate the sensitivity of the calculated failure probabilities to the assumed variability in K_{Ic} and K_{Ia}.

H.3 Results

This section presents results of a reference case and of certain sensitivity studies performed using the simulation model described in Section H.2. As stated earlier, due to uncertainties in the input data, it is suggested that the results be considered in a relative rather than an absolute sense. The sensitivity studies performed identify important parameters and their interaction and suggest how sensitive the reference case failure probabilities are to uncertainties in the input data. The results presented are conditional probabilities; that is, the probability of failure of a RPV weld given that the pressurized thermal shock transient under consideration occurs. To convert the results presented here into failure rates, the frequency of occurrence of the transient considered must be defined. Since the results presented are for an individual weld in the RPV beltline, the total conditional failure probability of the RPV beltline welds is the appropriate summation of the failure probabilities for each weld. If these values are sufficiently low and independence is assumed, the failure probabilities for the six welds can simply be summed. If the failure probabilities become high, the intersection of the weld failure probabilities must be subtracted.

H.3.1 Reference Case

The reference case analysis is defined as follows:

- The Rancho Seco transient (Figure H-7)
- The OCTAVIA flaw distribution,
- Copper $\sim N(\mu, 0.025\%)$,
- $RT_{NDTo} \sim N(\mu, 15°F)$,
- FLUENCE $\sim N(\mu, 30\%)$,
- ΔRT_{NDT} - HEDL mean curve, and
- K_{Ic} and K_{Ia} treated as random variables.

Figures H-8 through H-12 present the conditional failure probabilities calculated for the reference case condition. Each figure presents the failure probability versus the mean fluence for a specified mean copper content and three mean values of RT_{NDTo}. Also, plotted across the top of each figure, is the ΔRT_{NDT} calculated using the mean HEDL curve. These shifts are based on the mean copper content and fluence value in each figure. These curves make it possible to estimate the failure probability for the beltline region of a PWR for which the mean values of the random variables can be estimated.

Several important observations can be made regarding Figures H-8 through H-12. The first observation is that no failure probabilities less than 10^{-5} are calculated for any combination of mean fluence copper content, or RT_{NDTo}. This result occurs because the Rancho Seco transient will result in an applied K_I value greater than the assumed mean upper shelf toughness of 200 $KSI\sqrt{in.}$ for flaws of 3.0 inches or greater depth, and the probability of such flaws existing is nearly 10^{-5} in the flaw distribution assumed. Therefore, the lower limit on calculated failure probabilities would change for different transients, flaw distributions, or assumptions about the upper shelf toughness.

The second important observation is that any specified value of failure probability corresponds within a few degrees to a specific mean value of RT_{NDT}, independent of the copper content and fluence by which the RT_{NDT} value was achieved. For example in Figure H-8, based on a copper content of 0.34%, a failure probability of 2×10^{-5} corresponds to a mean RT_{NDT} value of approximately 255°F to 260°F for the three values of RT_{NDTo}. Similarly, in Figure H-11, based on a mean copper content of 0.28%, a failure probability of 2×10^{-5} corresponds to a mean RT_{NDT} of approximately 255°F for the two values of RT_{NDTo}. These results demonstrate that RT_{NDT} is in fact an excellent criterion for evaluating reactor pressure vessel integrity under specified thermal shock conditions. The mean RT_{NDT} value corresponding to a specific failure probability will, of course, be different for different pressurized thermal shock transients.

H.3.2 Reference Case Sensitivity Studies

Sensitivity studies were conducted on the distribution for copper content, initial RT_{NDT}, fluence, and fracture toughness. In addition, conditional failure probabilities were calculated assuming that specific flaw sizes exist with a probability of 1.0. Finally, a sensitivity study was conducted for a set of hypothetical transients with assumed expontial temperature decays and constant pressures. These cases are intended to provide insight into how sensitive RPV failure calculations are to thermal hydraulic parameters such as temperature, pressure, rate of cooldown, and heat transfer coefficient.

H.3.2.1 Copper Content

Figure H-13 illustrates the results of the sensitivity study on copper content. When the standard deviation for the copper distribution was increased from 0.025% to 0.07%, the calculated failure probabilities increased by approximately a factor of 5.

H.3.2.2 Initial RT_{NDT}

Figure H-14 illustrates the results of the sensitivity study on RT_{NDTo}. When the standard deviation for the RT_{NDTo} distribution was increased from 15°F to 25°F, the calculated failure probabilities were increased by a factor of approximately 3.

H.3.2.2 Fluence

Figure H-15 illustrates the results of the sensitivity study on fluence. The standard deviation for the fluence distribution was increased from 30% to 50% and decreased to 15%. The increased standard deviation resulted in approximately a factor of three increase in calculated failure probabilities, while the decrease in the standard deviation had little effect on the calculated failure probabilities.

H.3.2.4 Fracture Toughness

Figure H-16 illustrates the results of the sensitivity study on fracture toughness. Three different representations of the fracture toughness distribution were considered. In the first two cases the normal distribution about the mean fracture toughness values for K_{Ic} and K_{Ia} was maintained but the standard deviation was increased to 15% and then 20% of the mean value. In the third case K_{Ic} and K_{Ia} were treated deterministically using the lower bound fracture toughness curves from Section XI of the American Society of Mechanical Boiler and Pressure Vessel Code (see Figures H-3 and H-[illegible] The sensitivity study was conducted for a mean copper content of 0.34% and a mean initial RT_{NDT} of 0°F. Assuming the larger standard deviations resulted in less than a factor of three difference from the reference case failure probabilities for a mean RT_{NDT} of 236°F or less. At higher values of RT_{NDT} the calculated failure probabilities for the assumed standard deviation of 15% and 20% were a factor of 50 and over an order of magnitude greater than the reference case, respectively. When the lower bound fracture toughness curves from Section XI of the Code were used, the calculated failure probabilities were one order of magnitude to almost two orders of magnitude higher than the reference case.

Results of intermediate scale tests conducted at Oak Ridge National Laboratory suggest that long cracks in large reactor vessels may exhibit "lower bound" fracture toughness. Several points should be made regarding this hypothesis. First, the "lower bound" performance was relative to fracture toughness data generated from small specimens not all of sufficient size to qualify as valid in accordance with ASTM-E-399 criteria. Second, cracks that exhibited "lower bound" performance in the ORNL tests were long flaws ($\sim$38 inches), and shorter more realistic flaws are expected to exhibit toughness more closely represented by the toughness distribution assumed in the reference case. Finally, the intermediate scale tests performed have exhibited statistical variability in fracture toughness, but none of them have demonstrated fracture toughness as low as the ASME Code Section XI toughness curves.

The results of this sensitivity study show that the failure probabilities are sensitive to the distribution in fracture toughness, especially for mean values of RT_{NDT} greater than approximately 240°F. Thus, an effort should be made to better define this distribution. Experience to date suggests that fracture toughness may be a function of crack length as well as other parameters, and that in analyses assuming a bivariate flaw distribution of depth and length, it may also be appropriate to consider a relation between crack length and fracture toughness.

H.3.2.5 Simultaneous Increase in the Variability of All Random Variables

Figure H-17 presents the failure probabilities calculated when all the random variables were assumed to show the increased variances used in sensitivity studies, including one case where K_{Ic} and K_{Ia} were treated as random variables and one case where they were modelled using the lower bound curves. For the first case the calculated failure probabilities were approximately an order of magnitude greater than the reference case, while for the second case (lower bound K_{Ic} and K_{Ia}) the calculated failure probabilities were almost three orders of magnitude higher.

H.3.2.6 Flaw Distribution

Figure H-18 presents the conditional failure probabilities calculated assuming that flaw sizes ranging from 0.125 inches to 2.0 inches exist with a probability of 1.0 and for several different mean fluence values and values of RT_{NDT}.

The curves presented in Figure H-18 are useful because they can be used to calculate failure probabilities for different crack depth distributions. In Table H-1 the conditional failure probability is calculated for a reactor pressure vessel with mean copper content of 0.34% and mean initial RT_{NDT} of 0°F, assuming a flaw distribution less severe than the OCTAVIA distribution assumed in the reference case. The estimated failure probability for the less severe flaw distribution is 4.7×10^{-5} compared to 7.5×10^{-5} for the OCTAVIA distribution. The relatively small difference in the estimated failure probabilities results because the flaw distributions considered are not significantly different in the range of flaw depths that contribute most to the failure probability. An advantage of this approach to evaluating sensitivity to the assumed flaw distribution is that it allows easy identification of the range of flaw depths that contribute most signficantly to the failure probability.

H.3.2.7 Shift in RT_{NDT}

A sensitivity study was conducted using the fluence versus ΔRT_{NDT} relation from Regulatory Guide 1.99, "Effects of Residual Elements in Predicted Damage to Reactor Vessel Materials." Use of the upper bound trendlines presented in Regulatory Guide 1.99 is not considered appropriate in a probabilistic analysis but was considered in this sensitivity study in an effort to quantify the effect of differences in assumed trendlines. Figure H-19 presents the results generated assuming ΔRT_{NDT} as predicted by the HEDL trendlines and the Regulatory Guide

1.99 trendlines. Assuming the more severe Regultory Guide 1.99 trendlines increased the calculated failure probabilities by a maximum of nearly two orders of magnitude.

H.3.2.8 Upper Shelf

As discussed in Section H.2.1, the results presented in this study are based on linear elastic fracture mechanics analysis. In the transients of interest, however, linear elastic fracture mechanics may not be valid when cracks are predicted to run deep into the vessel wall where the material is operating in the upper shelf temperature regime. In the upper shelf temperature regime, crack extension generally occurs in a ductile mode referred to as tearing rather than in a cleavage mode as predicted by linear elastic fracture mechanics. In the reference case analysis, the mean fracture toughness curves were truncated at an upper value of 200 KSI$\sqrt{\text{in}}$., and it was assumed that if crack arrest did not occur before the applied K_I reached 200 KSI$\sqrt{\text{in}}$., the crack would tear through the wall. In reality this problem requires an elastic-plastic or tearing instability type of analysis which has not yet been fully developed and validated for pressurized thermal shock conditions. A study was conducted to evaluate the sensitivity of the calculated failure probabilities to the assumed upper shelf value. In this study the mean upper shelf value was increased to 300 KSI$\sqrt{\text{in}}$., 400 KSI$\sqrt{\text{in}}$. and infinity and a check was incorporated for plastic instability of the remaining section. The assumed higher upper shelf toughness values all resulted in the same calculated failure probabilities, as illustrated in Figure H-20. The calculated failure probabilities with the increased upper shelf values are more than an order of magnitude less than the reference case failure probabilities for mean values of RT_{NDT} less than approximately 240°F. At a mean RT_{NDT} value of 250°F the failure probability associated with the increased upper shelf toughnesses is approximately a factor of four less than the reference case; and at a mean RT_{NDT} value of 275°F or greater the calculate failure probabilities are the same. Thus upper shelf material behavior may decrease the probability of catastrophic vessel failure for mean RT_{NDT} values of 250°F or less but provides very little additional margin at higher values of RT_{NDT}. Two notes of caution are in order. First, recent information suggests that the gradient in fluence attenuation may not be as steep as assumed in these analyses, and a different model assuming greater radiation damage deeper in the vessel wall may bring the reference case and increased upper shelf toughness failure probabilities closer together at a lower value of mean RT_{NDT}. Second, the calculated probabilities of crack initiation, which are significant from an economic point of view, are unaffected by the assumption regarding upper shelf toughness.

H.3.2.9 Cladding

For surface cracks as assumed in this evaluation, the stainless steel cladding will increase the applied stress intensity value due to differential thermal expansion between the clad and base metal. This effect has not yet been included in the fracture mechanics code used in the probabilistic analysis, although it has been evaluated deterministically. A study was conducted to estimate the magnitude of the effect of the increased K_I due to cladding on the calculated failure probabilities. In this study the thermal component of the applied stress intensity factor, K_{It}, was increased by 10% and 20%. This is a gross approximation since the actual increase in K_I will be a function of crack depth and time in the transient. However, calculations indicate that for the Rancho Seco transient the maximum contribution to the thermal component of the applied K_I is less than 10%. Therefore, the case of a 10% increase in K_{It} should be bounding for the Rancho Seco transient as analyzed deterministically. The case of a 20% increase in K_{It} gives some insight into sensitivity of the assumptions regarding initial stress in the cladding at normal operating temperature. The results of the study are presented in Figure H-21. For an increase in K_{It} of 10% there is essentially no change in the calculated failure probabilities for mean surface RT_{NDT} values less than approximately 250°F. Above a mean RT_{NDT} of 250°F the failure probabilities increase by less than a factor of three. Figure H-22 illustrates the factor of increase in conditional failure probability assuming a 10% increase in the thermal component of the applied stress intensity factor due to the affect of cladding. For a 20% increase in K_{It} the calculated failure probabilities increase by a maximum factor of approximately 4.

It should be noted that the differential thermal effect between the cladding and base metal may be more significant for more severe thermal shocks, and caution must be exercised in extending the results of this study to those transients.

H.3.3 Transient Sensitivity Studies

In addition to the reference Rancho Seco transient, postulated MSLB and turbine trip with stuck-open bypass valve transients were evaluated using the probabilistic code. The same transients were analyzed deterministically by ORNL in Reference H.1 and were selected for probabilistic analysis to provide some estimate of the conservatisms in the deterministic calculations. Also, a set of hypothetical pressurized thermal shock transients with assumed exponential temperature decays and constant pressure levels was analyzed to determine the sensitivity of failure probability to the minimum temperature reached in the transient, rate of temperature drop, pressure level, and heat transfer coefficient.

H.3.3.1 Main Steamline Break and Turbine Trip With Stuck Open By-pass Valve Transients

Figures H-23 and H-24 present the pressure and temperature time histories associated with the postulated MSLB and stuck-open bypass valve transients, respectively. The solid lines in the figures represent the pressure and temperature time histories calculated by Brookhaven National Laboratory using the IRT Code. Reference H.1 provides details of the assumptions made in performing the thermal hydraulic calculations. The solid lines in each figure represent the pressure and temperature time histories calculated by the IRT analysis. The dashed lines represent the fourth order polynomial fits to the IRT pressure and temperature time histories used for performing closed form heat transfer and stress analyses. The applied stress intensity values resulting from these polynomial fits agree well with those calculated by ORNL using the OCA-1 numerical heat transfer analysis. Figures H-25 and H-26 present the calculated failure probabilitie for the MSLB and stuck-open bypass valve, respectively, for a longitudinal beltline weld with a mean initial RT_{NDT} of 0°F and mean copper contents of 0.22% and 0.34%. The failure probabilities are very high for both of these severe thermal transients.

H.3.3.2 Hypothesized Transients with Exponential Cooldowns and Constant Pressures

Table H-2 presents the failure probabilities for a set of hypothesized pressurized thermal shock transients. The temperature time history in each transient is assumed to follow an exponential decay defined by

$$T(t) = T_f + (550 - T_f)e^{-\beta t}$$

where T is the temperature in °F, t is time in minutes, T_f is the final temperature of the transient in °F, and β is the decay constant in min^{-1}. Three values of T_f, 150°F, 225°F, and 300°F; three values of β, 0.05 min^{-1}, 0.15 min^{-1}, and 0.50 min^{-1}; and five constant pressure levels, 0 psig, 500 psig, 1000 psig, 1500 psig, and 2000 psig were considered for a total of 45 different transients. Each of these transients was then evaluated for five levels of fluence, 0.5 10^{19} neut/cm^2, 1.0 x 10^{19} neut/cm^2, 2.0 x 10^{19} neut/cm^2, 3.0 x 10^{19} neut/cm^2, and 4.0 x 10^{19} neut/cm^2 assuming a mean copper content of 0.30% and a mean initial RT_{NDT} of 20F. The data presented in Table H-1 have been used to evaluate the sensitivity of failure probability to the normalizing factor T_f - RT_{NDT}, β, and pressure.

H.3.3.2.1 T_f-RT_{NDT} Sensitivity Study

Figure H-27 presents failure probability versus T_f - RT_{NDT} for the three different values of β considered and a constant pressure of 1000 psig. An ideal normalizing factor would combine the significant transient parameters in such a way that one curve of failure probability versus the normalizing factor could be used to estimate the probability of failure for any arbitrarily defined transient. Several factors combining T_f, β, pressure, total temperature drop. and RT_{NDT} were considered but no combination of these factors yielded a perfect normalizing factor. However, for the range of transients considered here, T_f - RT_{NDT} is a fairly effective normalizing factor for any specific β and constant pressure level. Figure H-28 indicates that failure probability is highly sensitive to the value of T_f -RT_{NDT}. For example, considering a β of 0.15 min^{-1}, a decrease in T_f - RT_{NDT} from -20°F to -70°F results in a factor of approximately 150 increase in failure probability.

H.3.3.2.2 Cooldown Rate Sensitivity Study

Figure H-27 indicates a much greater increase in failure probabilities when β is increased from 0.05 to 0.15 than when β is increased from 0.15 to 0.50. This observation is more clearly illustrated in Figure H-28 where failure probability is plotted as a function of β for several values of T_f-RT_{NDT} and 1000 psig constant pressure. The curves illustrate that failure probability is very sensitive to β in the range below 0.15 min^{-1} while increasing β beyond 0.15 min^{-1} increases the failure probability by less than a factor of five. This is most likely a result of the assumed thermal inertia of the system, and the sensitivity curves will change if different thermal characteristics are assumed in the heat transfer analysis.

H.3.3.2.3 Pressure Sensitivity Study

Figure H-29 is a plot of failure probability versus pressure for several values of the parameter T_f - RT_{NDT}. The failure illustrates increasing sensitivity to pressures as the parameter T_f - RT_{NDT} increases. For example, for a T_f - RT_{NDT} value of -25°F an increase in pressure from 500 psig to 2000 psig results in approximately a factor of 200 increase in failure probability while a similar pressure increase for a T_f - RT_{NDT} value of -120°F increases the failure probability by only a factor of 5. Thus pressure is a more important parameter in the transients where the minimum temperature is near the value of RT_{NDT} rather than well below it. It should be noted that for a pressure level of 0.0 psig, the failure probability is zero. Thermal Shock Experiment 6 recently completed at ORNL demonstrated that although severe cracking may occur under the condition of no pressure, thermal stresses alone are not sufficient to drive a crack through the RPV wall.

H.3.3.2.4 Heat Transfer Coefficient Sensitivity Study

Figure H-30 presents the results of a sensitivity study conducted on heat transfer coefficient. The two curves in the figure present RPV failure probability versus heat transfer coefficient, h in BTU/hr/ft^2 °F, for two different hypothetical exponential cooldowns. One has a final transient temperature of 150°F while the other has a final transient temperature of 200°F. A constant pressure level of 1000 psig was assumed and the RPV material was assumed to have an adjusted RT_{NDT} of 250°F. When the thermal conductivity of the cladding is considered, the range of the effective heat transfer coefficient for the thermal hydraulic transients under consideration is between 200 BTU/hr/ft^2 °F and 400 BTU/hr/ft °F. The results indicate that over that range, the assumed heat transfer coefficient can make as much as an order of magnitude difference in the calculated RPV failure probabilities. The results presented in this study were generated assuming an effective heat transfer coefficient of approximately 300 BTU/hr/ft^2. The assumed thermal diffusivity in this study was 0.98 in^2/min and a constant value of 0.332 was used for the parameter $(\frac{E\alpha}{1-\nu})$. Where E is Young's Modulus, α is the coefficient of thermal expansion, and ν is Poisson's ratio.

H.3.4 Inservice Inspection Sensitivity Study

Sensitivity studies were conducted using Figure H-18 to evaluate the effect of various levels of non-destructive examination (NDE) reliability on reactor pressure vessel failure probability. Three different functions of flaw non-detection probability were considered. The first function for probability of flaw non-detection was taken from Reference H.2. This function was based on a survey of NDE experts. The other two flaw nondetection probability functions assumed probabilities of non-detection of 0.5 and 0.05, respectively, over the entire range of crack depths. The latter two functions were selected primarily for the purpose of evaluating the sensitivity of failure probability to a wide range of NDE reliabilities. However, they were also intended to correspond to condition of rough surface finish and smooth surface finish, respectively. It was assumed for all functions of NDE reliability that cracks of greater than 2.0 inches in depth would be detected with certainty. The results of these evaluations are presented in Tables H-3A through H-3C. The first column in these tables gives the flaw depth, a, in inches; the second column is the probability of existence of a crack of depth a as estimated by the OCTAVIA flaw distribution; column three is the probability of non-detection; column four is the probability of existence of a crack of depth a after performing an NDE (the product of columns two and three); column five is the conditional probability of failure given the Rancho Seco transient and existence of a crack of depth a; and column six is the contribution to the conditional failure probability of the reactor vessel weld for each crack depth (the product of columns four and five). The conditional failure probability of the reactor vessel weld given that the Rancho Seco transient occurs is given by the sum of the probabilities in column six.

The conditional failure probabilities of a reactor pressure vessel weld following inservice inspection can be compared to the conditional failure probability of 7.5×10^{-5} before the inservice inspection, from Figure H-8. This comparison indicates inservice inspections conducted with reliabilities corresponding to the Reference H.2 report probability of non-detection function or the constant 0.5 probability of non-detection problem will do very little to improve reactor pressure vessel reliability under pressurized thermal shock conditions. However, if a probability of non-detection of 0.05 can be achieved, even for small flaws, then a substantial decrease in failure probability, approximately a factor of 20, will result.

H.3.5 Warm Prestressing Sensitivity Study

A study was conducted to determine the effects of warm prestressing on the calculated conditional failure probabilities for the idealized Rancho Seco transient that was considered as the reference transient in Section H.3.1. The warm prestress phenomenon was modelled by simply not allowing crack initiation at any time step in the transient for which the applied K value for the simulated crack depth was greater at the previous time step. No allowance was made for a possible increase in the allowable K_I to K_{Ic} ratio above 1.0 resulting from warm prestressing.

For the Rancho Seco transient warm prestressing was very effective in inhibiting crack extension. The conditional failure probabilities calculated assuming warm prestressing were less than 10^{-5} for mean RT_{NDT} values less than 290°F (See Table H-3.)

TABLE H-3A: A Marshall Report Probability of Nondetection

a	P(a)	P(Non-Detection)	P(a Inspection)	P(Failure)	P(Failure)
0.125	0.83	.69	.57	0	0
0.25	0.16	.49	.78	$5x10^{-5}$	$3.9x10^{-6}$
0.50	$4.2x10^{-3}$	.24	$1.0x10^{-3}$	$1.0x10^{-2}$	$1.0x10^{-5}$
1.00	$4.1x10^{-4}$	.061	$2.5x10^{-5}$	$5.4x10^{-2}$	$1.4x10^{-6}$
1.50	$1.3x10^{-4}$	.018	$2.3x10^{-6}$	$5.6x10^{-2}$	$1.3x10^{-7}$
2.00	$4.2x10^{-5}$	$8.1x10^{-3}$	$3.4x10^{-7}$	$4.5x10^{-2}$	$1.5x15^{-8}$
			CONDITIONAL FAILURE PROBABILITY		$1.5x10^{-5}$

TABLE H-3B Constant 0.50 Probability of Non-Detection

a	P(a)	P(Non-Detection)	P(a 1 Inspection)	P(Failure)	P(Failure)
0.125	0.83	0.50	0.42	0	0
0.25	0.16	0.50	0.08	$1.5x10^{-4}$	$1.2x10^{-5}$
0.50	$4.2x10^{-3}$	0.50	$2.1x10^{-3}$	$1.0x10^{-2}$	$2.1x10^{-5}$
1.00	$4.1x10^{-4}$	0.50	$2.1x10^{-4}$	$5.4x10^{-2}$	$1.1x10^{-5}$
1.50	$1.3x10^{-4}$	0.50	$6.5x10^{-5}$	$5.6x10^{-2}$	$3.6x10^{-6}$
2.00	$4.2x10^{-5}$	0.50	$2.1x10^{-2}$	$4.5x10^{-2}$	$9.5x10^{-7}$
			CONDITIONAL FAILURE PROBABILITY		$4.9x10^{-5}$

TABLE H-3C: Constant 0.05 Probability of Non-Detection

a	P(a)	P(Non-Detection)	P(a 1 Inspection)	P(Failure)	P(Failure)
0.125	0.83	0.05	$4.2x10^{-2}$	0	0
0.25	0.16	0.05	$8.0x10^{-3}$	$1.5x1^{-4}$	$1.2x10^{-6}$
0.50	$4.2x10^{-3}$	0.05	$2.1x10^{-4}$	$1.0x10^{-2}$	$2.1x10^{-6}$
1.00	$4.1x10^{-4}$	0.05	$2.1x10^{-6}$	$5.4x10^{-2}$	$1.1x10^{-7}$
1.50	$1.3x10^{-4}$	0.05	$6.5x10^{-6}$	5.6x10-2	$3.6x10^{-7}$
2.00	$4.2x10^{-5}$	0.05	$2.1x10^{-6}$	$4.5x10\text{-}10^{-2}$	$9.5x10^{-8}$
			CONDITIONAL FAILURE PROBABILITY		$3.8x10^{-5}$

Based on the above studies it can be concluded tha+ for transients whose thermal hydraulic characteristics ensure warm prestressin ditions, the probability of RPV failure can be significantly reduced.

H.3.6 Flaw Orientation Sensitivity Study

Results presented thus far have concentrated on the longitudinally oriented beltline welds. The volume and orientation of weld material in the reactor vessel beltline region depends on whether the beltline shell was fabricated from rolled plates or forged rings as illustrated in Figure H-1. Several operating vessels are fabricated from ring forgings or have limiting values of RT_{NDT} associated with circumferential welds.

The orientation of the beltline welds is significant in the evaluation of pressurized thermal shock transients because flaws oriented in a circumferential direction have a lower propensity for extension than those oriented parallel to the longitudal axis of the vessel. The circumferentially oriented crack has a lower propensity for crack extension because it is subject to a pressure stress only half as great as the longitudinal flaw and because the applied stress intensity factor is lower due to the increased bending stiffness of the cylinder about its azimuthal axis. In addition, these two factors also create a greater propensity for crack arrest in a circumferentially oriented flaw. Because flaws in the weld material are generally assumed to be oriented in the direction of the weld, reactor vessels fabricated from forged rings with circumferential welds are expected to have a greater tolerance for pressurized thermal shock loadings than reactor vessels fabricated from rolled plates with longtudinal welds.

A study was conducted to evaluate the relative differences in integrity between longitudinally and circumferentially oriented welds. Both determinstic and probablistic calculations were performed for two different transients. The transients were the idealized Rancho Seco Transient illustrated in Figure H-7 and the MSLB accident illustrated in Figure H-24. Two dimensional (infinitely long longitudinal and 360° circumferential) flaws were evaluated using linear elastic fracture mechanics analysis.

The resu s of the deterministic calculations indicate that the Rancho Seco transient will not cause catastrophic failure of the reactor pressure vessel for a surface RT_{NDT} less than 350°F (calculated by R.G. 1.99). For a surface RT_{NDT} of 350°F or lower, the deterministic calculations predict crack arrest less than halfway through the vessel wall in the linear elastic regime. For a surface RT_{NDT} of 370°F, crack arrest is predicted approximately three-fourths of the way through the vessel wall. Although some margin still exists for circumferentially oriented flaws, this depth of crack extension is approaching the condition where the vessel would fail due to plastic instability of the remaining ligament. Furthermore, this amount of crack extension leaves little margin for tearing of the crack which could occur in low upper shelf materials.

The probabilistic analysis of the Rancho Seco transient generally supports the conclusions from the deterministic calculations. The failure probabilities calculated for the Rancho Seco transient assuming that a 1.0-inch flaw existed with certainty were less than 10^{-5} for a mean surface RT_{NDT} values of 275°F or less and approximately 3.2×10^{-5} for mean surface RT_{NDT} of 290°F. Comparable failure probabilities for longitudinally oriented flaws were 7.5×10^{-4}, 10^{-2}, and 4.5×10^{-2} for mean surface RT_{NDT} values of 250°F, 275°F, and 290°F, respectively. Thus, for the Rancho Seco transient, the failure probability of a circumferentially oriented flaw is at least three orders of magnitude less than that of a longitudinally oriented flaw for mean surface RT_{NDT} values of 290°F or less. A comparison of the crack initiation probabilities for longitudinal and circumferential flaws indicated that the probability of initiation of a circumferential flaw ranges by approximately a factor of 1000 to 25 less than that of a longitudinal flaw for a corresponding range in mean surface RT_{NDT} values of 215°F to 290°F.

Deterministic calculations for the MSLB indicate that vessel failure due to extension of circumferential cracks will occur at RT_{NDT} surface values of 226°F (calculated by Regulatory Guide 1.99) or greater. Since 226°F was the lowest RT_{NDT} evaluated, vessel failure might be predicted at even lower values of RT_{NDT}. The probabilistic analysis of the MSLB indicated that the probability of failure of a circumferentially oriented flaw can be as little as a factor of 12 to 3 less than that for a longitudinal flaw for a corresponding range in mean surface values of RT_{NDT} between 250°F and 290°F. Figure H-32 presents the factor decrease in failure probability for circumferentially versus longitudinally oriented flaws of 1.0-inch and 0.5-inch depths. For a mean surface RT_{NDT} value of 215°F, no failures were generated in the simulation analysis. The probability of crack initiation for the postulated MSLB accident was essentially equal over a range in mean surface RT_{NDT} values of 215°F to 290°F for the flaw sizes considered.

In summary, both deterministic and probabilistic evaluations indicate that for transients as severe as those which have been observed (the Rancho Seco transient being considered the most severe) circumferential flaws will not lead to catastrophic vessel failure for relatively high values of RT_{NDT}. Furthermore, the probability of initiation of circumferentially oriented flaws is significantly less than that of longitudinal flaws until relatively high values of RT_{NDT} are reached. However, for much more severe postulated transients, deterministic analyses predict that catastrophic vessel failure can result from circumferentially oriented flaws at relatively low values of RT_{NDT}. In addition, probabilistic analyses indicate a relatively small difference in failure probabilities between circumferential and longitudinal flaws and essentially no difference in the probability of crack initiation for more severe transients.

H.4 Application of Probabilistic Analyses in Establishing Regulatory Criteria

Probabilistic analysis is a very powerful technique for gaining insight and understanding of complex technical issues and when used correctly can result in effective regulation without excessive conservatism. However, misapplication of the results of probabilistic analyses which may occur due to inadequate understanding of the bases upon which they were developed could compromise safety and economic objectives. In this context, the purpose of this section is to identify some of the limitations of the work performed and to define how the results presented previously can be most approximately used in developing a regulatory position on the pressurized thermal shock issue.

H.4.1 Limitations of Probabilistic Fracture Mechanics Analyses

As indicated in Section H.2.3, the statistical distributions used to generate the results presented in Section H.3 are based largely on expert opinion and are subjective in nature. Efforts are currently in progress to assemble improved data bases and develop more rigorous statistical distributions. However, results generated using improved input data will not be available to assist in developing a short-term position on the pressurized thermal shock issue. Uncertainty in the statistical distributions used in the model is one of the main reasons for conducting the sensitivity studies presented in Section H.3.2. The results of these sensitivity studies, in which the variability and form of the statistical distributions were varied, indicate that uncertainties in the statistical distributions for copper content, initial RT_{NDT}, and fluence could contribute as much as an order of magnitude uncertainty to the results presented in Section H.3.

Flaw depth is the random variable with the greatest uncertainty. The sensitivity studies on flaw depth distribution and inservice inspection indicate that the calculated failure probabilities for the Rancho Seco transient are relatively insensitive to changes in the distribution for crack depths greater than approximately one inch. This is because relatively small flaws can dominate the failure probability due to the nature of the stresses and toughness gradient associated with pressurized thermal shock events. The sensitivity studies also indicate that the calculated failure probabilities could change substantially given a significant change in the distribution of crack depths. When the probabilities of all crack depths are altered by a constant factor, the calculated failure probabilities change by approximately the same factor. Thus, the uncertainty in the calculated failure probabilities is directly related to the uncertainty in the same crack distribution. Unfortunately, little data exist from domestic operating reactor vessels that would allow a rigorous determination of the flaw depth distribution, particularly in the range of crack depths less than one inch. The distribution of crack depths has large uncertainty associated with it and could easily contribute plus or minus an order of magnitude or more uncertainty to the calculated failure probabilities.

The sensitivity studies conducted on fracture toughness indicate that the calculated failure probabilities are very sensitive to the assumed variability in the fracture toughness data. At high values of RT_{NDT}, relatively small increase in the variabiilty of the fracture toughness can increase the calculated failure probabilities by well over an order of magnitude.

The above discussion suggests that the calculated failure probabilities could be underestimated due to uncertainties in copper content, initial RT_{NDT}, and fluence; the calculated failure probabilities could be over or underestimated due to uncertainty in the crack depth distribution; and the calculated failure probabilities could be underestimated due to uncertainties in the fracture toughness distribution. In addition to these uncertainties, there exist uncertainties due to elements not considered in the probabilistic model. Specifically, the toughness of the stainless steel cladding which may be great enough to inhibit the initiation of small flaws and warm prestressing which may inhibit crack extension were not considered. If, in fact, the vessel cladding does maintain high toughness in the range of fluence levels of interest, the extension of finite cracks could be inhibited and the failure probabilities may be greatly overestimated. Similarly, warm prestressing which will be effective for a large class of pressurized thermal shock events would greatly reduce the calculated failure probabilities for such events.

Work is continuing to better quantify the confidence levels that can be associated with the calculated failure probabilities. However, based on the currently available data and analysis, it appears that plus or minus two orders of magnitude is a reasonable estimate of the uncertainty associated with the calculated failure probabilities.

H.4.2 Application of the Probabilistic Fracture Mechanics Results

The discussion of the previous section suggests that the results which have been presented are most appropriately used in a relative sense for identifying significant variables and variable interactions. Because of the uncertainties associated with the calculated failure probabilities, use of the results in an absolute sense to establish an RT_{NDT} screening limit would be inappropriate. Nonetheless, there does exist a tendency to view the results in an absolute sense when evaluating proposed regulatory requirements. Furthermore, there is a desire to view the results in an absolute sense when performing a probabilistic risk assessment. Utilization of the results in these manners is useful in evaluating a regulatory position, but the limitations of the analysis as discussed in the previous section must be kept in mind.

In conclusion, it is suggested that the regulatory criteria should be based on deterministic fracture mechanics analyses and that the probabilistic analyses not be used as the basis for developing such criteria until such time as greater confidence in the probabilistic analyses can be attained. It is suggested, however, that the probabilistic analyses be used, with caution, to check deterministically derived criteria relative to desired margins of safety.

REFERENCES

H.1 Kryter, R. C. et al. 1981. Evaluation of Pressurized Thermal Shock. NUREG/CR-2083, ORNL/TM-8072, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

H.2 Study Group Report, "An Assessment of the Integrity of PWR Pressure Vessels," UKAEA Report, October 1976.

Table H-1 Flaw distribution sensitivity study

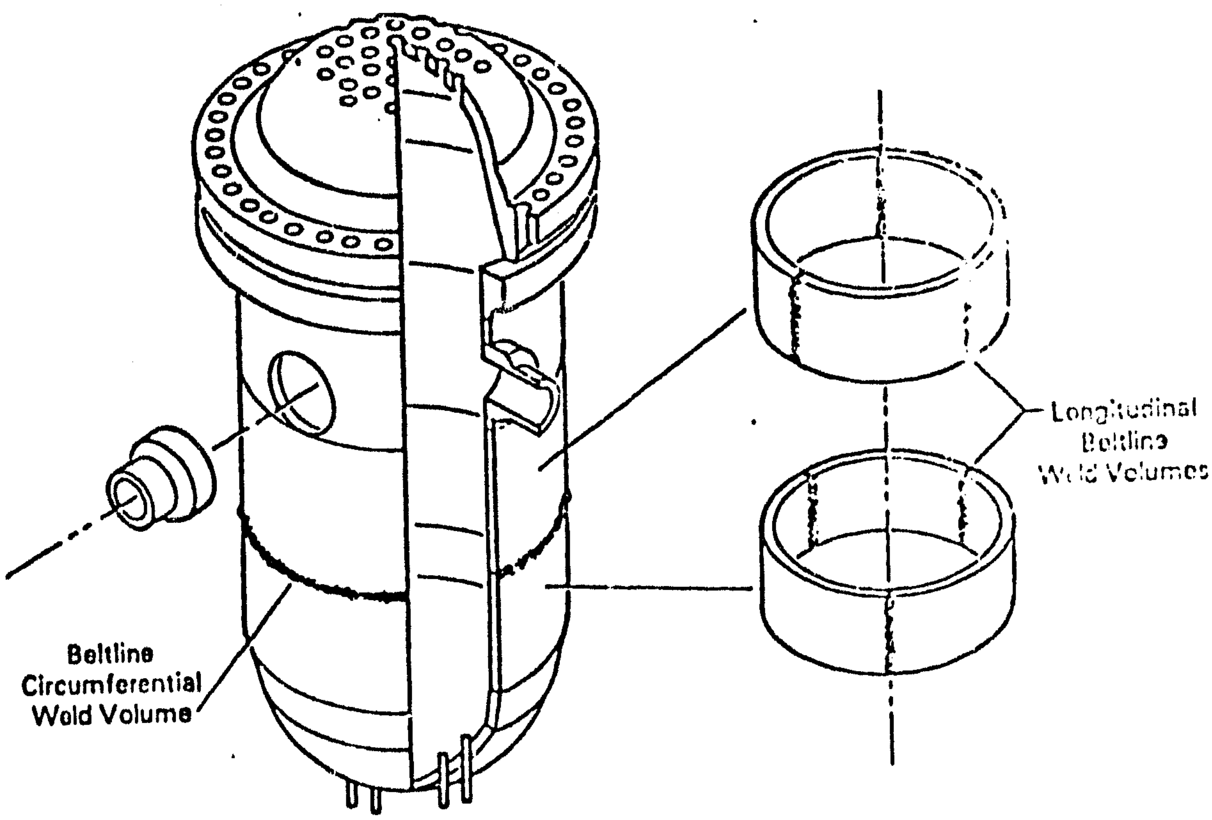
a(inches)	P(a)	P(F|a)	P(F) = P(F|a)P(a)
0.125	-	0	0
0.25	4×10^{-2}	1.5×10^{-4}	4×10^{-6}
0.50	3×10^{-3}	1.0×10^{-2}	3×10^{-5}
1.00	2×10^{-4}	5.4×10^{-2}	1×10^{-5}
1.50	1×10^{-5}	5.6×10^{-2}	6×10^{-7}
2.00	1×10^{-6}	4.5×10^{-2}	5×10^{-8}

Table H-2 $Cu_{\mu} = 0.30$ $Cu_{\sigma} = 0.025$ $F_{\sigma} = 30\%$ Ni = .75

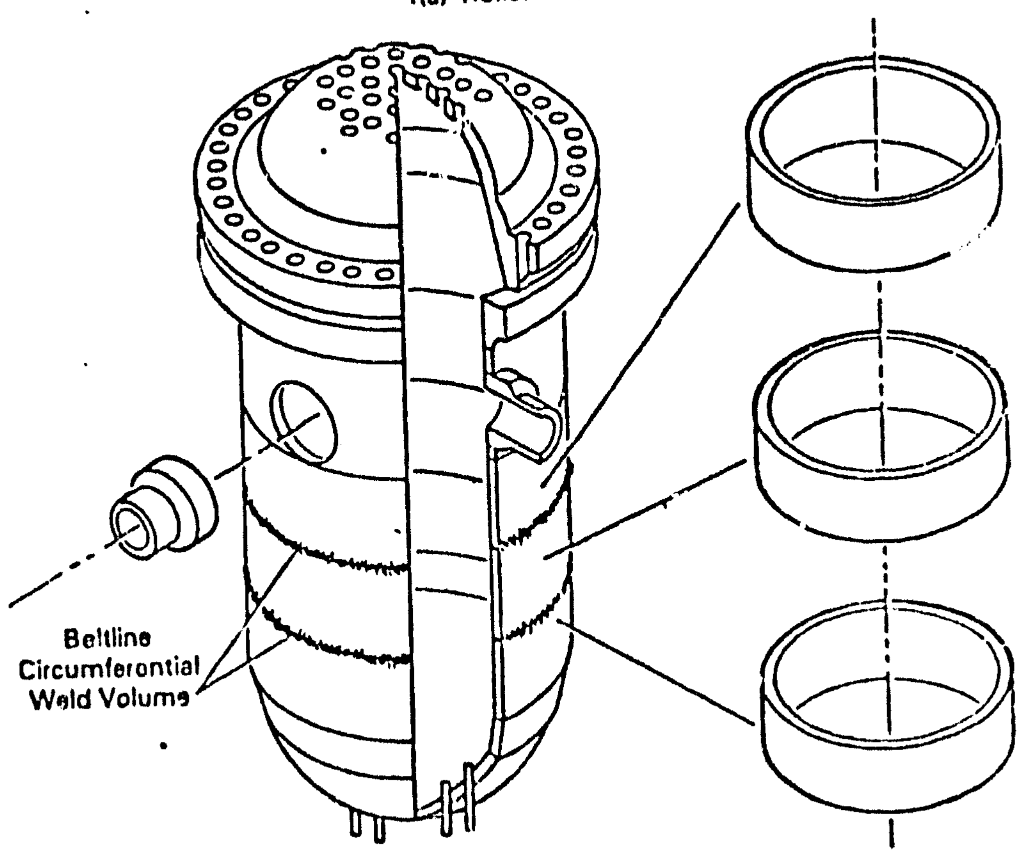
$IRT_{NDT\mu} = 20°F$ $IRT_{NDT\sigma} = 15°F$ OCTAVIA FLAW DISTRIBUTION

HEDL MEAN ΔRT_{NDT} $T = T_f + (T_i - T_f)e^{-\beta t}$

Pressure (psi)	Fluence (neut/cm²)	T_f 150°F, β .05	.15	.50	225°F .05	.15	.50	300°F .05	.15	.50
0	0.5×10^{19}	0			0	0	0	0	0	0
	1.0×10^{19}	0			0	0	0	0	0	0
	2.0×10^{19}	$2 \times 10^{-[illegible]}$			0	0	4×10^{-6}	0	0	0
	3.0×10^{19}	$5.8 \times 10^{-[illegible]}$			0	0	1×10^{-5}	0	0	0
	4.0×10^{19}	2.1×10^{-4}	3.1×10^{-2}	9.1×10^{-2}	0			0	0	0
500	0.5×10^{19}	0		8.3×10^{-2}	0	0	0	0	0	0
	1.0×10^{19}	2.2×10^{-5}	2.6×10^{-3}	1.1×10^{-2}	0	0	0	0	0	0
	2.0×10^{19}	2.9×10^{-4}	2×10^{-2}	6.3×10^{-2}	0	1×10^{-5}	4.8×10^{-5}	0	0	0
	3.0×10^{19}	9.3×10^{-4}	5×10^{-2}	1.3×10^{-1}	0	9×10^{-5}		0	0	0
	4.0×10^{19}	1.7×10^{-3}	8.3×10^{-1}	2.1×10^{-1}	0	4.3×10^{-4}	2.3×10^{-3}	0	0	0
1000	0.5×10^{19}	2.4×10^{-5}	1.2×10^{-3}	4.1×10^{-3}	0	2×10^{-6}	4×10^{-6}	0	0	0
	1.0×10^{19}	2.6×10^{-4}	1×10^{-2}	3×16^{-2}	0	2×10^{-6}	8×10^{-6}	0	0	0
	2.0×10^{19}	1.7×10^{-3}	4.9×10^{-1}	1.2×10^{-1}	4×10^{-6}	1.3×10^{-4}	5×10^{-4}	0	0	0
	3.0×10^{19}	5.1×10^{-3}	9.8×10^{-2}	2.2×10^{-1}	3.2×10^{-4}	8.1×10^{-4}	3.4×10^{-3}	0	0	0
	4.0×10^{19}	1.1×10^{-2}	1.5×10^{-1}		7×10^{-5}	2.6×10^{-3}	9.6×10^{-3}	0	0	2×10^{-6}
1500	0.5×10^{19}	1.9×10^{-4}	3.8×10^{-3}	1×10^{-2}	4×10^{-6}	1×10^{-5}	1.4×10^{-5}	0	2×10^{-6}	4×10^{-6}
	1.0×10^{19}	1.1×10^{-3}	2.5×10^{-2}	5.6×10^{-2}	0	2.8×10^{-5}	6.6×10^{-5}	0	2×10^{-6}	4×10^{-6}
	2.0×10^{19}	$6.2 \times 10^{-[illegible]}$	$9.[illegible] \times 10^{-2}$	1.9×10^{-1}	2.8×10^{-5}	6.1×10^{-4}	2×10^{-3}	0	2×10^{-6}	4×10^{-6}
	3.0×10^{19}	1.7×10^{-2}	$1.[illegible] \times 10^{-1}$		1.8×10^{-4}	3×10^{-3}	9.1×10^{-2}	0	4×10^{-6}	8×10^{-6}
	4.0×10^{19}	3.4×10^{-2}	2.4×10^{-1}		4.4×10^{-4}	8.2×10^{-3}	2.3×10^{-2}	0	1.2×16^{-5}	3.2×10^{-5}
2000	0.5×10^{19}	5.5×10^{-4}	1.0×10^{-2}	2.1×10^{-2}		2×10^{-5}	3.6×10^{-5}	2×10^{-6}		1×10^{-5}
	1.0×10^{19}	2.7×10^{-7}	4.9×10^{-2}	9.4×10^{-2}	1.8×10^{-5}	9×10^{-5}	2.7×10^{-4}	2×10^{-6}	1×10^{-3}	1.4×10^{-5}
	2.0×10^{19}	1.6×10^{-2}	1.5×10^{-1}	2.8×10^{-1}	1.7×10^{-4}	2.1×10^{-3}	5.1×10^{-3}	2×10^{-6}	1×10^{-5}	1.4×10^{-5}
	3.0×10^{19}	4.0×10^{-2}	2.6×10^{-1}		6.7×10^{-4}	8.7×10^{-3}	2×10^{-3}	2×10^{-6}	2×10^{-3}	3×10^{-5}
	4.0×10^{19}	6.6×10^{-2}	2.6×10^{-1}		1.6×10^{-3}	2.1×10^{-2}	4.4×10^{-2}	1×10^{-5}	6.8×10^{-3}	1.2×10^{-4}



1(a) Rolled and Welded Beltline Shell



1(b) Welded-Ring-Forging Beltline Shell

FIGURE H-1 PWR Beltline Shell Fabrication Configurations

RESIDUAL = PREDICTED ΔRTNDT - OBSERVED

ΔRTNDT IN °F

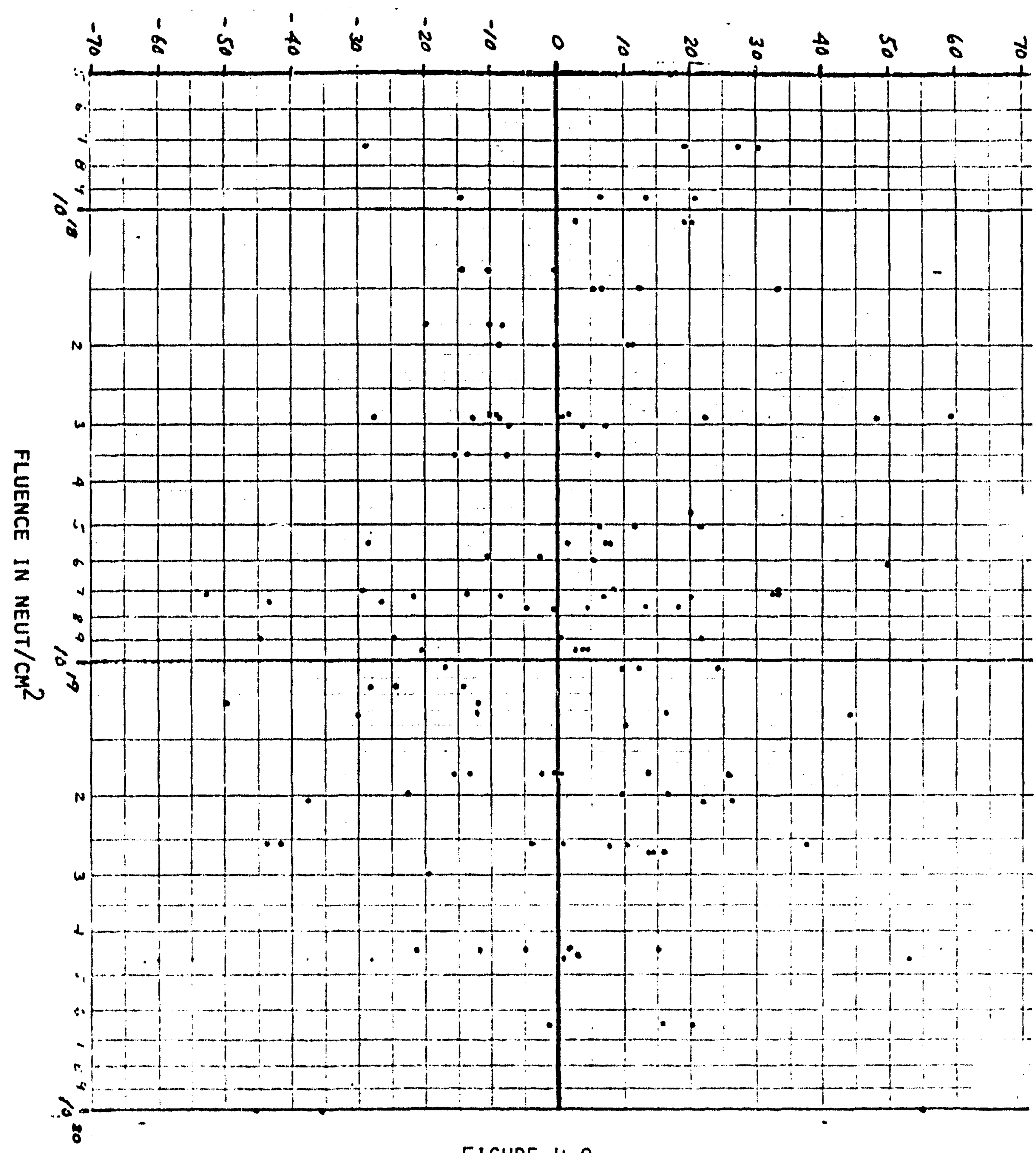


FIGURE H-2

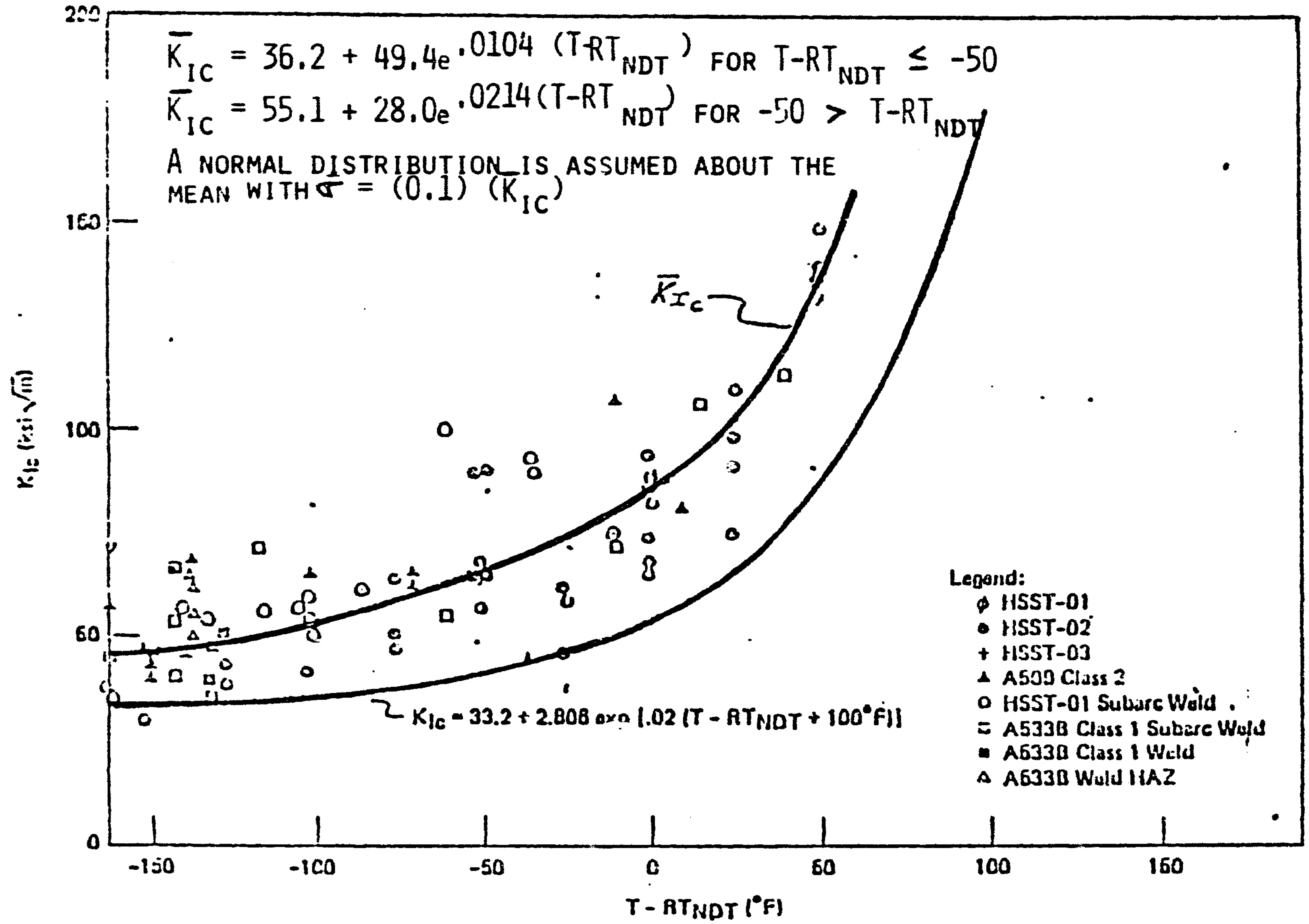


FIGURE H-3 K_{Ic} Reference Toughness Curve with Supporting Data*

*SEE A.B. LIDIARD, "A SIMPLIFIED ANALYSIS OF PRESSURE VESSEL RELIABILITY," J. BR. NUCL. ENERGY SOC. JULY 1977.

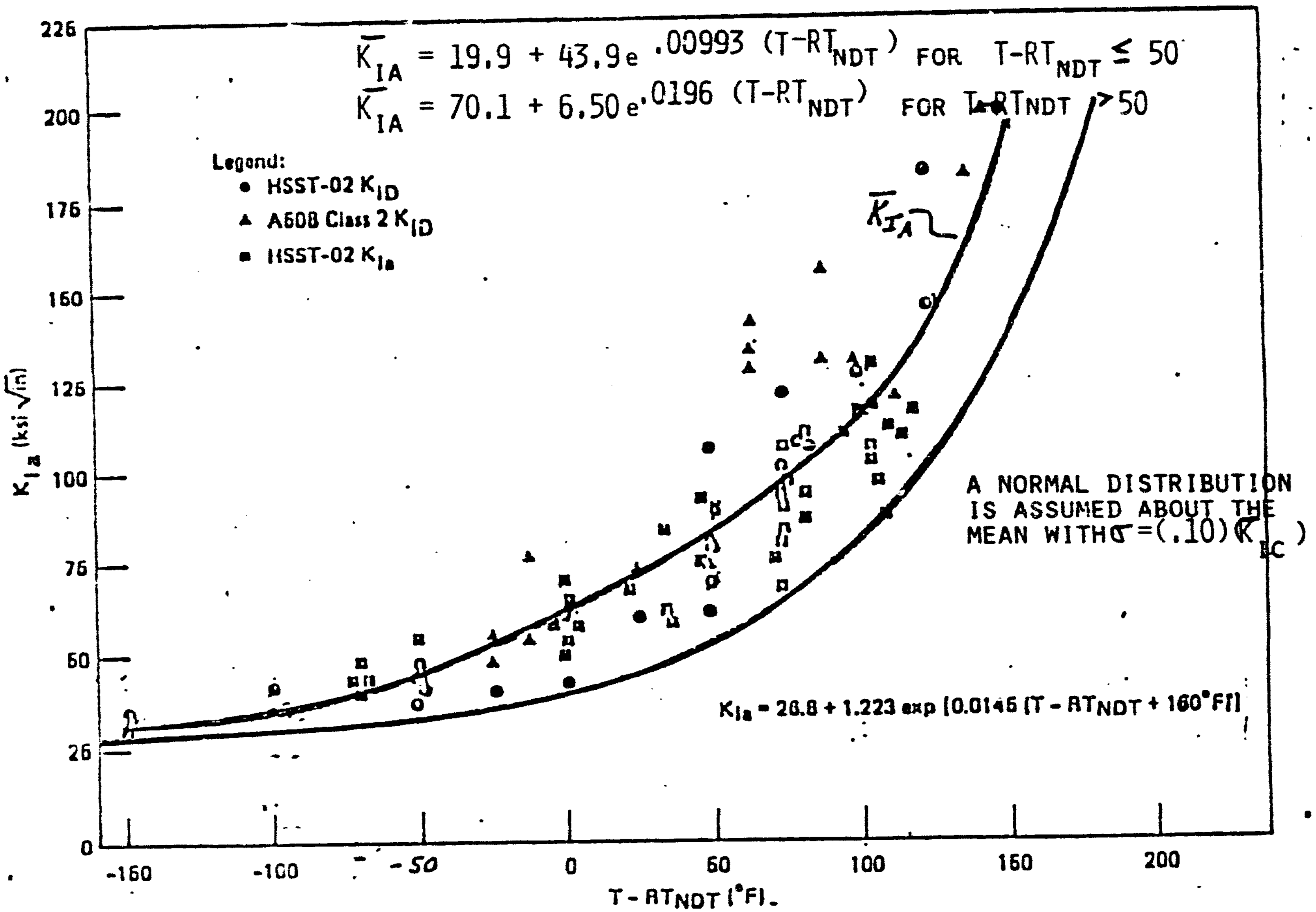
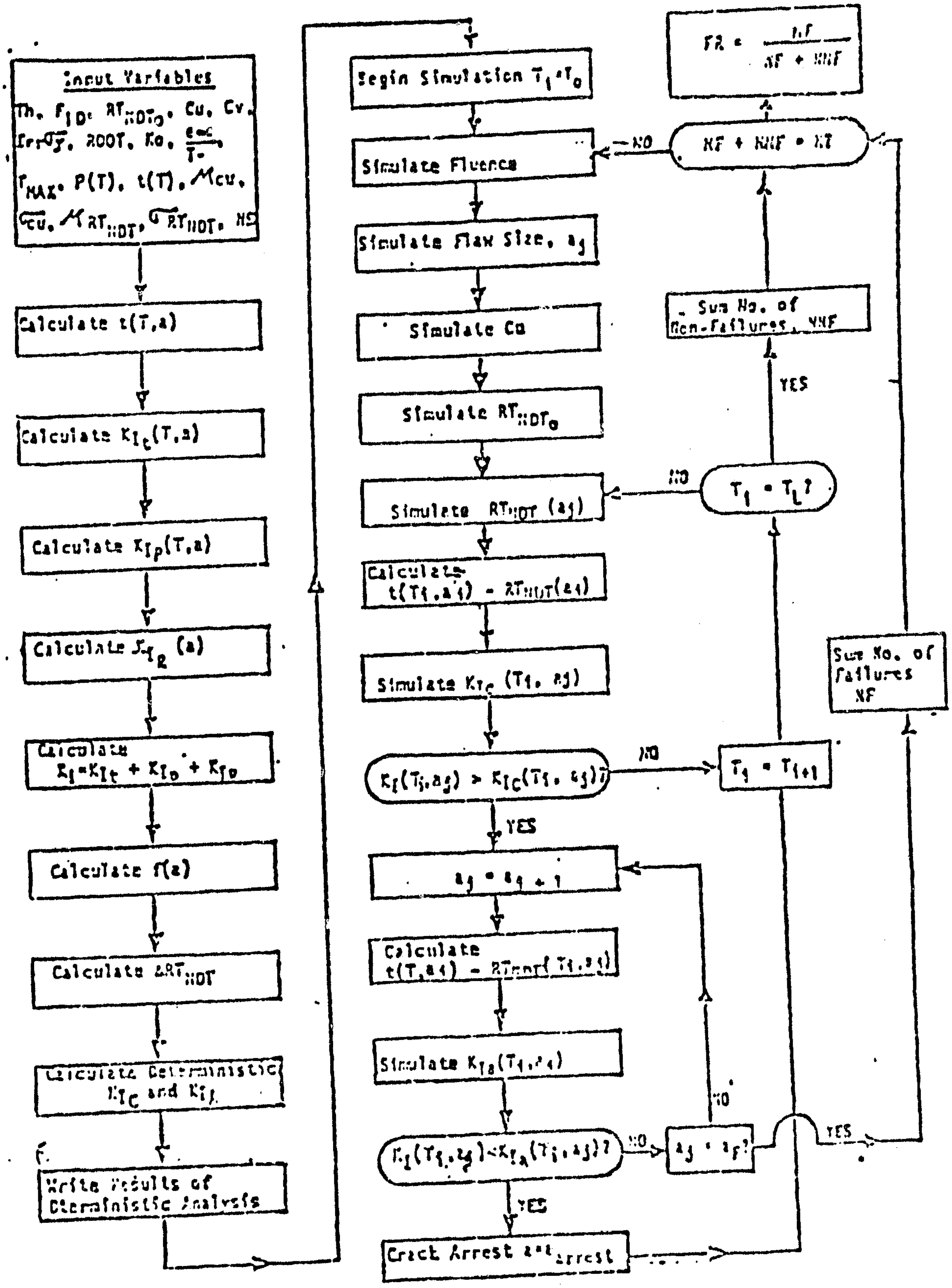


FIGURE H-4 K_{Ia} Reference Toughness Curve With Supporting Data'



Input Variables
Calculate t(T,a)
Calculate K(T,a)
Calculate Deterministic
Write Results of Deterministic Analysis
Begin Simulation
Simulate Fluence
Simulate Flaw Size
Simulate Cu
Simulate RT
Calculate
Simulate
YES
NO
Crack Arrest
Sum No. of Non-Failures, NNF
Sum No. of Failures NF

FIGURE H-5

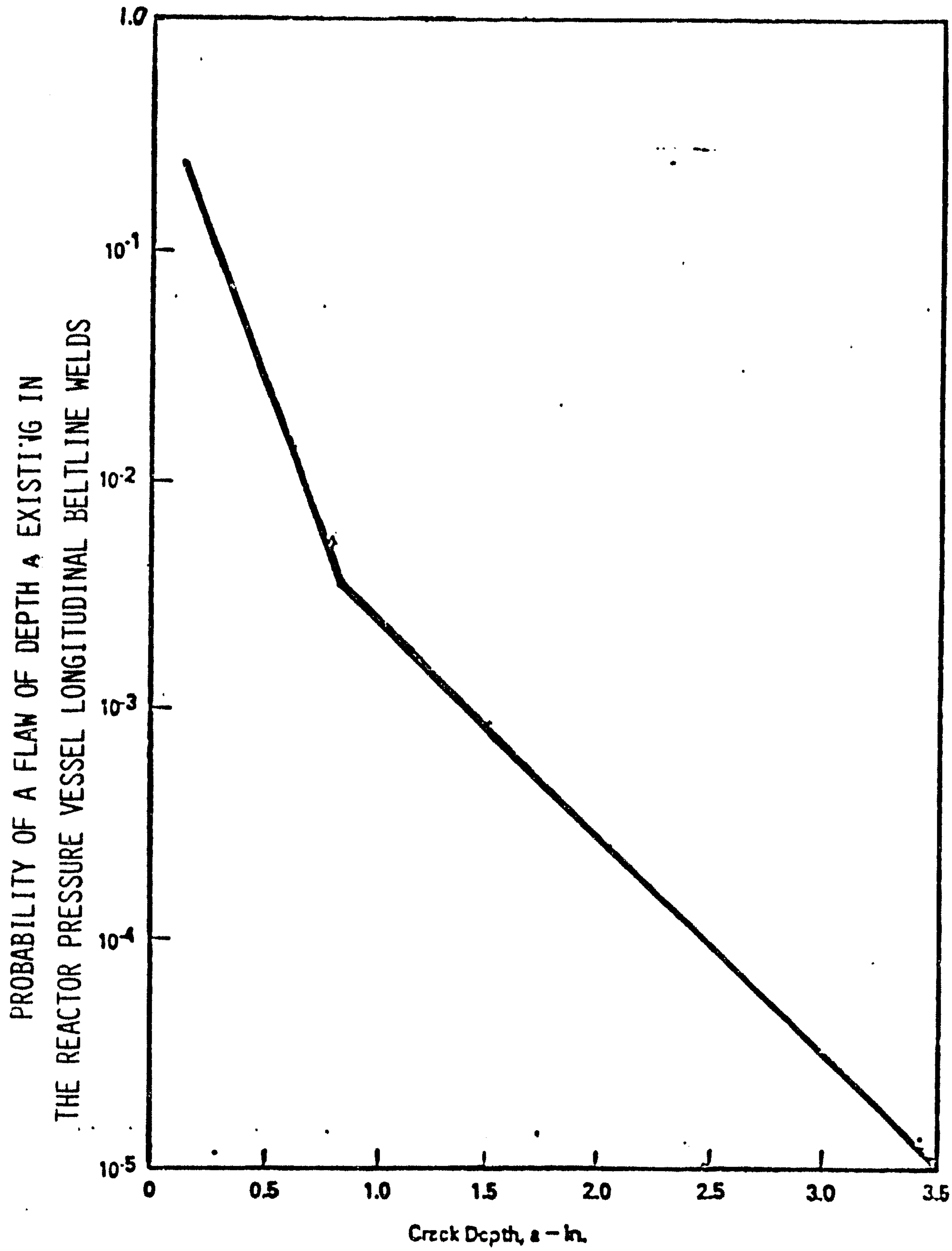


FIGURE H-6 Reactor-Vessel-Beltline Flaw Distributions

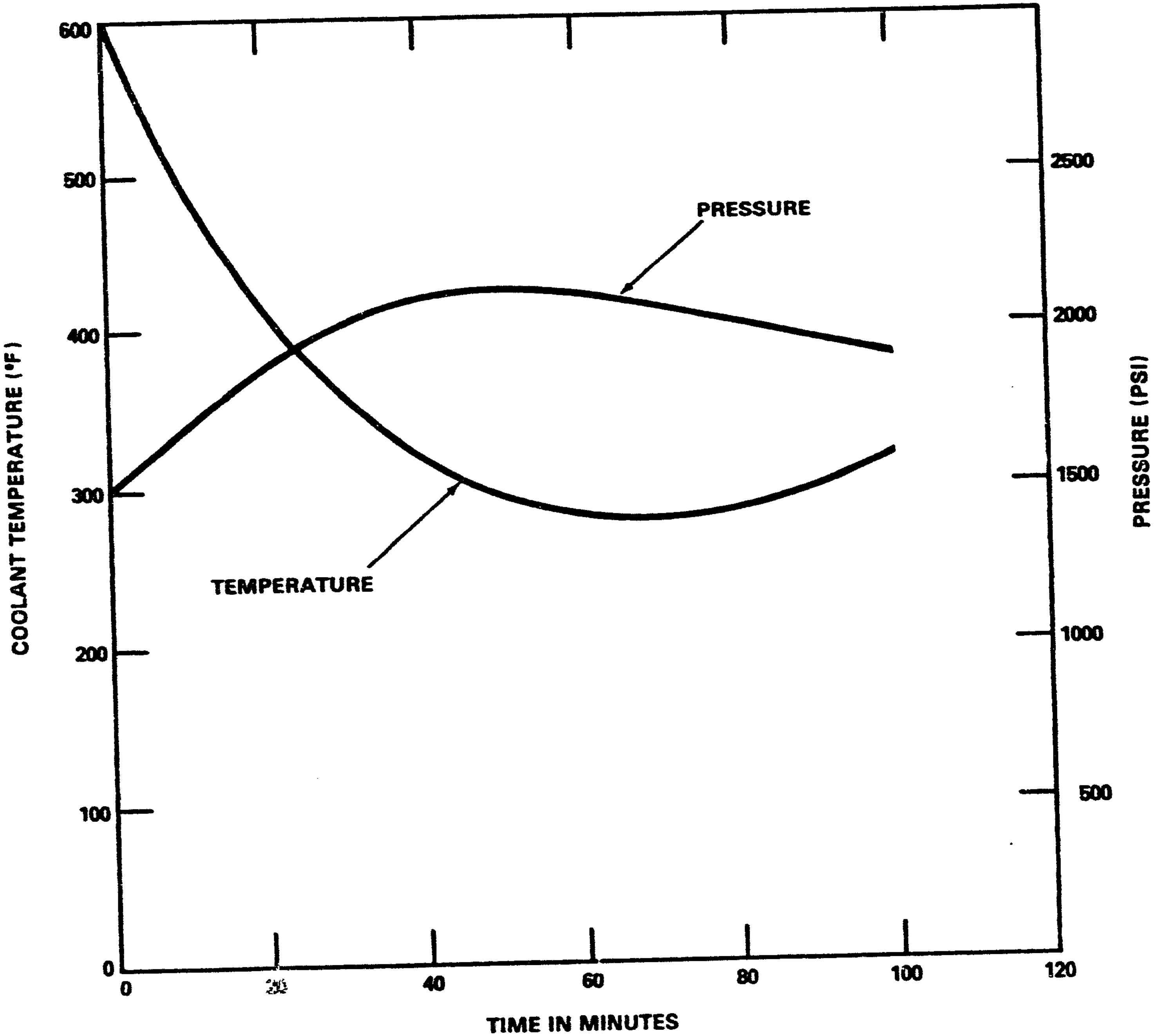


FIGURE H-7: RANCHO SECO TRANSIENT IDEALIZED PRESSURE AND TEMPERATURE TIME HISTORIES

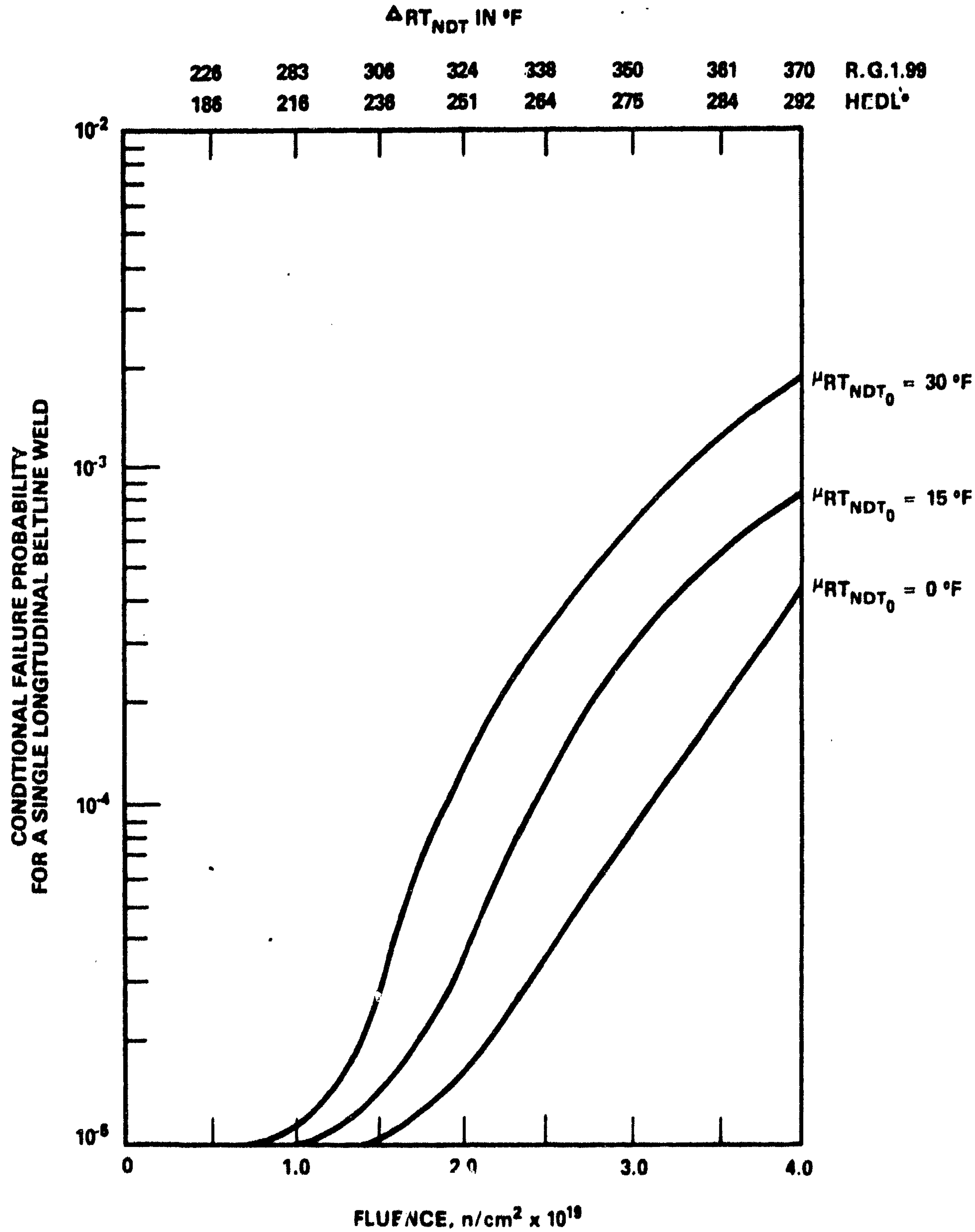


FIGURE H-8: CONDITIONAL FAILURE PROBABILITY FOR THE RANCHO SECO TRANSIENT MEAN Cu = 0.34%

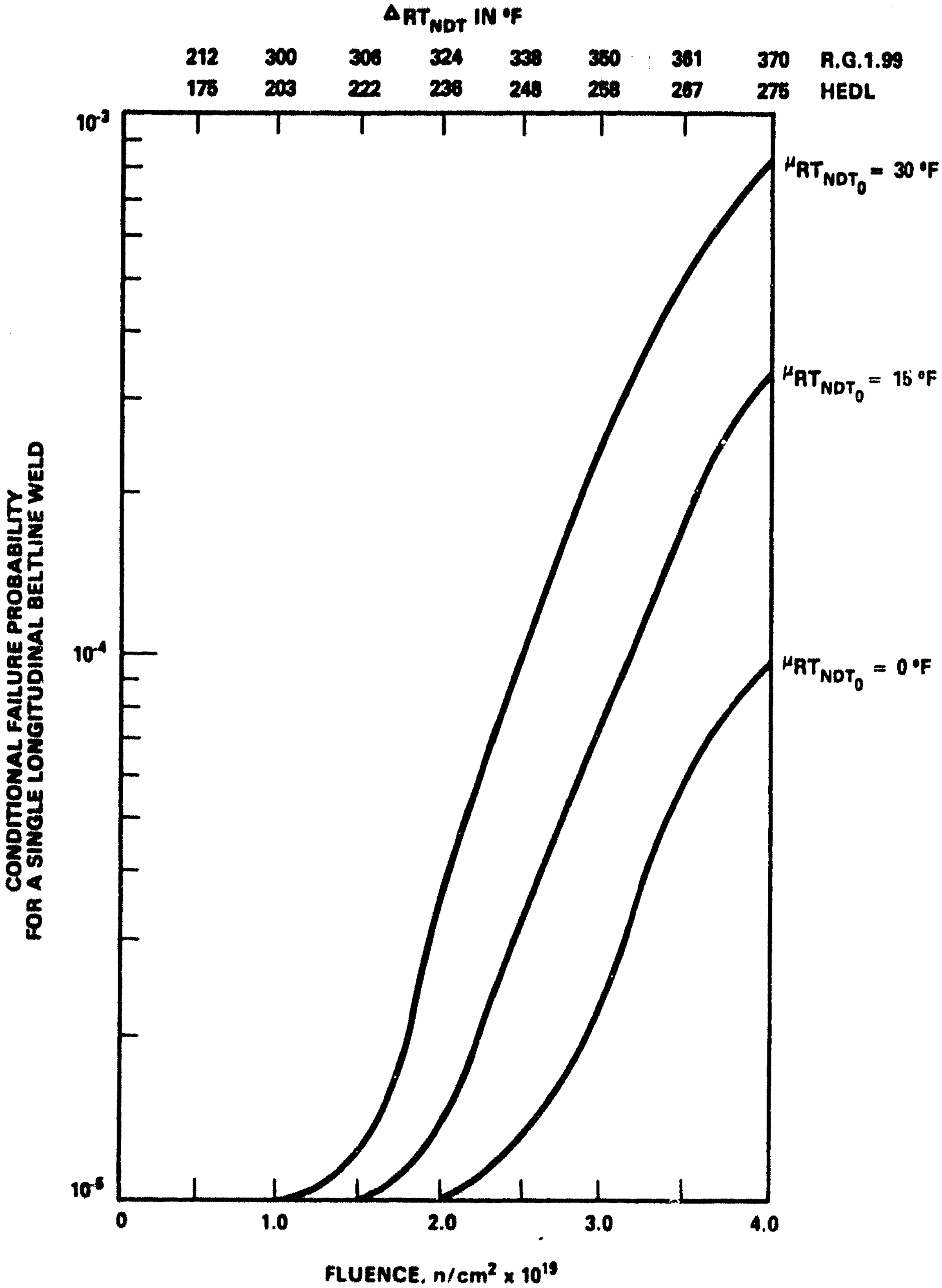


FIGURE H-9: CONDITIONAL FAILURE PROBABILITY FOR THE RANCHO SECO TRANSIENT MEAN Cu = 0.32%

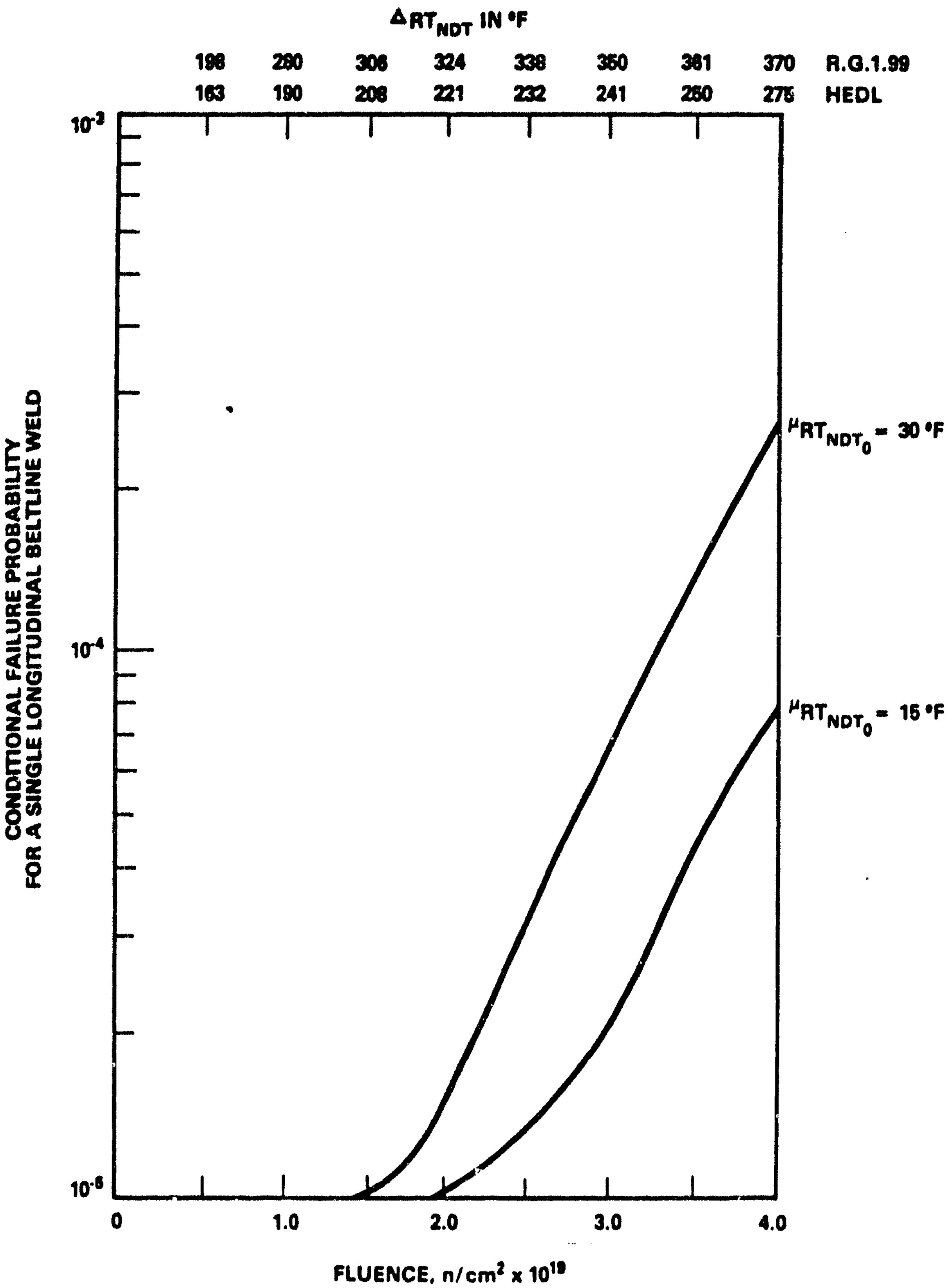


FIGURE H-10: CONDITIONAL FAILURE PROBABILITY FOR THE RANCHO SECO TRANSIENT MEAN Cu = 0.30%

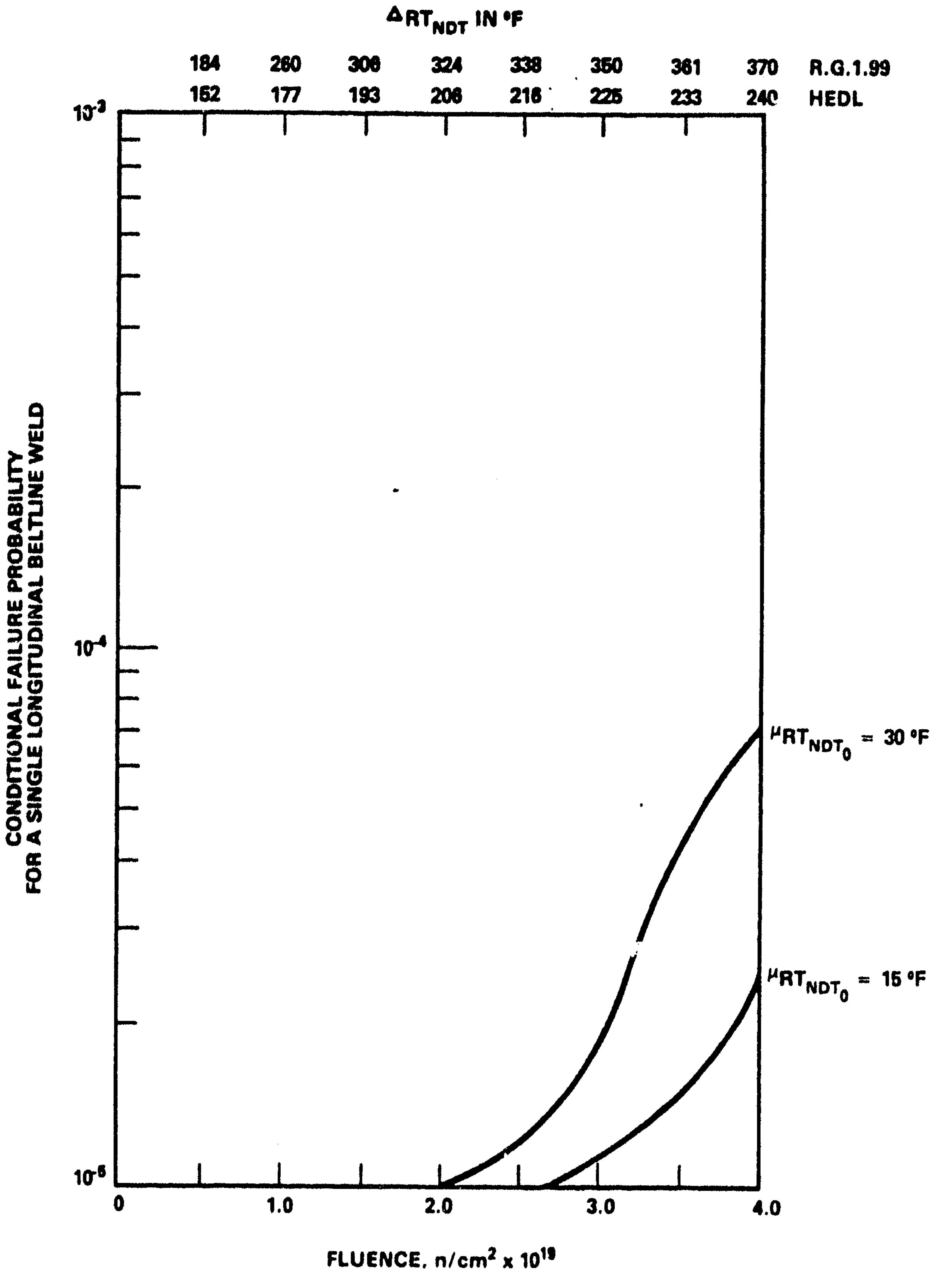


FIGURE H-11: CONDITIONAL FAILURE PROBABILITY FOR THE RANCHO SECO TRANSIENT MEA?! Cu = 0.28%

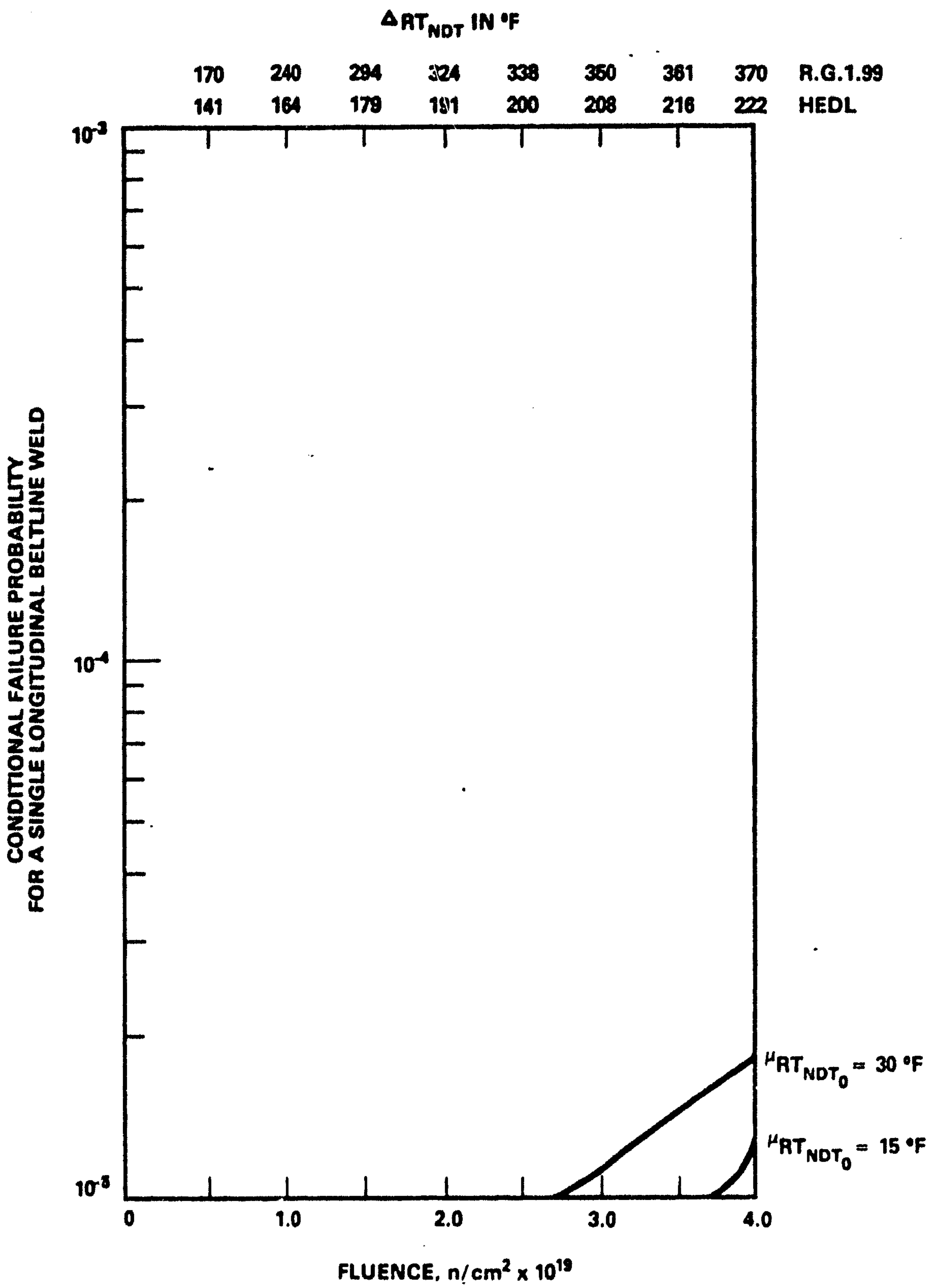


FIGURE H-12: CONDITIONAL FAILURE PROBABILITY FOR THE RANCHO SECO TRANSIENT MEAN Cu = 0.26%

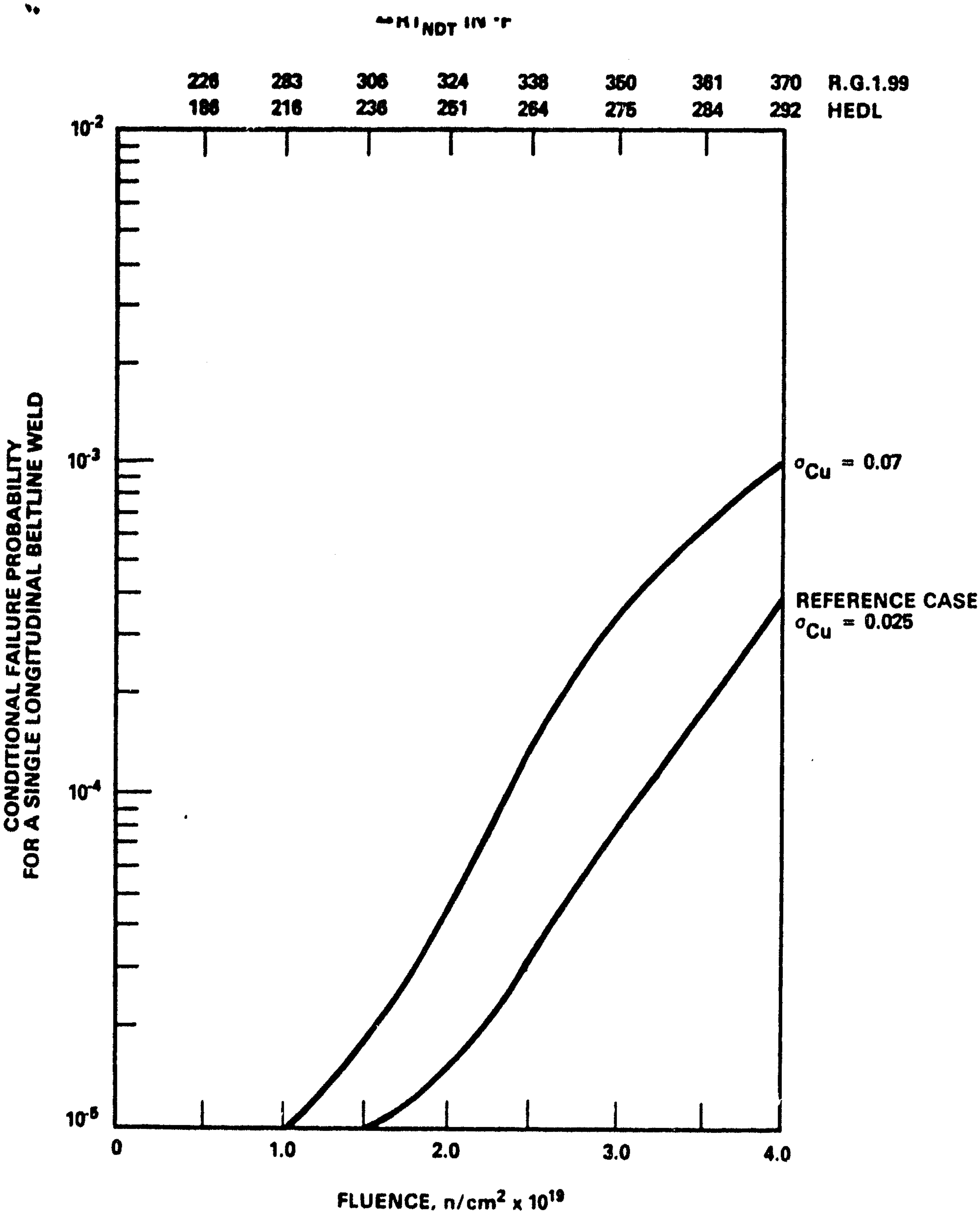


FIGURE H-13: COPPER CONTENT SENSITIVITY STUDY
MEAN Cu = 0.34% MEAN RT_{NDT_0} = 0°F

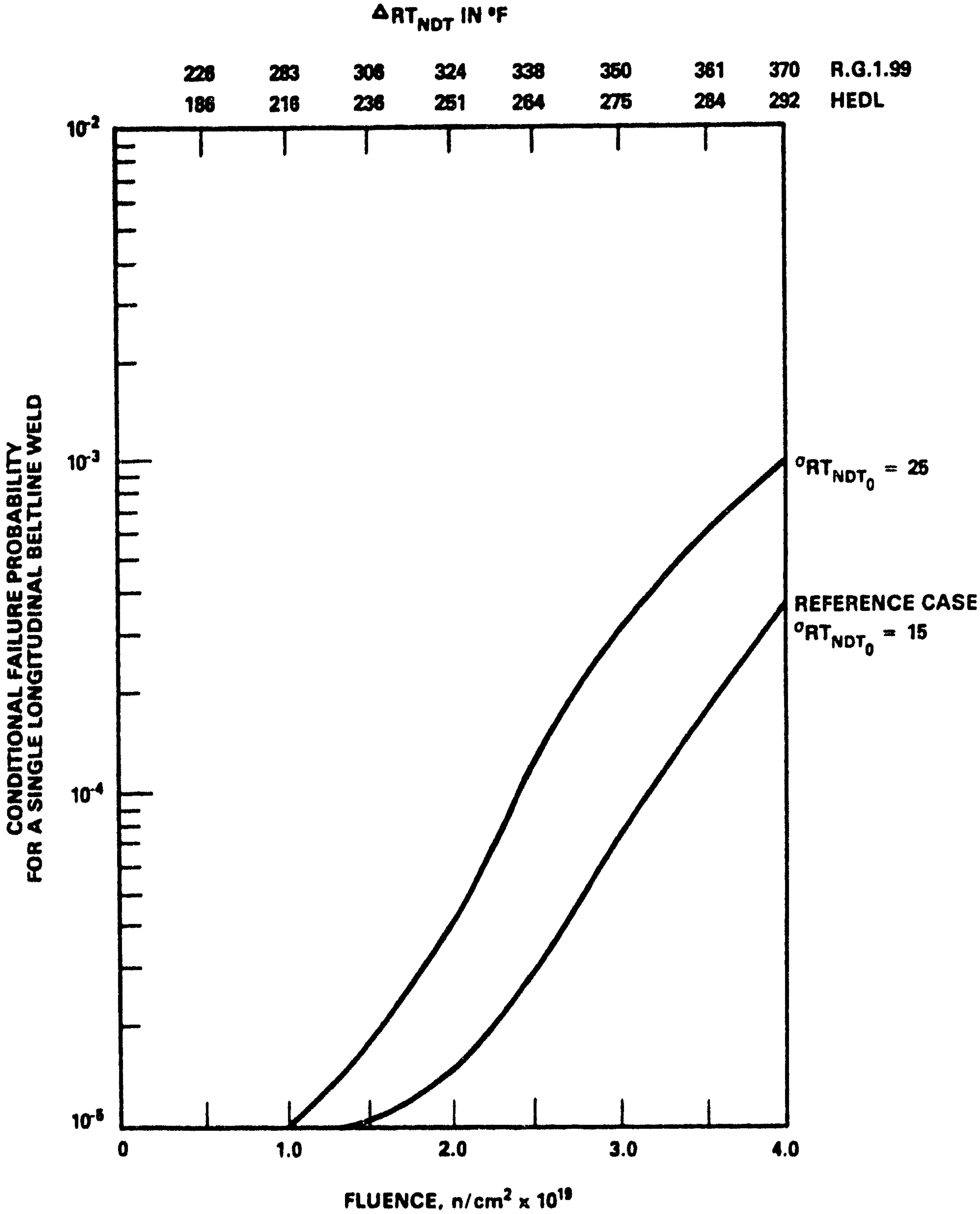


FIGURE H-14: INITIAL RT_{NDT} SENSITIVITY STUDY
MEAN Cu = 0.34% MEAN RT_{NDT_0} = 0 °F

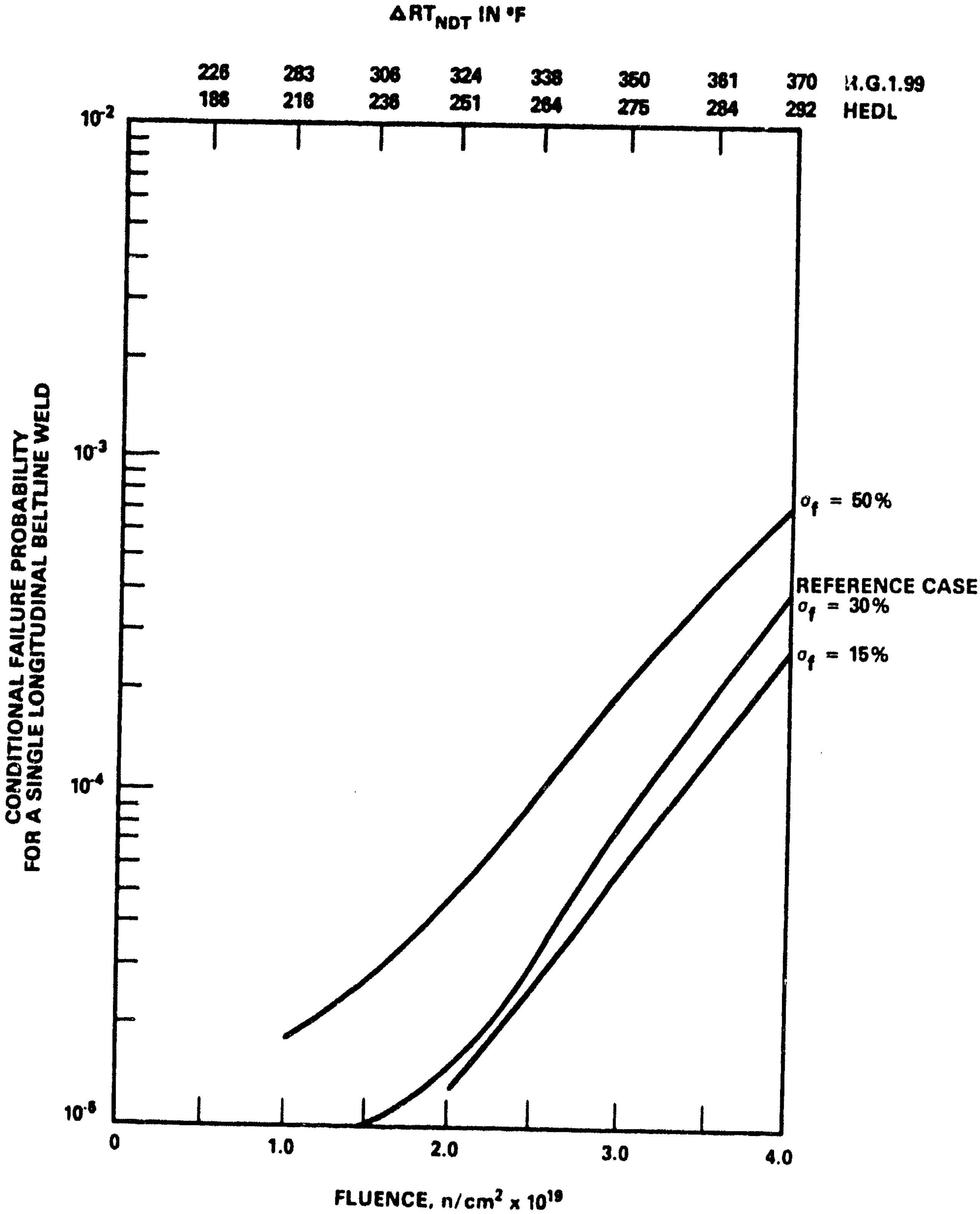


FIGURE H-15: FLUENCE SENSITIVITY STUDY
MEAN Cu = 0.34% MEAN RT_{NDT_0} = 0°F

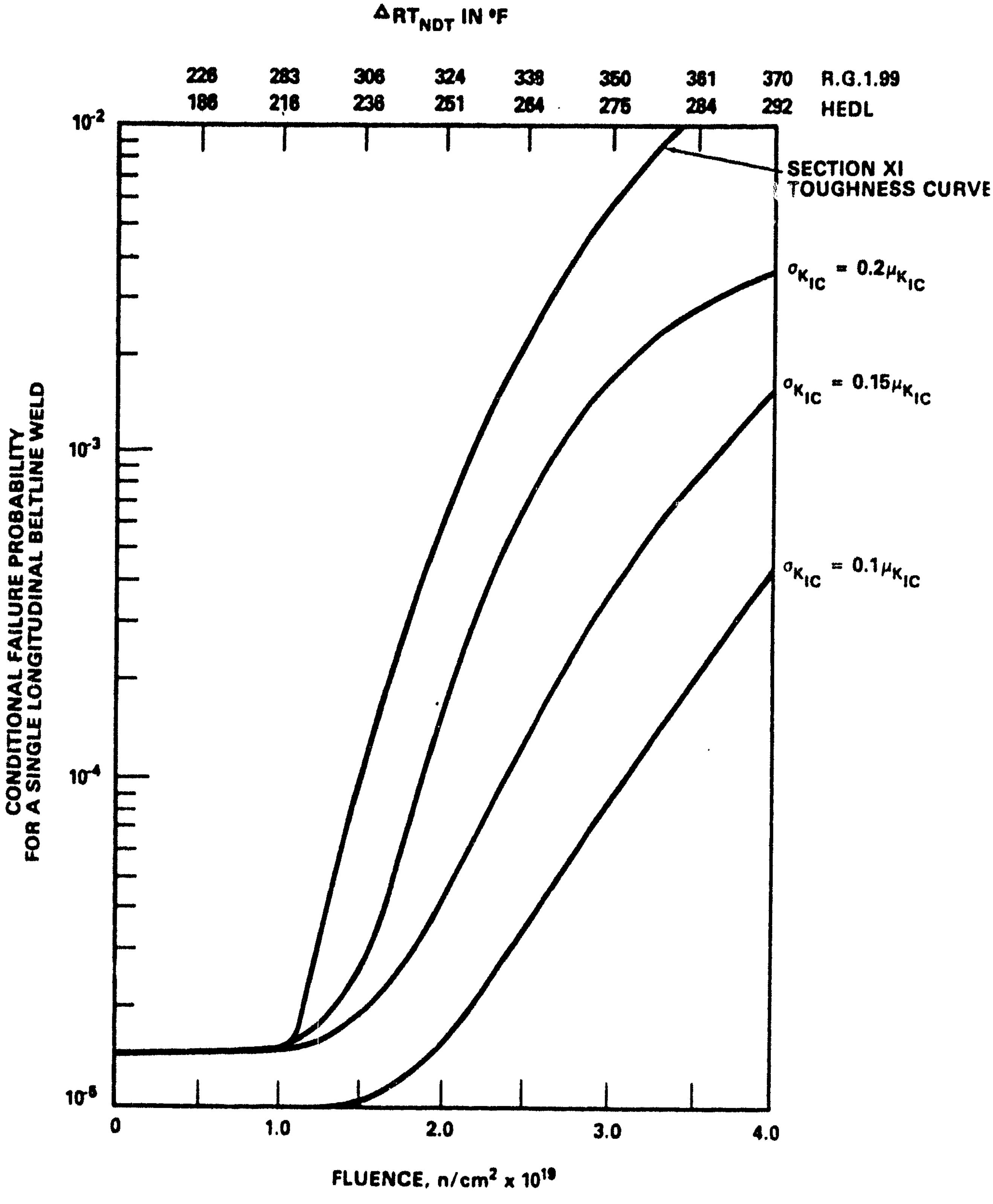


FIGURE H-16: FRACTURE TOUGHNESS DISTRIBUTION SENSITIVITY STUDY
MEAN Cu = 0.34%
MEAN RT_{NDT_0} = 0°F

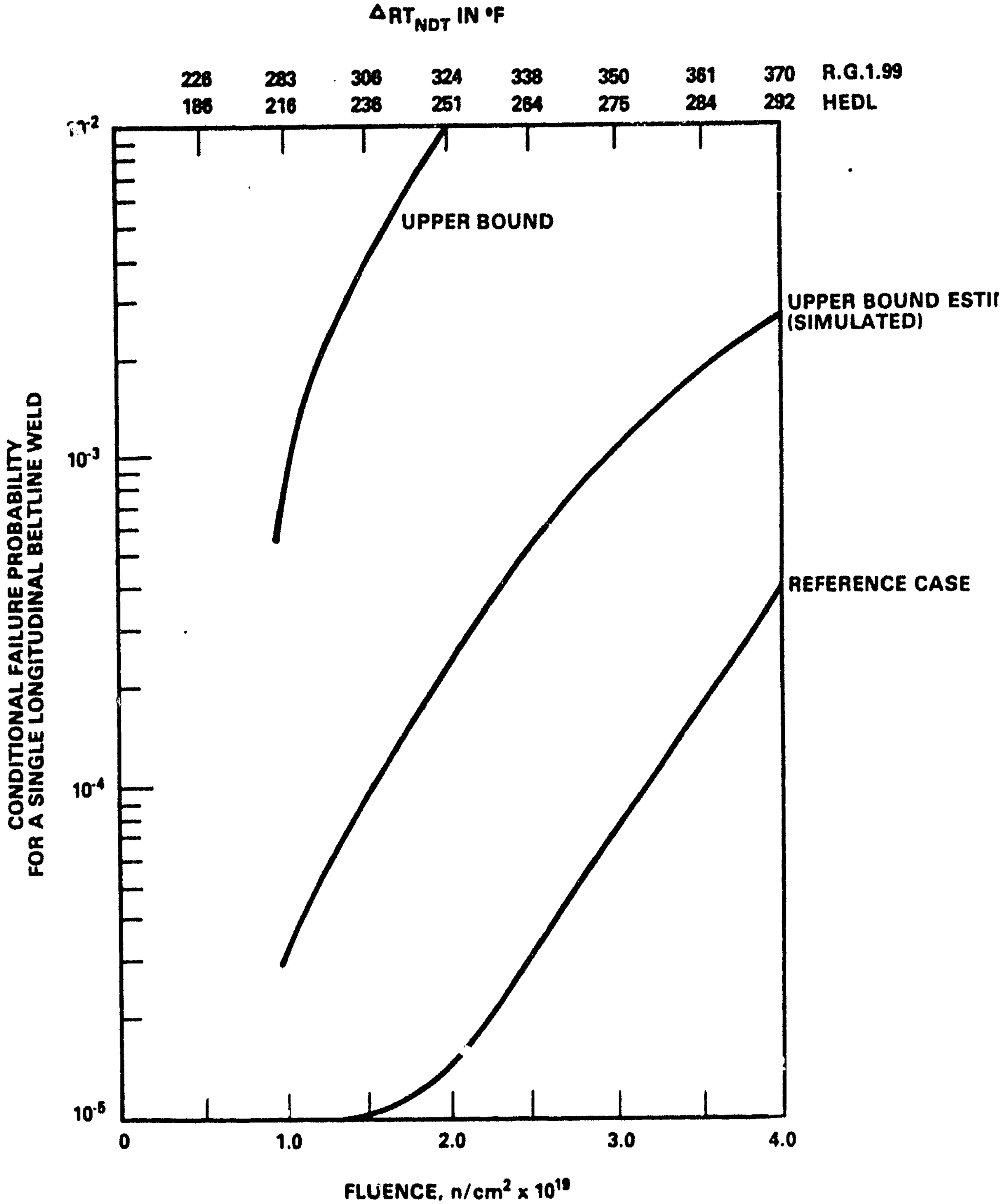


FIGURE H-17: SIMULTANEOUS INCREASE IN THE VARIABILITY OF THE RANDOM VARIABLES MEAN Cu = 0.34% MEAN RT_{NDT_0} = 0°F

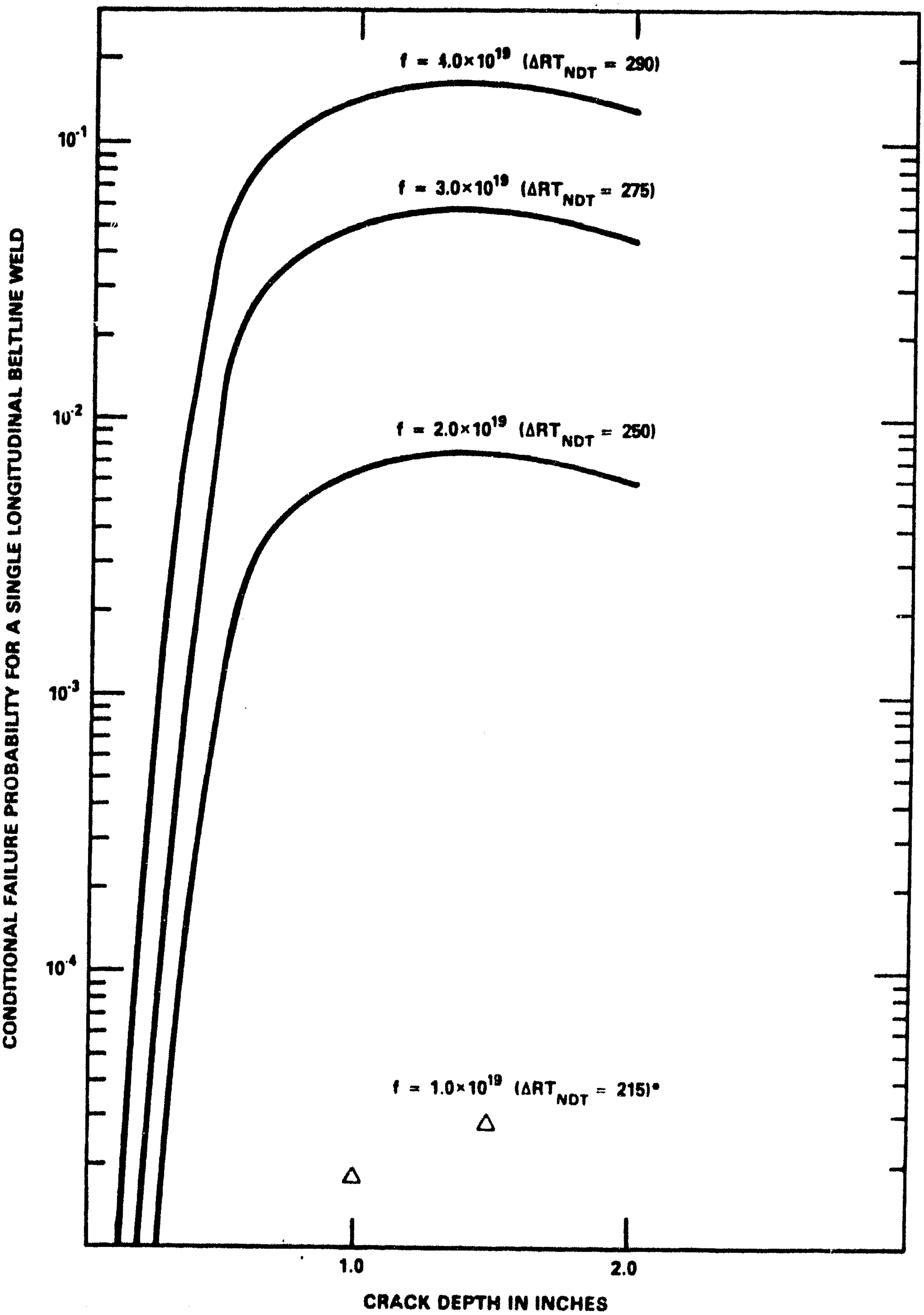


FIGURE H-18: CONDITIONAL FAILURE PROBABILITY P(F/a) FOR THE RANCHO SECO TRANSIENT MEAN Cu = 0.34% MEAN RT_{NDT_0} = 0°F

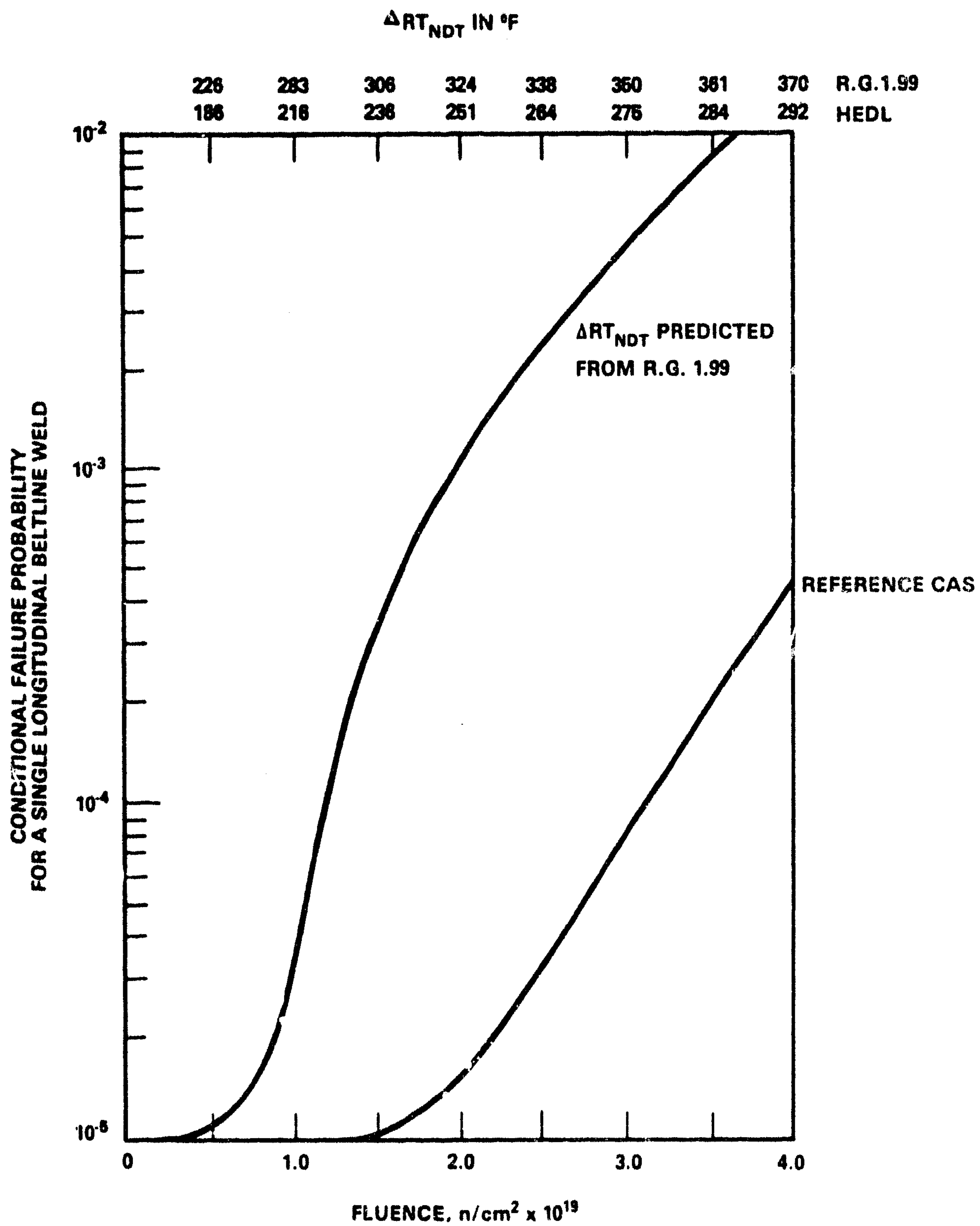


FIGURE H-19: SHIFT IN RT_{NDT} SENSITIVITY STUDY
MEAN Cu = 0.34% MEAN RT_{NDT_0} = 0°F

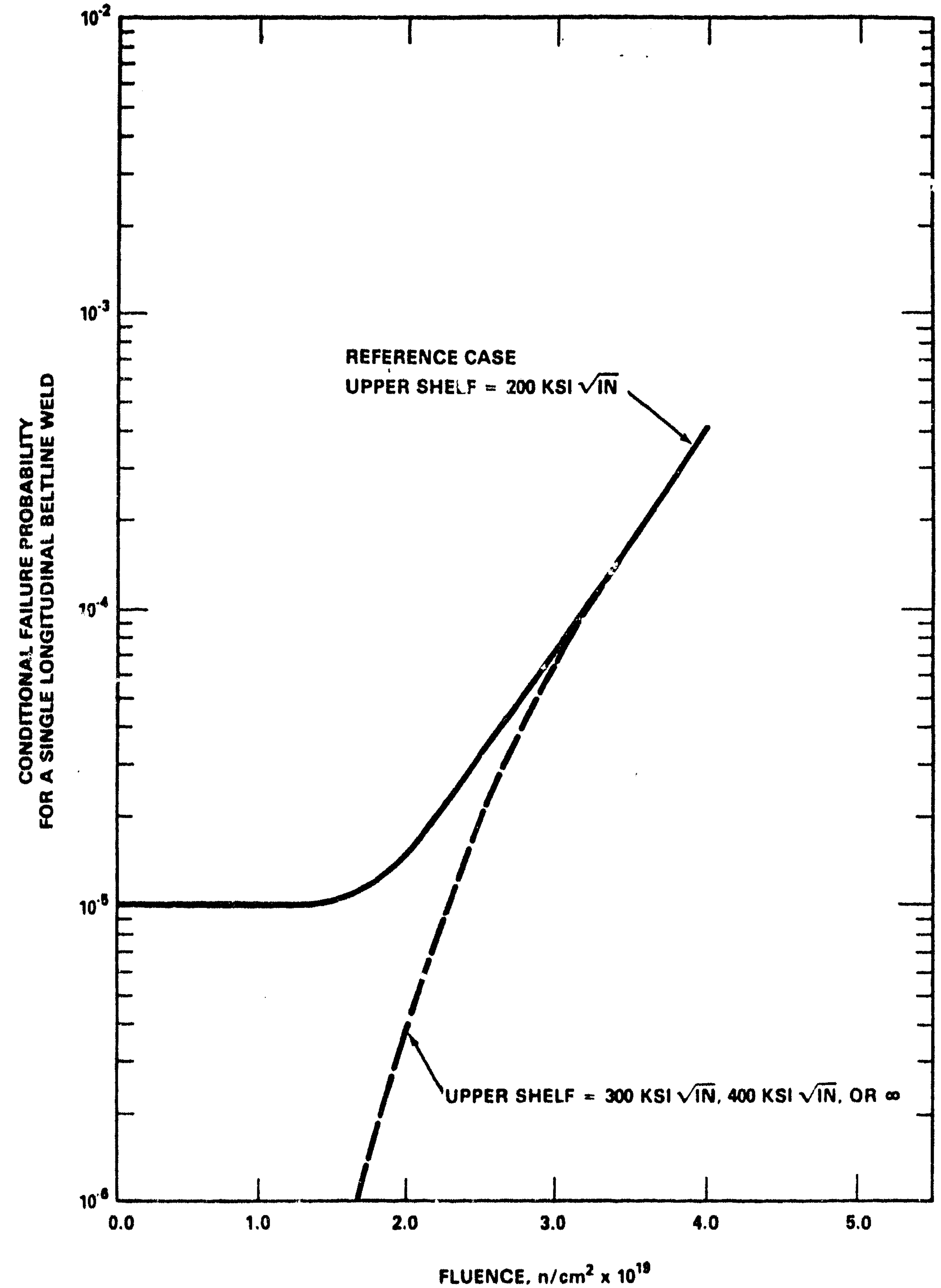


FIGURE H-20: UPPER SHELF TOUGHNESS SENSITIVITY STUDY
MEAN Cu = 0.34%
MEAN RT_{NDT_0} = 0°F

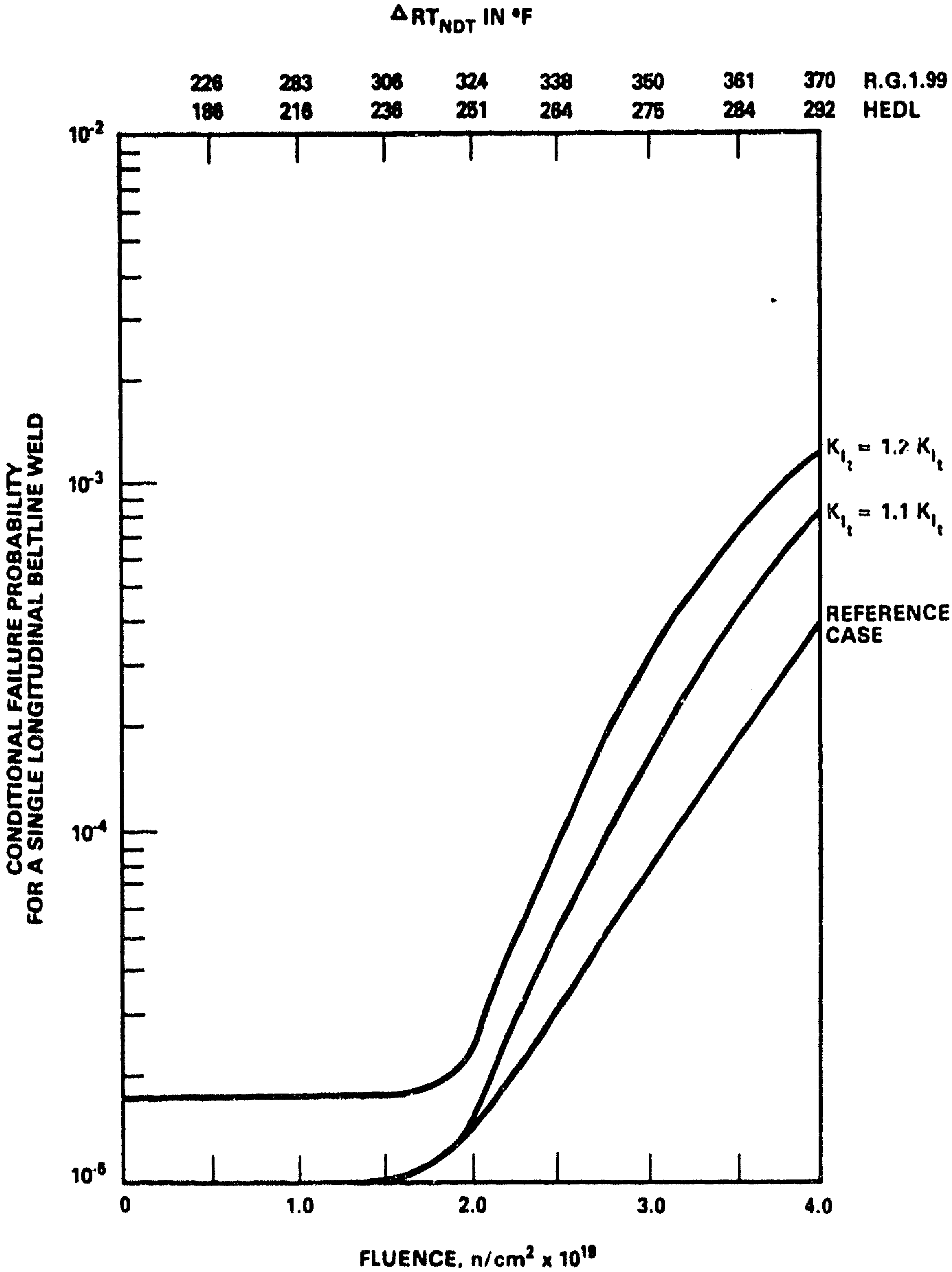


FIGURE H-21: CLADDING SENSITIVITY STUDY MEAN Cu = 0.34%
MEAN RT_{NDT_0} = 0°F

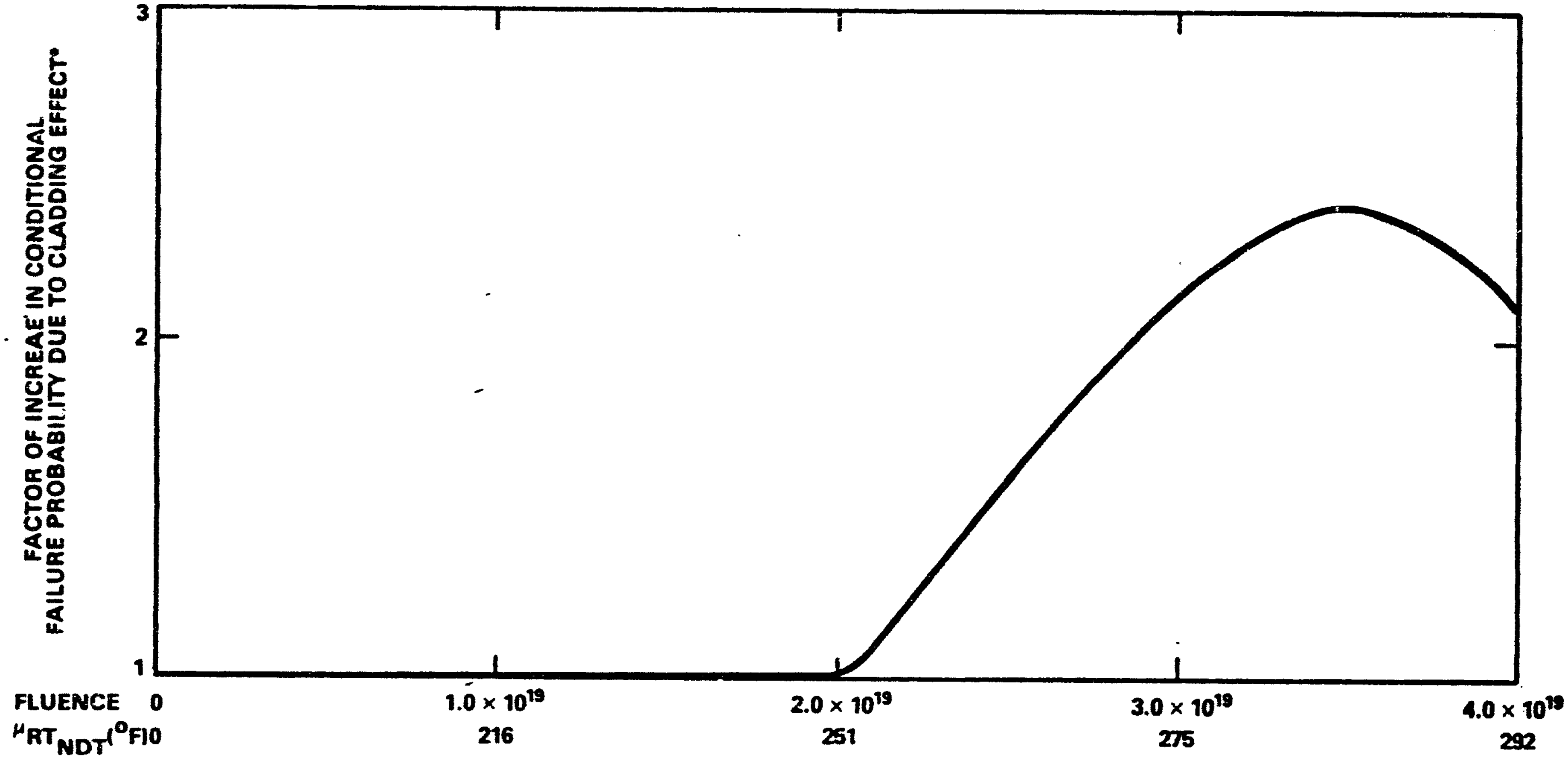


FIGURE H-22: FACTOR OF INCREASE IN CONDITIONAL FAILURE PROBABILITY DUE TO CLADDING FOR THE RANCHO SECO TRANSIENT

*BASED ON 10% INCREASE IN K_{I_t} AND RANCHO SECO TRANSIENT

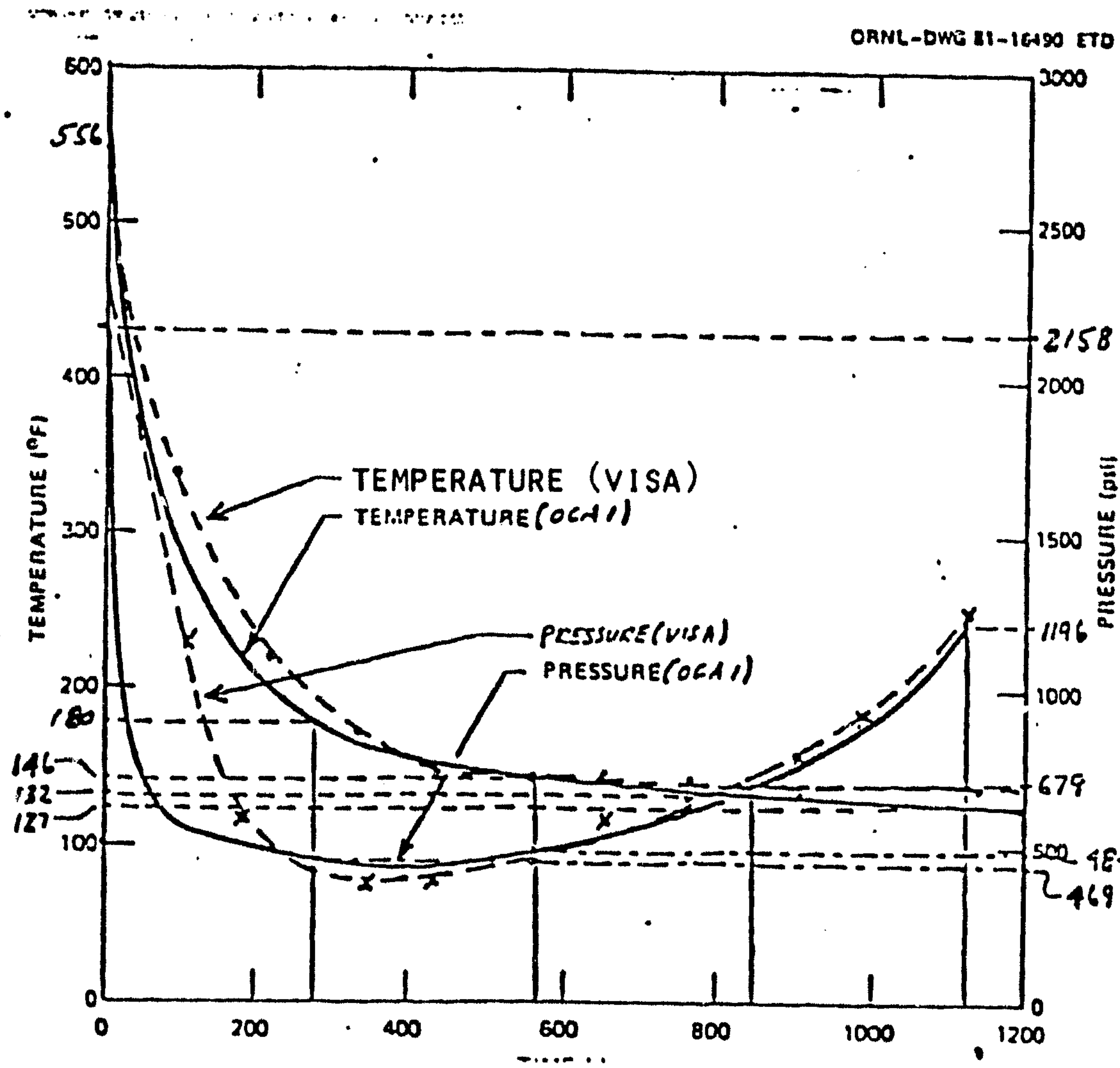


FIGURE H-23 IRT MAIN STEAM LINE BREAK PRESSURE AND TEMPERATURE TIME HISTORIES

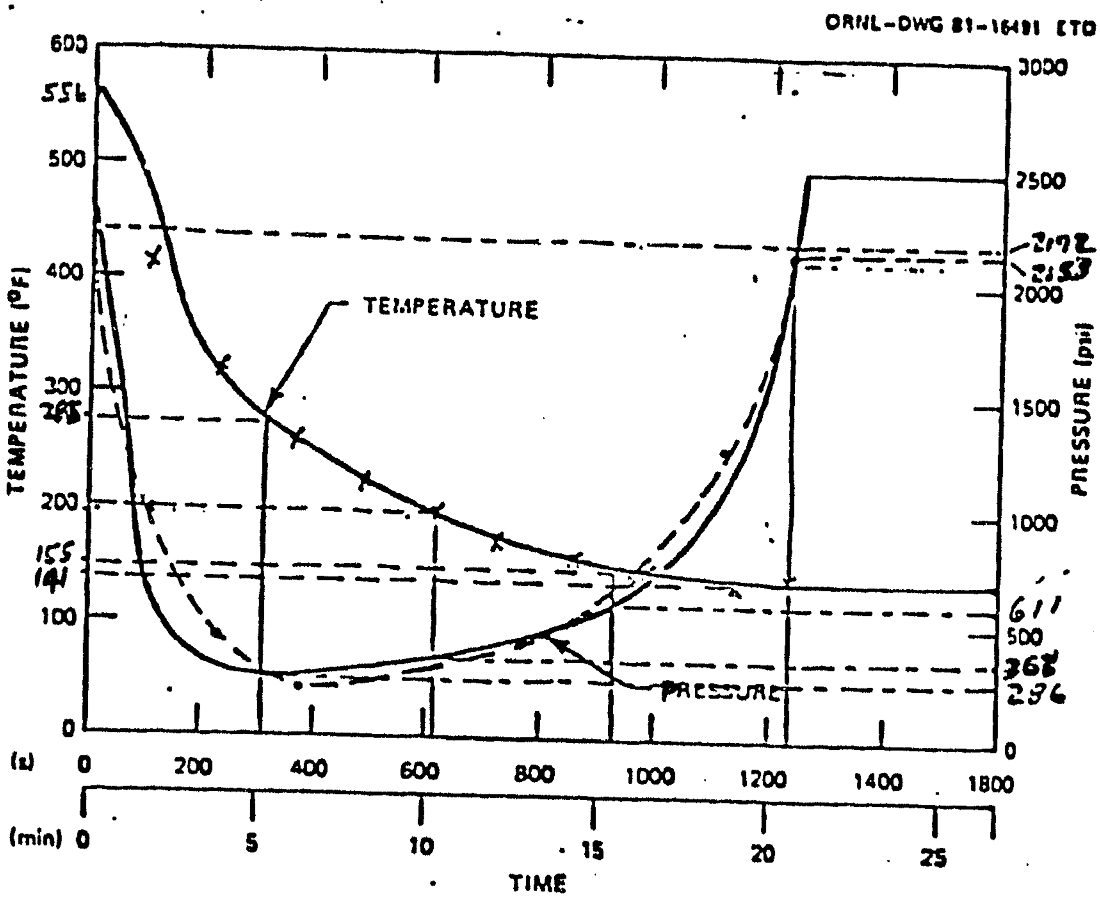


FIGURE H-24 TURBINE TRIP WITH STUCK OPEN BYPASS VALVES (SCRAM NORTH = 0.061 $\Delta k/k$) TEMPERATURE AND PRESSURE TIME HISTORIES

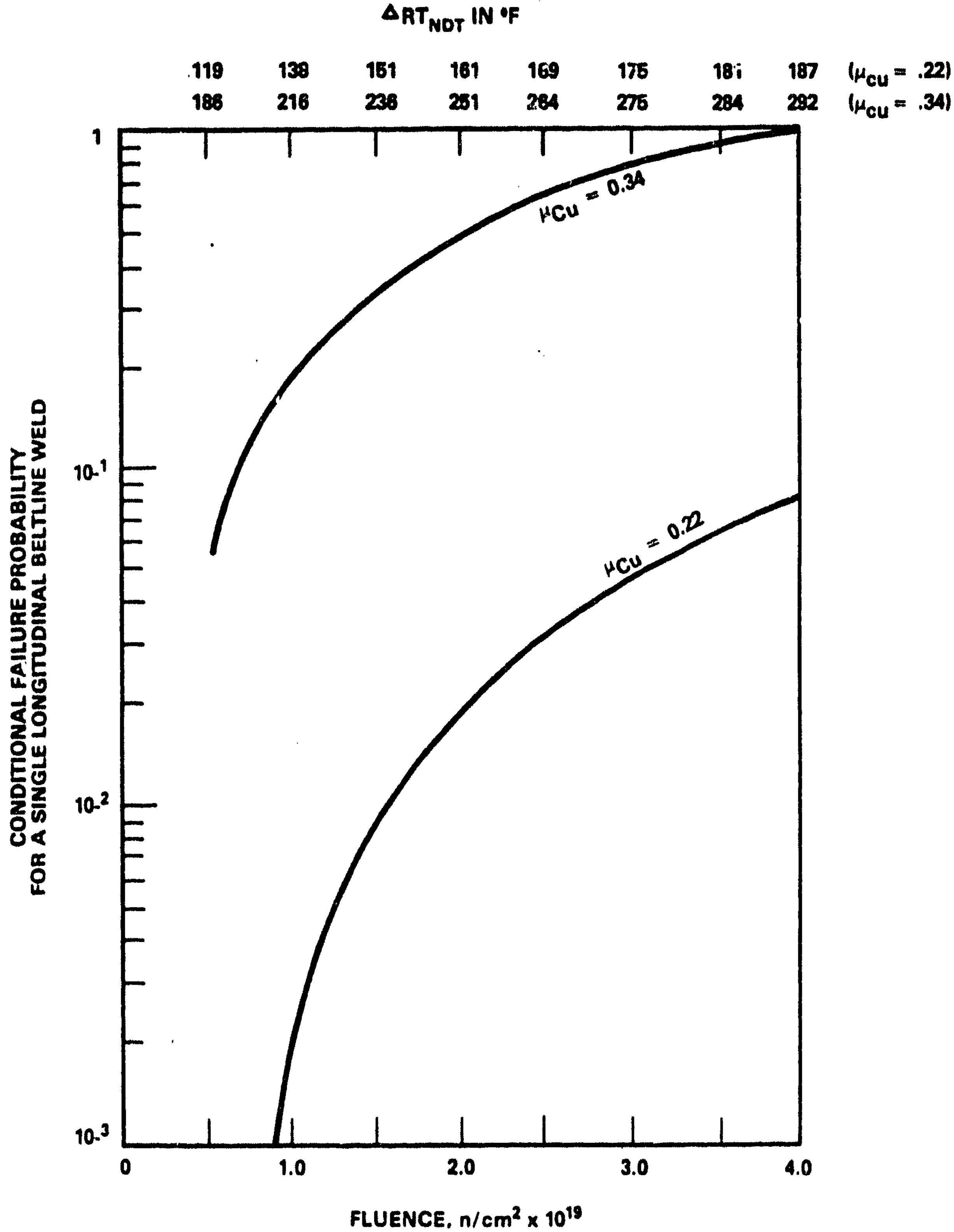


FIGURE H-25: CONDITIONAL FAILURE PROBABILITY FOR THE IRT MSLB MEAN RT_{NDT_0} = 0 °F

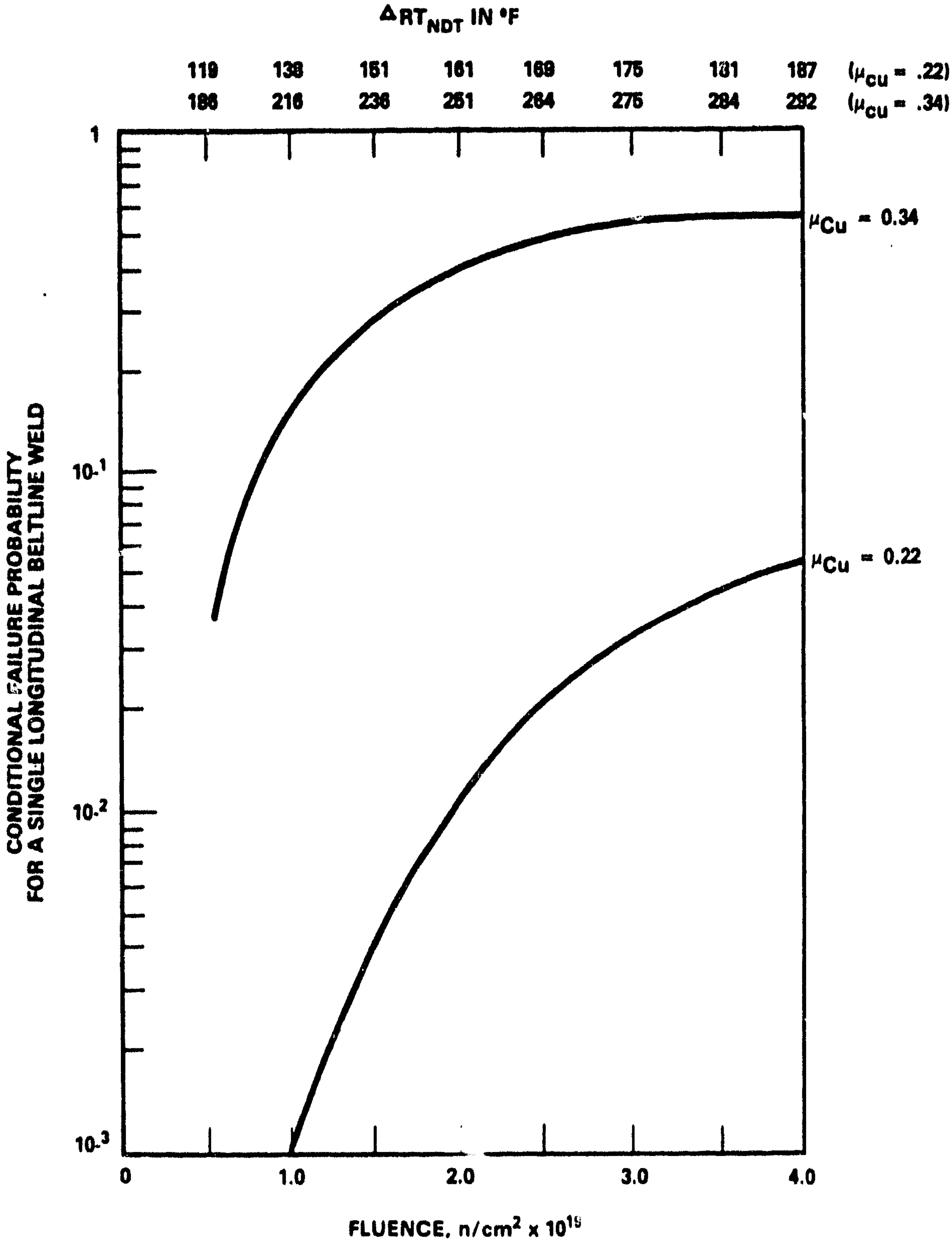


FIGURE H-28: CONDITIONAL FAILURE PROBABILITY FOR TURBINE TRIP/STUCK OPEN BYPASS VALVE MEAN RT_{NDT_0} = 0 °F

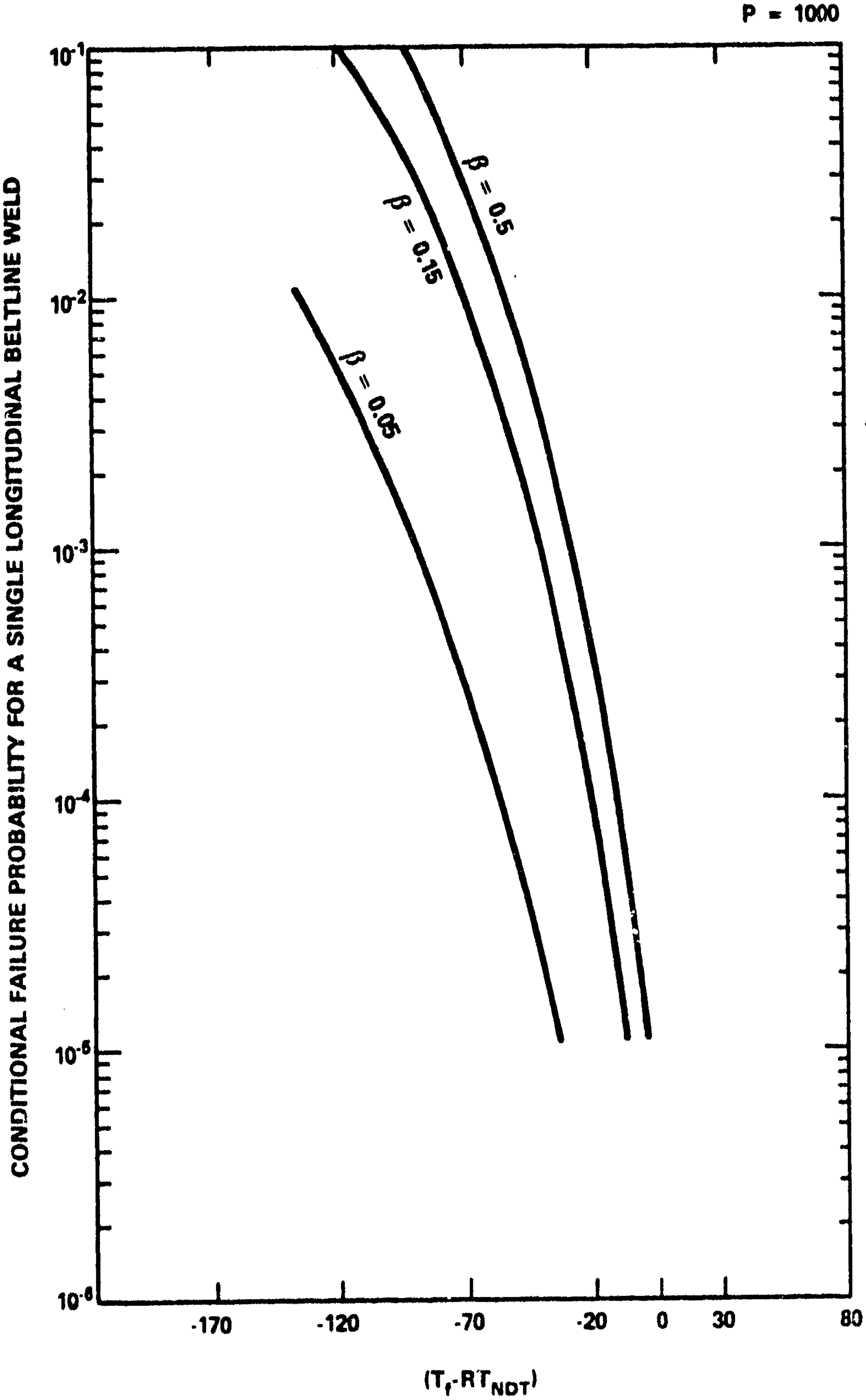


FIGURE H-27: SENSITIVITY OF CONDITIONAL FAILURE PROBABILITY TO T_f-RT_{NDT}

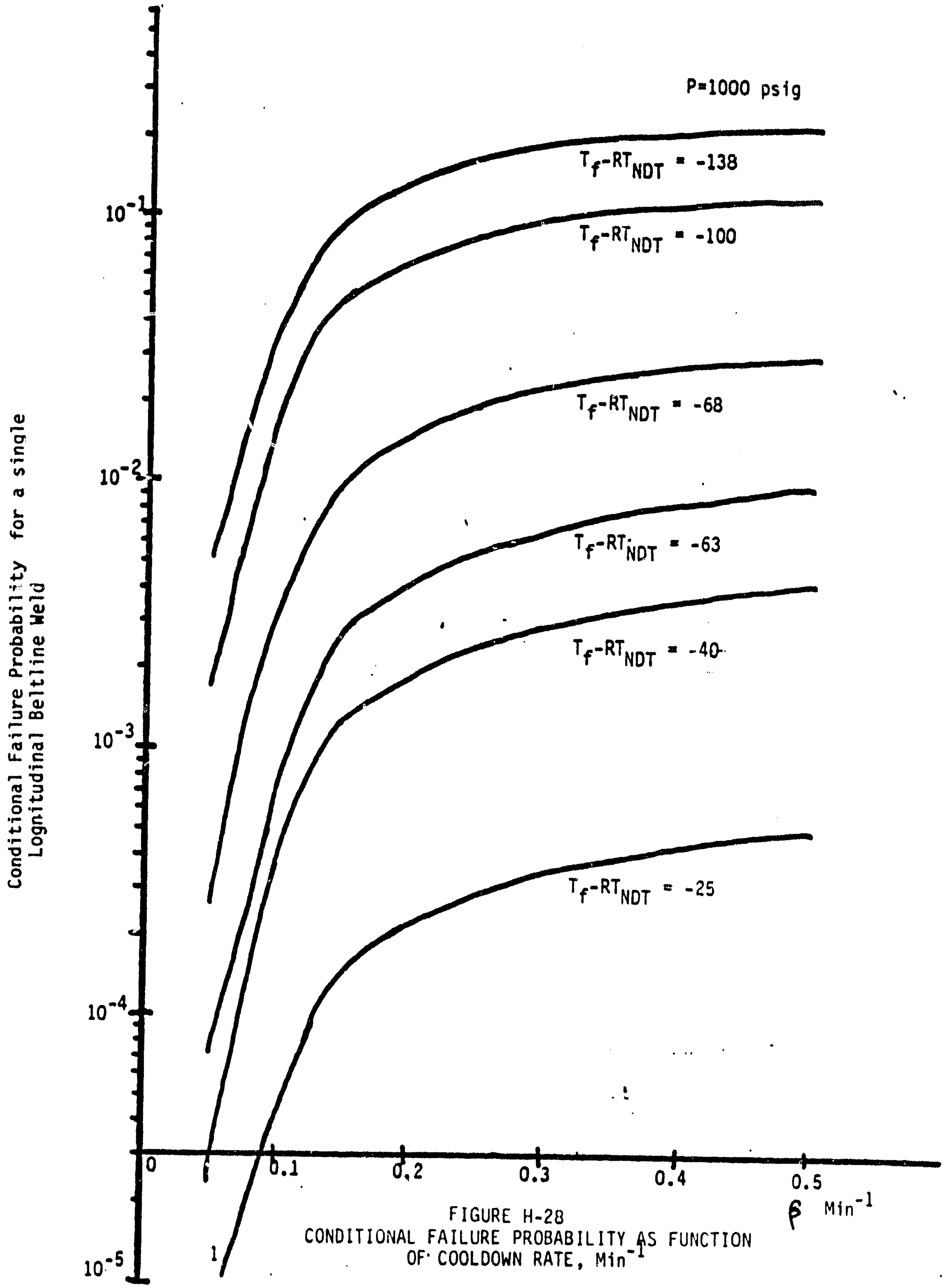


FIGURE H-28
CONDITIONAL FAILURE PROBABILITY AS FUNCTION OF COOLDOWN RATE, Min^{-1}

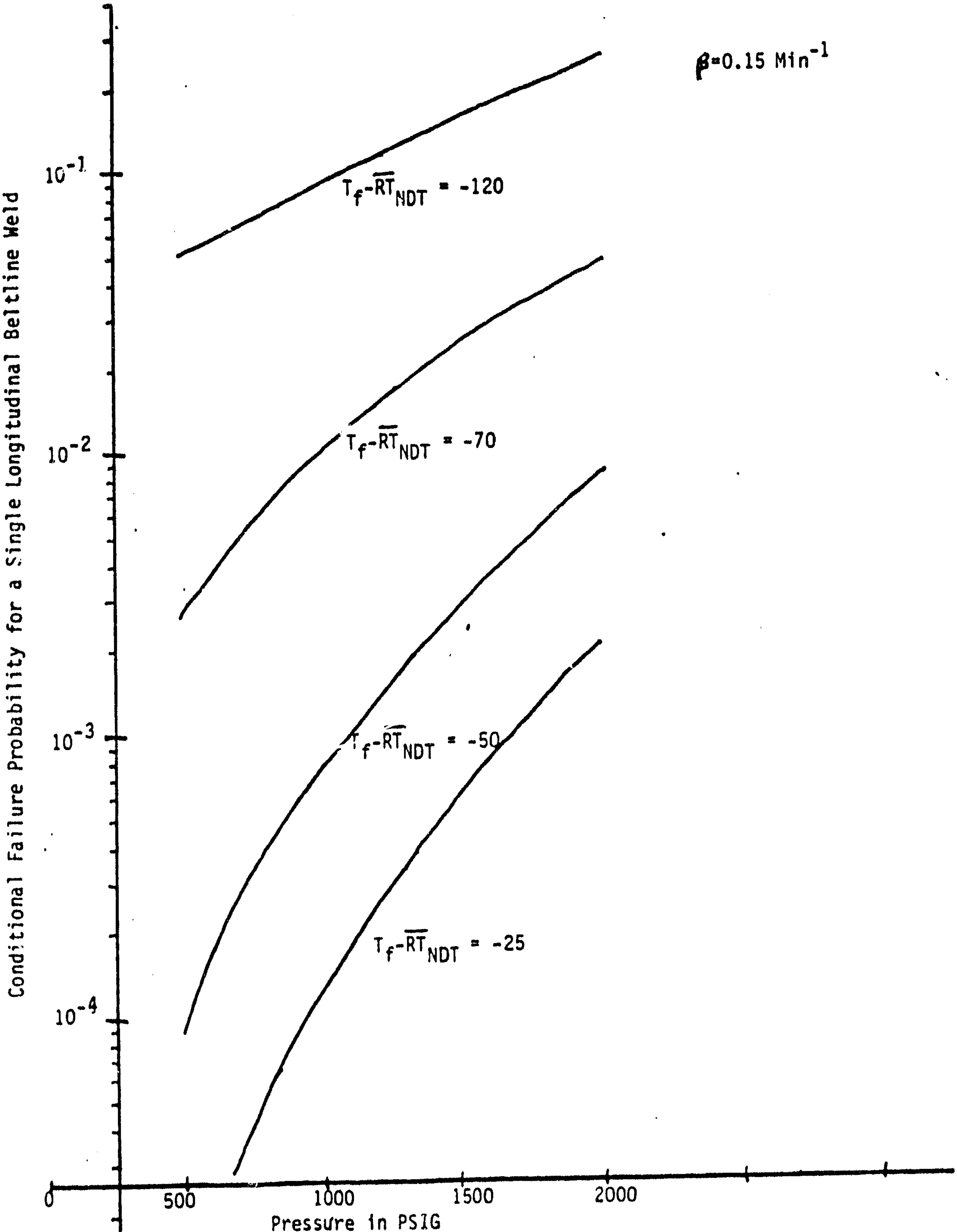


FIGURE H-29

CONDITIONAL FAILURE PROBABILITY AS A FUNCTION OF PRESSURE

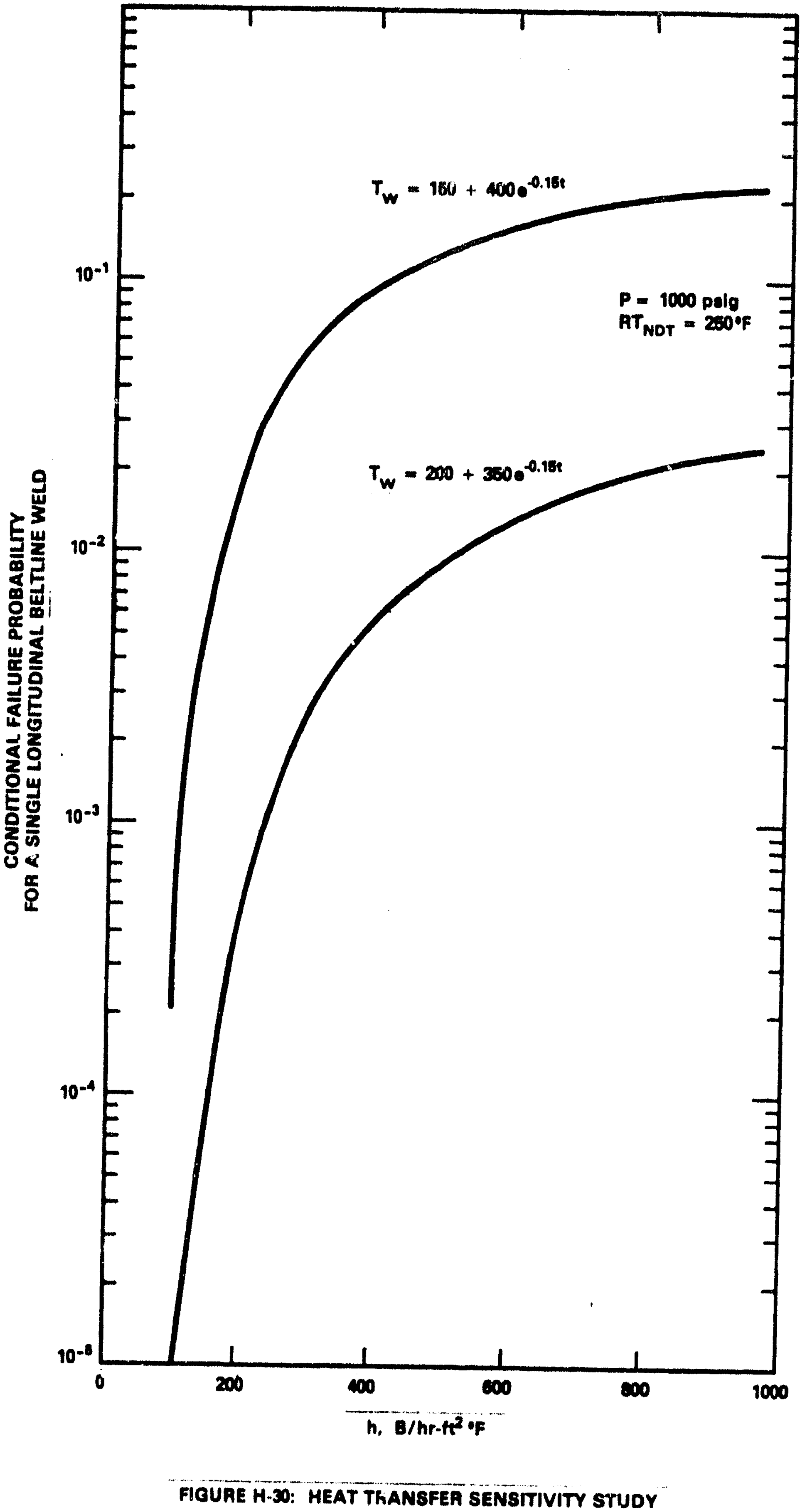


FIGURE H-30: HEAT TRANSFER SENSITIVITY STUDY

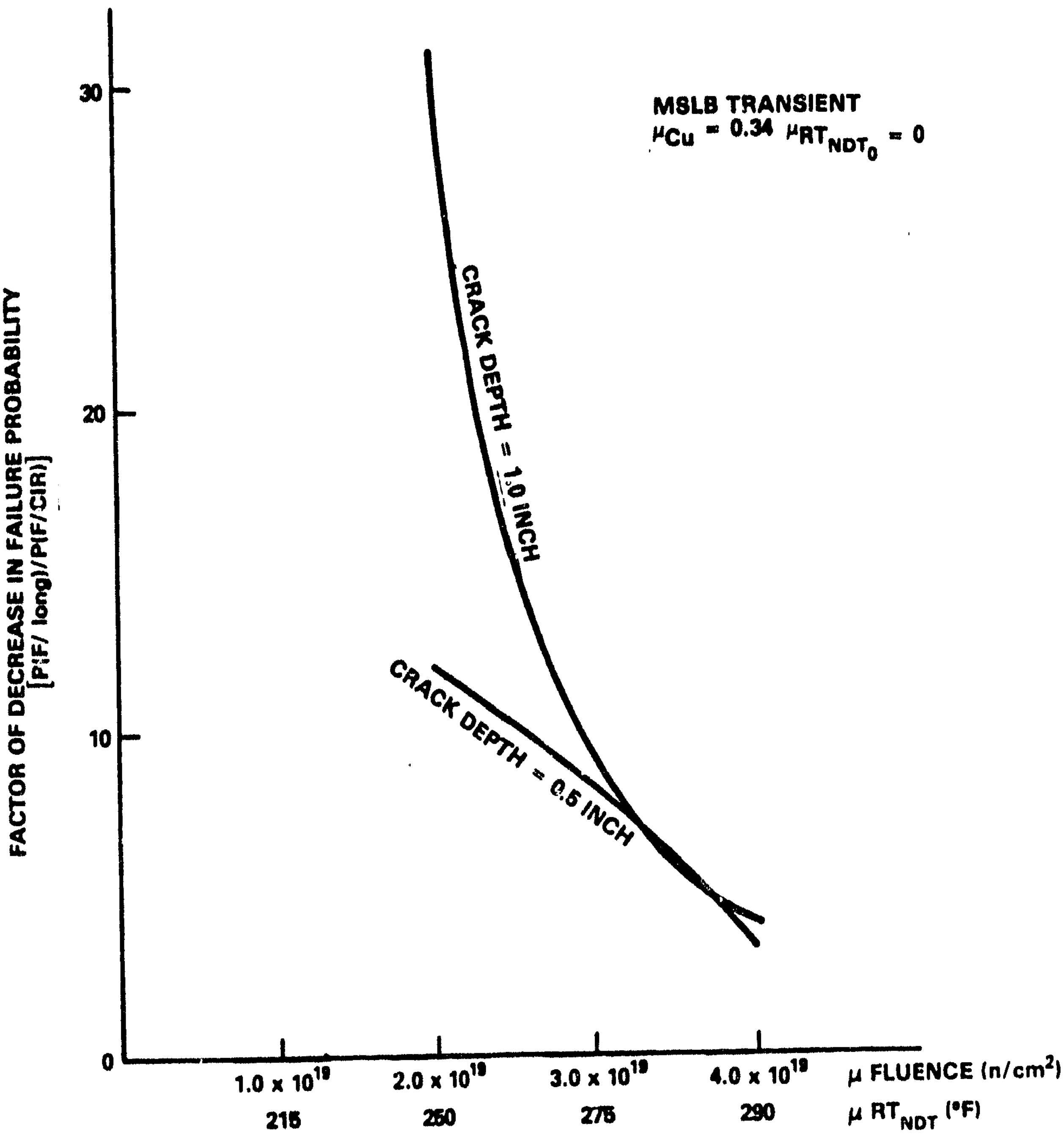


FIGURE H-31: CRACK ORIENTATION SENSITIVITY STUDY

APPENDIX I

FLUENCE RATE REDUCTION TO PWR PRESSURE VESSELS

I.1 Introduction

The NRC staff, as part of its evaluation of the Pressurized Thermal Shock (PTS) problem for PWR pressure vessels (PV), has undertaken a survey of domestic and foreign PTS experience and an evaluation of various fast neutron fluence rate reduction concepts (Ref. I.1). The survey included all three PWR vendors and the eight most affected PWR plants,* that is, those with significant vessel fluence. The staff found general agreement among those surveyed as to the techniques available for fast fluence rate reduction. The reason for this agreement is the generic similarities of the PWR plants of different manufacture and limited number of options which are considered viable.

The staff evaluation includes concepts for: (1) fluence rate reduction (by factors of 2 to 3) employing low leakage fuel loading, and (2) reductions (by factors of 10 or more) using select fuel assembly replacement on the core periphery with nonfueled assemblies containing stainless steel rods. The impact of implementing any of these schemes is so plant dependent that it was not possible to do more than estimate the impact on the total peaking factor as part of this study.

The low leakage fuel loading schemes are also characterized as an "in-out" fuel loading scheme in contrast to the usual "out-in" loading scheme. The "in-out" ("out-in") refers to fuel assembly movement during refueling from the core interior (periphery) to the core periphery (interior). In a low leakage fuel loading scheme, therefore, twice or thrice burned fuel assemblies (or even poisoned fuel assemblies) are placed on the core periphery. In our evaluation

*Fort Calhoun, San Onofre, Oconee-1, Maine Yankee, Calvert Cliffs-1, H. B. Robinson-2, Turkey Point-4 and Three Mile Island-1.

We used stainless steel rods in the nonfueled assemblies. Other choices, however, could be made for the stainless steel rods.

This report also includes a survey of foreign reactor experience with respect to fluence rate reduction to the pressure vessel.

I.2 Survey of Licensees, Owners' Groups, and Vendors

The staff visited Combustion Engineering (CE) and Westinghouse (W). Lengthy discussions were held with cognizant personnel in reactor physics, thermal-hydraulics, fuel management, and licensing. A visit could not be arranged with Babcock & Wilcox (B&W) so that information was obtained with a telephone conference call. The vendors were asked to discuss (1) the reduction of peak and longitudinal weld steam fluence accumulation rates by factors of 2, 3, 5, or 10, (2) the corresponding impact of fluence rate reduction schemes, and (3) the estimated cost of implementation of various schemes. The same questions that we asked the vendors were also asked the licensees of the eight most affected plants through the NRC project managers (Ref. I.2). These licensees had little information to offer and, generally, referred us to the respective vendors.

Limited cost estimate data was obtained from our survey. Low leakage fuel loading schemes (in-out) may result in overall cost savings to licensees because of the benefits of extended cycle operation which could accompany such schemes. However, extremes of low leakage loading schemes could cost from 1 to \$5 million. Replacement of fuel assemblies with stainless steel rodded assemblies on the core periphery could cost up to \$20 million per year due to derating plus a one-time engineering cost of \$15 to \$25 million.

I.3 Survey of Foreign Experience

Several foreign reactor plants with radiation-induced pressure vessel embrittlement have been modified or modifications are planned. Such modifications include raising the temperature of the high pressure injection water and reducing the fluence accumulation rate (i.e., lowering the fast flux to weld seems or plate material of the vessel). In the following we will deal with modifications related to fluence rate reduction to the pressure vessel. The information gathered was the result of a questionnaire directed to several countries around the world in the summer of 1981 shortly after the PTS task force was formed by NRC (Ref. I.3).

I.3.1 Finland

Loviisa-1 The Soviet built, 420 MWe Finnish reactor was put into operation in 1977 (Ref. I.4). The loading consists of about 360 hexagonal fuel assemblies. The reactor had operated for about 3 years when it was determined that the radiation-induced weld seam embrittlement was higher than originally estimated. In 1980, with only 3 years of operation, the estimated nil-ductility transition temperature increase was 76°C. The originally predicted increase for 40 years of operation was 85°C.

It was decided to remove 36 fuel assemblies on the periphery of the core and replace them with hollow steel rods in the hexagonal shroud identical to that of the fuel assemblies. The assemblies that were removed represented 10% of the core inventory. However, there was no reduction in the power level because the plant had adequate thermal margin. Due to the hexagonal shape of the fuel assemblies the azimuthal flux distribution was fairly uniform varying from .73 to 1.00. The peak fast neutron flux decreased by a factor of about .7. The new flux peak appeared in the location of the previous minimum, reduced by a factor of 2.8 from .73 to about .25 (estimated nonpeak value between .22 to .30). This modification along with an increase in the temperature of the emergency injection water and changes to the emergency operating procedures is expected to provide adequate protection for the remaining life of the plant.

Loviisa-2 This is a sister plant to Loviisa-1 that was put into operation in 1980. The Finns could not decide from cost-effectiveness considerations whether a modification similar to that for Loviisa-1 should have been implemented during the first cycle in Loviisa-2. Nevertheless, the same modifications could be made in a later cycle.

No information is available to us on surveillance programs, neutron transport calculations, uncertainties, or specific fluence values.

I.3.2 Germany (Obrigheim and Stade)

The PWR plants in the Federal Republic of Germany (FRG) have pressure vessels with only horizontal weld seams, hence, the azimuthal position of the peak fluence is immaterial* and the concern is in the irradiation of the base metal (Ref. I.8). An extensive surveillance program has been instituted in all FRG PWRs. Present estimates indicate that at the end of the 40 EFPY of operation there would be excessive irradiation of the pressure vessel of Stade and Obrigheim and that fuel assembly substitutions to lower the projected peak fluence would be needed. These reactors are very similar to Westinghouse plants, hence, we surmise that the azimuthal distribution has localized peaks. Becuase there is no discussion of potential consequences we assume that element substitution will be of a limited extent with no power derating. The stade reactor has been using a low leakage loading (Ref. I.6) for the last few cycles. The estimated end of pressure vessel life fluence for Obrigheim is somehwat higher than that estimated for Fort Calhoun and for Stade is considerably lower than most American pressure vessels (Ref. I.5). The Federal Ministry of Internal Affairs of Germany in its August 10, 1981 reply to the NRC questionnaire indicated that nonfueled assembly replaced was contemplated (Ref. I.7) for these two reactors.

I.3.3 France

Recent information received from the French (Central Service for the Safety of Nuclear Installations) (Ref. I.8) indicates that a program for the study of material embrittlement was instituted about 10 years ago. This program has only recently been expanded to include pressure vessel dosimetry. No definitive plans are known at this time for pressure vessel fluence rate reduction modifications.

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*The PWR at Gundremmingen, currently under construction, has longitudinal welds.

I.3.4 Other Countires

Replies to the NRC questionnaire have been received also from Italy, Spain, Sweden, Korea, and Japan. However, none of the operating utilities have taken any steps to lower fluence rate to the pressure vessel. All show awareness of the problem. Surveillance programs have been established in Sweden and Japan.

I.4. Evaluation of Fast Fluence Rate Reduction Schemes

In order to assess independently a number of fluence rate reduction schemes, an evaluation was performed for the staff by its consultants at BNL (Ref. I.9). From the eight most affected PWR reactors, three plants, Oconee-1, Fort Calhoun, and Robinson-2 (one from each PWR vendor), were selected for the staff evaluation. These plants were selected because of the availability of plant-specific data and the relatively large vessel fluence. Table I-1 presents some pertinent information concerning these plants. Included in the table are the vendors' and our consultant's estimate of the present and end of vessel life fluence.

The approach taken by the staff in performing this evaluation was:

(a) To use the transport theory code DOT 3.5 to calculate the fast fluence to the pressure vessel. The calculations were two-dimensional and used 16 neutron energy groups. The BNL methods have been benchmarked to a number of tests and are comparable to those used by the vendors.

(b) To use as-built dimensions, material compositions, and measured values of the neutron source to evaluate H. B. Robinson-2, Oconee-1, and Fort Calhoun.

(c) To calculate for each of these plants the (1) current values of the peak fluence at the longitudinal welds, (2) projected value of the peak fluence to the end of 32 effective full power years (EFPY), (3) fluence attenuation through the pressure vessel, (4) fluence time spectra at various wall thicknesses, (5) pressure vessel flueoce azimuthal distribution, and (6) end of vessel life fluence value for various fluence rate reduction schemes.

(d) To evaluate the impact of these modifications in terms of the potential increase in the total peaking factor.

(e) To compare the staff's calculations to similar calculations from the licensee or vendor when possible.

Since the fluence to the pressure vessel is caused primarily by the fast neutrons in the peripheral fuel assemblies, schemes for reducing fluence accumulation rate to the pressure vessel fall into two main classes. The first class is designed to lower the neutron leakage from the periphery of the core by lowering the power level of the peripheral fuel assemblies. The second class is designed to lower the fluence rate to the pressure vessel by placing a thick metal shield between the core periphery and the pressure vessel. This second class of fluence rate reduction schemes will not, however, be discussed further because of the lack of space between the core and vessel to accommodate large thicknesses of metal.

The first class of fluence rate reduction schemes consider the lowering of the peripheral fuel assemblies' powers by (1) using low leakage fuel loadings, and (2) removal of fuel assemblies and replacement with assemblies containing stainless steel rods. Note that the use of nonfueled assemblies contains elements of both classes of fluence rate reduction schemes. The power of the reactor could also be lowered in order to reduce the peripheral assemblies' powers. This power derating was not considered in our evaluation. Instead, the assumptions in the staff evaluation are (1) the total power of the reactor is constant, (2) the shape of the power distribution remains the same from the periphery toward to center of the core, (3) the maximum linear heat generation rate is assumed constant, and (4) the core flow is assumed constant.

Since the PTS problem solution will be plant-specific, no attempt was made to optimize core fuel loading patterns on a cycle-by-cycle basis to lessen the impact on fuel cycle economics or to assess the impact on normal operation, transients, and accidents. Only a rough estimate was made of the impact in terms of a potential increase in the total peaking factor.

Some of the plant-specific factors include, among others things, the location of the weld seams in the pressure vessel, the cooper and nickel content of the weld seams, the core power and size, the peripheral fuel location, the presently accumulated fluence, the fuel management scheme presently employed, and the location of the peak fluence on the vessel. An example of one of these plant-specific items is the weld seam location for the three plants. Figure I-1 shows the three weld seams at Oconee-1 folded onto a quarter core. The Fort Calhoun weld seams are shown in Figure I-2 folded onto an eighth of the core. Figure I-3 shows the weld seams at H. B. Robinson-2 folded onto an eighth of the core.

Calculations were performed by BNL for the three plants to evaluate the low leakage fuel loadings and peripheral element replacement with assemblies containing stainless steel rods. Similar results were obtained for all three plants. The conclusions of our analysis agreed with statements made by the parties we talked to in our sruvey. These BNL calculations will be reported in a forthcoming BNL-NUREG report (Ref. I.13).

Table I-2 shows some results from the BNL calculations for Oconee-1 for three cases in which the ratio of the peripheral assembly power to the core average power was varied. One should roughly assume the 0.910 power ratio to be representative of normal out-in fuel assembly loading, the 0.527 ratio to represent in-out low leakage fuel loading using partially burned or poisoned assemblies, and the zero power ratio to represent peripheral fuel assemblies for which the fission source was artificially zeroed (not achievable in practice). Table I-3 shows the same results in a different format giving the fluence for the remaining 28 EFPY in terms of the relative fluence rate to the peak longitudinal weld seam for the original out-in fuel loading. Two additional cases are also shown in Table I-3. Both of these cases are representative of fuel element removal and replacement with stainless steel assemblies. In one of the cases the stainless steel rods are spaced in the same way as fuel rods while in the other case the rods are more closely packed with additional stainless steel rods. Both Tables I-2 and I-3 clearly demonstrate the fluence rate reduction factors that are possible for the two fluence rate reduction schemes. Table I-3 further demonstrates the effectiveness of including stainless steel rods in the replacement assemblies.

Figure I-4 shows in graphical form the fluence rate reduction factor for the remainins 28 EFPY (data from Table I-2) for the original peak vessel fluence location as a function of peripheral fuel assembly power. The results are linear as a consequence of our assumptions and modeling.

The fase neutron flux attenuation through the pressure vessel is shown in Figure I-5. The curve is nonlinear but shows more than a factor of 10 reduction in flux on the outside wall of the pressure vessel. This figure allows the estimation of fluence rate accumulation at various positions within the pressure vessel when the fluence rate is known on the inside wall of the vessel.

Figure I-6 provides a summary of the staff's evaluation for Oconee-1 showing results for a number of fluence rate reduction schemes as a function of effective full power years of operation. Shown in the figure are the licensee's FSAR value as well as the vendor's (B&W) estimate of the vessel fluence for the current in-out low leakage fuel loading scheme. Note that the staff's evaluation for low leakage fuel loadings closely agrees with the B&W results and both results are about half of the FSAR estimate. Three other evaluations for element removal and replacement with stainless steel assemblies are shown in the figure. Pattern 1 refers to the removal and replacement with stainless steel elements of the entire peripheral row of elements. Pattern 2 was chosen so that the fluence rate to the weld seams could be reduced to a minimum number of assembly substitutions. Pattern 3 was chosen to reduce the power peak at the core flats caused by Pattern 2. For Patterns 1, 2, and 3 fuel assembly removals and substitutions numbered 40, 20, and 32, respectively.

Figure I-7 provides a summary similar to that of Figure I-6 of the staff's evaluation for Fort Calhoun. Shown in the figure are the licensee's FSAR value as well as the vendor's (CE) estimate of the vessel fluence for the current fuel loading scheme. Note that the staff's evaluation for the current fuel loading scheme is in reasonable agreement with the CE results and both results are about a factor of 2 larger than the FSAR estimate. Staff results are also shown for two in-out low leakage schemes; in one the peripheral power is 0.41 of the core average power and in the other the peripheral power is zero. Three additional cases are shown in the figure for fuel assembly removal and replacement with stainless steel assemblies. The three cases are for the removal and replacement of 40, 24, and 16 assemblies.

Figure I-8 provides a summary similar to that of Figure I-6 of the staff's evaluation for H. B. Robinson-2. Shown in the figure is the vendor's (W) estimate of the vessel fluence for the current fuel loading scheme. Note that the staff's evaluation for the current fuel loading scheme is a factor of about 1.14 larger than the vendor's evaluation. Staff results are shown for the out-in loading scheme for which the peripheral power to the core average power ratio is 0.89 as well as for two in-out loading schemes for which this ratio is 0.45 and zero.

Table I-1 summarizes the staff survey and evaluation of the peak vessel fluence for the three plants for various schemes and the associated decrease in the total peaking factor. The table also gives present and end-of-life estimates of vessel fluence by BNL, the vendors, and the FSAR value for the current fuel management scheme.

I.5 Cost Estimate for Fuel Assembly Substitution to Lower Neutron Flux to the Pressure Vessel

Cost estimates of fuel assembly substitution require consideration of plant-specific factors such as, azimuthal flux distribution, longitudinal weld seam location, thermal margin, operating history, and future fuel management. A proper and accurate cost estimate study of fuel substitution requires core redesign, fuel management analysis, and fuel management decisions by the licensee. The staff has not attempted plant-specific analyses, but has approached the problem on a generic basis, using the following assumptions:

(1) The peak fluence to the critical weld will be the criterion for flux reduction requirements. This is a reasonable assumption for the older reactors under consideration because the peak fluence location has received considerably more irradiation than other azimuthal locations. For the newer plants, methods exist which allow fuel assembly management

with low leakage schemes to even out the fluence accumulation in the belt region or minimize fluence to the welds. The latter considerations are beyond the scope of this study.

(2) We assume that a core redesign could take advantage (in part or in total) of some relaxation of Appendix K, lowering of MDNBR, or power derating depending on the required flux reduction and the plant-specific parameters.

(3) To estimate the effect of 10 CFR 50.46 and Appendix K, we assume that the linear heat generation rate could be increased by 20 percent if a best estimate evaluation model were used rather than the conservative evaluation models specified by Appendix K (Ref. I.14). The 20 percent figure may be true for some plants (those limited by decay heat or stored energy) but is not necessarily true for all plants to the same extent. No effort has been made to distinguish whether specific plants are Appendix K limited or not.

(4) To estimate the effect of the DNBR limits (for those plants which are DNBR rather than Appendix K limited) we assume that lowering the DNBR criterion by 10 percent would allow raising the average heat generation rate by about 20 percent (Ref. I.15). This assumption is based on a sensitivity analysis of MDNBR to eight DNB parameters, one of which is power. A standard Westinghouse, Combustion Engineering, and Babcock and Wilcox plants were used with the corresponding correlations. The suggested relationship is accurate to better than 1.5 percent. The study was carried out by Battelle, Pacific Northwest Laboratories.

(5) We assume that the daily power replacement cost (hence, plant shutdown cost) is $0.3 M. Daily power derating costs are derived from this by prorating. It is assumed that fuel cycle costs caused by the new core (due to increased enrichment, more frequent refuelings, possible redesign of the control rods, etc.) are included in the above cost. Plant redesign, power replacement, etc., are plant-specific quantities, and the above figure is for reference purposes only.

(6) A one-time core redesign cost of $20 M is assumed for any core needing fuel assembly replacement. This cost will depend on the number of fuel assemblies to be replaced, the size of the core, the objectives of the redesign, etc. This cost is only an approximate figure and should be appropriately adjusted for individual plants. It includes fabrication of dummy replacement assemblies and other necessary core changes; for example, flow adjustments.

Table I.4, Operating Reactor RT_{NDT} Evaluation, was compiled using currently available information (Ref. I.16). The table shows the critical element of the pressure vessel, i.e., circumferential weld, axial weld, or the plate and the remaining increase in RT_{NDT} before the screening criteria are reached. The EFPYs of operation and the fluence at the end of 1981, and the total fluence at which the screening criteria would be reached are also listed. Using the above information, the required annual fluence to just reach the screening criteria at the end of 32 EFPYs was computed. Finally, from this information, the factor by which the flux to the pressure vessel must be reduced so as not to exceed the screening criteria at the end of the 32 EFPY was calculated.

The Flux Reduction Factor (FRF) permits the staff to distinguish the following categories of plants (a) those with FRF<1 which indicates that no flux reduction would be needed; (b) plants with 1< FRF <2, which indicates that the flux reduction necessary not to exceed the screening criteria before the end of life of the plant can be achieved with ordinary low leakage fuel management (which can result in economic benefits due to longer fuel cycles and increased burnup); (c) plants with 2< FRF <3, which indicates that combined low leakage loading and azimuthal flux distribution management can achieve the goal of meeting the criteria at 32 EFPY (depending on the particular plant under consideration, loss of thermal margin and operational flexibility may result, which may offset part of the economic gains of low leakage fuel loading); and (d) plants with FRF >4. There are four plants in this last category, i.e., H. B. Robinson 2, Turkey Point 3 and 4, and Fort Calhoun. Due to the large flux reduction required to meet the screening criteria at the end of the 32 EFPY, these plants are discussed separately.

The effect of low leakage and of fuel assembly substitution has been studied by the staff in some detail. The azimuthal variation of the end of life fluence in a variety of peripheral assembly substitutions has been studied, including zero power assemblies, nominal stainless steel rods, and closed packed stainless steel rods. These studies were carried out for Maine Yankee, H. B. Robinson-2, Oconee-1 and Fort Calhoun (Ref. I.17). Analysis of San Onofre, Turkey Point, and Three Mile Island is well under way.

H. B. Robinson 2

The applicant's end of life fluence to the pressure vessel for H. B. Robinson-2 was 6.4×10^{19} n/cm^2. Staff sponsored calculations indicated a corresponding value of 7.4×10^{19} n/cm^2. This brought the estimated fluence to the end of 1981 to 1.64×10^{19}. In the meantime H. B. Robinson-2 has refueled in 1982 with a low leakage core (extended burnup) which lowered the flux to the peak location by about 50 percent. The current cycle will last about 1.0 EFPY with an accumulated total of 8.0 EFPY. The peak fluence location will be at 1.76×10^{19} n/cm^2 which brought the available margin at the end of 1982 to $.31 \times 10^{19}$ n/cm^2, i.e., RT_{NDT} = 290°F (10°F remaining) and at the end of the current cycle to $.17 \times 10^{19}$ n/cm^2, i.e., 296°F (4°F remaining).

The required peak flux reduction factor to meet the screening criteria at the end of the plant life >9.5 with the current low leakage scheme or >20 without a low leakage core. The maximum obtainable with the removal of the outer row of assemblies is about 9.5

Assuming removal of the outer row of assemblies and substitution with close packed stainless steel assemblies, the plant will reach screening criteria at 14.2 EFPY. At the current rate of fluence accumulation the plant will reach the screening criteria in 2.8 EFPY from the end of 1981, or 1.7 EFPY from the end of the current cycle.

Removal of all peripheral assemblies and substitution with closed packed stainless steel rods will bring the end-of-life vessel fluence to about 1.9×10^{19}, i.e., at the required value, shown in Table I.4. The removals will amount to 36 peripheral assemblies and will result in an increase of about 30 percent in the absolute peak power assuming the same power level and fuel distribution. Recent experience indicates that a factor of two in the peak power difference can be accommodated with fuel loading measures and careful fuel management. For the remaining 15 oercent there are three options or a combination thereof:

(a) If the plant is decay heat or stored energy limited, relaxation of the Appendix K requirements would allow an increase of up to 10 percent;

(b) If the plant is limited by MDNBR; a reduction of the criterion by 0.10 would allow increasng the linear power by about 20 percent; and

(c) Derating by 15 percent.

Options (a) and (b) will cost about $20 M for the design of the new core and will shorten refueling intervals by about 24 percent, if the maximum enrichment remains the same. Increased fuel enrichment might add to the new core costs. Option (c) will cost about $20 M for the design of the new core and an additional cost of $9.9 M per EFPY for power replacement.

Turkey Point 3 and 4

The required flux reduction factor is 4.5 from the present fuel loading mode. Since these plants are identical with respect to core arrangement, size, dimensions, material and neutron source distribution with H. B. Robinson-2, we have used the results of the Robinson study. Removal of the three assemblies at the flats with careful fuel management, will reduce the flux by the required amount. It is unlikely that derating would be required. The maximum number of removed peripheral assemblies is 12, i.e., 7.6 percent of the fuel inventory, which produce about 4.3 percent of the power.

The cost is estimated at $20 M for core redesign. Power replacement cost is not significant.

Fort Calhoun

Total required flux reduction factor is 4.0. This can be accomplished by the removal of two central assemblies in the flats or the periphery for a total of 8 assemblies. (However, while removal of the central assemblies would reduce the flux by the required amount, the remaining two assemblies may not be functional in their position with the central assemblies removed.) The 8 assemblies represent 6 percent of the fuel inventory and about 5.5 percent of the power. Actual power derating may not be necessary.

The cost is estimated at $20 M for core redesign. Power replacement cost is not significant.

I.6 Conclusions

(1) All vendors and licensees provided similar responses to our survey inquiries.

(2) Presently employed in-out, low leakage loading schemes provide about a 30% reduction of the fast neutron fluence rate as a side benefit derived from extended cycle core designs and may represent overall cost savings to licensees.

(3) In-out, low leakage loading schemes using twice or thrice burned fuel assemblies on the core periphery can provide a factor of 2 to 3 reduction in the fluence rate to the pressure vessel.

(4) If in addition to twice or thrice burned fuel, peripheral assemblies loaded with burnable poisons are used, a factor of 5 reduction in fluence rate can be achieved with in-out, low leakage loading schemes.

(5) If one attempted to maintain the core power rating while implementing low leakage schemes, the power distribution would become more centrally peaked and would require core redesign and fuel rearrangement to flatten power and probably would result in plant derating depending on available plant thermal margin.

(6) In-out, low leakage schemes can be used to reduce locally fluence rates in areas of peak welds, but may result in slightly higher fluence elsewhere and the appearance of peaks at new locations.

(7) The exact impact of in-out, low leakage loading schemes is plant dependent and cannot be generalized.

(8) The effectiveness of in-out, low leakage loading schemes is greatest for plants with large azimuthal flux peaks (CE & W). Implementation in B&W plants would probably involve a large numer of assemblies because of the more uniform azimuthal flux distribution.

(9) Reduction of the fluence by factors of 10 or higher can be affected by peripheral assembly replacement with nonfueled assemblies (e.g., stainless steel). This can be done locally or uniformly, as needed, depending on the azimuthal flux distribution and location of weld seams.

(10) use of nonfueled assemblies would result in significant loss of heat transfer area (10-15%), reduced core size, increased thermal peaking, increased linear power generation rates, and increased rod worths. It would require a new core design, with different fuel enrichment and new transient and accident analyses. New limiting safety system setpoints would have to be generated and fuel management philosophies would change.

(11) Selected replacement would provide local reductions of fluence by a factor of 10 or more. If core symmetry is not maintained, however, the normal means of monitoring core power distribution based on neutron detectors and 1/8 core symmetry would have to be changed.

(12) Use of nonfueled assemblies could result in power derating of perhaps 30%.

(13) Effectiveness of any of these fluence reduction schemes depends on previous vessel exposures and materials and is less significant once significant fluence has been accumulated.

(14) The staff is continuing to study the effect of low leakage core and fuel assembly substitution on the fracture toughness properties of operating reactor pressure vessels. There have been four vessels identified which may require large flux reduction to prevent exceeding the proposed screening criteria prior to the projected EOL. Our cost estimate for core redesign and fuel assembly substitution is $20 M plus power replacement cost.

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I.2 Letter from T. Novak to Licensees, August 21, 1981.

I.3 Letter from T. LaFleur to: Belgium, Finland, France, Germany, Italy, Japan, Korea, Netherlands, Spain, Sweden and Switzerland, with whom we have bilateral information exchange agreements, July 29 and August 20, 1981.

I.4 Institute of Radiation Protection, "Embrittlement of Loviisa-1 Reactor Pressure Vessel," Confidential Memorandum, May 15, 1981.

I.5 MPA Stuttgart, Replies to NRC Questionnaire, September 18, 1981.

I.6 Reactor Safety Commission, "Safety Against Brittle Fracture of the Reactor Pressure Vessel," NRC Translation 8-23-A, January 1981.

I.7 Letter from Schnurer of the Federal Ministry of Internal Affairs, West Germany, to LaFleur, NRC, August 10, 1981.

I.8 Letter and Attachments to the NRC, March 19, 1982.

I.9 Letter from R. Mattson to Gregory Ogeka of DOE, "BNL Technical Assistance to the Division of Systems Integration, NRR, NRC, Pressure Vessel Radiation Embrittlement Calculations" (FIN A3387), March 19, 1982.

I.10 Letter, W. V. Parker to H. R. Denton, January 15, 1982 (Oconee Nuclear Station, Docket Nos. 50-269, 50-270, and 50-287).

I.11 Letter, W. C. Jones (Omaha Public Power District) to T. M. Novak, November 13, 1981; Letter, W. C. Jones to R. A. Clark, November 12, 1981; Letter, W. C. Jones to H. R. Denton, January 23, 1981.

I.12 H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261, License No. DPR-23, Letter, E. G. Utley to D. G. Eisenhut entitled, "Thermal Shock to Reactor Pressure Vessels," January 25, 1982 and subsequent communications.

I.13 Report on the PTS Audit Calculations by BNL in a BNL-NUREG report to be published.

I.14 H. Richings, N. Lauben, personal communication, October 13, 1982.

I.15 G. M. Hesson and J. M. Cuta, "Analysis of the Sensitivity of Calculated MDNBR to Eight Selected DNB Parameters," Battelle, PNL, FATE-79-101, March 1979.

I.16 P. N. Randall, personal communication, October 14, 1982.

I.17 A. Aronson, et al., "Evaluation of Methods for Reducing Pressure Vessel Damage Fluence," to be issued by BNL as a NUREG.

Table I.1 Vessel fluence for Oconee-1, Ft. Calhoun, and H. B. Robinson-2

	Oconee-1	Ft. Calhoun	Robinson-2
Total Effective Full Power Years of Operation (EFPY) as of 12/81	5.1	5	7.1
(a) Out-in loading EFPY	4	5	7.1
(b) In-out low leakage EFPY	1.1	-	-
Present Vessel Fluence Using Current Fuel Loading Scheme ($\times 10^{18}$ n/cm^2)			
(a) BNL calculation	2.70	7.24	18.8
(b) Vendor calculation	2.55 (Ref. I-10)	6.60 (Ref. I-11)	13.8 (Ref. I-12)
End of Vessel Life Fluence ($\times 10^{18}$ n/cm^2)			
I. Using Current Fuel Loading Scheme			
(a) BNL calculation	12.1*	45.9	85.0
(b) Vendor calculation	12.5 (Ref. I-10)	42.0 (Ref. I-11)	65.6 (Ref. I-12)
(c) FSAR value	22.0	20.0	51.0
*II. Using In-out Low Leakage Loading Scheme/ (Increase in Total Peaking Factor (%))	11.2 (7)	30.0 (17)	53.7 (17)
**III. Using Stainless Steel Assemblies on Periphery/ (Increase in Total Peaking Factor(%))	1.90 (23)	12.7 (30)	28.1 (23)

*Out-in loading scheme value is 18.5×10^{18} n/cm^2.
**BNL calculation.

Table I-2 Staff evaluation of flux and fluence to weld seams and peak fluence location for Oconee-1

	Weld* SA-1430	Weld SA-1493	Weld SA-1073	Peak Wall Location
I. Flux ($x10^{10}$ n/cm^2-sec)				
Fuel Loading Method				
(a) Out-in, $P/\bar{P}$ = 0.910	1.59	1.37	1.45	1.84
(b) Low leakage, $P/\bar{P}$ = 0.527	.984	.915	.911	1.11
(c) Low leakage, $P/\bar{P}$ = 0.0	.125	.188	.125	.188
II. Fluence for 28 EFPY ($x10^{18}$ n/cm^2)				
Fuel Loading Method				
(a) Out-in, $P/\bar{P}$ = 0.910	13.5	11.7	12.3	15.6
(b) Low Leakage, $P/\bar{P}$ = 0.527	8.35	7.76	7.73	9.41
(c) Low Leakage, $P/\bar{P}$ = 0.0	1.05	1.60	1.06	1.60
III. Fluence for 32 EFPY ($x10^{18}$ n/cm^2)				
Fuel Loading Method				
(a) Out-in, $P/\bar{P}$ = 0.910	16.1	13.9	14.6	18.5
(b) Low leakage, $P/\bar{P}$ = 0.527	9.93	9.23	9.20	11.2
(c) Low leakage, $P/\bar{P}$ = 0.0	1.25	1.90	1.26	1.90

*See Figure I-1 for weld seam location.

Table I-3 Staff evaluation of the ratio of the fluence rate to the weld seams and the original peak fluence location to the fluence rate of Weld Seam SA-1430 for the remaining 28 EFPY for Oconee-1

Fuel Loading Method	Relative Fluence and Fluence Rate Reduction Factor*			
	Weld SA-1430	Weld SA-T493	Weld SA-1073	Original Peak Fluence Location
Out-in, $P/\overline{P}$ = 0.91	1.00 / 1.00	.863 / 1.16	.909 / 1.10	1.15 / .87
In-out, low leakage $P/\overline{P}$ = 0.527	.618 / 1.62	.575 / 1.74	.572 / 1.95	.697 / 1.43
In-out, low leakage $P/\overline{P}$ = 0.0	.078 / 12.8	.118 / 8.47	.078 / 12.8	.098 / 10.2
Stainless Steel Assemblies, $P/\overline{P}$ = 0.0	.049 / 20.4	.103 / 9.71	.057 / 17.5	.077 / 13.0
Stainless Steel Assemblies,** $P/\overline{P}$ = 0.0	.033 / 30.3	.081 / 12.3	.040 / 25.0	.061 / 16.4

*First number is the fluence rate ratio; the second number is the fluence rate reduction factor.

**This case includes nonfueled assemblies with additional stainless steel rods in a close packed array.

As of 12/31/81

Plant, Vendor/ PV Fabricator/ MWe	Controlling element Axial or Circumferential weld or plate	RT_{NDT} Below Screening Criteria (°F)	Total Fluence E>1 MeV to Meet Screening Criteria $n/cm^2 x10^{19}$	EFPY	Fluence $(n/cm^2 x10^{19})$	Fluence per EFPY $(n/cm^2 x10^{19})$	Additional Fluence To Reach Screening Criteria $n/cm^2 x10^{19}$	Remaining EFPY in Plant Life (32 EFPY)	To Reach Screening Criteria at 32 EFPY: Fluence per EFPY $(n/cm^2 x10^{19})$	To Reach Screening Criteria at 32 EFPY: Flux Reduction Factor	$FRF < 1$	$1 < FRF < 2$	$2 < FRF < 3$	$FRF > 4$
Robinson-2	circ.	19	1.95	7.1	1.41	.199	.54	24.9	.0217	9.2[(1)]				
W/CE/665		12	1.95	7.1	1.64	.230	.31	24.9	.0124	18.5[(1)]				X
Fort Calhoun CE/CE/486	axial	44	1.18	5.07	.51	.100	.67	26.93	.0249	4.0				X
Turkey Pt.-4 W/B & W/666	circ.	41	1.85	5.67	.91	.160	.94	26.33	.0357	4.5[(2)]				X
Turkey Pt.-3 W/B & W/666	circ.	41	1.85	5.67	.91	.160	.94	26.33	.0357	4.5[(2)]				X
Maine Yankee CE/CE/825	axial	54	1.18	5.90	.41	.069	.77	26.10	.0295	2.3			X	
Calvert Cliffs-1 CE/CE/850	axial	131	8.22	4.65	.68	.146	7.54	27.35	.276	.53[(3)]	X			
Indian Pt.-3 W/CE/965	plate	58	1.04	2.98	.167	.056	.873	29.02	.0301	1.9		X		
Yankee Rowe W/B & W/175	plate	59	4.48	14.56	1.14	.078	3.34	17.44	.1915	.4	X			
Rancho Seco B & W/B & W/913	axial	63	.77	3.54	.205	.059	.565	28.46	.0199	2.9			X	
Three Mile Island-1 B & W/B & W/792	axial	66	.75	3.52	.187	.053	.563	28.48	.0198	2.7			X	
Oconee-2 B & W/B & W/860	circ.	69	.99	4.71	.287	.061	.703	27.29	.0258	2.4			X	
Zion-1 W/B & W/1040	circ.	75	1.25	4.97	.313	.063	.937	27.03	.0347	1.8		X		
Point Beach-1 W/B & W/497	axial	72	3.48	8.07	.734	.091	2.750	23.93	.1149	.8	X			
Oconee-1 B & W/B & W/860	axial	73	1.33	5.04	.273	.054	1.057	26.96	.0392	1.38		X		
Indian Pt.2 W/CE/873	axial	81	1.18	4.40	.22	.050	.960	27.60	.0348	1.4		X		

As of 12/31/81

Plant, Vendor/ PV Fabricator/ MWe	Controlling element Axial or Circumferential weld or plate	RT_{NDT} Below Screening Criteria (°F)	Total Fluence E>1 MeV to Meet Screening Criteria $n/cm^2 x10^{19}$	EFPY	Fluence $(n/cm^2 x10^{19})$	Fluence per EFPY $(n/cm^2 x10^{19})$	Additional Fluence To Reach Screening Criteria $n/cm^2 x10^{19}$	Remaining EFPY in Plant Life (32 EFPY)	To Reach Screening Criteria at 32 EFPY: Fluence per EFPY $(n/cm^2 x10^{19})$	To Reach Screening Criteria at 32 EFPY: Flux Reduction Factor	FRF<1	1<FRF<2	2<FRF<3	FRF>4
Ginna W/B & W/490	circ.	87	4.95	8.18	.949	.116	4.000	23.82	.1679	.7	X			
Point Beach-2 W/B & W, CE/497	circ.	85	4.65	7.54	.935	.124	3.697	24.46	.1511	.8	X			
Arkansas, ANO-1 B & W/B & W/836	axial	82	1.23	4.42	.199	.045	1.031	27.58	.0374	1.2		X		
San Onofre-1 W/CE/436	axial	89	12.23	9.04	2.71	.300	9.520	22.96	.4146	.7	X			
Zion-2 B & W/B & W/1040	axial	93	.77	4.49	.09	.02	.680	27.51	.0247	.8	X			
Palisades CE/CE/740	axial	93	2.33	4.12	.478	.116	1.852	27.88	.0664	1.7		X		
Crystal River-3 B & W/B & W/825	axial	93	1.18	2.48	.136	.055	1.044	29.52	.0354	1.6		X		
Surry-1 W/B & W/775	circ.	100	5.50	4.88	.761	.159	4.74	27.12	.1748	.9	X			
Cook-1 W/CE/1054	circ.	100	1.94	4.56	.287	.063	1.653	27.44	.060	1.05		X		
North Anna-1 W/RG/865	plate	108	11.57	2.41	.442	.183	11.13	29.59	.376	.486	X			
Beaver Valley W/CE/833	circ.	118	2.06	1.87	.316	.169	1.744	30.13	.058	2.91			X	
North Anna-2 W/RD/890	plate	118	10.1	.77	.138	.179	9.962	31.23	.319	.56	X			
Salem-1 W/CE/1090	axial	130	3.68	2.26	.148	.065	3.532	29.74	.119	.55	X			
Oconee-3 B&W/B&W/860	circ.	129	5.04	4.78	.292	.061	4.748	27.22	.174	.35	X			
Surry-2 W/B&W/775	axial	130	14.8	4.83	.754	.156	14.05	27.17	.517	.30	X			

As of 12/31/81

Plant, Vendor/ PV Fabri- cator/ MWe	Controlling element Axial or Circumferential weld or plate	RT_{NDT} Below Screening Criteria (°F)	Total Fluence E>1 MeV to Meet Screening Criteria $n/cm^2 x10^{19}$	EFPY	Fluence $(n/cm^2 x10^{19})$	Fluence per EFPY $(n/cm^2 x10^{19})$	Additional Fluence To Reach Screening Criteria $n/cm^2 x10^{19}$	Remaining EFPY in Plant Life (32 EFPY)	To Reach Screening Criteria at 32 EFPY: Fluence per EFPY $(n/cm^2 x10^{19})$	To Reach Screening Criteria at 32 EFPY: Flux Reduction Factor	FRF< 1	1< FRF< 2	2< FRF< 3	FRF>4
St. Lucie-1 CE/CE/777	axial	135	3.02	3.52	.222	.063	2.798	28.48	.098	.64	X			
Calvert Cliffs-2 CE/CE/850	axial	140	8.48	3.63	.534	.147	7.946	28.37	.280	.53	X			
Trojan W/CBI/1130	plate	147	16.2	3.00	.207	.069	16.00	29.00	.552	.13	X			
Davis Besse-1 B&W/B&W/906	circ.	156	5.25	1.68	.111	.066	5.14	30.32	.170	.39	X			
Haddam Neck W/CE/582	axial	161	36.48	10.92	1.190	.109	35.29	21.08	1.674	.07	X			
Kewaunee W/CE/535	circ.	168	17.14	5.87	.786	.134	16.35	26.13	.626	.21	X			
Farley-1 W/CE/829	axial	168	11.61	2.19	.370	.169	11.24	29.81	.377	.45	X			
Millstone-2 CE/CE/870	circ.	183	7.62	3.91	.219	.056	7.40	28.09	.263	.21	X			
Prairie Island-2 W/SFAC/520	axial	216	90.7	5.62	.753	.134	89.95	26.38	3.41	.04	X			
Prairie Island-1 W/SFAC/520	axial	237	292.1	5.90	.790	.139	291.3	26.10	11.16	.01	X			
										Summary	23	8	5	4

(1) The lower line in H. B. Robinson lists the staff's calculations. The differences have not been resolved as of 11/10/82. Note that the current cycle of the Robinson plant (after 12/31/81) is a low leakage cycle, therefore, the FRF would currently be lower by a factor of 2, i.e., 4.6 and 9.2 for licensee and NRC calculations.

(2) The staff has been notified by Florida Power and Light (Ref. 5) that the fluence at 5.67 EFPY was $.91 \times 10^{19}$ n/cm^2 and the projected fluence for the next 3 EFPY will be 1.39×10^{19} n/cm^2 i.e., the fluence accumulations rate will remain the same. These values have not been reviewed by the staff.

(3) The value of the fluence required to reach the screening criteria is based on .21% copper content in the weld seam (Ref. 6).

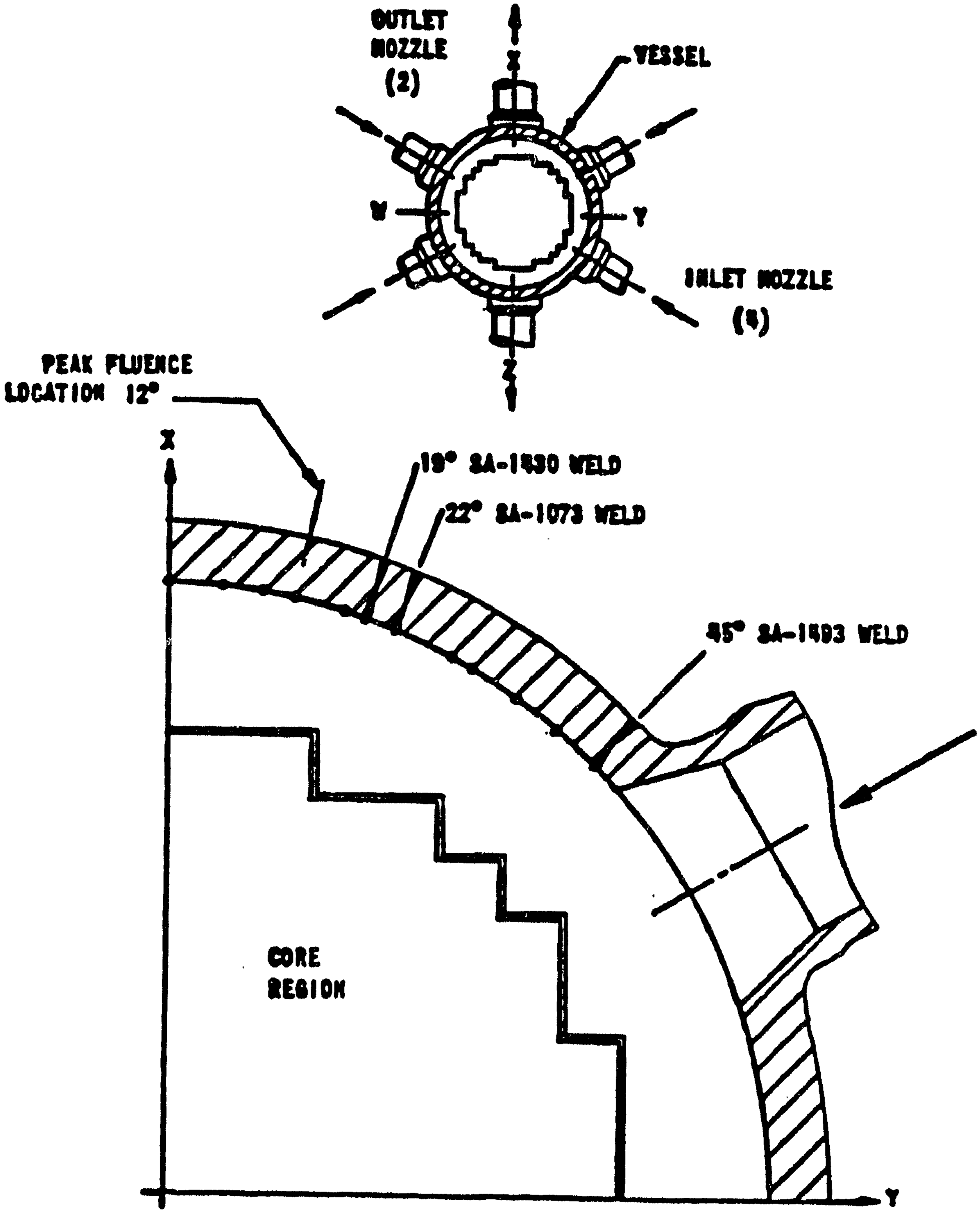

FIGURE 1-1. LONGITUDINAL WELD LOCATIONS FOR OCONEE-1

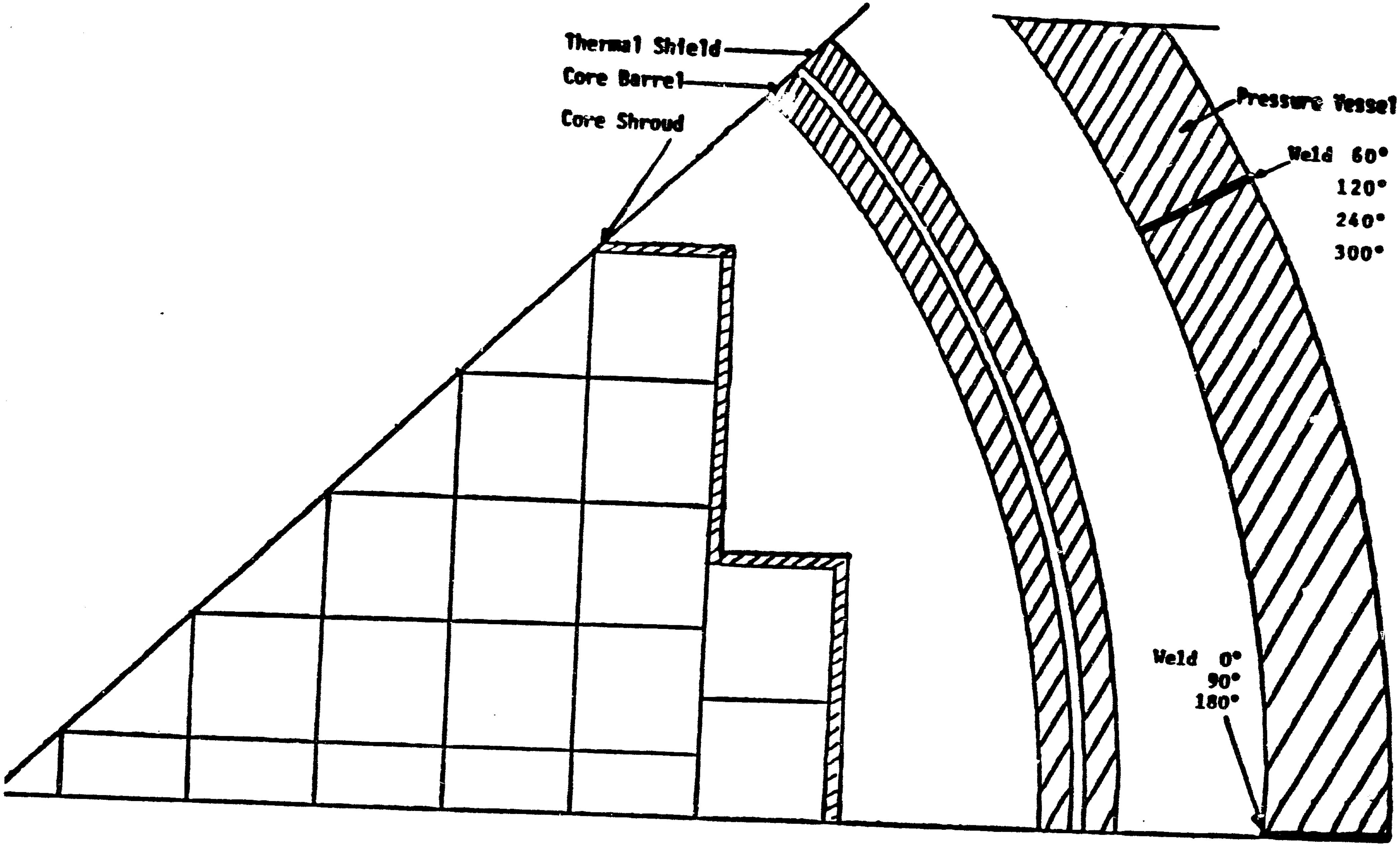

FIGURE I-2 FUEL ELEMENT GRID AND LONGITUDINAL WELD SEAMS FOR FORT CALHOUN

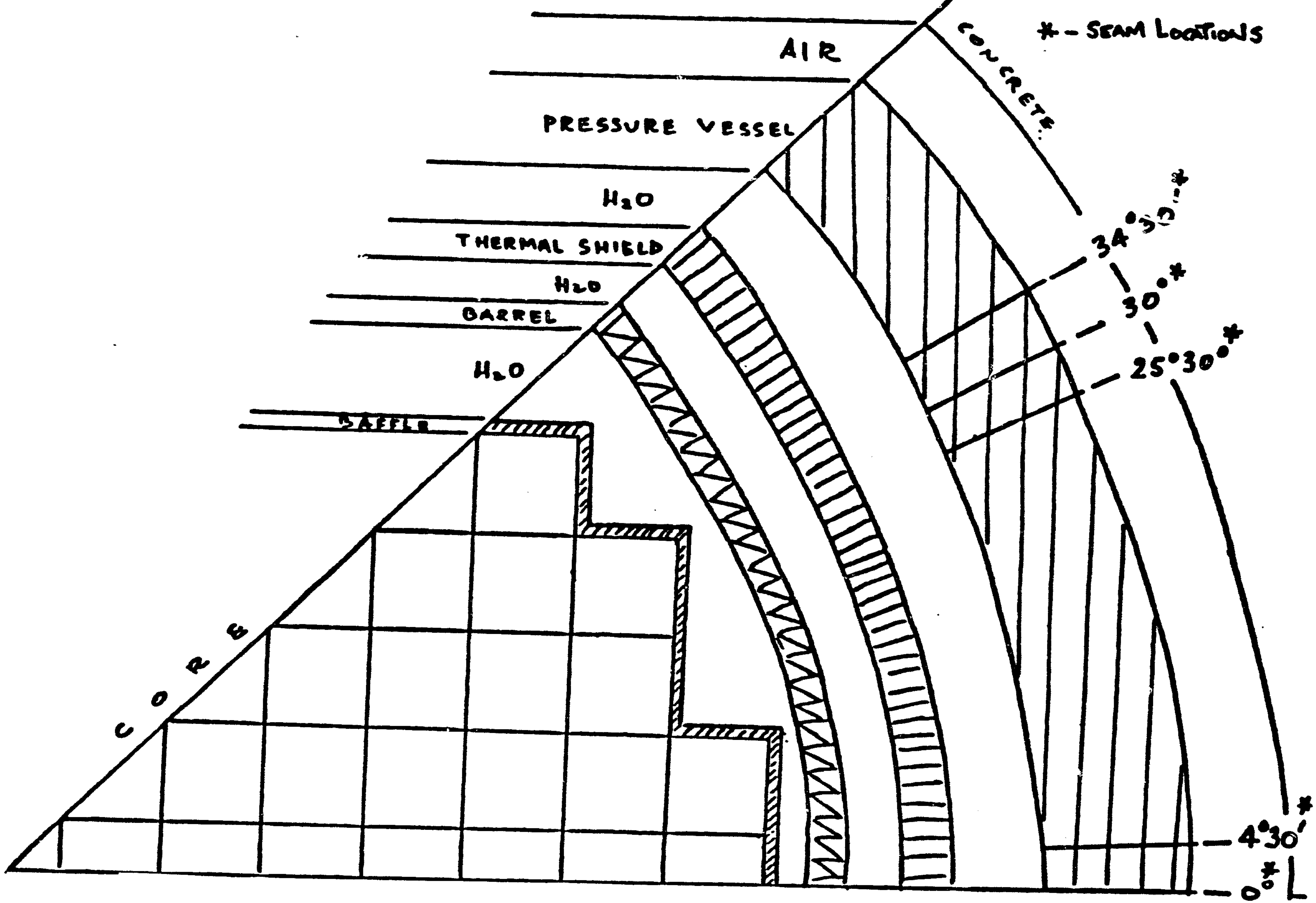

FIGURE I-3 FUEL ELEMENT GRID AND LONGITUDINAL WELD SEAM LOCATIONS FOR H. B. ROBINSON-2

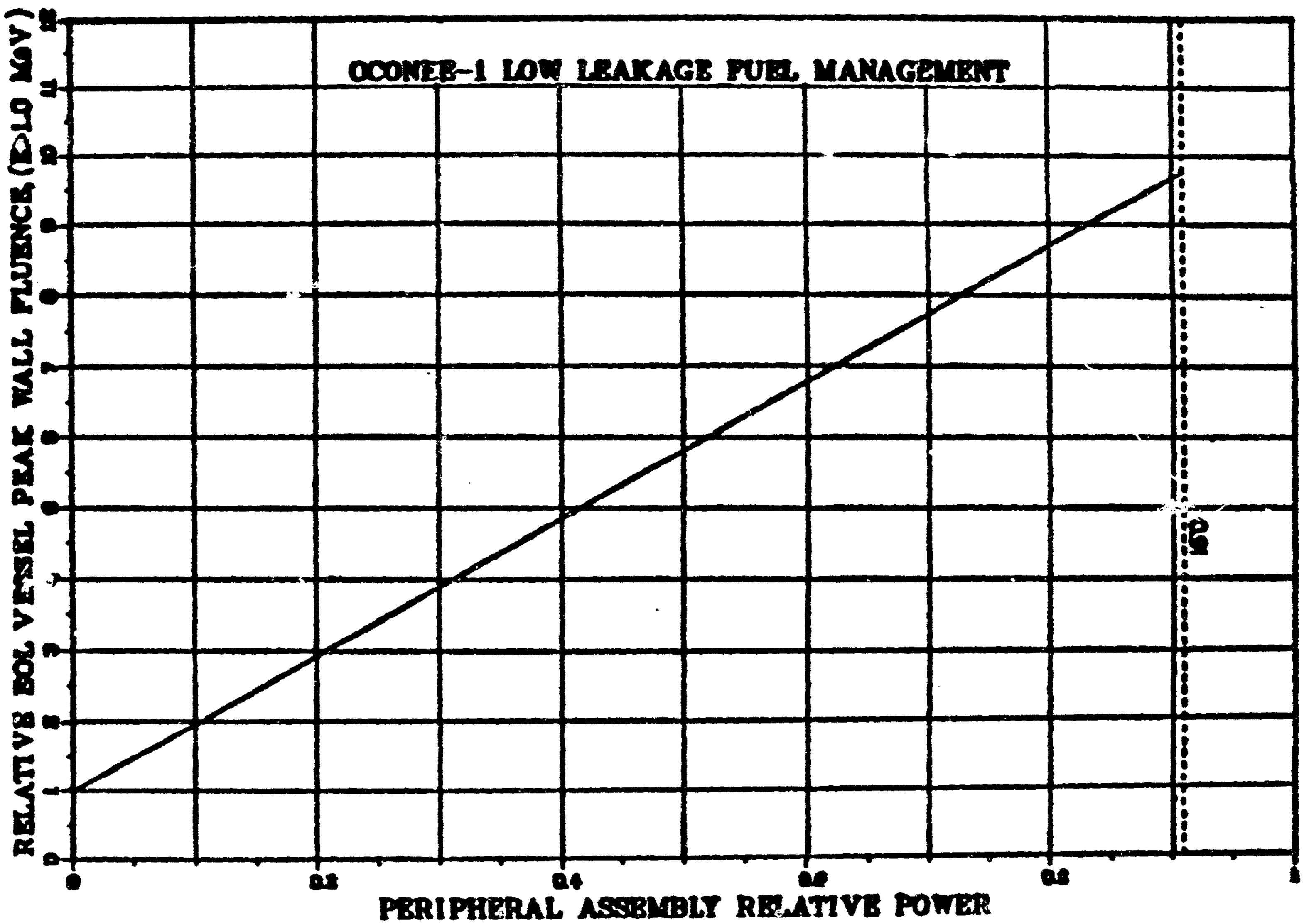

FIGURE I-4 PEAK WALL FLUENCE AS A FUNCTION OF PERIPHERAL ASSEMBLY POWER FOR OCONEE-1 (FOR REMAINING 28 EFPYS)

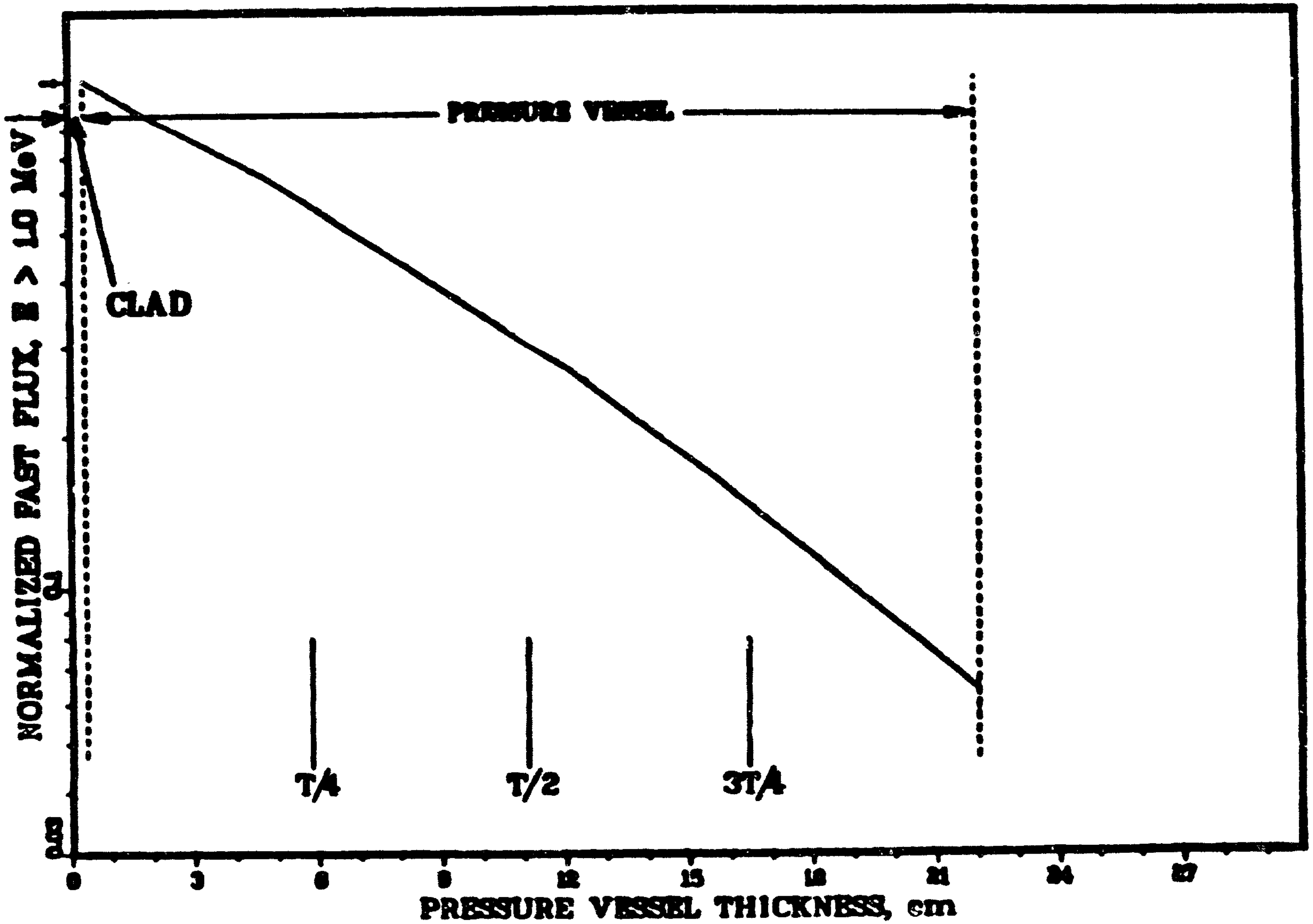

FIGURE I-5 FAST FLUX ATTENUATION THROUGH THE PRESSURE VESSEL WALL FOR OCONEE-1

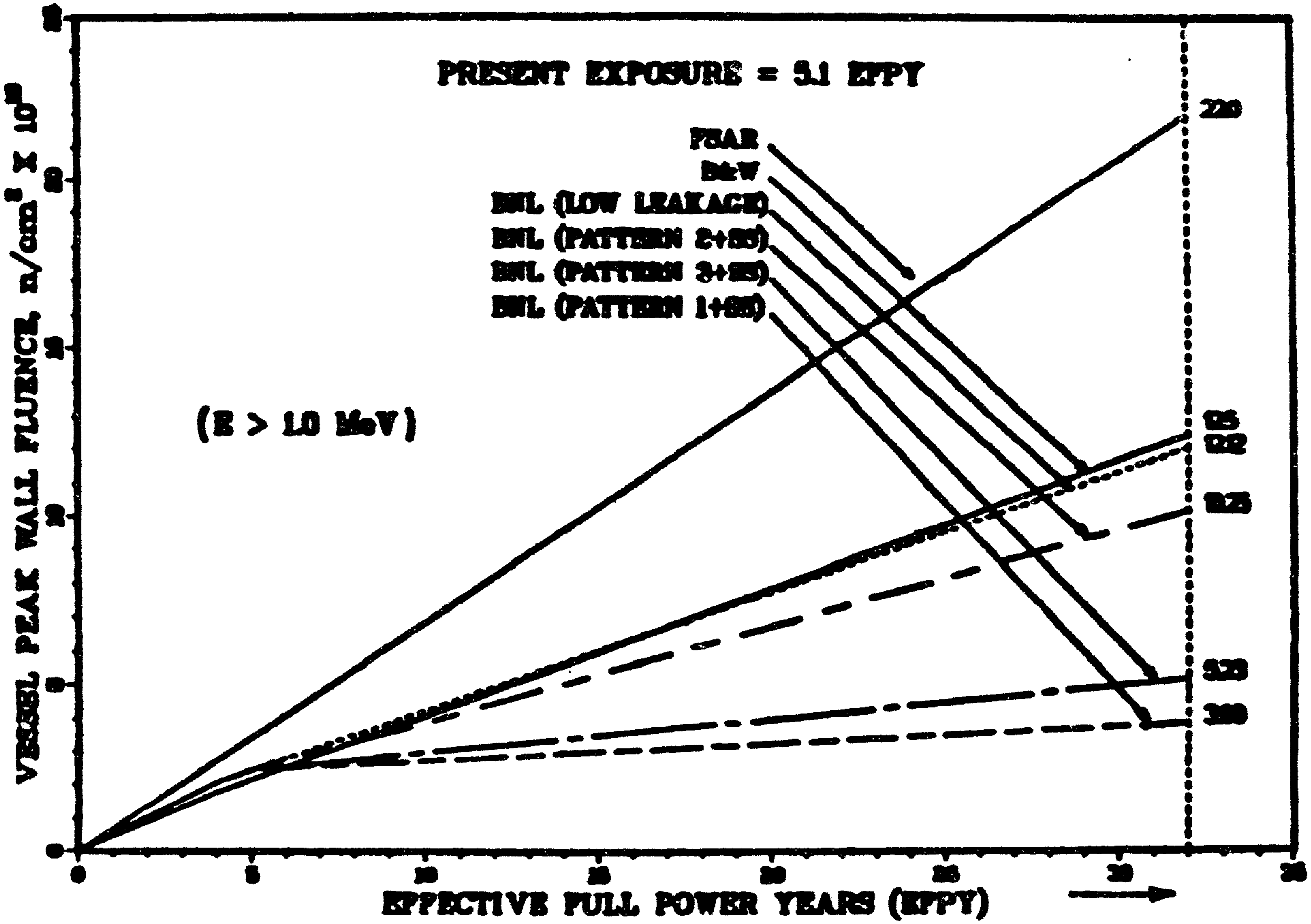

FIGURE I-6 Peak Fluence As A Function of EFPY For Various Fluence Rate Reduction Schemes For Oconee-1

Pattern 1 Removal and replacement of outer row of assemblies with 40 stainless steel assemblies
Pattern 2 Removal and replacement of 20 assemblies opposite weld seams
Pattern 3 Modification of Pattern 2 with additional removals and replacement of peripheral assemblies (32 total)

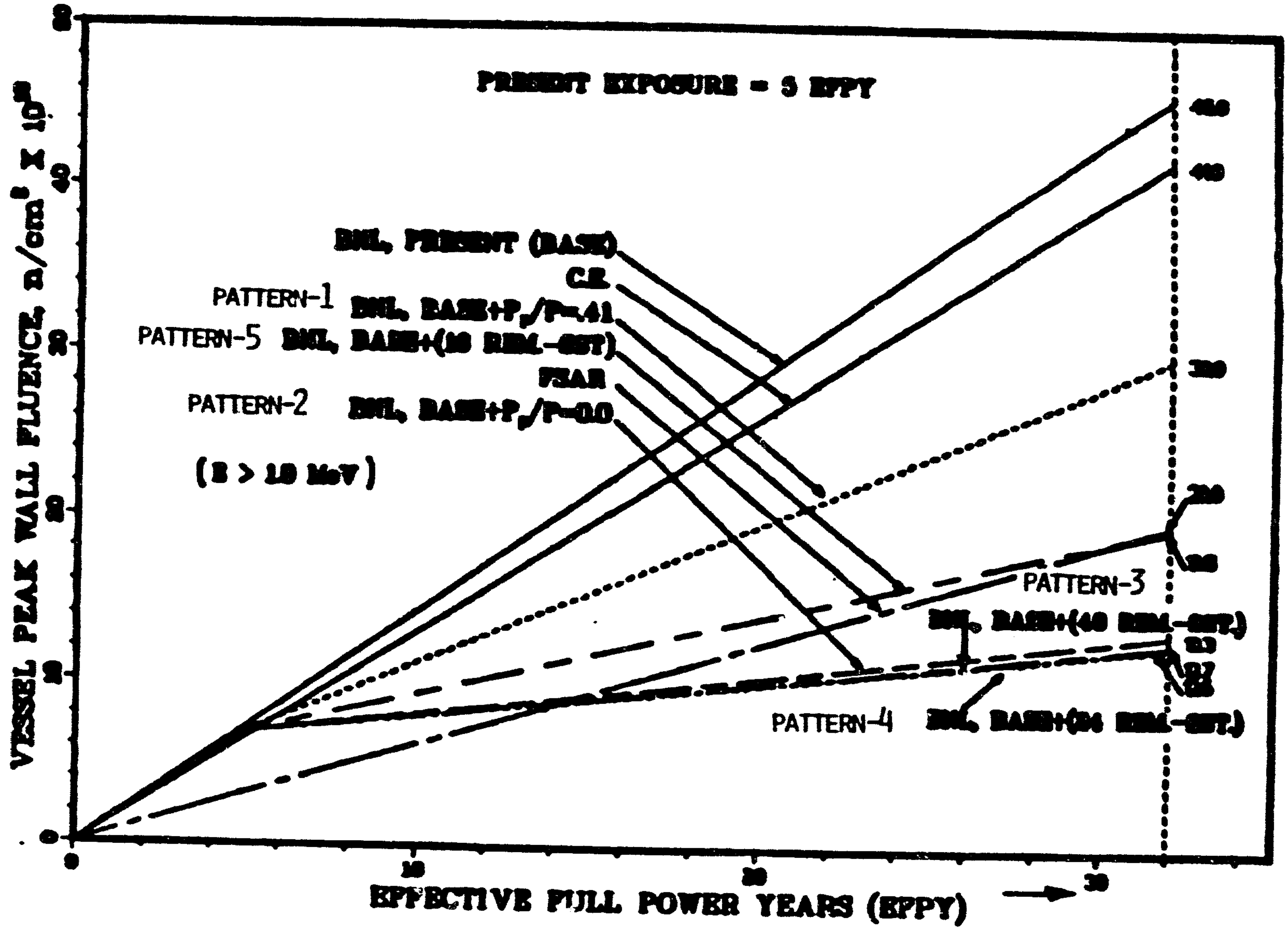

Figure I-7 Peak Fluence As A Function of EFPY For Various Fluence Rate Reduction Schemes For Ft Calhoun

Pattern 1 Low leakage scheme(peripheral to core average power = 0.41)
Pattern 2 Low leakage scheme(zero peripheral power)
Pattern 3 Assembly replacement with 40 stainless steel assemblies
Pattern 4 Assembly replacement with 24 stainless steel assemblies
Pattern 5 Assembly replacement with 16 stainless steel assemblies

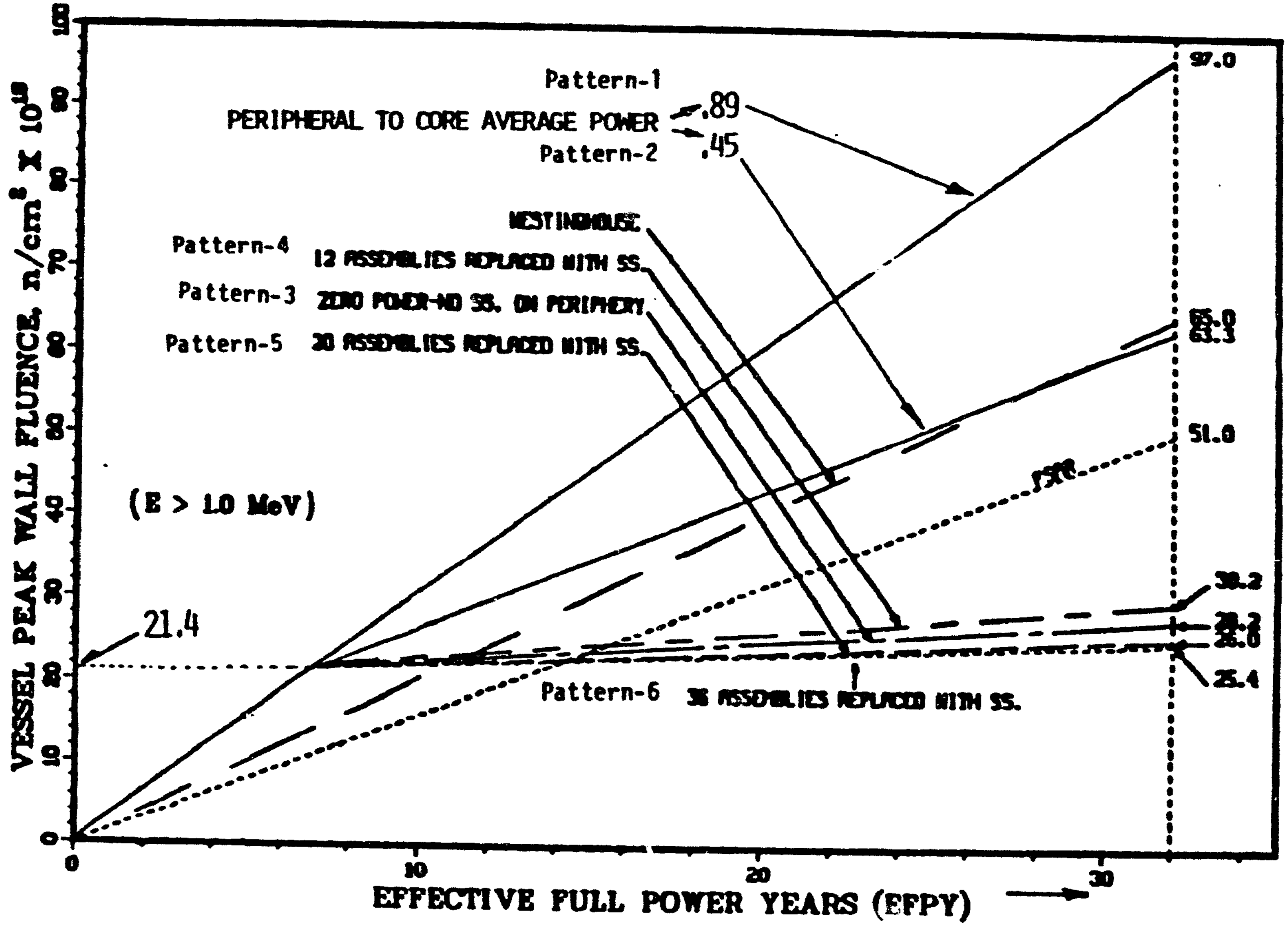

Figure I-8 Peak Fluence As A Function of EFPY for Various Fluence Rate Reduction Schemes for Robinson-2

Pattern-1 Out-in fuel loading, $P/\bar{P} = 0.89$
Pattern-2 Low leakage loading, $P/\bar{P} = 0.45$
Pattern-3 Low leakage loading, $P/\bar{P} = 0.0$
Pattern-4 Fuel assembly replacement with 12 stainless steel assemblies
Pattern-5 Fuel assembly replacement with 20 stainless steel assemblies
Pattern-6 Fuel assembly replacement with 36 stainless steel assemblies

APPENDIX J

SUMMARY OF NRC STAFF POSITION ON REVIEW OF CONTROL SYSTEMS

J.1 Introduction

The following summarizes the staff philosophy on the review of control and protection systems and delineates actions completed or planned to address the effects of control systems on plant safety. The following also specifically discusses the possible impact of control system failures on pressurized thermal shock and actions which should be considered to minimize the possibility of control system failures resulting in an excessive plant cooldown transient.

J.2 Philosophy of Separation of Protection and Control Systems

The philosophy on the separation of protection systems and control systems was developed in the 1960's and early 1970's through interactions between the regulatory staff and industry. The interactions occurred primarily through the development of industry standards such as IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations." The staff did not dictate a particular philosophy, but rather explored through the standards committees and early plant licensing reviews various approaches which could be taken toward reactor protection.

A brief, simplified description of the approach toward protection and control is as follows. A nuclear power plant must satisfy utility requirements for the economic production of power. These requirements include plant operation with a limited number of operators, high plant availability with few unplanned shutdowns, and the ability to follow the utility grid load demand. The requirements for operation are based largely on matching the capabilities of nonnuclear plants. Plant control systems to accomplish the desired economic operational characteristics are established. The control systems, of course, have to be capable of allowing the plant to perform normal operations with margin to plant safety limits.

To assure that safety limits are not exceeded should any system used for normal operation fail, various protective functions such as reactor trip and decay heat removal have been established in the Commission regulations. Systems whose primary purpose is to accomplish the protective functions are provided to fulfill these requirements.

One, thus, has two somewhat differing objectives. The first is to allow normal plant operation within a utility grid which is also supplied by many non-nuclear plants. For this, control functions have been established. The second objective is to ensure that even with failures of the operational equipment, safety limits are not exceeded. For this, protective functions have been established to assure plant safety.

Once control functions and protective functions are defined, a decision has to be made as to whether the same systems should be used for both or whether separate systems should be used. The philosophy developed through the standards committees was one in which the protection systems were treated separately. This allowed a set of guidelines to be established with the intent of ensuring that protection functions are accomplished with a very high degree of reliability. Having a specific, well-defined group of protection systems to accomplish required safety functions allows both industry and the regulatory agency to concentrate their efforts and make effective use of limited resources in accomplishing safety goals.

In development of the philosophy, it was recognized that some limited ties between protection systems and control systems are approp iate and even unavoidable. For example, the systems will always be interrelated through the fluid process systems. Additional interfaces such as the use of the same sensors for protection and control were considered acceptable providing appropriate rules are followed. General Design Criterion 24 and IEEE-279 permit limited interconnections between protection and control tems and define rules for implementing these interconnections.

J.3 NRC Staff Reviews of Control Systems

NRC staff reviews have been performed on currently licensed plants with the goal of ensuring that control system failures will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or maintain the plant in a safe shutdown condition following any "anticipated operational occurence" or "accident." The approach has been to either provide independence between safety and nonsafety systems or to require isolating devices such as isolation amplifiers between safety and nonsafety systems such that failures of nonsafety system equipment cannot propagate through the isolating devices to impair operation of the safety system equipment. In addition, a specific set of "anticipated operational occurrences" and "accidents" have been analyzed to demonstrate that plant trip and/or safety system equipment actuation occurs with sufficient capability and on a time scale such that the consequences are within specified acceptable limits. In these analyses, conservative initial plant conditions, core physics parameters, equipment availability, and instrumentation setpoints have been assumed. Conservative parameters (for example, heat fluxes, temperatures, pressures, and flows) which could result in core or reactor coolant system pressure boundary damage are also assumed. Where active control system operation would mitigate the consequences of a transient, in general, no credit is taken for the control system operation. In some cases, credit has been allowed for the operation of specific control systems in mitigating the consequences of particular "anticipated operational occurrences." Where this has been allowed, special design features and/or technical specification requirements such as periodic testing have been provided.

Where active control system operation would not mitigate the consequences of a transient, no penalties are taken in the analyses for incorrect control system actions caused by control system failures. In the case of control systems, for example, the loss of forced reactor flow is analyzed assuming the reactivity control systems either operate properly or do not operate at all, whichever is the worst case. A loss of forced reactor flow occurring simultaneously with an inadvertent rod withdrawal is not considered. Among the specified set of "anticipated operational occurrences" analyzed are occurrences resulting from both mechanistic and nonmechanistic control system failures. The conservative analyses performed are intended to demonstrate that the potential consequences to the health and safety of the public are within acceptable limits for a wide range of postulated events even though specific actual events might not follow the same assumptions made in the analyses.

In general, until approximately one year ago systematic evaluations of control systems designs had not been performed to determine whether single event induced multiple control system actions could result in a transient such that core or reactor coolant system pressure boundary limits established for "anticipated operational occurrences" are exceeded. Single failures or events which could include multiple control system actions such as discussed above do indeed exist, but experience with operating plants indicates that incidents resulting in transients more severe than currently analyzed as "anticipated operational occurrences" have a low probability. Recent operating plant license applicants have been required to address the possibility of multiple control system actions caused by certain specified events such as a power supply failure or sensor impulse line failure.

The applicants have been required to identify any power sources, sensors, or sensor impulse lines which provide power or signals to two or more control systems and demonstrate that failures of these power sources, sensors, or sensor impulse lines will not result in consequences more severe than those bounded by the analyses of "anticipated operational occurrences" in Chapter 15 of the FSAR. At this time, similar reviews have not been required of operating plant licensees. However, the effort on the current license applications will provide general guidance on whether significant problems may exist on operating plants.

Until approximately two and one-half years ago systematic evaluations of control system designs had not been performed to determine whether postulated accidents could cause control system failures resulting in control actions which would make accident consequences more severe than presently analyzed. Accidents could cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. Licensees have, however, now reviewed the possibility of consequential control system failures which exacerbate the effects of high energy line breaks and taken action, where needed, to assure that the postulated events would be adequately mitigated. Similar efforts are also being performed on plants currently under operating license review.

It should be emphasized that the issue is not whether reactor trip or safety system equipment action would be defeated by control system failures, but whether control system failures could cause a transient or accident to proceed in a manner potentially more severe than currently analyzed. Systematic reviews of safety systems have been performed with the goal of ensuring that control system failures (single or multiple) will not defeat trip or safety system action, and both industry standards and staff regulatory guides are quite clear that this is a design requirement for safety systems including those used for reactor trip.

As part of the staff's ongoing activities to access the adequacy of non-safety grade control systems, Unresolved Safety Issue (USI) A-47 "Safety Implications of Control Systems" was established to perform an indepth evaluation of the control systems that are typically used during normal plant operation and (1) to verify the adequacy of current licensing design requirements or (2) to propose (if necessary) additional guidelines and criteria to assure that nuclear power plants do not pose an unacceptable risk due to inadvertent non-safety grade control system failures. This activity will review the plant designs of the manual and automatic control systems for each of the four nuclear steam system supplier designs (B&W, CE, GE and W).

The activity will evaluate and identify any non-safety grade control systems whose failure (1) may lead to transients or accidents more severe than those analyzed in Chapter 15 of the plant FSAR, or (2) could produce unacceptable frequency of occurrence of those transients bounded by Chapter 15 analysis. As specific subtasks of these reviews, failures that could lead to steam generator or reactor vessel overfill or overcooling transients will be evaluated.

It is anticipated that the technical resolution of this safety issue will be completed in 1984.

J.4 Instrumentation and Control System Impact on Pressurized Thermal Shock

Control system failures can cause inadvertent reactor coolant system cooldowns and inadvertent increases in reactor coolant system pressure. Whether any credible control system failures can cause unacceptable reactor coolant system temperature/pressure combinations, however, requires further analyses.

There are control system failures which can cause excessive feedwater flow or abnormally low feedwater temperature, either of which could lead to reactor coolant system cooldown. If it is found necessary through review of limiting transients, feedwater flow can be terminated automatically with safety-grade equipment following detection of an excessive cooldown. If the problems of concern are found to be only with the control system (and not, for example, with feedwater valve failures) then safety-grade interlocks could be used to redundantly override the control system and terminate feedwater. If there is a concern with excessive feedwater caused by valve malfunction (such as a feedwater control valve failing open) feedwater could be terminated with safety-grade equipment by closing redundant valves or by tripping feedwater pumps and closing a single set of valves for redundancy. This method of terminating feedwater flow could, however, require the addition of expensive equipment on some plants. Also, analyses would have to be performed to determine if feedwater pump trip or valve closure could be accomplished sufficiently rapidly to mitigate any transient of concern.

There are control system failures which can cause excessive steam flow through electric, air, or hydraulic operated steam valves which could lead to reactor coolant system cooldown. As with the feedwater flow, steam flow could be terminated with safety-grade interlocks or safety-grade isolation valves following detection of excessive cooldown. If a cooldown transient, however, is initiated by a "stuck open" safety valve, it could not be terminated by safety system equipment since design codes prohibit isolation valves in series with safety valves.

Inadvertent reactor coolant system pressure increases caused by control system failure can be terminated by redundantly turning off pressurizer heaters or redundantly terminating charging flow if shown to be necessary. However, it should be noted that inadvertent cooldowns of sufficient magnitude will, in general, result in eventual automatic initiation of safety injection which, in turn, results in an increase in reactor coolant pressure if operator action is not taken.

A number of plants currently employ interlocks and valves which are redundant and at least "quasi-safety-grade" to automatically terminate feedwater flow and/or steam flow under conditions which could lead to inadvertent cooldown, overfill of steam generators (PWRs), or overfill of reactor vessels (BWRs).

In addition to inadvertent cooldowns or increases in pressure which can be caused by control system failures, actuation of certain emergency safeguards systems can cause inadvertent cooldown and consequential increase in reactor coolant pressure. For example, actuation of auxiliary feedwater on a PWR following a reactor trip can cause an inadvertent reactor coolant system cooldown, contraction of water in the reactor coolant system, depressurization of the reactor coolant system, automatic actuation of safety injection, and then a repressurization of the reactor coolant system. This could occur if operator action is not taken to manually control auxiliary feedwater after its automatic initiation. During recent operating license reviews, the Instrumentation and Control Systems Branch has been reviewing the circuits, equipment, and indications used by the operator to control auxiliary feedwater after automatic initiation with the goal of ensuring that a single failure will not cause uncontrolled auxiliary feedwater flow. A staff position on the design of the auxiliary feedwater system, including instrumentation and controls, has been proposed and is currently under review by the Division of Safety Technology. Implementation of this position would significantly improve the failure tolerance of the auxiliary feedwater system from the standpoint of failures which could result in excessive plant cooldown.

J.5 Actions Completed or Underway to Detemine Potential Consequences of Control System Failures

The consensus judgment of the NRC staff continues to be that the risk associated with control system failures is not sufficient to require immediate corrective actions. However, to provide added assurance, the following actions are being or have been taken:

(1) The resolution of Unresolved Safety Issue A-47, "Safety Implications of Control Systems" will systematically determine if current licensing practices with respect to control systems are adequate. The plan for resolution of this issue specifically addresses evaluations to determine any actions required to prevent control system failures from causing unacceptable reactor coolant system cooldown or overfill of a steam generator (PWR) or reactor vessel (BWR).

(2) Standard Review Plan Section 7.7 calls for staff reviews to assure that failures of control systems will not impair the capability of the protective system in any significant manner or cause plant conditions more severe than those for which the plant safety systems are designed. The staff has pursued these reviews primarily to ensure that electrical interconnections between protection systems and control systems are implemented such that failures in control system equipment cannot impair the operation of protection system equipment. The Chapter 15 design-basis event analyses have also been reviewed to assure that sufficient conservatism has been assumed so that these analyses adequately bound the consequences of single control system failures. The Instrumentation and Control Branch has been reviewing control system designs of operating license applicants to confirm that the Chapter 15 design bases analyses also bound multiple control system failures initiated by credible failures of common power sources, sensors, or sensor impulse lines. In addition, operating license applicants have been requested to review the potential for control system malfunctions caused by high energy line breaks. Section 7.7 of the Standard Review Plan was revised in 1981 to be more explicit on criteria applicable to control systems. Specifically, the criteria shown in Table J.1 are now delineated in Section 7.7 and reviews of plants currently under licensing review are performed with the goal of verifying that the criteria are met.

(3) In September 1979, all licensees were asked to review the possibility of consequential control system failures which could exacerbate the effects

of high energy line breaks and identify appropriate actions, where needed, to assure that these events would be adequately mitigated. The review was requested as a result of postulated scenarios involving consequential control system failures identified by Westinghouse. All licensees responded to the request and the responses were screened. On the basis of the review, no specific event leading to unacceptable consequences was identified and, in general, control equipment locations were such that consequential failures would be unlikely. Some licensees, however, did make changes to operating procedures to address the possibility of control failures.

(4) I&E Bulletin 79-27 was issued to licensees requesting that evaluations be performed to ensure the adequacy of plant procedures for accomplishing shutdown upon loss of power to any electrical bus supplying power for instruments and controls. In their response to the bulletin, licensees have indicated that corrective action has been taken including hardware changes and revised procedures, where required to assure that the loss of any single instrument bus would not result in the loss of instrumentation required to mitigate such an event. As part of operating license reviews, we are requesting similar verification by operating license applicants.

(5) Implementation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following An Accident," and NUREG-0737, "Clarification of TMI Action Plan Requirements," will significantly upgrade both the quantity and quality of information available to the operator to diagnose and respond to control system failures.

(6) In 1979 B&W completed a failure modes and effects analysis and review of operating experience for their Integrated Control System (ICS) and reported the results in B&W Report BAW-1564, "Integrated Control System Reliability Analysis." B&W made several recommendations regarding control system improvements which could be made to improve overall plant performance. Licensees with B&W plants were requested to evaluate the B&W recommendations and report their follow-up actions to the staff. Responses were received and reviewed. Based on the review of BAW-1564 and the responses

to the B&W recommendations, the staff has not identified any specific control system failures or actions that would lead to unacceptable consequences.

(7) The Office of Nuclear Regulatory Research is coordinating efforts with the IEEE to establish design criteria for systems important to safety which are not covered by and do not need to meet all of the rigorous standards for safety system equipment but nevertheless may be sufficiently important to safety to be included in the NRC review process.

J.6 Conclusions

At this time, the staff knows of no specific control system failures or actions which would lead to unacceptable consequences. A variety of efforts are still underway to determine the potential safety consequences of control system failures including their impact on pressurized thermal shock. Should these reviews indicate that additional criteria for control system designs are necessary or that specific problems require resolution, appropriate action will be taken for plants in the licensing process and for plants now in operation.

Table J.1

Standard Review Plan Guidance for Control System Review

(1) Confirm That The Plant Accident Analyses in Chapter 15 of the SAR Do Not Rely On The Operability Of Control Systems To Assure Safety.

(2) Confirm That The Safety Analyses Include Consideration Of The Effects Of Both Control Systems Action And Inaction In Assessing The Transient Response Of The Plan For Accidents And Anticipated Operational Occurrences.

(3) Confirm That Consequential Effects Of Anticipated Operational Occurrences And Accidents Do Not Lead To Control Systems Failures Which Would Result In Consequences More Severe Than Those Bounded By The Analyses In Chapter 15 Of The SAR.

(4) Confirm That The Failure Of Any Control System Component Or Any Auxiliary Supporting System For Control Systems Will Not Cause Plant Conditions More Severe Than Those Bounded By The Analyses Of Anticipated Operational Occurrences In Chapter 15 Of The SAR (The Evaluation Of Multiple Independent Failures Is Not Intended).

APPENDIX K

EFFECTS OF HEATING ECCS WATER ON PRESSURIZED THERMAL SHOCK

Increasing the temperature of the ECCS water can have a positive effect on PTS for LOCA events, where the dominant overcooling results from the injection of the cold ECCS water.

K.1 Large- and Small-Break LOCAs and Secondary Side Effects

It can be shown by analysis that large-break LOCAs are not considered to be a serious PTS problem. This is because in the unlikely event of a large break in the primary system, high pressure cannot be maintained in the reactor pressure vessel. Small-break LOCAs (less than two inches equivalent diameter) also are not a PTS problem because breaks in this size range result in cooldown rates of less than 100°F per hour. Such transients do not cause large thermal stresses. Breaks in the range of two up to possibly as large as six inches are of concern. These breaks are capable of removing all of the decay heat generated in the core and do not require or establish natural circulation for decay heat removal. Mixing of ECCS water in the downcomer is minimized in this case. (See Section K.3.) In addition, reactor system pressure can remain relatively high (~1200 psi) for the considerable amount of time required to uncover the break (i.e., steam discharge out of the break), or repressurization can occur after initiation of the break for some plants with high head HPI pumps. This scenario, loss of natural circulation with high pressure, at present appears to be the one most likely to benefit from heating ECCS water in order to reduce the PTS problem. For secondary side events (e.g., main steam line breaks), rapid cooldown and depressurization of the primary system can occur. ECCS actuation will repressurize the primary system. However, since there is no primary system LOCA, only a limited volume of ECCS water will be injected into the primary system by the operator to make up for shrinkage due to cooldown. Therefore, as far as PTS is concerned, the cooldown is not affected as much by ECCS injection as by primary to secondary heat transfer. However, for certain secondary side events (e.g., steam and feedline breaks) including steam generator tube rupture, interruption of circulation and consequent temperature transients that could be influenced by ECCS water temperature could be possible. At this time, sufficient analysis has not been done, and conclusions regarding these events would be premature.

K.2 Plants Which Have Raised ECCS Water Temperature

Several plants have the capability to heat the ECCS water. Connecticut Yankee heats the refueling water storage tank (RWST) to 50°F in the winter to prevent water in the outdoor tank from freezing. Maine Yankee has a technical specification to maintain the RWST at a minimum temperature of 40°F. The water currently is heated no higher than 80°F. Yankee Rowe is the only U.S. plant that heats its ECCS water substantially above normal ambient temperatures, even in the summer. The safety injection tank water temperature is maintained at 120°F (130°F maximum) to minimize any PTS problem. A review has been conducted by Yankee Atomic Electric Company to ensure that the increased water temperature would not adversely impact postulated accidents.

The Loviisa plant in Finland maintains the ECCS water temperature between a minimum of 113°F and a maximum of 140°F. One of the reasons for this is because the low pressure ECCS system injects through nozzles directly into the reactor vessel. There is no mixing in the cold leg, so the ECCS water is heated to minimize the thermal shock.

K.3 Mixing of ECCS Water

Mixing of ECCS water with water in the reactor vessel has been and continues to be evaluated through analysis and experimentation (Ref. K.1-K.3). As long as adequate reactor coolant flow is maintained, good mixing of ECCS water in the cold leg downcomer is expected, and heating the ECCS water is expected to be of little benefit from a PTS standpoint. In the event that loop flow stagnated, the degree of mixing of ECCS water injected into the cold leg is less certain. If mixing were minimal, colder ECCS water could contact the reactor vessel wall, and therefore, heating the ECCS water would be beneficial in reducing thermal stresses.

Investigations of the mixing phenomena under stagnant loop flow conditions are underway in order to better quantify the degree of mixing.

K.4 Maximum ECCS Water Temperature

The maximum heating that could be allowed without causing other problems with ECCS operation has not been calculated. The impact on containment sprays, pump net positive suction head, and ECCS performance are examples of factors that could limit the water temperature. Evaluations such as these would have to be done on a plant-specific basis.

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K.2. Rothe, P. H., and M. F. Ackerson, Fluid and Thermal Mixing in a Model Cold Leg and Downcomer with Loop Flow, EPRI NP-2312, April 1982.

K.3. Levy, S., An Appropriate Prediction of Heat Transfer During Pressurized Thermal Shock, SLI-8213, June 1982.

APPENDIX L

NONDESTRUCTIVE EVALUATION METHODS

L.1 Detectability of Underclad Cracks

In order to have confidence that an inservice inspection (ISI) could detect near surface flaws in reactor pressure vessels that would be of interest in a pressurized thermal shock incident, it is necessary to demonstrate high probabilities of detection for 6.0 mm and larger cracks. Cracks of interest are those normal to the inside surface and oriented both parallel and perpendicular to the direction of clad lay. Weld defects within the first 25 mm as well as cracks resulting from clad deposition are of interest. European techniques using longitudinal waves are generally accepted as providing optimum detection results and have been shown to be effective in detecting 3.0 mm or smaller underclad cracks under the more ideal conditions of smooth clad and cracks predominantly perpendicular to the clad lay found in European pressure vessels. Most circumferential welds in U.S. pressure vessels have been clad using the manual metal arc (MMA) process. This welding process creates rough and noisy inspection conditions that inhibit inspection effectiveness. The NRC has, therefore, requested the Pacific Northwest Laboratory (PNL) to evaluate the reliability and effectiveness of these techniques for inspecting U.S. vessels. (See Section L.2.) Results of tests show that light grinding of the clad surface (specifically improving the surface roughness by a factor of 2, from 0.012 in. RMS to 0.006 in. RMS) improves the crack detectability confidence level from low to very high.

Further work is planned to refine the measurement methods for clad conditions, develop appropriate examination (and calibration) methods, determine crack detection probabilities for various inspection techniques, and to establish performance of techniques for crack sizing. Hence, the surface roughness and cladding noise under field conditions could be quantified, a criteria established for determining if the cladding conditions permit a valid inspection to be performed, and a procedure given for an effective inspection.

L.2 Influence of Improved NDE Techniques

PNL has developed estimates to predict the influence of improved vessel examination techniques on vessel failure and allowable RT_{NDT}.

Table L-1 summarizes the results of this investigation. Using "best estimates" on probability of flaw detection, we have attempted to provide bounds for adjustments in allowable RT_{NDT} to reflect the benefit of optimized vessel inspection techniques. Table L.1 shows that the probability of flaw detection using optimized techniques varies from 50 to 95%, depending on clad type and surface finish. The corresponding benefit from inspection expressed as an increase in allowable RT_{NDT} varies from 10 to 33°F. In addition, we have provided supporting material for fracture mechanics and NDE in Sections L.2.1 and L.2.2 that indicate methodology used to derive Table L-1.

L.2.1 Fracture Implications of Improved Inservice Inspection

The results of probabilistic fracture mechanics calculations were available to PNL from the work of Mr. J. Strosnider of NRC (see Appendix H). These results were used to estimate an allowable increase in RT_{NDT} which could be justified on the basis of the estimated probability of crack detection for inservice inspection (ISI).

Figure L-1 shows trends of the NRC results for failure probability as a function of RT_{NDT}. Results for the NRC cooling rate curves for parameters $\beta = 0.051$, 0.15, and 0.50 are shown along with other results for the temperature/pressure curves of the Rancho Seco transient. It was assumed that the range of interest was a failure rate of 10^{-4} given the occurrence of a transient. In Figure L-1, Po is the probability of failure at a given transient and RT_{NDT}, and P is the probability of failure for the same transient but increased value of RT_{NDT}. All calculations here were based on the $\beta = 0.15$ cooling rate parameter.

<u>TABLE L-1</u>

ESTIMATED DETECTABILITY OF UNDERCLAD CRACKS
AND ESTIMATED INCREASES IN ALLOWABLE RT_{NDT}

CLAD	FINISH	FLAW DIRECTION WITH RESPECT TO CLAD	PROBABILITY OF DETECTION	FACTOR OF IMPROVEMENT (1) IN RELIABILITY	ALLOWABLE INCREASE IN RT_{NDT}, °F
Strip	Smooth	Perpendicular and Parallel	95%	20 to 40	27 to 33
Single Wire Strip	Smooth Unground	Perpendicular Perpendicular	85%, 0.5"-1.0" Flaw 90%, 1.0" or Greater Flaw	7.4 to 14.8	17 to 24
Single Wire Strip	Smooth Unground	Parallel Parallel			
Manual	Ground	Perpendicular and Parallel	75%, 0.5"-1.00" Flaw	4.3 to 8.6	13 to 19
Single Wire	Unground	Perpendicular and Parallel	80%, 1.0" or Greater Flaw		
Manual	Unground	Perpendicular and Parallel	50%, 0.5"-1.0" Flaw 75%, 1.0" or Greater Flaw	2.8 to 5.6	10 to 15

(1) Factor of Improvement = Probability of Failure <u>without</u> Inspection/Probability of Failure <u>with</u> Inspection.

Lower bound assumes flaws are isolated and independent occurrences. Upper bound assumes possible occurrence of multiple flaws in a given weld.

TABLE L-2

ESTIMATE OF FAILURE PROBABILITY WITH AND WITHOUT INSERVICE INSPECTION

A	P(A)	P_{ND}	P(F/A)	Failure Probability $P(A) \cdot P(F/A)$ (without ISI)	Failure Probability $P(A) \cdot P_{ND}$ (with ISI)
0.125	8.3×10^{-1}	0.5	0	0	0
0.25	1.6×10^{-1}	0.05	1.5×10^{-4}	2.4×10^{-5}	1.2×10^{-6}
0.50	4.2×10^{-3}	0.5	1.0×10^{-2}	4.2×10^{-5}	2.1×10^{-6}
1.0	4.1×10^{-4}	0.05	5.4×10^{-2}	2.2×10^{-5}	1.1×10^{-5}
1.5	1.3×10^{-4}	0.05	5.6×10^{-2}	7.3×10^{-6}	3.6×10^{-7}
2.0	4.2×10^{-5}	0.05	4.5×10^{-2}	1.9×10^{-6}	9.5×10^{-8}
2.5	1.3×10^{-5}	0.05	-	-	-
3.0	5.0×10^{-6}	0.05	-	-	-
3.5	3.3×10^{-6}	0.05	-	-	-
				$Po(F) = 9.7 \times 10^{-5}$	$P(F) = 4.8 \times 10^{-6}$

Notes: (1) Based on data from status report by Jack Strosnider on "Failure Probability of a RPV Subject to Pressurized Thermal Shock," March 5, 1982

(2) For "Rancho Seco Transient Reference Case," mean copper = 0.34, mean RTN_{DT} = 0.0 and mean fluence = 3.0×10^{19}

(3) Probability of flaw nondetection (P_{ND}) for smooth strip clad

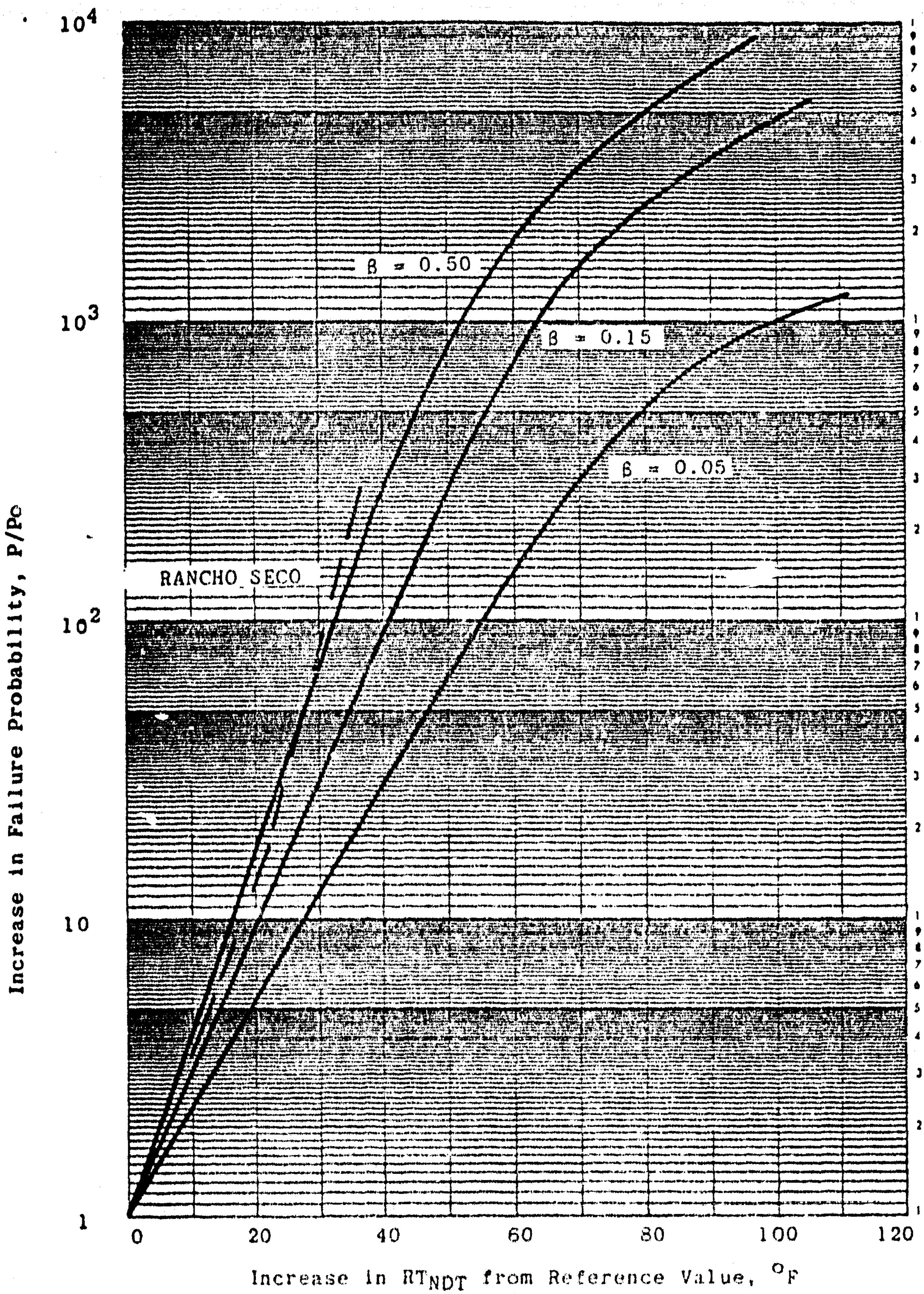

Figure L-1 Relationship Between Failure Probability and RT_{NDT} as Indicated by NRC Analysis of Pressurized Thermal Shock of Reactor Vessels.

In PNL's calculations, the decrease in vessel failure probability due to ISI was first estimated. A trade-off between this decrease with an offsetting increase in failure probability due to relaxation in RT_{NDT} requirements was then performed. Table L-2 illustrates the estimate of failure probability as a function of probability of nondetection of a flaw (P_{ND}). In Table L-2:

A = Flaw depth

$P(A)$ = Probability of a flaw of depth A in the critical weld

$P(F/A)$ = Probability of failure for the Rancho Seco transient given the presence of a flaw of depth A

$P_{ND}(A)$ = Probability of not detecting a flaw of depth A based on PNL estimates

$P(A)\cdot P(F/A)$ = Probability of failure without ISI given the occurrence of the Rancho Seco transient

$P(A)\cdot P(F/A)\cdot P_{ND}$ = Probability of failure with ISI given the occurrence of the Rancho Seco transient

Table L-2 used the best detection capability corresponding the more favorable conditions of PNL's flaw detection studies. Results for other inspection conditions are given in Table L-1. The ratio of failure probabilities in Table L-2 was 20:1 for the no ISI case versus the ISI case. Turning to Figure L-1, an increase in RT_{NDT} of 27°F will give an offsetting 20:1 factor in failure probability. Therefore, it is estimated for this particular example that the allowable RT_{NDT} can be increased by 20°F with no net increase in failure probability provided that inservice inspection is performed.

It is recognized that the flaw size distribution in the NRC probabilistic analyses is subject to considerable uncertainties. Therefore, the estimated flaw size distribution as modified by ISI is subject to the same uncertainties. However, the relative improvement in reliability due to ISI is believed to be significantly more accurate than the absolute values of failure probability.

The upper bound estimate of the allowable increases in RT_{NDT} is an attempt to consider the statistical nature of underclad cracks. Evidence suggests that one can expect either no cracks at all or a large number of cracks. Given a large number of cracks is indeed very small ($P_{ND}\eta = 0.05^{10} = 10^{-13}$), thus, one can arrive at vastly different conclusions regarding the benefits of ISI, depending on the assumption on the stochastic structure of the flaw distribution. The upper bound estimate as shown in Table L-1 on the benefit of ISI conservatively assumes that half the flaws in vessels are random occurrences and that the remaining flaws occur in groups so to be readily detectable. The assumption that all flaws are random occurrences will tend to greatly underestimate the potential benefits of ISI. On the other hand, it is unreasonable to assume that random flaws will not occur, since one can be led to accept any level of embrittlement in a vessel provided that an ISI reveals no flaws.

L.3 Flaw Detectability Measurements

Flaw detectability experiments have been carried out on strip clad, single wire sub arc clad, and manual clad. Both ground and unground surfaces were evaluated. The test blocks used for this evaluation were: a 750-mm-dia. strip clad pressurizer dropout, two 600-mm square blocks with strip and single wire clad with one side ground and the other as welded*, two small blocks with ground and unground manual clad. The pressurizer dropout contained through clad notches as well as actual thermal fatigue underclad cracks. The two EPRI blocks contained unclad notches and the manual clad samples contained two reference reflectors for evaluation of general noise level. The measurements reported here were taken using a 2-MHz dual beam longitudinal (SEL) 70° transducer, with 10- by 15-mm elements and focal cross over point of 17 mm. This unit was considered optimum for the clad conditions and thicknesses (6 to 9 mm) tested. All measurements were performed manually.

The results of signal amplitudes compared to the signal amplitude of a 3 mm flat bottom reference reflector are shown in Table L-3. In addition, a blind test was conducted. This blind test used the pressurizer dropout sample that contained nine actual underclad cracks generated by a thermal fatigue process. The

*Access to these two samples was made possible through J. R. Quinn, Electric Power Research Institute (EPRI), Palo Alto, CA.

cracks were oriented both parallel and perpendicular to the direction of the cladding. The cracks ranged in depth from 0.25 to 0.75 inch through the wall. Although none of the three operators had prior knowledge of crack location, each operator detected every crack. The probability of detection data reported in Table L-1 are estimates based on an optimized inspection system, our flaw amplitude measurement and our blind test.

TABLE L-3

FLAW AMPLITUDE RESPONSE

SENSITIVITY STANDARD: 3MM FLAT BOTTOM REFERENCE REFLECTOR

SAMPLE TYPE	FLAW DEPTH RANGE	FLAW RESPONSE RANGE (+) GREATER REFERENCE REFLECTOR
Ground; Strip Clad; Underclad Notch	5mm to 18mm	0 to +9dB
Unground; Strip Clad; Underclad Notch	5mm to 18mm	0 to +8dB
Ground, Single Wire; Underclad Notch	5mm to 18mm	-1 to +10dB
Unground; Single Wire; Underclad Notch	5mm to 18mm	-1 to +12dB
Ground; Strip Clad Pressuirzer Dropout Underclad Cracks	5mm to 18mm	0 to +11 dB

APPENDIX M

IN SITU ANNEALING

Annealing of the beltline region of reactor vessels is a potential remedial measure for the PTS problem for vessels that have suffered considerable radiation embrittlement.

Time-Temperature Effects on Recovery of Properties

There is a fairly good experimental basis for choosing the annealing temperature and time. From the Naval Research Laboratory, research funded by the NRC has revealed the effects of annealing at 650°F and 750°F and the effects of reirradiation and reannealing (Ref. M.1-M.3). Research at Westinghouse funded by the Electric Power Research Institute (EPRI) has revealed the effects of annealing at temperatures of 650, 700, 750, 800 and 850°F (Ref. M.4). As expected, there is a clear trend toward better recovery of properties at the high temperatures and at longer times up to one week. In this discussion, therefore, we will assume that annealing would be done at 850°F for one week, and that the resulting recovery of fracture toughness properties would be about 80 percent.

Reirradiation Effects

With regard to the rate at which ΔRT_{NDT} increases upon reirradiation, the data are scattered and somewhat conflicting. The rate of reembrittlement should be as low as that just prior to annealing, and is almost certainly significantly lower than that at the start of life. Thus, a plant that annealed its vessel after, say 8 EFPY should expect much more than 8 additional EFPY before reaching the same ΔRT_{NDT}. Obviously, a better estimate of the reirradiation rate is desired for economic considerations before undertaking annealing; but for purposes of safe operation in later years, there will be additional information from test reactor programs and from plant surveillance data.

One technical question that has yet to be thoroughly investigated is the verification test program for a specific plant, which will be required to measure the effects of the annealing operation and the reirradiation.

Appendix G of 10 CFR 50 requires that the degree of recovery be measured "...by testing additional specimens that have been withdrawn from the surveillance program capsules and that have been annealed under the same time-temperature conditions as those given the beltline material."

The specimens in most capsules have been irradiated substantially more than the vessel; hence, measurement of ΔRT_{NDT} for those specimens after annealing should give a conservative estimate of the condition of the vessel. Their use as a guide to the rate of reembrittlement is not well understood. One alternative is to test "reconstituted" Charpy specimens from earlier surveillance capsules, i.e., fabricate Charpy specimens by welding ends on the broken halves of specimens that have lower fluences because they were withdrawn from the vessel early in life. Another alternative is to irradiate archive material to the desired fluence in test reactors and then check the effects of annealing and reirradiation.

With regard to the feasibility of annealing, NRC staff has the results of the EPRI study (Ref. M4) and the (potential) advice of vessel fabricators who have experience in post-weld heat treatment after field fabrication and after repairs. The EPRI study developed a means of heating by electric resistance elements supported on a frame that would be lowered into the vessel before the water is removed. No insurmountable difficulties were reported, but many engineering details remain to be resolved.

From the standpoint of risk, the main concern seems to be the potential for distortion of the vessel and the economic risks associated with problems in reinstallation of the core support structure and the closure head. At 850°F, some creep and relaxation could occur at regions where there are significant stresses caused by differential expansion during heatup and cooldown, by residual stresses, and by the stresses near the supports caused by the dead weight of the vessel. These problems have not been dealt with very carefully or completely as yet. From what has been done, it dos not appear that the piping would have to be separated from the vessel. Again, the experience of field fabricators of vessels must be tapped.

Other components that require study of the risks of annealing are the vessel insulation, the adjacent concrete and the supports. The movement of the vessel relative to the support when heated to 850°F will of course be greater than that at the design temperature of 650°F. Also, for those supports where the concrete is only a short distance below the vessel nozzle that must carry the load, the structural integrity of the concrete must not be impaired.

In conclusion, it appears that from the safety standpoint the benefits of annealing are quite clearcut and the risks are low. The risks of annealing are economic risks. There is, of course, a cost in man-rem and dollars if everything goes as planned. The largest uncertainty remains the economic and exposure risks associated with correction of distortion of the vessel or other damage if things do not go as planned.

REFERENCES

M.1. J. R. Hawthorne, "Status of Knowledge of Radiation Embrittlement in USA Reactor Pressure Vessel Steels," NUREG/CR-2511, NRC Memo Rept. 4737.

M.2. C. Z. Serpan, H. E. Watson, and J. R. Hawthorne, "Interaction of Neutron and Thermal Environmental Factors in the Embrittlement of Selected Structural Alloys for Advanced Reactor Applications," Nuclear Engineering and Design, V.II, 1970, pp. 368-380.

M.3. C. Z. Serpan and J. R. Hawthorne, "Yankee Reactor Pressure Vessel Surveillance: Notch Ductility Performance of Vessel Steel and Maximum Service Fluence Determined from Exposure During Cases 2, 3, and 4. ASME Journal of Basic Engineering, V. 89, 1967, pp. 897-910.

M.4. EPRI Thermal Annealing Program, Research Project 1021-1. Development of a Generic Procedure for Thermal Annealing on Embrittled Reactor Vessel Using a Dry Annealing Method. EP Interim Report July, 1982, T. R. Mager and R. D. Rishel, PN-2483. Only drafts of other parts of the final report were available as of August 15, 1982.

APPENDIX N

FUTURE CONFIRMATORY STUDIES

N.1 Introduction

The following issues relating to pressurized thermal shock require confirmatory study.

1. Applicability of Linear Elastic Fracture Mechanics (LEFM) for initiation, propagation and arrest for reactor pressure vessels subjected to a pressurized thermal shock scenario.

2. Effectiveness of Warm Prestress

3. Vessel failure under nonpressurized thermal shock conditions.

4. Behavior of small finite flaw when subject to PTS conditions.

5. Cladding-flaw interaction; bimetallic effects.

6. Irradiated cladding material and fracture properties.

7. Arrest on the upper shelf.

8. Postarrest performance for a deep crack in upper shelf material toughness.

9. Definition of margin when using RT_{NDT} to set fracture toughness curves.

10. Variation of through-wall fracture toughness degradation.

11. Validation of fracture toughness degradation as a function of fluence for ferritic welds.

12. Effect of trace elements (copper, nickel, phosphorus) on the embrittlement rate of RPV steels at reactor operating conditions.

13. Effectiveness of thermal annealing on fracture toughness recovery and reembrittlement rate.

14. Establishments of criteria and standards to be applied to any proposed, in situ thermal annealing of operating reactor vessels.

N.2 Summary of Prior Studies

Thick section pressure vessel materials have been characterized to form the basis for fracture toughness and crack growth data in the ASME B&P codes. Crack arrest methodology has been extensively evaluated and preliminary specimen designs developed. Methods of elastic-plastic fracture analysis have been developed and evaluated. Irradiation effects on pressure vessel plate, forging and weldments, including low-shelf thoughness weldments, have been studied using compact specimens up to 4 inches thick. Thirteen intermediate tests have been performed on nine vessels to validate methods of fracture-failure analyses, to demonstrate the capability of NDE methods and repair procedures in thick sections. Seven thermal shock (unpressurized) experiments have been performed on thick-section cylinders to demonstrate the applicability of LEFM in predictions of flaw behavior and to establish the applicability of small specimen toughness determinations in fracture analysis. Unique crack arrest data have been developed in these tests. Small scale stainless steel cladding tests have been performed to determine the influence of cladding on flaw development. Computer codes have been developed to evaluate fracture potential to define and quantify the principal variables that need to be considered in operating systems. The effect of trace elements, such as copper, nickel and phosphorus, on the embrittlement potential of commonly used reactor pressure vessel steels when subject to different levels of neutron bombardment has been determined. The effect of thermal annealing, at various temperatures on the fracture toughness recovery of neutron embrittlement steels has been defined and quantified. Elastic-plastic material fracture toughness testing procedures have been developed and elastic-plastic fracture data basis are being developed for unirradiated and irradiated reactor pressure vessel steels. Extensive participation with NRR, code writing bodies (ASME, ASTM), information dissemination through formal and informal exchanges, and international cooperative efforts have been maintained.

N.3 Present Programs Addressing Issues

The following programs are underway or are planned to address the issues identified in Section N.1. The numbers in parentheses refer to issues in Section N-1.

- Complete 3 dimensional finite element fracture computer codes [ORFLAW-3D and ORVIRT-3D] (4)(5)(6)
  - completion date: March 1983

- Complete evaluation of finite flaw behavior (4)
  -completion date: December 1982

- Complete development of unified LEFM-EPFM methodology considering all regimes of toughness (1)(2)(4)(5)(7)(8)
  -completion date: September 1983

- Complete testing of low-shelf weldments (8)(11)
  -completion date: December 1983

- Complete testing 1TCT irradiated specimens of present practice steel (11)
  -completion date: September 1983

- Complete irradiation of cladding material (6)
  -completion date: June 1983

- Complete testing of irradiated cladding material (6)
  -completion date: December 1984

- Complete material procurement for K_{Ic} (4T) study (7)(9)(11)
  -completion date: December 1983

- Complete irradiation for K_{Ic} (4T) study (7) (9)(11)
  -completion date: June 1985

- Complete testing for K_{Ic} (4T) study (7)(9)(11)
  -completion date: September 1985*

- Complete development of irradiated crack ar. ,t data base (6)(7)(9)(10)(11)
  -completion date: September 1986

- Complete probabilistic fracture mechanics version of Computer Code OCA-2 (9)
  -completion date: September 1982

- Complete Thermal Shock Experiment TSE-7 (1)(3)(4)
  -completion date: March 1983

- Complete Thermal Shock Experiment TSE-8 (1)(3)(4)(5)
  -completion date: March 1984

- Complete Thermal Shock Experiment TSE-9 (1)(3)(4)(5)
  -completion date: March 1985

- Complete feasibility study and system design for Pressurized Thermal Shock Experiments (1)(2)(4)(5)(7)(9)
  -completion date: September 1982

- Complete PTSE facility construction checkout (1)(2)(4)(5)(7)(9)
  -completion date: April 1983

- Complete PSTE-1 (1)(2)(4)(5)(7)(9)
  -completion date: March 1983

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*Interim data from testing program will be available at earlier dates in 1984 and 1985.

- Complete PSTE-2 (1)(2)(4)(5)(7)(9)
  -completion date: March 1984

- Complete development of crack arrest specimen and test procedures (1)(7)
  -completion date: March 1983

- Complete construction of capsules and begin irradiation of specimens in dose rate study. (11)(12)(13)
  -completion date: October 1982

- Complete dose rate study [show irradiation more closely simulating operating reactor experience] (11)(12)(13)
  -completion date: October 1985

- Complete testing of SSC-2 and PSF dosimetry specimens (10(11)(12)
  -completion date: March 1983

- Complete variable radiation sensitivity study (11)(12)
  -completion date: May 1983

- Complete high temperature (454°C) annealing study (13)(14)
  -completion date: March 1985

- Complete high temperature (454°C) annealing
  -reembrittlement rate study (12)(13)(14)
  -completion date: May 1986

- Complete study on the effectiveness of drop weight method of determining NDT and applicability of RT_{NDT} (9)
  -completion date: October 1984

- Complete program on System Requirements and Standards development for annealing of reactor pressure vessels (12)(13)(14)
  -completion date: October 1984

- Completion of testing irradiated material from KRB Block A pressure vessel wall (10)(11)(13)(14)
  -completion date: October 1983

N.4 Applicability of Research

The research program is integrated with the needs of NRC licensing in addressing the issue of pressure vessel integrity, both under normal and accident or upset operating conditions. Every element of the described program is based upon the need of NRR to define and quantify methods use for evaluating pressure vessel safety issues. Every element of the described program is reviewed frequently by NRR, from the U.S. industry, and American and International technical community for its appropriateness and applicability to known or anticipated safety issues. The timeliness of this ongoing research is such that approximately 70 percent of the issues to be resolved in PTS will be addressed and results obtained by the research effort within Fiscal Year 1983. The remaining 30 percent of the intially needed information should be available as follows: 20 percent in FY 1984, 5 percent in FY 1985, and 5 percent in FY 1986. The planned funding effort is as follows:

FY 1982	FY 1983	FY 1984	FY 1985	FY 1986
\$6,650K	\$6,850K	~\$6,000K	~\$6,000K	~\$6,000K

It should be noted that though most of the initial data will be developed as described above, a considerable confirmatory effort must be continued during the years 1983-1986 to ensure that the results obtained are statistically valid. Another reason for the extension of the program through FY 1986 is the time required to carry out an effective irradiation study.

The funding shown above is committed to four contracts through FY 1983 and thereafter to three contracts.

1. HSST program (ORNL)
2. Pressure Boundary Integrity for Water Reactor (ENSA)
3. Pressure Vessel Simulation (ORNL)
4. Systems Requirements for Annealing (EG&G/INEL, terminates FY 1983)

N.5 Confirmatory Studies on Fluence Trend Curves

N.5.1 Refinement of Chemistry and Fluence Factors

Immediate steps must be taken to scrub down the PWR surveillance data base and add data from BWR surveillance. Then a reanalysis is required to refine the copper and nickel terms and determine what the exponent on fluence should be, and whether it should be constant over the whole fluence range. Probably, test reactor data should be omitted until later when a time-temperature parameter is better understood. This represents a change in attitude from that on which Regulatory Guide 1.99 and the MPC trend curves are based. The change reflects the increasing number of surveillance reports in recent years, more than it reflects any increased suspicion that test reactor data and surveillance data are separate populations. There is now an EPRI data base in which the Charpy curves have been fitted by a hyperbolic tangent function and new values of Charpy shift calculated. These values must be compared with the existing data base, which was obtained from curves drawn by eye, and differences reconciled where possible. After these steps are taken; and the new regression analysis is performed, the results will be incorporated in Revision 2 of Regulatory Guide 1.99.

N.5.2 Long-Range Effort

There are two refinements that require further input from research efforts before incorporation in further revisions of Regulatory Guide 1.99. One is the change from fluence measured in terms of neutrons/square centimeter, ($E > 1$ MeV) to fluence measured in terms of a damage function that considers the effects of different energy spectra, probably displacements per atom (dpa). The other refinement to be expected is a time-temperature parameter that accounts for irradiation temperature and exposure time. Both refinements are needed to permit the inclusion of test reactor and surveillance data in the same data base with complete confidence that they belong in the same population.

N.6 Study of PTS Event Scenarios

N.6.1 Objective and Scope

The objective of this study is to provide an independent probabilistic analysis of PTS at a representative B&W, CE, and W PWR. The results will estimate the likelihood of vessel cracking due to PTS, identify what is important (dominant sequences, important operation actions, etc.) and will identify major uncertainties. The results will also provide a comparison of the risk-reduction effectiveness of alternative corrective actions.

The scope of the study is limited to addressing the reliability of pressure vessel integrity and does not address the consequences of vessel failure. The study of the three plants, Oconee 1, Calvert Cliffs 1, and H. B. Robinson 2, will be plant specific. Extension of this study to a generic analysis of classes of plants is beyond the scope of this study.

The study will support resolution of USI A-49 in four ways:

(1) Confirm understanding of PTS; e.g., how likely is vessel failure? What are the important event sequences, operator actions, and control features? How effective are various proposed measures for reducing the likelihood of vessel failure?

(2) Improve methods for analyzing PTS.

(3) Provide a plant-specific analysis of PTS for three plants.

(4) Provide an improved basis for staff evaluation of plant-specific analyses.

N.6.2 Study Plan

The study will use a functional approach rather than a detailed component-by-component approach. Conceptually the plan is first to identify phenomena that could cause overcooling such as too much feedwater or too much ECC; second, to identify initiating events; and then to analyze the reliability of functions that prevent overcooling.

The study involves the following steps for each of the three plants.

First, the analysts (ORNL for probabililistic and fracture-mechanics analysis and LANL/INEL for thermal hydraulic analysis) obtain information on the plant and understand how the plant operates regarding overcooling transients.

Then ORNL performs an event-tree analysis to systematically delineate event sequences that could lead to overcooling and estimates the frequency of occurrences of these sequences.

About a dozen of these sequences are selected for detailed analysis by LANL using TRAC or by INEL using RELAP-5 to calculate temperature and pressure in the downcomer during the transient. These dozen cases are selected to cover a range of severity. Initially, in the Oconee study, both TRAC and RELAP-5 are used to compare and help check out the codes. Subsequently TRAC will be used to analyze Calvert Cliffs, and RELAP-5 will be used to analyze H. B. Robinson.

The method for assessing these TRAC and RELAP-5 models of specific plants (including secondary and control systems) is still being developed. Tentative plans are to calculate plant behavior during a transient such as a turbine trip and compare the results with plant data regarding behavior of turbine-bypass valves, feedwater flow, steam generator levels, reactor coolant temperatures, etc. The intent is to verify that the code behaves reasonably in transients of interest.

For each transient the coolant temperature and pressure calculated by TRAC or RELAP-5 will be used in a fracture mechanics calculation of the conditional probability of vessel failure given that transient occurs. Based on these results, ORNL will estimate the consequence to vessel integrity for each of the transient sequences in the event trees. Each of the sequences will then be sorted into one of a half dozen or so damage bins. These bins will be identified in terms of how many years the plant could operate before the transients in that bin could crack in the vessel. Bins, for example, would be 0-5 yrs., 6-10 yrs., etc. The likelihood of vessel cracking will be added up for all the sequences in each bin to obtain the frequency of vessel-cracking vs. effective-full-power years. Dominant sequences will be apparent in the results.

N.6.3 Status and Schedule

Following a preliminary survey of available information in the Summer of 1981, the Oconee probabilistic study started in FY 1982. The analysis is scheduled to be completed in February 1983, and the draft report in March 1983.

In July 1982, the owners of Calvert Cliffs and H. B. Robinson agreed to participate in the study. This Calvert Cliffs analysis began in August 1982 and should be completed in October 1983 with a draft report completed in January 1984. The schedule for the H. B. Robinson study has not yet been established.

N.7 Reactor Vessel Annealing Study

N.7.1 Status of NRC Annealing Program

In a letter from Harold R. Denton to Robert B. Minogue dated June 15, 1982, NRR requested RES to perform a study to determine the feasibility of conducting a demonstration of in situ annealing of an irradiation-embrittled reactor pressure vessel to restore the fracture toughness properties. The feasibility study would consider potential candidate vessels, the value and generic applicability of information to be gained from a demonstration, and the cost-effectiveness of such a program.

RES engaged EG&G Idaho, INEL, to conduct the study. A survey of the reactor vendors, architect engineering firms, and consultants was conducted by EG&G Idaho during the past two months. The tentative conclusion from the survey is given in the draft interim report EGG-FM-6083, October 1982, "Evaluation of a Reactor Pressure Vessel Anneal Demonstration." The consensus view is that in situ annealing of an irradiation-embrittled reactor vessel is practical.

However, a demonstration anneal is advisable only if the vessel selected possesses the appropriate geometry, materials of construction, and damage level of concern. The review has indicated that Humboldt Bay, Indian Point 1, Shippingport, and BR-3 vessels were not ideal for the demonstration, and therefore a final recommendation has not been made.

The work is continuing to meet the objective of the NRR request.

APPENDIX O

SUMMARY OF ORNL FRACTURE-MECHANICS ANALYSIS FOR SEVERAL PWR RECORDED OCA TRANSIENTS*

Fracture-mechanics calculations were made recently for several PWR overcoooling accidents (OCAs) that have occurred since 1970, including the 1978 Rancho Seco transient (Ref. O-1). Information pertaining to these transients is presented in Table O-1 and Figures O-1 to O-6.

TABLE O-1 PWR OCA DATA[a]

Plant	Date of Accident	Vessel Dimensions (in.) Inner radius	Wall thickness	RT_{NDTo} (°F) Cir. weld	Long. weld
H. B. Robinson	4/28/70	78	9.31	-20	-20
H. B. Robinson	11/5/72				
H. B. Robinson	5/1/75				
Rancho Seco	3/25/78	86	8.5	b	+60
TMI-2	3/28/79	86	8.5	b	+20
Ginna	1/25/82	66	6.5	+20	c

[a]Data obtained from Nuclear Reactor Regulation, NRC, 6/16/82.

[b]Data not available.

[c]Forged vessel (no longitudinal welds).

Figures O-1 to O-6 describe the primary-system-pressure transient and the coolant-temperature transient in the cold leg upstream of the point where the emergency core coolant (ECC) is injected. Because of the location of the temperature measurement, the recorded temperatures are not necessarily accurate indications of he coolant temperatures in the downcomer. For instance, the injection of ECC would result in a lower temperature, and recirculation of

*Contribution by R. D. Cheverton, D. G. Bolls, and S. K. Iskandera of ORNL.

core coolant through the vent valves in a B&W plant would result in higher temperatures. However, the fracture-mechanics calcuations have been made using the recorded temperatures in Figures 0-1 to 0-6 as downcomer temperatures. The curves in Figures 0-1 to 0-6 were digitized for input purposes, using enough time steps to describe the curves accurately; essentially no smoothing of the curves was necessary. Thus, the analysis reflects the effect of nearly all of the irregularities in the curves, except perhaps for the pressure curve in Figure 0-2. For this case it appears that the pressure dropped below 1700 psi but was not recorded. In the calculation, it was assumed for this particular case that the minimum pressure was 1700 psi.

The fracture-mechanics calculations were performed using OCA-II and the basic input data shown in Table 0-2.

In the process of making the fracture-mechanics calculations, a search was made for threshold values of the nil-ductility reference temperature at the inner surface (RT_{NDTS}) corresponding to incipient initiation (II). Results of the analysis are presented in the form of sets of critical-crack-depth curves for the threshold conditions (Figs. 0-7 to 0-18). A summary of the data is shown in Table 0-3.

In Figures 0-7 to 0-18, the existence of minimum points in the constant K_I curves indicates that the requisite conditions for warm prestressing (WPS) exist ($dK_I/dt < 0$). However, the existence of more than one minimum would indicate that K_I fluctuated with time, and under these circumstaces it is not clear that WPS would actually be effective. It was ignored, therefore.

Table O-2. Input Data for OCA Analyses

Parameter	Value	
Vessel dimensions[a]	See Table 1	
Cladding thickness, a/w	0.025	
Flaw Type	Long axial and continuous circumferential on inner surface and extending through cladding	
Range of flaw depths included in analysis, a/w	0.01-0.95	
Limits imposed on critical crack depths, a/w	0.025-0.15	
K_{IC} and K_{Ia}	ASME Section XI	
$(K_{Ia})max$, ksi $\sqrt{in.}$	200	
$\Delta RT_{NDT} = f(Cu, Ni, F)$	$\alpha F^{0.27}$	
Fast neutron fluence (F)	$F = F_0 \exp(-0.24a\ in.^{-1})$	
ΔRT_{NDT} (a)[b]	$= \Delta RTNDT_s{}^c e^{-0.065a\ in.^{-1}}$	
ΔRT_{NDTs}, °F	≤ 500	
Fluid-film heat transfer coefficient (h_f), Btu/hr·ft²·°F	300[d]	
	Cladding	Base Material
Thermal conductivity (k), Btu/hr·ft·°F	10	24
Thermal coefficient of expansion (α), $°F^{-1}$	10×10^{-6}	8.04×10^{-6}
Modulus of elasticity (E) lbs/in.²	28×10^6	28×10^6
Specific heat (c_p), Btu/lb·°F	0.12	0.12
Density (ρ), lbs/ft³	489	489
Poisson's ratio	0.3	0.3

[a]Sets of K_I^* values were calculated for each set of dimensions.

[b]ΔRTNDT at the tip of the flaw.

[c]ΔRTNDT at the inner surface of the vessel.

[d]Corresponds to main circulating pumps off.

TABLE O-3 Results of OCA Analyses

Transient			Weld[a]	$RTNDT_s$,[b] °F	a_c,[d] in
Robinson	1970	broken loop	L	321 (F)	0.93
Robinson	1970	broken loop	C	351 (A)	0.93
Robinson	1972		L	381 (F)	1.4
Robinson	1972		C	>480	--
Robinson	1975	loop C	L	354 (F)	1.4
Robinson	1975	loop C	C	372 (A)	0.93
Robinson	1975	loop B	L	395 (F)	1.4
Robinson	1975	loop B	C	440 (A)	1.2
Rancho Seco	1978		L	295 (F)	1.3
TMI-2	1979	loop A	L	209 (F)	1.3
TMI-2	1979	loop B	L	225 (F)[c]	1.3
Ginna	1982	loop B	C	378 (A)[c]	0.91

[a] L and C refer to longitudinal and circumferential.

[b] F and A in parentheses refer to failure and arrest.

[c] Small increase in $RTDNT_s$ would result in failure.

[d] Critical crack depth.

47c

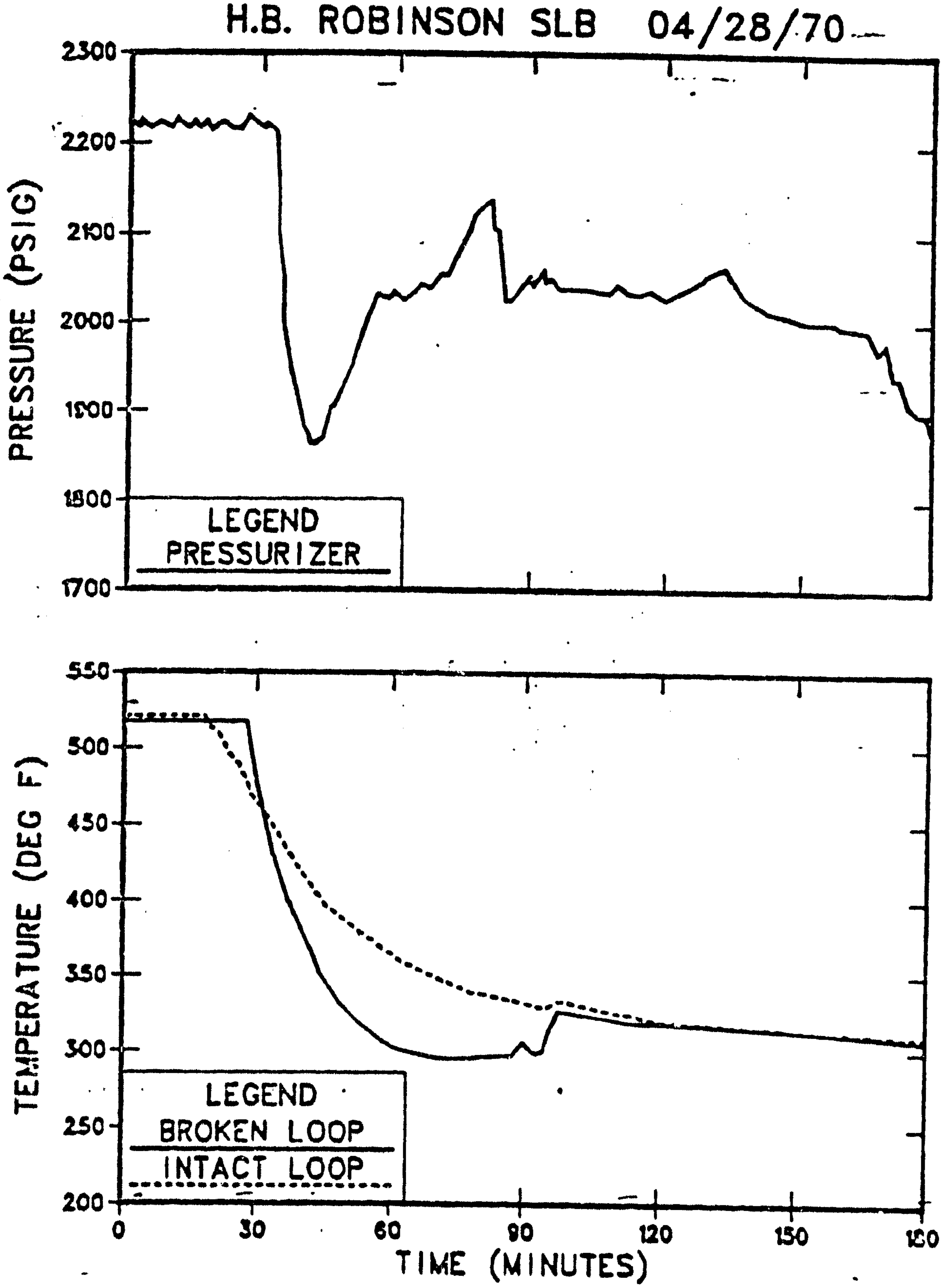
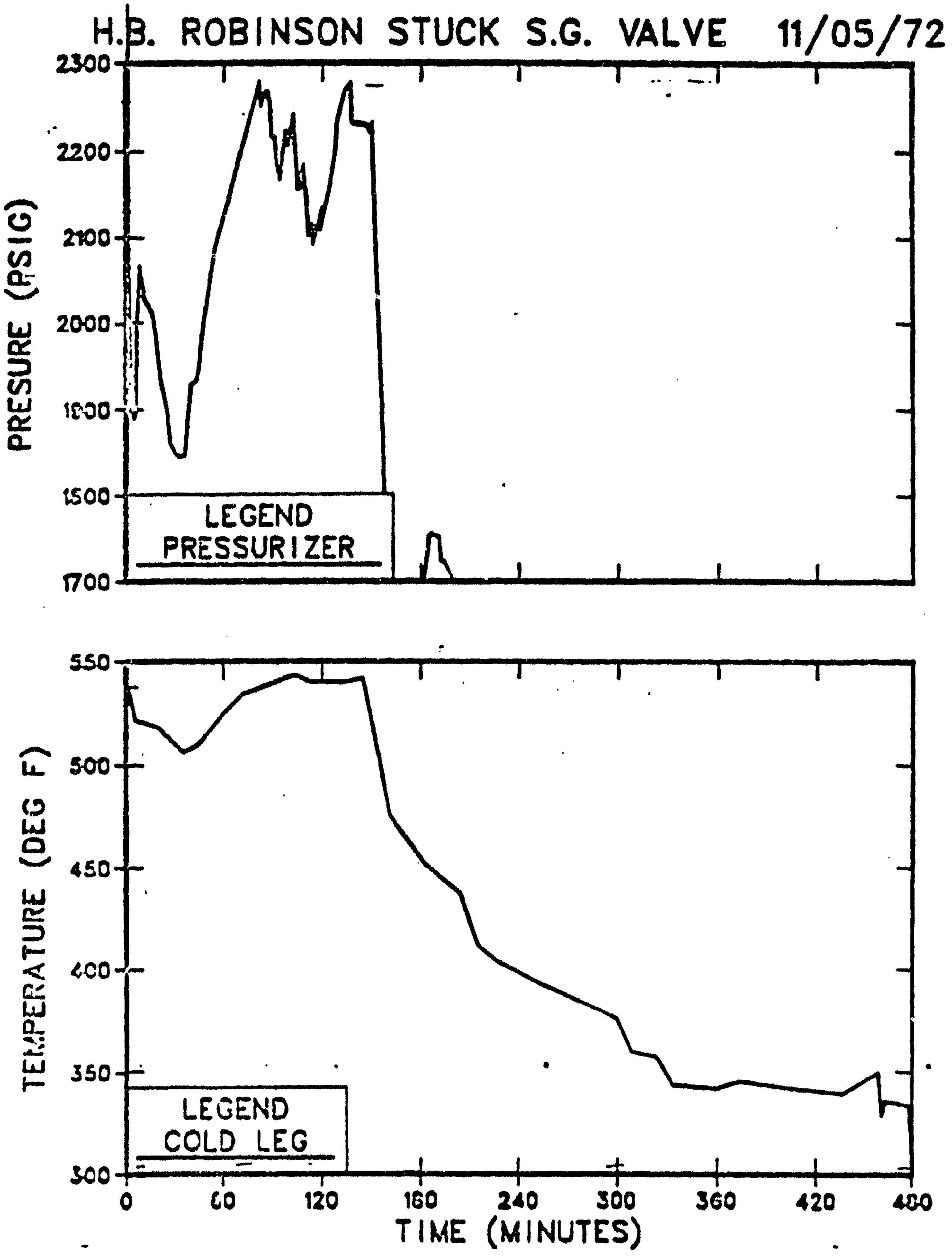


FIGURE 0-1



H.B. ROBINSON STUCK S.G. VALVE 11/05/72
PRESURE (PSIG)
2300
2200
2100
2000
1900
1800
1700
LEGEND
PRESSURIZER
TEMPERATURE (DEG F)
550
500
450
400
350
300
LEGEND
COLD LEG
0
60
120
180
240
300
360
420
480
TIME (MINUTES)

FIGURE 0-2

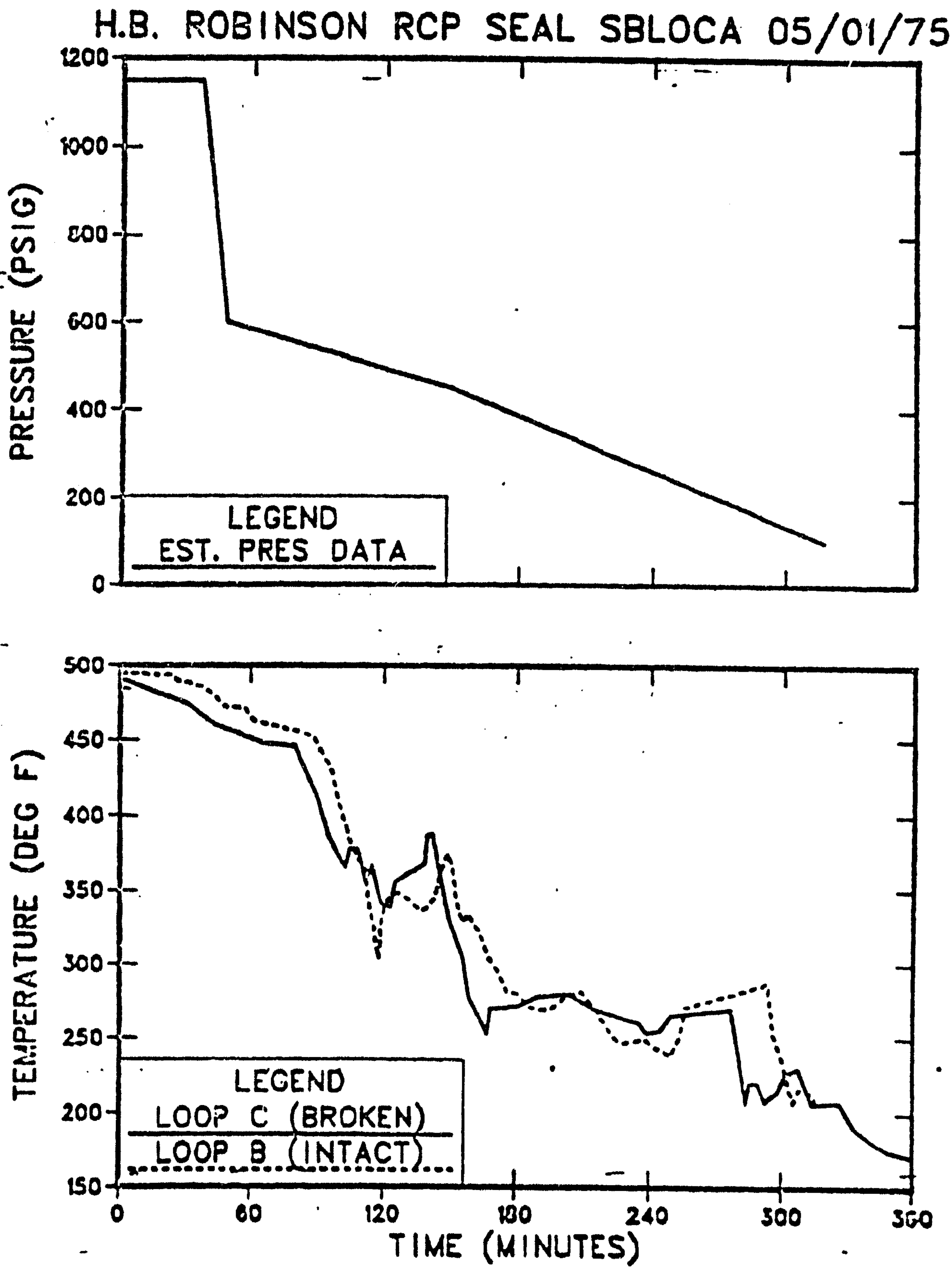

H.B. ROBINSON RCP SEAL SBLOCA 05/01/75
PRESSURE (PSIG)
1200
1000
800
600
400
200
0
LEGEND
EST. PRES DATA
TEMPERATURE (DEG F)
500
450
400
350
300
250
200
150
LEGEND
LOOP C (BROKEN)
LOOP B (INTACT)
0
60
120
180
240
300
360
TIME (MINUTES)

FIGURE 0-3

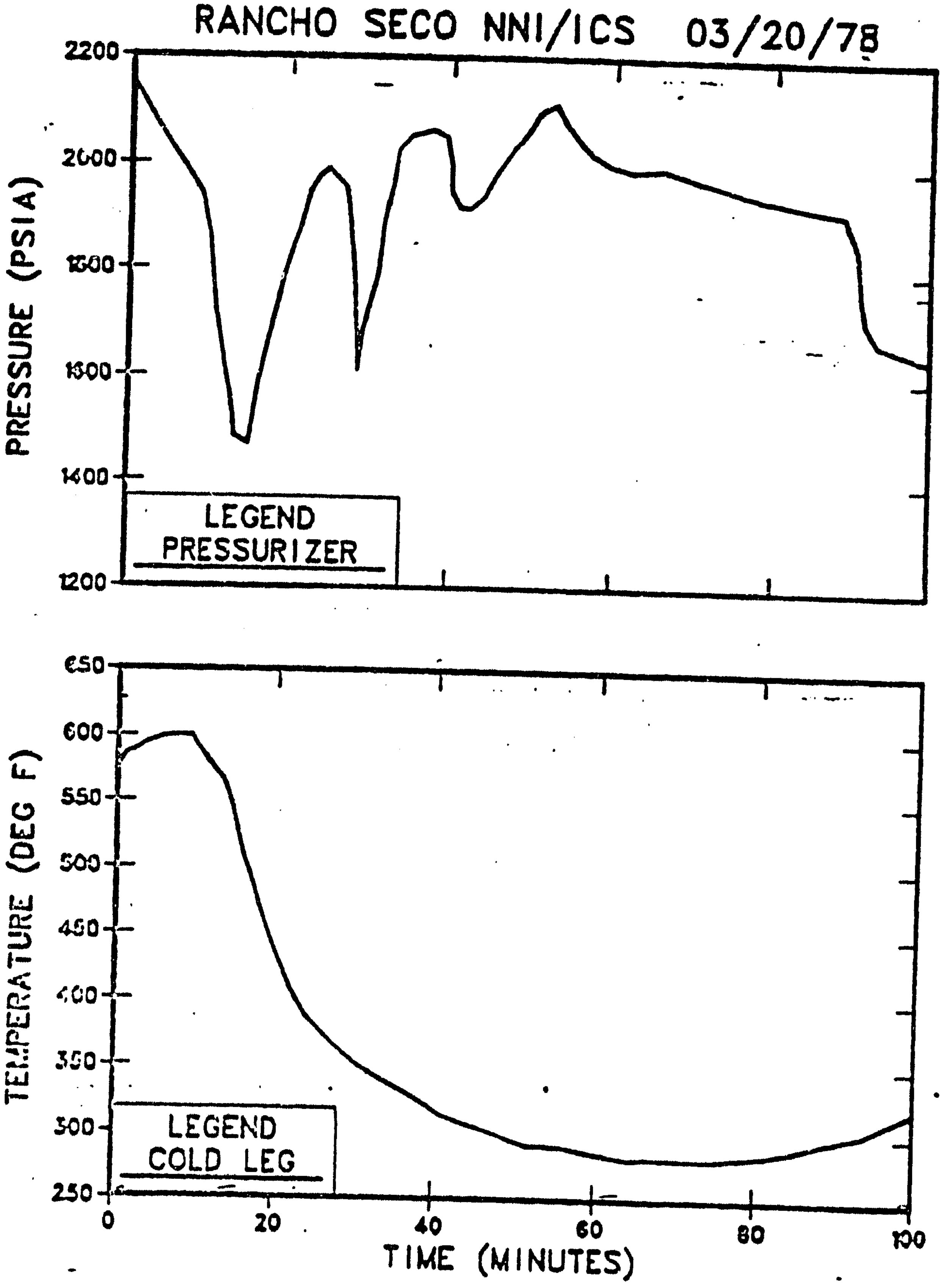

RANCHO SECO NNI/ICS 03/20/78
2200
2000
1800
1600
1400
1200
PRESSURE (PSIA)
LEGEND
PRESSURIZER
650
600
550
500
450
400
350
300
250
TEMPERATURE (DEG F)
LEGEND
COLD LEG
0
20
40
60
80
100
TIME (MINUTES)

FIGURE 0-4

THREE-MILE ISLAND 2 03/28/79

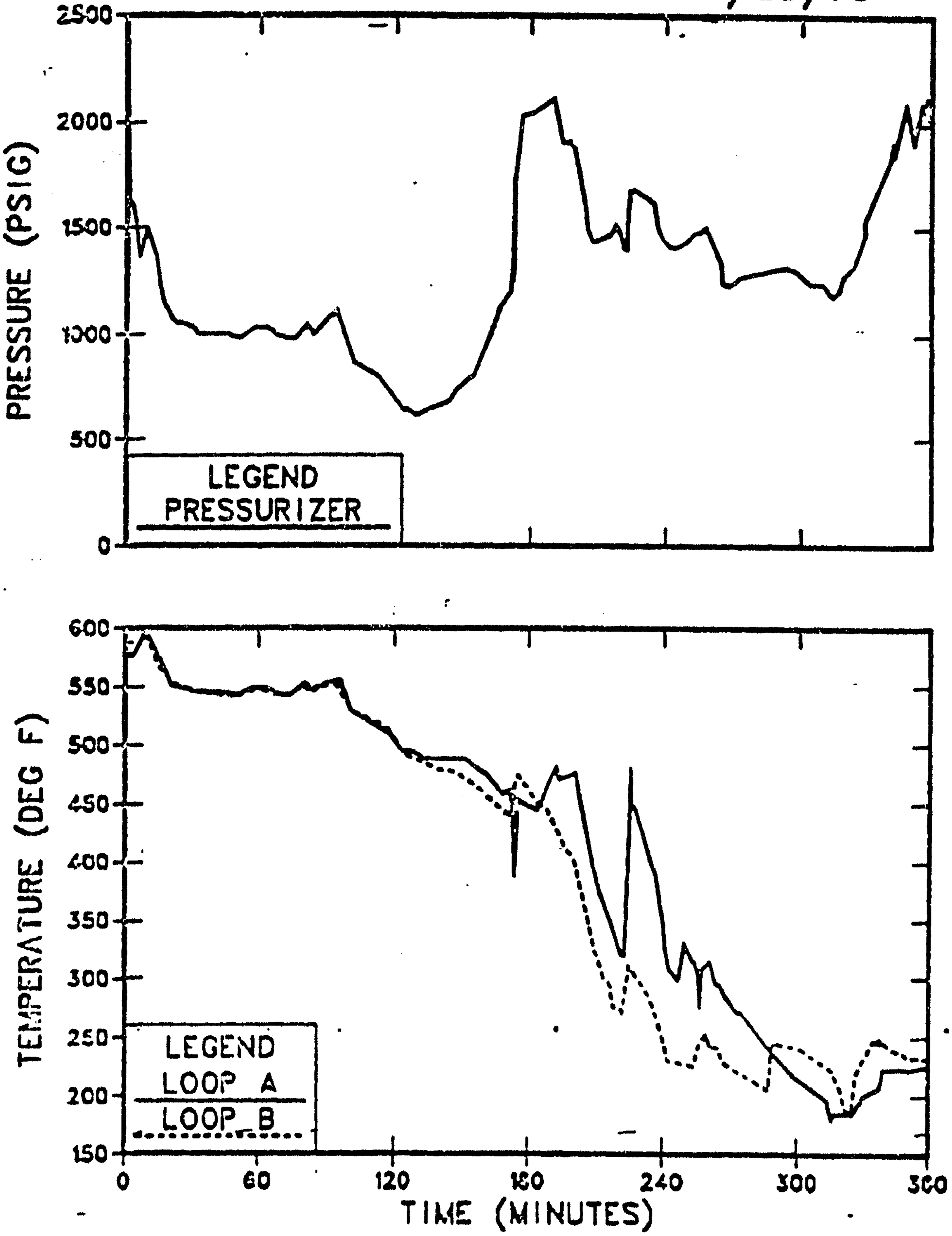


FIGURE 0-5

-121

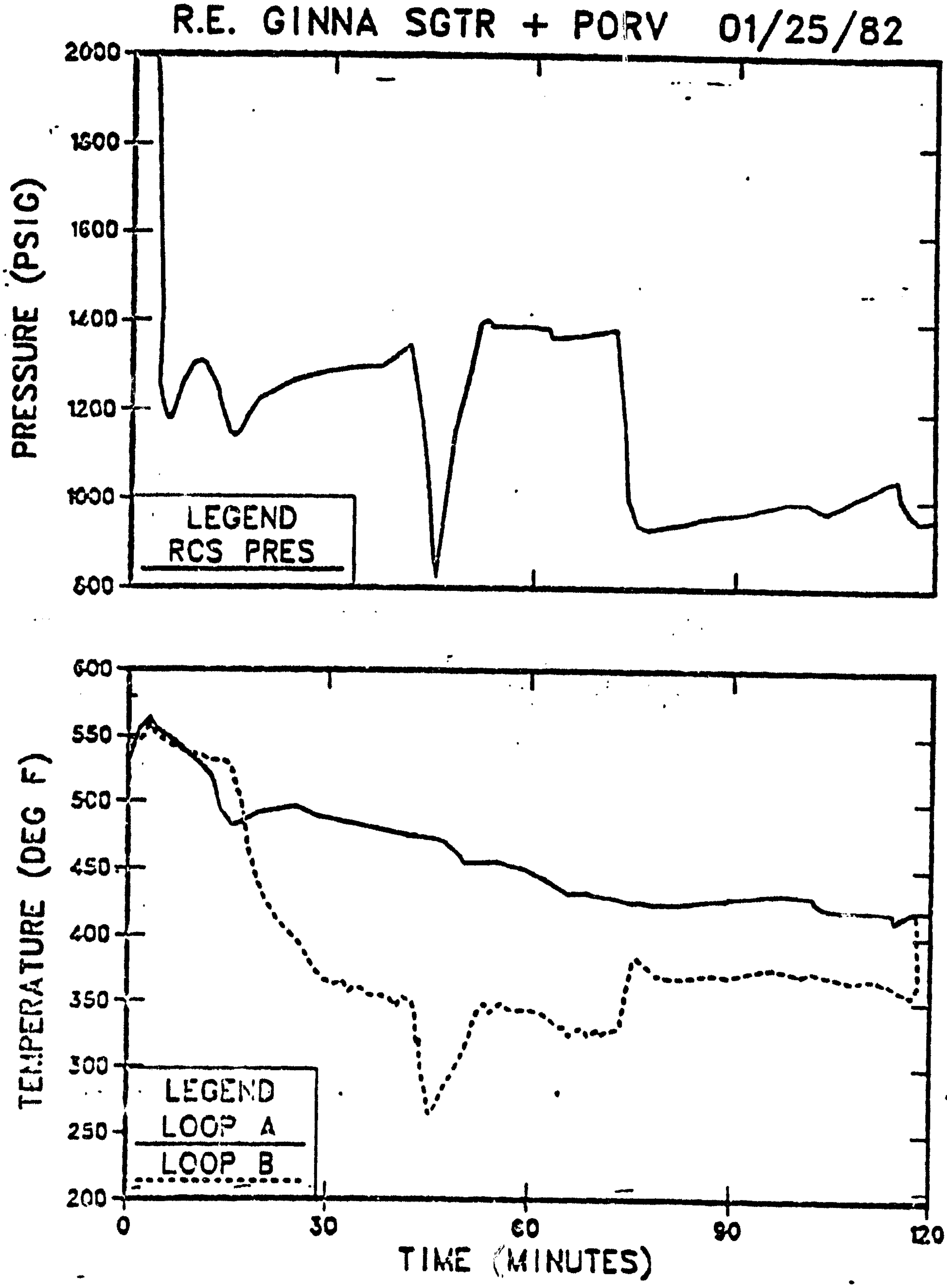


FIGURE 0-6

FIGURE 0-7

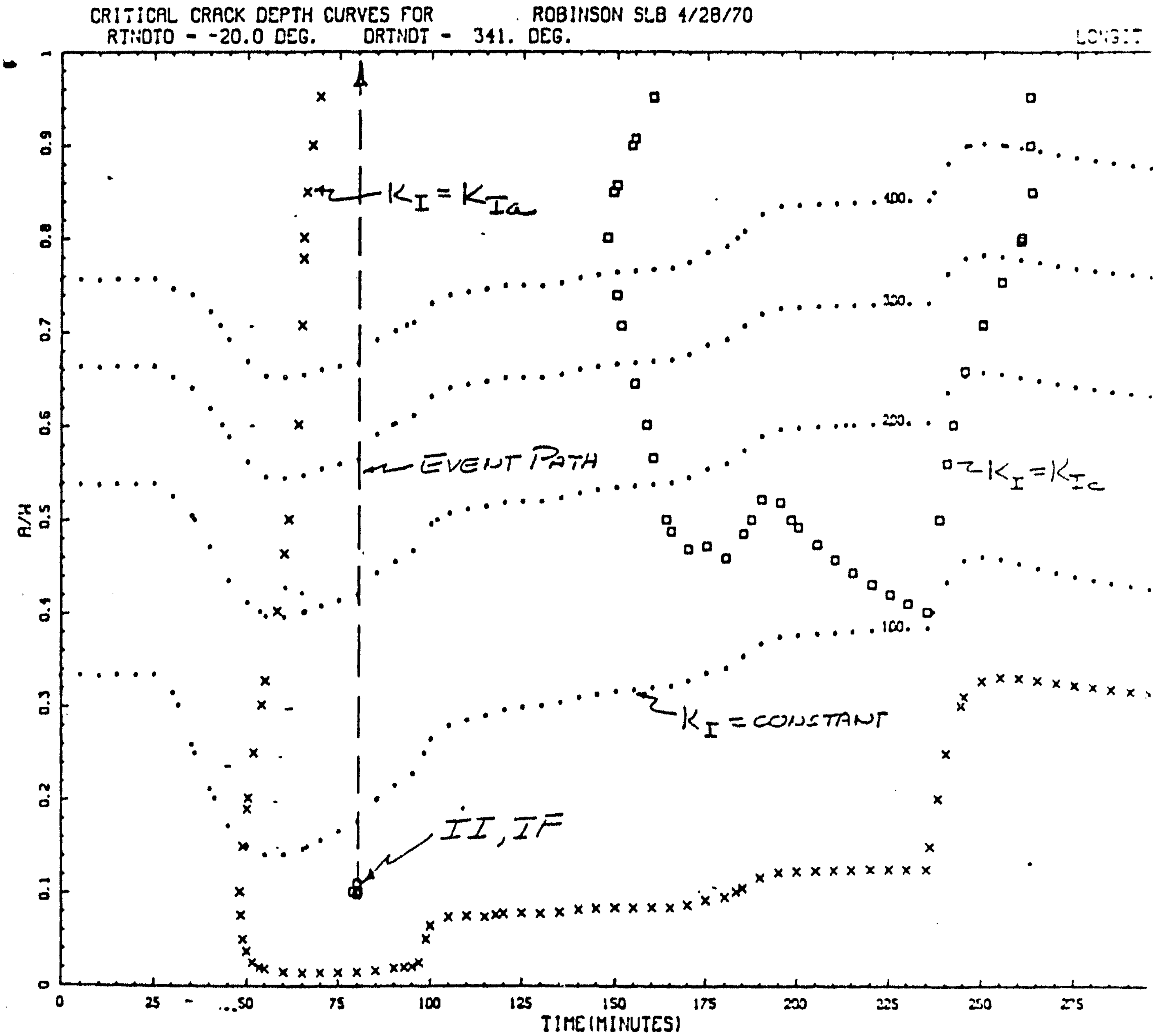

CRITICAL CRACK DEPTH CURVES FOR ROBINSON SLB 4/28/70
RTNDTO - -20.0 DEG. DRTNDT - 341. DEG.
LONGIT
A/H
TIME (MINUTES)
0
25
50
75
100
125
150
175
200
225
250
275
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
100.
200.
300.
400.
$K_I = K_{Ia}$
EVENT PATH
$K_I = K_{Ic}$
K_I = CONSTANT
II, IF

FIGURE 0-8

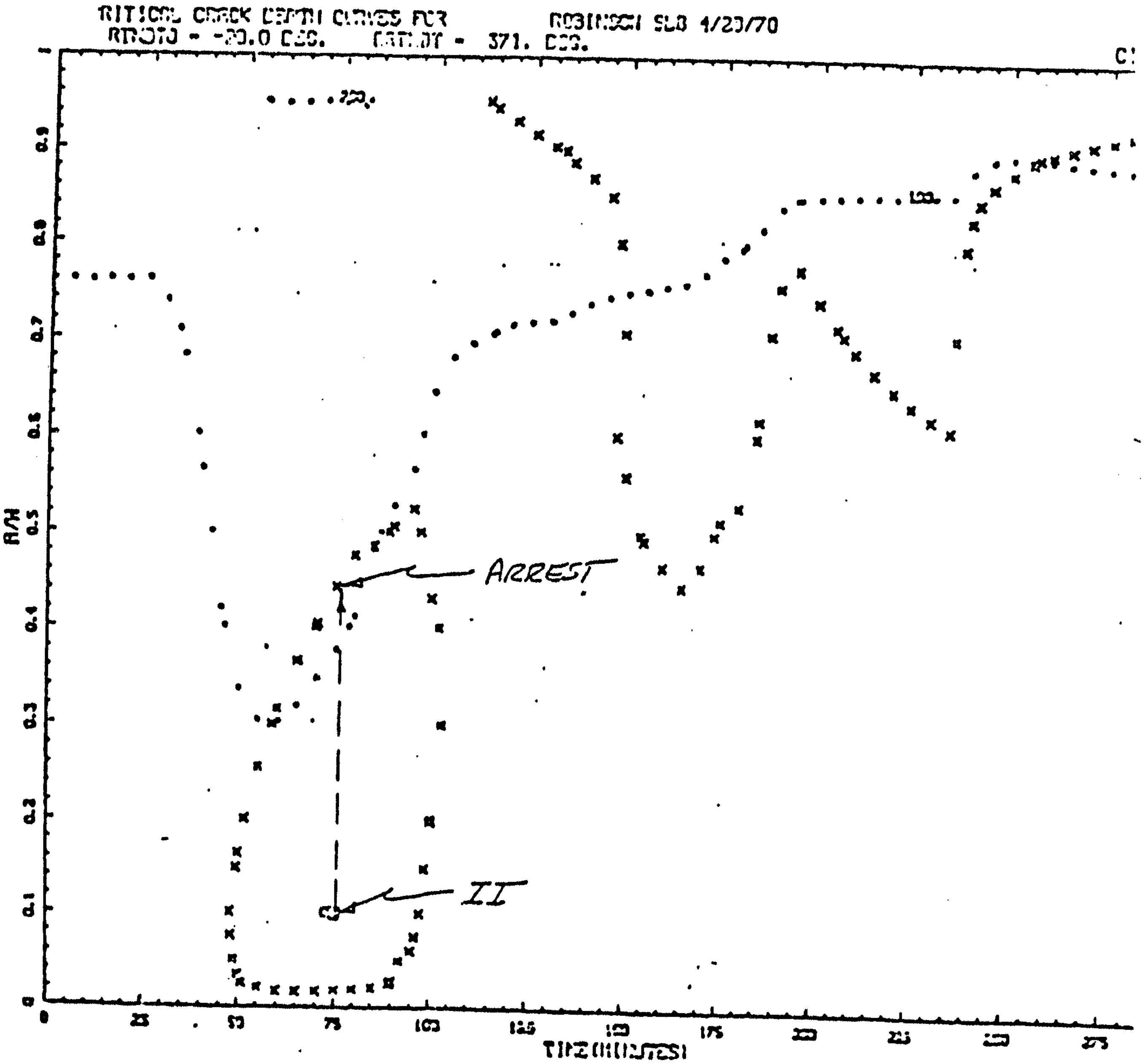
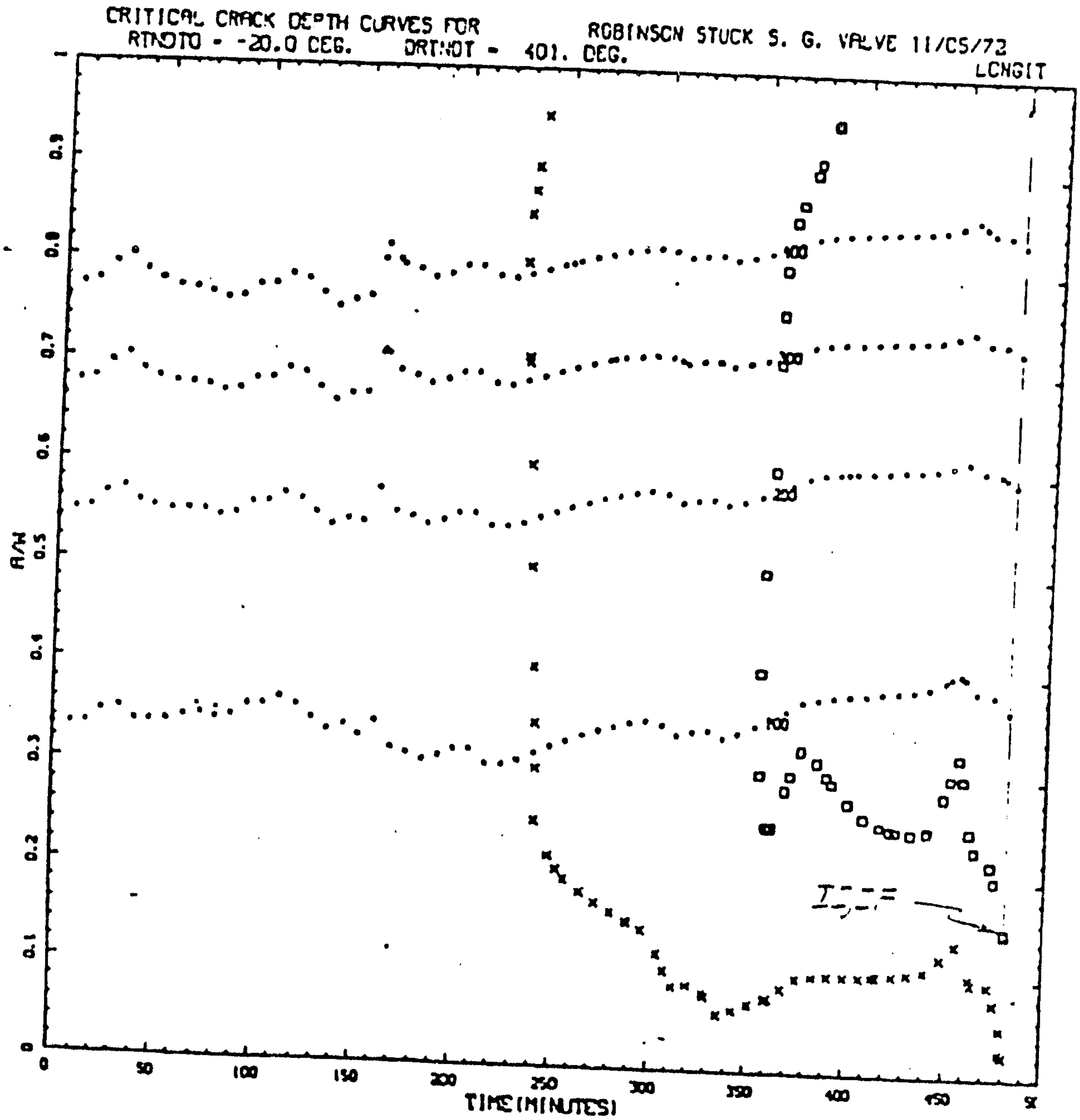


FIGURE 0-9



CRITICAL CRACK DEPTH CURVES FOR
RTNDTO - -20.0 DEG. DRTNDT - 401. DEG.
ROBINSON STUCK S. G. VALVE 11/05/72
LONGIT
A/W
TIME(MINUTES)

FIGURE 0-10

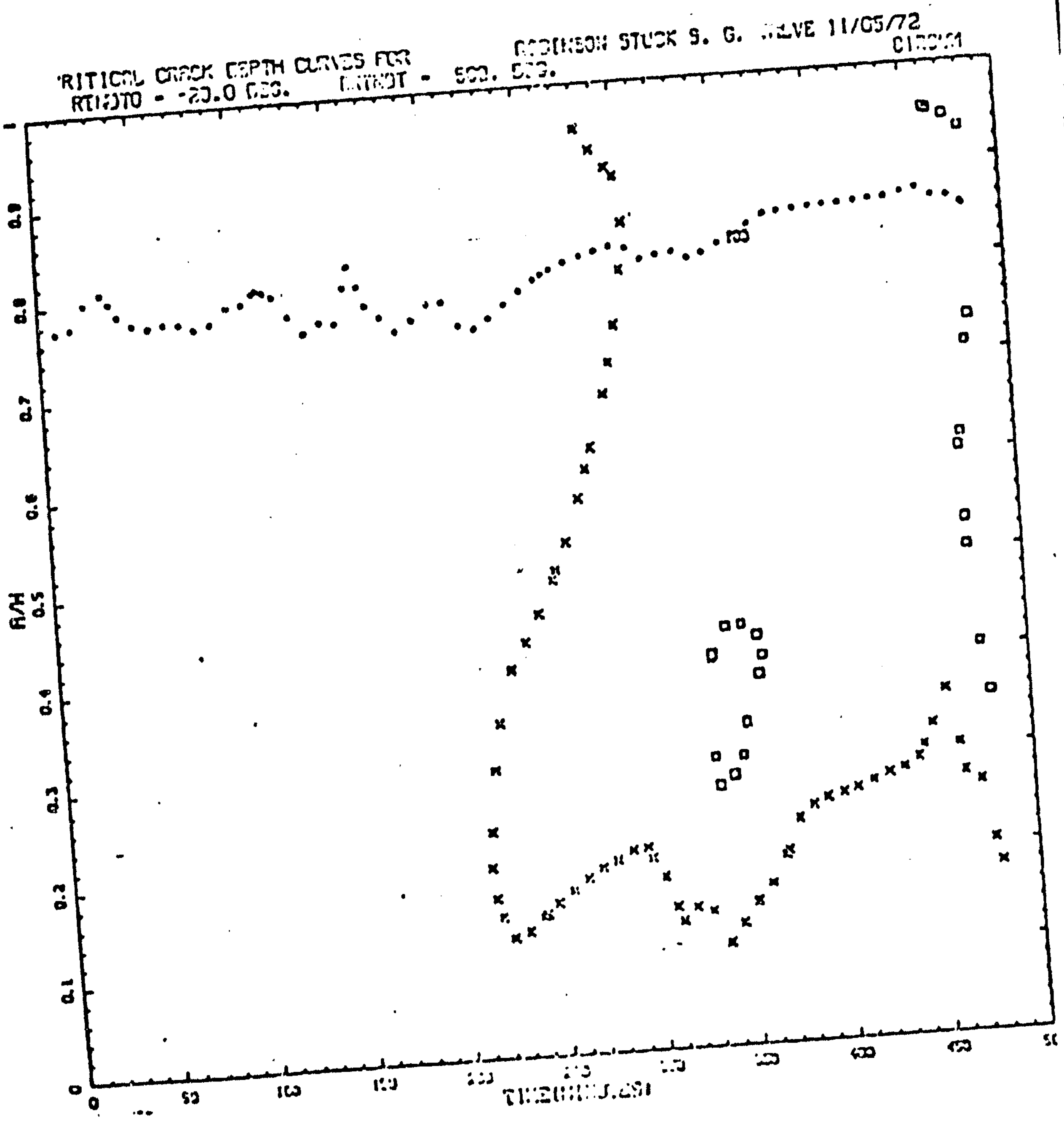

CRITICAL CRACK DEPTH CURVES FOR
RTNDTO = -20.0 DEG.
ROBINSON STUCK S. G. VALVE 11/05/72
A/H
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
0
50
100
200
400
450

FIGURE 0-11

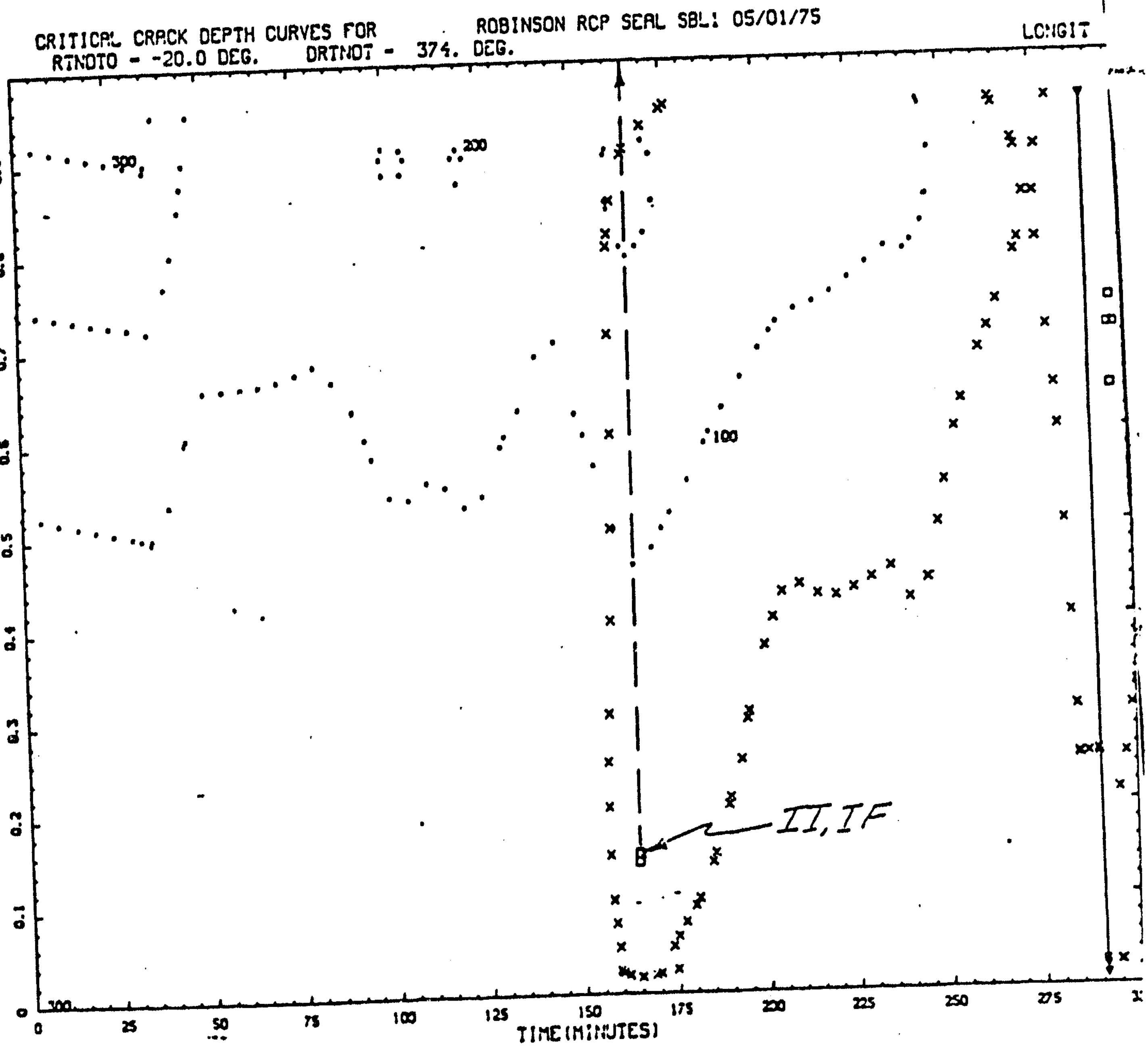

CRITICAL CRACK DEPTH CURVES FOR ROBINSON RCP SEAL SBL1 05/01/75 LONGIT
RTNDTO - -20.0 DEG. DRTNDT - 374. DEG.
A/H
TIME (MINUTES)
0
25
50
75
100
125
150
175
200
225
250
275
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
300
200
100
II, IF

FIGURE 0-12

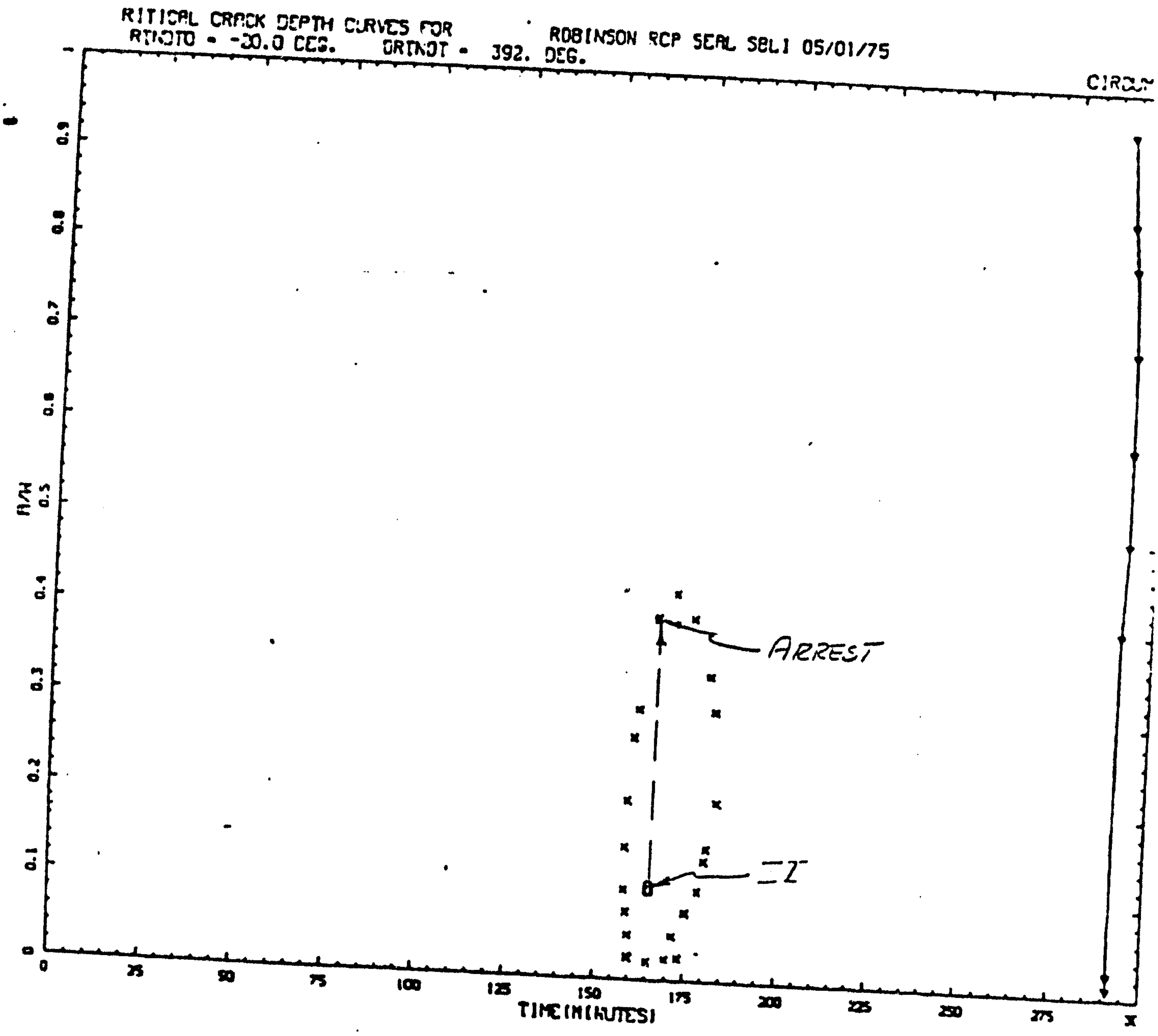

RITICAL CRACK DEPTH CURVES FOR
RTNOTO = -20.0 DEG. ORTNOT = 392. DEG.
ROBINSON RCP SEAL SBLI 05/01/75
CIRCUM
A/W
TIME(MINUTES)
ARREST
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
25
50
75
100
125
150
175
200
225
250
275

FIGURE 0-13

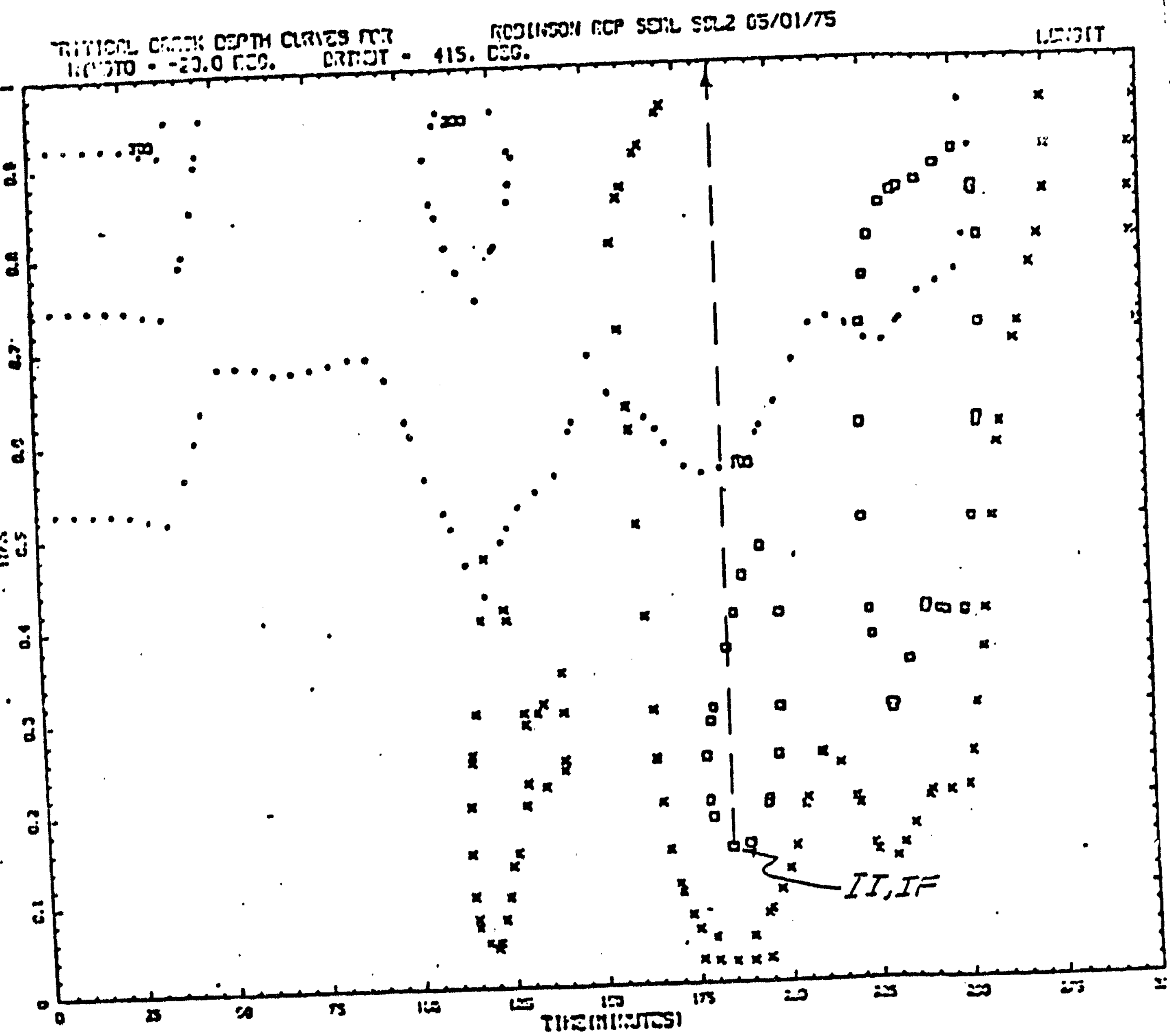

ROBINSON RCP
05/01/75
415. DEG.
-20.0
0
25
50
75
175
0.1
0.2
0.3
0.4
0.5
0.6
0.8
II,IF

FIGURE 0-14

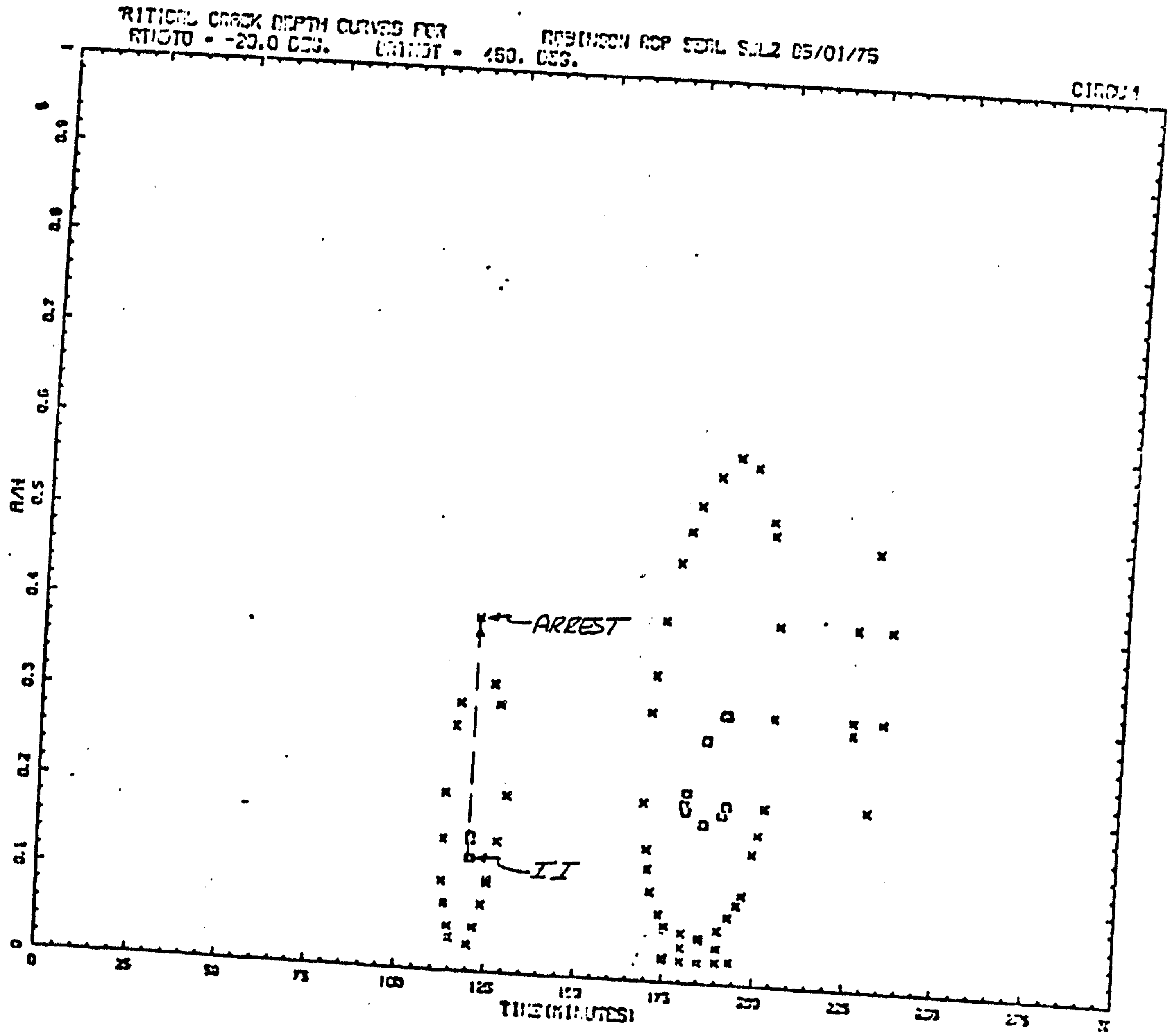
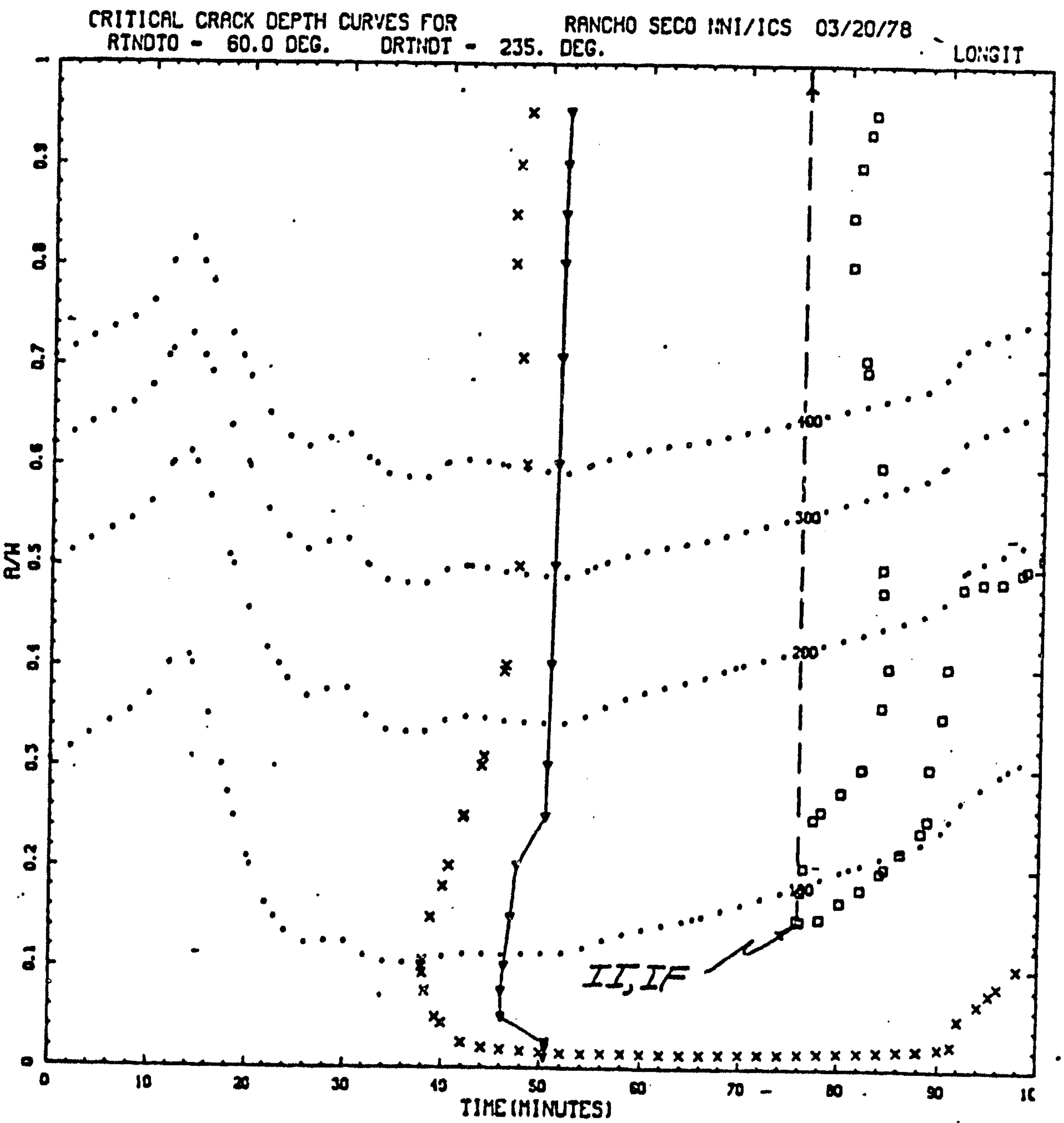
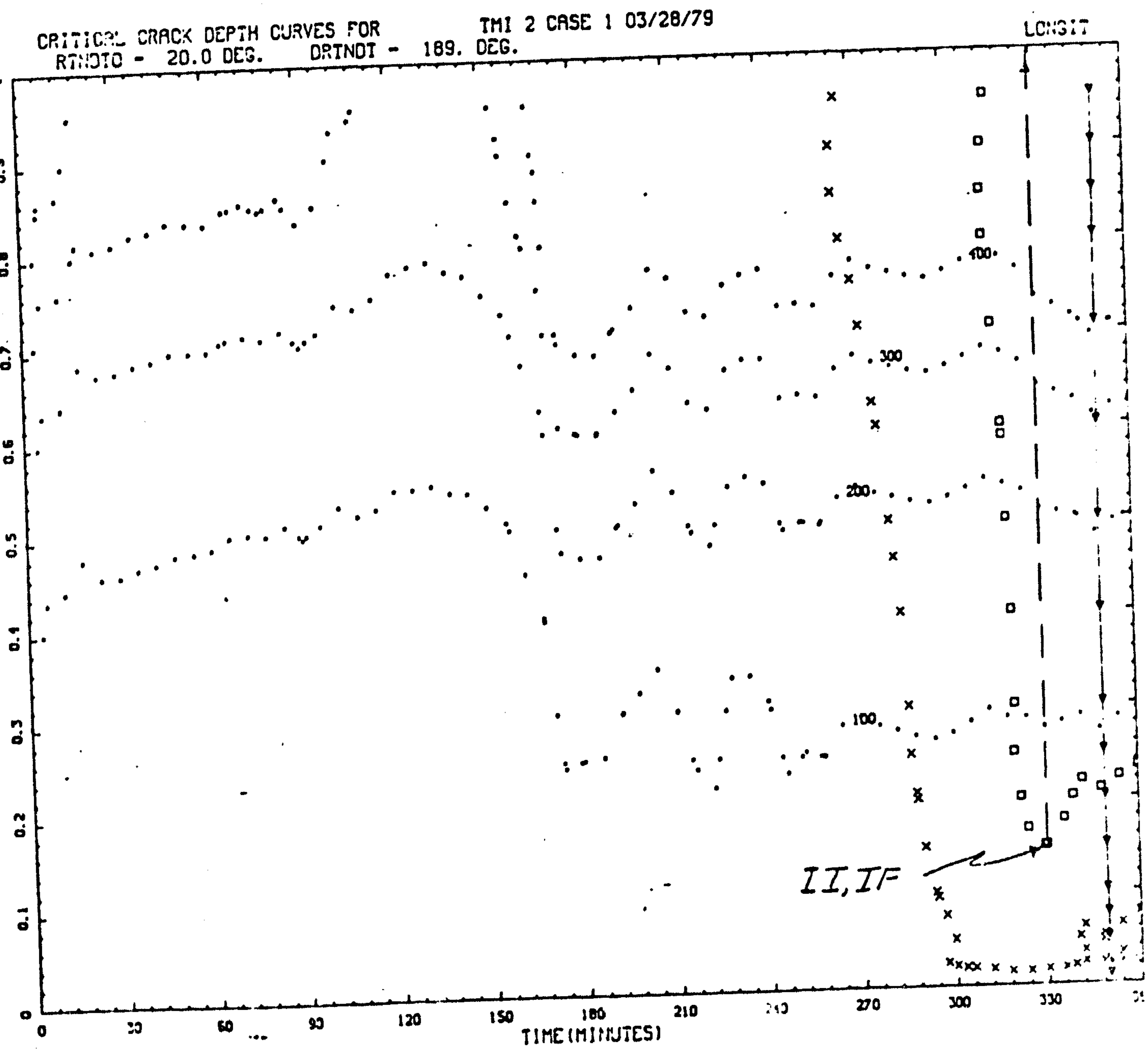


FIGURE O-15



CRITICAL CRACK DEPTH CURVES FOR
RTNDTO - 60.0 DEG. DRTNDT - 235. DEG.
RANCHO SECO NNI/ICS 03/20/78
LONGIT
A/W
TIME(MINUTES)
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
10
20
30
40
50
60
70
80
90
400
300
200
100
II,IF

FIGURE 0-16



CRITICAL CRACK DEPTH CURVES FOR
RTNDTO - 20.0 DEG. DRTNDT - 189. DEG.
TMI 2 CASE 1 03/28/79
LONGIT
A/W
TIME(MINUTES)
400
300
200
100
II, IF

ЧЗЗ

FIGURE 0-17

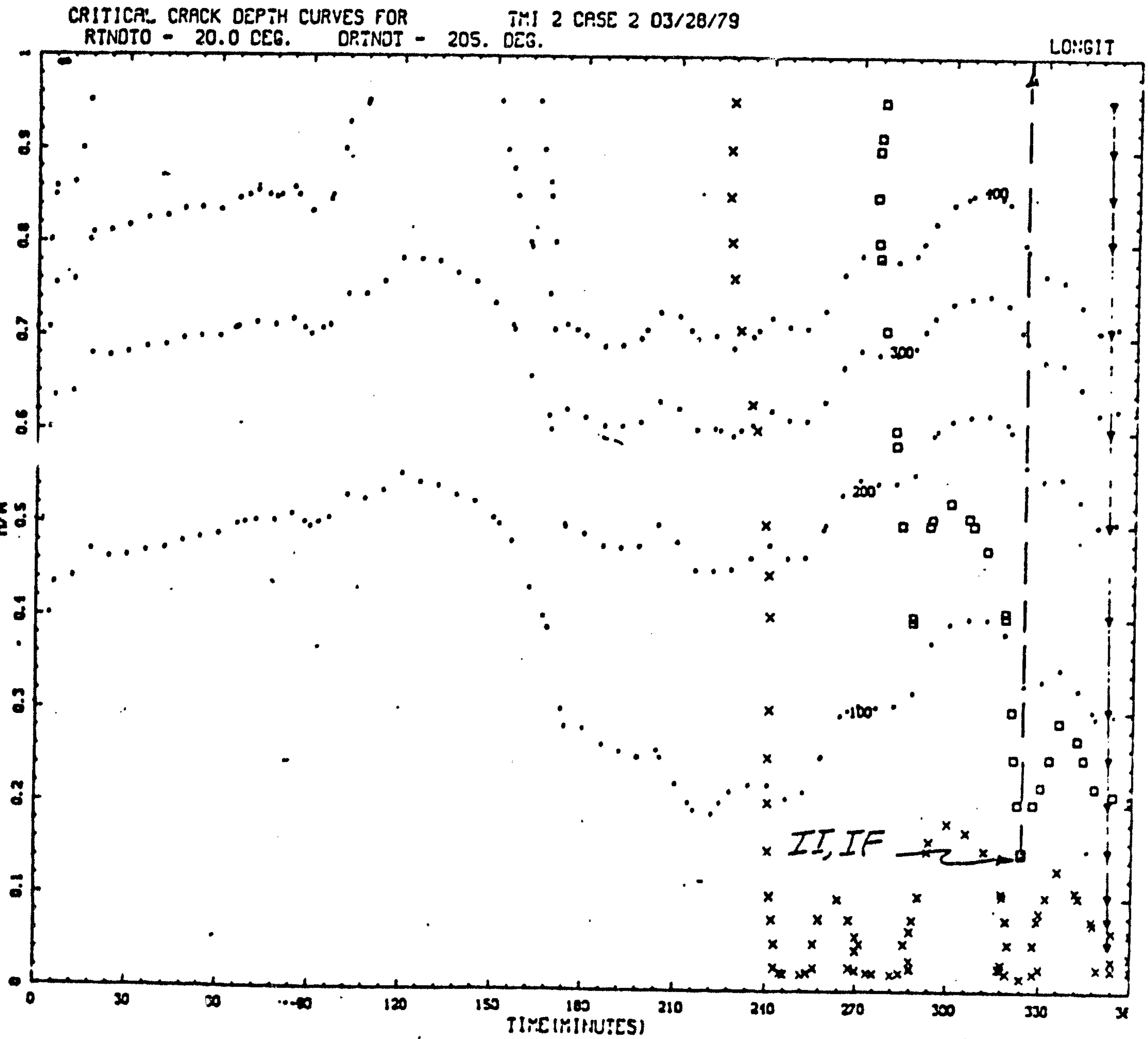
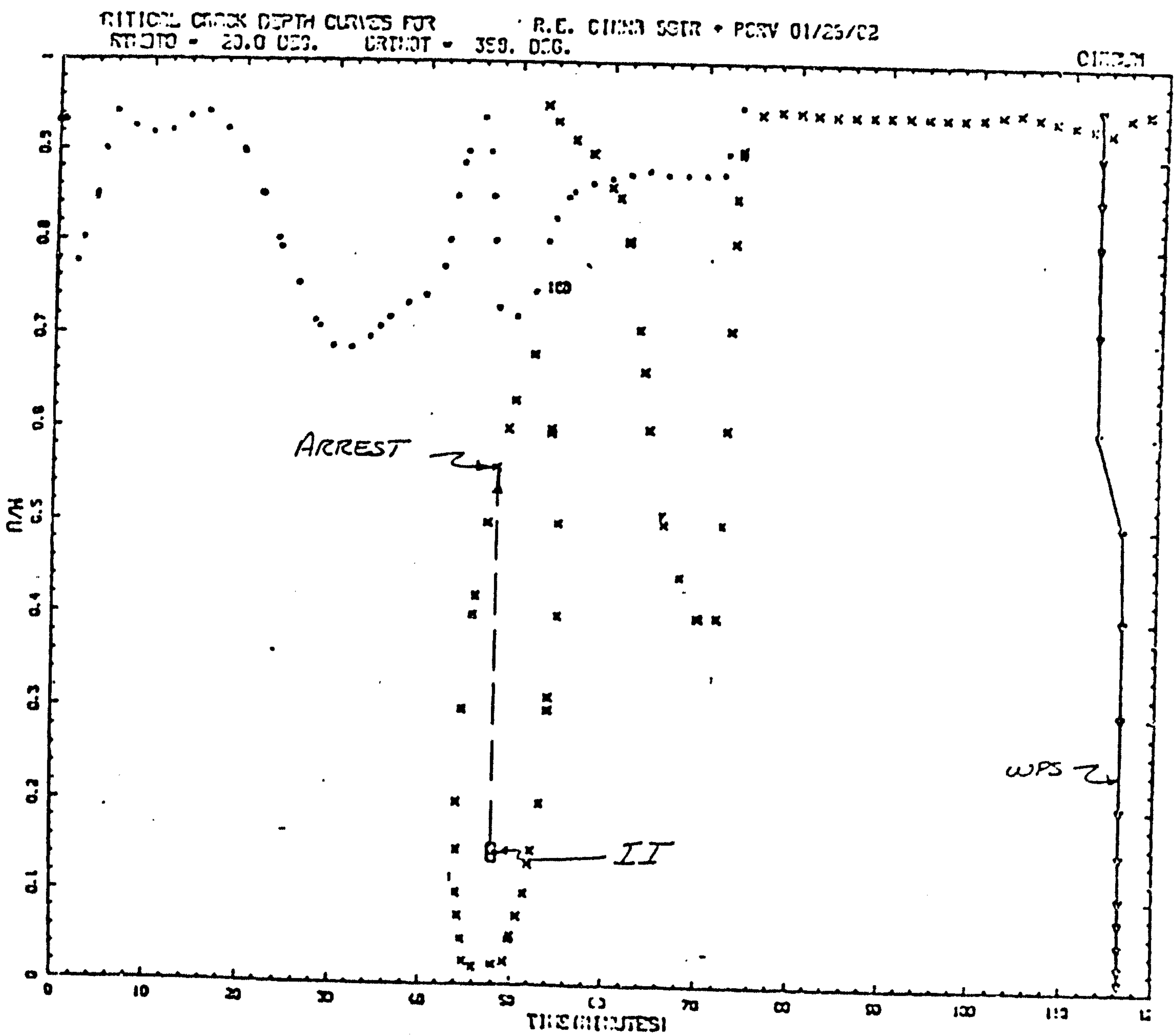


FIGURE 0-18


CRITICAL CRACK DEPTH CURVES FOR
R.E. GINNA SGTR + PCRV 01/25/82
ARREST
II
WPS
A/W
TIME (MINUTES)

REFERENCES

0.1 R. D. Cheverton et al., "Thermal-Shock Investigations," Heavy Section Steel Technology Program Quart. Prog. Rep. for July-September 1981, ORNL/TM-8145, pp. 69-86.

APPENDIX P
CALCULATED RT_{NDT} VALUES FOR PLANTS

P.1 RT_{NDT} Screening Values for All Plants

Table P.1 contains the results of the calculations described in Appendix E for 40 operating PWR plants comprising all of those having significant radiation damage plus all others for which information was readily available. As of June 28, 1982, there were 7 recently-licensed PWRs omitted.

In the column headed "Recommended RT_{NDT} Value for Screening" separate values are given for circumferential and axial welds, because the stress intensity factors produced by certain transients are different for the two cases. For many transients for which pressure is high, the critical value of RT_{NDT} is at least 30 degrees higher for circumferential cracks. Plants are listed in descending order of RT_{NDT}, taking that difference into consideration. For plants where the plate or forgoing governs, its RT_{NDT} value is listed in both columns. Repeating from Appendix E, the recommended RT_{NDT} is the sum of the mean initial RT_{NDT}, the mean ΔRT_{NDT} at the inner vessel wall and the "2-sigma" term.

The sources of information from the various plants are as follows. The EFPY ae calculated from data submitted monthly to the NRC for total megawatt hours thermal. This value is divided by the rated thermal power to get effective full power hours. For fluence, copper, and nickel content, the 8 plants that had been submitted reports containing the results of a recent review of all available data. These 8 plants can be identified in the Table by their values for "Licensee's RT_{NDT}." Most of the other plants had submitted detailed information on their vessel beltline materials and fluence at the critical locations in response to an inquiry from the NRC in May 1977. Finally, there were surveillance reports for a number of plants, which contained updated calculations of fluence at the vessel wall.

P.2 Comparison of NRC and Licensee's Values

For the 8 plants, Table P.1 shoes licensee's values of RT_{NDT}. For the three CE plants, Table P.1 also shows the values calculated by CE in Appendices to CEN-189 (Ref. E.2.). These CE values range from 28 to 39 degrees F above the licensee's values, largely because of differences in the estimates of initial RT_{NDT}. For the CE plants, the NRC value of RT_{NDT} fall, between the licensee's value and the CE value for Fort Calhoun, falls 18 degrees F above the CE value for Maine Yankee, and Falls far below the CE value for Calvert Cliffs 1, because we recently accepted a lower estimate of copper content. For the three Westinghouse plants, the NRC value of RT_{NDT} is 9 degrees lower for Robinson 2, and 12 degrees lower for San Onofre. For Turkey Point 4, the NRC value is 48 degrees higher, because the licensee used a surveillance value that happens to fall well below the Guthrie mean trend curve. For the B&W plants, the NRC value is 48 degrees higher, because the licensee mean trend curve. For the B&W plants, the NRC value of RT_{NDT} gives a higher value than the trend curve from Regulatory Guide 1.99, which B&W used. For Three Mile Island 1, the NRC value of RT_{NDT} was 59 degrees higher than the licensee's value because: (a) they used an initial value of RT_{NDT} of -14°F whereas the NRC used a mean value of 0°F and 2 sigma of 34 degrees F, and (b) they used Regulatory Guide 1.99 as described above for Oconee 1. Actually, B&W did not give the values quoted in the Table. Those values were calculated by the NRC, using copper and fluence values from proprietary references given by B&W. These differences will have to be resolve for those plants that fail the screening criterion.

Table P.1 RT_{NDT} Values for All Plants(1) Calculated Per the Recommendations of the Working Group on RT_{NDT} (2) for the Vessel Inside Surface.

Plant NSSS/Vessel Fabricators	EFPY as of 12/31/81	Fluence n/cm² x10¹⁸	Copper %	Nickel %	Mean Initial RT_{NDT}, °F	Mean ΔRT_{NDT} °F	$2\sqrt{\sigma_0^2+\sigma_\Delta^2}$ (5)	RT_{NDT}, °F, as of Dec 31, 1981(6) Circum.	Axial	Licensee's RT_{NDT}, °F
Robinson 2	7.10	(14.1)(3)(8)	(0.35)	(1.20)	(-56)	(303)(4)	34 (4)	281		290
W/CE		14.8 (3)(8)	0.27	0.20	-56	151	59		154	220
Turkey Point 4	5.67	9.1 (9)	(0.32)	(0.57)	(0)	(200)	59	259		211
W/B&W	No Axial Welds									
Turkey Point 3	5.67	(9.1)(9)	(0.32)	(0.57)	(0)	(200)	59	259		
W/B&W	No Axial Welds									
Fort Calhoun	5.07	(7.04)	(0.35)	0.99	(-56)	(2.64)(4)	34(4)	242		(7)
CE/CE		5.1	0.35	0.99	-56	248 (4)	34 (4)		226	209 (239)
Indian Point 3	2.98	(1.67)	(0.24)	(0.52)	(+74)	(90)	48	212		
W/CE	Plate Governs		0.24	0.52	+74	90	48		212	
Yankee Rowe	14.56	(11.35)	(0.24)	(0.52)	(+74)	(90)	48	212		
W/B&W/B&W	Plate Governs		0.20	0.63	+30	133	48			
Rancho Seco	3.54	(2.33)	(0.31)	(0.59)	(0)	(135)	59	194		
B&W/B&W		2.05	0.35	0.59	0	148	59		207	
Three Mile Island 1	3.52	(1.87)	(0.31)	(0.59)	(0)	(133)	59	192		(129)
B&W/B&W		(1.87)	0.35	0.60	0	145	59		204	145
Oconee 2	4.71	(2.87)	(0.35)	(0.71)	(0)	(172)	59	231		

See footnote(s), last page of table

These values are subject to change when plant-specific analyses yield better information.

Table P-1 (Continued)

Plant NSSS/Vessel Fabricators	EFPY as of 12/31/81	Fluence n/cm^2 x10^{18}	Copper %	Nickel %	Mean Initial RT_{NDT}, °F	Mean ΔRT_{NDT} °F	$2\sqrt{\sigma_0^2+\sigma_\Delta^2}$ (5)	RT_{NDT}, °F, as of Dec 31, 1981(6) Circum.	Axial	Licensee's RT_{NDT}, °F
Point Beach 1	8.07	(10.01)	(0.24)	(0.57)	(0)	(151)	59	210		
W/B&W		7.34	0.24	0.57	0	139	59		[illegible]	
Oconee 1	5.04	(2.32)	(0.26)	(0.61)	(0)	113	59	172		
B&W/B&W		2.73	0.31	0.55	0	138	59		197	160
Indian Point 2	4.40	No Circum Data								
W/CE		2.2	0.34	1.2	-56	211 (4)	34		189	
Arkansas ANO-1	4.42	(2.70)	(0.31)	(0.59)	(0)	129	59	199		
B&W/B&W		1.99	0.31	0.59	0	129	59		188	
Point Beach 2	7.54	(9.35)	(0.25)	(0.59)	(0)	(156)	59	215		
W/B&W, CE		No Axial Welds								
Ginna	8.18	(9.49)	(0.25)	(0.56)	(0)	(154)	59	213		
W/B&W		No Axial Welds								
San Onofre	9.04	(33.45)	(0.27)	(0.20)	(-56)	(188)	59	191		203
W/CE		27.12	0.27	0.20	-56	178	59		181	
Zion 2	4.49	(2.83)	(0.26)	(0.61)	(0)	(119)	59	178		
B&W/B&W		0.90	0.35	0.59	0	118	59		177	
Palisades	4.12	(4.78)	(0.25)	(1.2)	(-56)	(174)	59	177		
CE/CE		4.78	0.25	1.2	-56	174	59		177	
Crystal River 3	2.48	(1.44)	(0.35)	(0.59)	(0)	(134)	59	193		
B&W/B&W		1.36	0.31	0.61	0	118	59		177	

Table P-1 (Continued)

Plant NSSS/Vessel Fabricators	EFPY as of 12/31/81	Fluence n/cm² $\times 10^{18}$	Copper %	Nickel %	Mean Initial RT_{NDT}, °F	Mean ΔRT_{NDT} °F	$2\sqrt{\sigma_0^2+\sigma_\Delta^2}$ (5)	RT_{NDT}, °F, as of Dec 31, 1981(6) Circum.	Axial	Licensee's RT_{NDT}, °F
Surry 1	4.88	(7.61)	(0.25)	(0.51)	(0)	(141)	59	200		
W/B&W		1.66	0.21	0.59	0	81	59		140	
Cook 1	4.56	(2.87)	(0.40)	(0.82)	(-56)	(222) (4)	34	200		
W/CE		1.55	0.13	0.99	-56	58	59		61	
North Anna 1	2.41	(4.42)	(0.14)	(0.80)	(+38)	(76)	48	162		
W/RD		No Axial Welds		Forging	Governs	48			162	
Beaver Valley	1.87	(3.16)	(0.37)	(0.62)	(-56)	(179)	59	182		
W/CE		0.47	0.36	0.62	-56	104	59		107	
North Anna 2	0.77	(1.38)	(0.83)	(+56)	(52)	48		152		
W/RD		No Axial Welds		Forging	Governs				152	
Salem 1	2.26	(1.49)	(0.24)	(0.51)	(+51)	(87)	48	150		
W/CE			0.24	0.51	Plate Governs	87	48		150	
Oconee 3	4.78	(2.92)	(0.24)	(0.63)	(0)	(112)	59	(171)		
B&W/B&W		No Axial Welds								
Surry 2	4.83	(7.54)	(0.19)	(0.56)	(0)	(108)	59	167		
W/B&W, RD		1.64	0.21	0.59	0	81	59		140	
Calvert Cliffs 1	4.65	(6.84)	(0.30)	(0.18)	(-56)	(135)	59	138		(7)
CE/CE		6.84	0.21	0.85	-56	136	59		139	
St. Lucie	3.52	(2.22)	(0.31)	(0.11)	(-56)	(98)	59	101		
CE/CE		2.22	0.30	0.64	-56	132	59		135	

Table P-1 (Continued)

Plant NSSS/Vessel Fabricators	EFPY as of 12/31/81	Fluence n/cm^2 x10^{18}	Copper %	Nickel %	Mean Initial RT_{NDT}, °F	Mean ΔRT_{NDT} °F	$2\sqrt{\sigma_0^2+\sigma_\Delta^2}$ (5)	RT_{NDT}, °F, as of Dec 31, 1981(6) Circum.	Axial	Licensee's RT_{NDT}, °F
Calvert Cliffs 2	3.63	(5.34)	(0.30)	(0.18)	(-56)	(127)	59	130		
CE/CE			0.30	0.18	-56	127	59		130	
Trojan	3.00	(2.07)	(0.16)	(0.62)	(+10)	(65)	48	123		
W CBI					Plate		48		123	
Davis Besse 1	1.68	(1.11)	(0.24)	(0.61)	(0)	(85)	59	144		
B&W/B&W		No Axial Welds								
Haddam Neck	10.92	(14.30)	(0.22)	(0.10)	(-56)	(111)	59	114		
W/CE		11.90	0.22	0.10	-56	106	59			
Kewaunee	5.87	(7.86)	(0.20)	(0.77)	(-56)	(129)	59	132		
W/CE		No Axial Welds								
Farley 1	2.19	(3.70)	(0.24)	(0.60)	(-56)	(117)	59	120		
W/CE		0.83	0.27	0.60	-56	89	59		92	
Millstone 2	3.91	(2.19)	(0.37)	(0.60)	(-56)	(117)	59	117		
		No Data for Axial Welds								
Prairie Island 2	5.62	(7.53)	(0.19)	(0.13)	(-56)	(81)	59	84		
W/SFAC		No Axial Welds								
Prairie Island 1	5.90	(7.90)	(0.14)	(0.17)	(-56)	(60)	59	63		
W/SFAC		No Axial Welds								

Footnotes

(1) Arranged in descending order of the RT_{NDT} as of December 31, 1981 considering circumferential to be 30°F less severe than axial orientations.

(2) Memorandum, M. Vagins to S. Hanauer, August 30, 1982.

(3) Values shown in parentheses on top line are for circumferential welds, bottom line is for axial welds. When plate governs--both lines.

(4) Determine by Reg. Gude 1.99, Rev. 1, Upper Limit Line, $\sigma_{\Delta} = 0$.

(5) σ_0 (17°F) and σ_{δ} (24°F) are the standard deviations of the initial RT_{NDT} and ΔRT_{NDT}, respectively. If palte or forging governed, acutal initial RT_{NDT} w available and $\sigma_0 = 0$

(6) The sum of the Mean Initial RT_{NDT}, the mean WRT_{NDT} and $2\sqrt{\sigma_0^2 + \sigma_{\Delta}^2}$, as of Dec. 31, 1981.

(7) Initial RT_{NDT} assumed by licensee to be -50°F and by CE to be -20°F. Values in parentheses are by CE.

(8) Fluence is per letter from CP&L Co., Sept. 24, 1982, pending agreement on final value.

(9) Fluence reduced from 11.16 n/cm² per letter from FPL Aug. 31, 1982, on TP 4. TP 3 tentatively assumed to be the same as TP 4.

(10) Fluence reduced to 0.73 x peak per letter from Omaha PPD, Sept. 1, 1982.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

November 5, 1982

IE INFORMATION NOTICE NO. 82-42: DEFECTS OBSERVED IN PANASONIC MODEL 801 AND MODEL 802 THERMOLUMINESCENT DOSIMETERS

Addressees:

All NRC licensees

Description of Circumstances:

Many licensees use thermoluminescent dosimeters (TLDs) for personnel radiation monitoring. One of the major suppliers of these TLD systems is the Panasonic Industrial Company. It has come to the attention of the NRC that Panasonic Model 801 and Model 802 dosimeters have exhibited a defect that could reduce the response of affected elements. Panasonic has sent a letter to its customers describing the problem and providing actions to be taken in case the defects are found in the customers' dosimeters. A copy of this letter is attached. The purpose of this Information Notice is to inform NRC licensees of the defect and its significance and how to respond if their dosimeters are affected.

Discussion:

In September 1982, during an interlaboratory calibration check, the NRC noted a decrease in the response of a number of the dosimeters which it uses in its TLD Direct Radiation Monitoring Network. Inspection of the dosimeters indicated that the affected elements had bubbles on the carbon-coated polyamide backing material and the aluminum substrate. The bubbles affect the heat transfer and thus the calibration. When these defects were discussed with the manufacturer, NRC staff learned that four other customers had also observed the same defect in their dosimeters. The precise cause of the defect is not yet known with certainty, but the following characteristics have been observed thus far:

1. Affected dosimeters apparently lose sensitivity to a variable degree, thus underestimating radiation exposure. The NRC has observed response reductions as high as a factor of two and other users have reported even greater reductions.

2. The effect appears to occur randomly. Though two dosimeters may be handled identically and have the same production date, one may produce a bubble in one or more elements and the other may not have any.

3. The bubbles usually appear on the carbon-coated polyamide backing material but have also been seen on the aluminum substrate. Changes in element calibration factors may also indicate the presence of a defect; however, bubbles have been observed that have not significantly affected element calibration factors. The bubbles are large and easily seen.

4. The effect has been seen on both lithium borate and calcium sulfate elements, but more frequently on the latter.

Although Panasonic believes that this defect is the result of manufacturing problems at the factory, the possibility that the defect may be reader-related has not been ruled out, because of possible TLD element overheating.

Guidance:

NRC licensees are required to perform personnel radiation monitoring in accordance with 10 CFR 20.202. Licensees who use Panasonic Model 801 or Model 802 TLD systems should verify that their dosimeters continue to respond accurately and should be prepared to take compensatory actions if the defects described herein appear on their dosimeters. Other Panasonic models that employ calcium sulfate elements may also be affected and should be checked as well. The only way to ensure that one's dosimeters are not affected is to visually examine each dosimeter. Therefore, recalibrations and visual examination of dosimeters should be sufficient to identify damaged dosimeters. The dosimeters should be recalibrated whenever bubbles are observed. Where damaged dosimeters are detected, personnel exposure should be re-evaluated using data from undamaged elements or by using other available dosimetry such as self-reading dosimeters.

No written response to this information notice is required. If you need additional information regarding this matter, contact the Administrator of the appropriate NRC Regional Office.

L. I. Cobb, Director
Division of Fuel Facilities, Materials
and Safeguards
Office of Inspection and Enforcement

Technical Contact: J. Metzger
301-492-9747

Attachments:
1. Ltr. fm Panasonic Industrial Co. to its customers signed by Panasonic Natl. Sales Manager.
2. List of Recently Issued Information Notices

4. The effect has been seen on both lithium borate and calcium sulfate elements, but most frequently on the latter.

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Guidance

NRC licensees are required to perform personnel radiation monitoring in accordance with 10 CFR 20.202. Licensees who use Panasonic Model 801 or Model 802 TLD systems should verify that their dosimeters continue to respond accurately and should be prepared to take compensatory actions if the defects described herein appear on their dosimeters. Other Panasonic models which employ calcium sulfate elements may also be affected and should be checked as well. The only way to ensure that one's dosimeters are not affected is to visually examine each dosimeter. Therefore, recalibrations and visual examination of dosimeters should be sufficient to identify damaged dosimeters. The dosimeters should be recalibrated whenever bubbles are observed. Where damaged dosimeters are detected, personnel exposure should be re-evaluated using data from undamaged elements or by using other available dosimetry such as self-reading dosimeters.

No written response to this Information Notice is required. If you need additional information regarding this matter, contact the Administrator of the appropriate NRC Regional Office.

L. I. Cobb, Director
Division of Fuel Facilities, Materials
and Safeguards, IE

Technical Contact: J. Metzger
301-492-9747

Attachments:
1. Ltr. fm Panasonic Industrial Co. to its customers signed by Panasonic Natl. Sales Manager.
2. List of recently issued Information Notices

bcc: G. Wayne Kerr, SP

IE:FFMB	IE:FFMB:C	IE:FFMSD
J.Metzger/DM	L.Higginbotham	L.I.Cobb
10/19/82	10/21/82	10/3/82

4. The effect has been seen on both lithium borate and calcium sulfate elements, but more frequently on the latter.

Although Panasonic believes that this defect is the result of manufacturing problems at the factory, the possibility that the defect may be reader-related has not been ruled out, because of possible TLD element overheating.

Guidance:

NRC licensees are required to perform personnel radiation monitoring in accordance with 10 CFR 20.202. Licensees who use Panasonic Model 801 or Model 802 TLD systems should verify that their dosimeters continue to respond accurately and should be prepared to take compensatory actions if the defects described herein appear on their dosimeters. Other Panasonic models that employ calcium sulfate elements may also be affected and should be checked as well. The only way to ensure that one's dosimeters are not affected is to visually examine each dosimeter. Therefore, recalibrations and visual examination of dosimeters should be sufficient to identify damaged dosimeters. The dosimeters should be recalibrated whenever bubbles are observed. Where damaged dosimeters are detected, personnel exposure should be re-evaluated using data from undamaged elements or by using other available dosimetry such as self-reading dosimeters.

No written response to this information notice is required. If you need additional information regarding this matter, contact the Administrator of the appropriate NRC Regional Office.

L. I. Cobb, Director
Division of Fuel Facilities, Materials
and Safeguards
Office of Inspection and Enforcement

Technical Contact: J. Metzger
301-492-9747

Attachments:
1. Ltr. fm Panasonic Industrial Co. to its customers signed by Panasonic Natl. Sales Manager.
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bcc: G. Wayne Kerr, SP

*See previous sheet for concurrences.

WPU:JD	IE:FFMB*	IE:FFMB:C*	IE:FFMS:D
11/01/82	J.Metzger/DM	L.Higginbotham	L.I.Cobb
5520	10/ /82	10/ /82	11/ /82

Attachment 1
IN 82-42
November 5, 1982

Panasonic

Panasonic Industrial Company
Division of Matsushita Electric
Corporation of America

One Panasonic Way
Secaucus, New Jersey 07094
201.348.6200
Cable: Mecaseca NJ

Executive Offices

Dear Health Physicist:

Recently four of more than 35 owners of the Panasonic TLD System have notified us that the carbon coated polyamide backing material in the elements of our dosimeters have developed bubble type imperfections. Our factory engineers traced the problem back to the production of the dosimeter inserts. The production environment should be virtually humidity free at this time, however, the air system was shut down at the end of each work shift and the humidity that built up during this time affected the interface between the carbon coated polyamide and the aluminum substrate holding the phosphor. In the heating cycle the element is heated very quickly creating water vapor instantaneously which stresses the layers present in the element structure beyond their limits creating a bubble condition.

Panasonic has analyzed the situation and environmental conditions have been resolved in the production process. Also, a new badge has been developed which contains a single layer where no water could be trapped. This new badge contains a single polyamide/carbon backing which is a new material. There will no longer be an aluminum substrate. These new badges will be completely compatible with the old badges. The target date for the availability of these badges is February 1983.

Panasonic recommends that dosimeters be stored and read in a controlled environment of 50% or less relative humidity. If the dosimeters are worn in a high humidity area especially during the summer months a 12 hour or longer storing period in a 50% relative humidity area before reading is recommended.

Several of the facilities have had some of the 150,000 dosimeters sold for as long as five years without problems. Therefore, this might be an isolated case from one of two production runs. If you find this has happened to your dosimeters, please notify us as soon as possible and we will replace the dosimeters free of charge. We suggest, however, that you wait until the new badges are available before replacement is requested if this is possible.

David Katzman
National Sales Manager
Industrial Equipment Department
Industrial Sales Division

Attachment 2
IN 82-42
November 5, 1982

LIST OF RECENTLY ISSUED IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
82-41	Failure of Safety/Relief Valves to Open at a BWR	10/22/82	All power reactor facilities holding an OL or CP
80-35	Leaking and Dislodged Iodine-125 Implant Seeds	10/6/82	Medical licensees holding specific licenses for human use of byproduct material in sealed sources
82-40	Deficiencies in Primary Containment Electrical Penetration Assemblies	09/22/82	All power reactor facilities holding an OL or CP
82-39	Service Degradation of Thick Wall Stainless Steel Recirculation System Piping at a BWR Plant	9/21/82	All BWR facilities holding an OL or CP
82-38	Change in Format and Distribution System for IE Bulletins Circulars, and Information Notices	9/22/82	All NRC licensees
82-34 Rev. 1	Welds in Main Control Panels	09/17/82	All power reactor facilities holding an OL or CP
82-37	Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating Pressurized Water Reactor	9/14/82	All power reactor facilities holding an OL or CP
82-36	Respirator Users Warning for Certain 5-Minute Emergency Escape Self-Contained Apparatus	9/2/82	All power reactor facilities holding an OL or CP, fuel facilities and Priority I material licensees

OL = Operating License
CP = Construction Permit

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GROUP # 01   TYPE LABELS - 3    SELECTION - 000000   SETS - 02   WO# 999999 WO DATE - 110982
     99999                                                                                             # LABELS   2,592
                                                                                               DATE COMPLETED - 11/10/82
                                                                                               TIME COMPLETED - 1254
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     99999                                                                                             # LABELS   2,592
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     99999                                                                                             # LABELS   2,592
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070119007040 1 99999 GARY L GROESCH 2257 BAYRON ROAD NEW ORLEANS LA 70119	029201007018 1 99999 ROBERT GUILD ESQ 314 PALL MALL COLUMBIA SC 29201
097258006279 1 99999 RE WILLIAM L HALLMARK ESQUIRE ONE SOUTHWEST COLUMBIA SUITE 800 PORTLAND OR 97258	002111006563 1 99999 HENRY HERRMANN 151 TREMONT RM 1045 BOSTON MA 02111
033138000804 1 99999 MARTIN H HODDER ESQUIRE 1131 N E 86TH STREET MIAMI FL 33138	085040005306 1 99999 PATRICIA L HOURIHAN 6413 SOUTH 26TH ST PHOENIX AZ 85040
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RADNOR PA 19087

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KENNETH SQUARE PA 19348

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014618005398 1 99999 MICHAEL SLADE 12 TRAILWOOD CIRCLE ROCHESTER NY 14618	019404001255 1 99999 JOSEPH A SMYTH ASSISTANT COUNTY SOLICITOR MONTGOMERY COUNTY COURTHOUSE NORRISTOWN PA 19404
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049503005388 1 99999 PETER W STEKETEE 505 PEOPLES BUILDING GRAND RAPIDS MI 49503	006114081138 1 99999 ROBYN TAYLOR 1947 BROAD ST HARTFORD CT 06114
028422006569 1 99999 FRANKY THOMAS CHAIRMAN-BD OF COMMISSIONERS PO BOX 249 BOLIVIA NC 28422	0CANAD079706 1 99999 CHRIS TOOMEY 37 LOCKDARE STREET--AGINCOURT ONTARIO M1S 2Z5 CANADA
094302006000 1 99999 PAUL C VALENTINE 321 LYTTON AVENUE PALO ALTO CA 94302	060602006323 1 99999 ROBERT J VOLLEN 109 NORTH DEARBORN CHICAGO IL 60602
060601005360 1 99999 N REGION ENVIR CONTROL DIV JOHN V VRANKEN 188 WEST RANDOLPH STREET CHICAGO IL 60601	020815029135 1 99999 SEYMOUR WENNER 4807 MORGAN DRIVE ASLBP CHEVY CHASE MD 20815
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008221006299 1 99999 JOHN WILLIAMSON 211 FOREST DRIVE LINWOOD NJ 08221	023883005666 1 99999 J L WILSON MANAGER PO BOX 315 SURRY VA 23883
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CALVERT CLIFFS NPS
S M DAVIS
SUPERVISOR-OPERATIONS CA
MARYLAND RTES 2 & 4
LUSBY MD 20657

020657007023 1 99999
BALTIMORE GAS & ELECTRIC CO
CALVERT CLIFFS NPS
R E DENTON
SUPERVISOR-TRAIN & TECH SERV
MARYLAND RTES 2 & 4
LUSBY MD 20657

021203005875 1 99999
BALTIMORE GAS & ELECTRIC CO
QUALITY ASSURANCE DEPT
R M DOUGLASS
FORT SMALLWOOD ROAD COMPLEX
BOX 1475
BALTIMORE MD 21203

021203006139 1 99999
BALTIMORE GAS & ELECTRIC CO
SUPPLY
A E LUNDVALL JR
VICE PRESIDENT - SUPPLY
PO BOX 1475
BALTIMORE MD 21203

020657005871 1 99999
BALTIMORE GAS & ELECTRIC CO
CALVERT CLIFFS NUC POWER PLANT
LEON B RUSSELL
PLANT SUPERINTENDENT
LUSBY MD 20657

021203079435 3 99999
BALTIMORE GAS & ELECTRIC CO
J A TIERNAN
MANAGER-NUCLEAR POWER
PO BOX 1475 CHARLES CENTER
LUSBY MD 21203

091749052940 1 99999
BARTON INSTRUMENTS
UNIT OF ITT
G R WELT
QA DIRECTOR
900 SOUTH TURNBULL CANYON RD
CITY OF INDUSTRY CA 91749

043201064637 1 99999
BATTELLE COLUMBUS LAB
LIBRARY
R A RAWLES
MGR ADMIN & INFO SERVICES
505 KING AVENUE
COLUMBUS OH 43201

099352006329 1 99999
BATTELLE PACIFIC NORTHWEST LAB
SPENCER H BUSH
PO BOX 999
ACRS
RICHLAND WA 99352

020657007021 1 99999
BALTIMORE GAS & ELECTRIC CO
CALVERT CLIFFS NPS
J A TIERMAN
MANAGER-NUCLEAR POWER DEPT
MARYLAND RTES 2 & 4
LUSBY MD 20657

021203073435 3 99999
BALTIMORE GAS & ELECTRIC CO
J A TIERNAN
MANAGER-NUCLEAR POWER
PO BOX 1475 CHARLES CENTER
LUSBY MD 21203

095060063660 1 99999
BARON & ASSOCIATES INC
STEPHEN L BARON
207 MISSION STREET
SANTA CRUZ CA 95060

091749052940 1 99999
BARTON INSTRUMENTS
UNIT OF ITT
G R WELT
QA DIRECTOR
900 SOUTH TURNBULL CANYON RD
CITY OF INDUSTRY CA 91749

043201082125 1 99999
BATTELLE COLUMBUS LAB
NUCLEAR TECH & PHYS SCI DEPT
WALTER J ELGIN
PROJECT MANAGER
505 KING AVE
COLUMBUS OH 43201

043201064637 1 99999
BATTELLE COLUMBUS LAB
LIBRARY
R A RAWLES
MGR ADMIN & INFO SERVICES
505 KING AVENUE
COLUMBUS OH 43201

099352006415 1 99999
BATTELLE PACIFIC NORTHWEST LAB
WALT APLEY
C/O MAX CLAUSEN
BATTELLE BLVD-SIGMA IV BLDG
RICHLAND WA 99352

099352006329 1 99999
BATTELLE PACIFIC NORTHWEST LAB
SPENCER H BUSH
PO BOX 999
ACRS
RICHLAND WA 99352

048106001113 1 99999
BECHTEL POWER CORP
CENTRAL LIBRARY
LIBRARIAN
P O BOX 1000
ANN ARBOR MI 48106

094119064724 1 99999
BECHTEL POWER CORP
NFO LIBRARY
PO BOX 3965
SAN FRANCISCO CA 94119

020760005872 1 99999
BECHTEL POWER CORP
J H BELL
CHIEF NUCLEAR ENGINEER
15740 SHADY GROVE ROAD
GAITHERSBURG MD 20760

094119064588 1 99999
BECHTEL POWER CORP
M H BRUINS
STAFF ASST ADMIN SVCS
PO BOX 3965
SAN FRANCISCO CA 94119

020760026694 1 99999
BECHTEL POWER CORP
NANCY G CHAPMAN
15740 SHADY GROVE RD
GAITHERSBURG MD 20760

048106052944 1 99999
BECHTEL POWER CORP
J I DUTSON
PROJECT MANAGER
POST OFFICE BOX 1000
ANN ARBOR MI 48106

090060001126 1 99999
BECHTEL POWER CORP
LOS ANGELES POWER DIVISION
JEAN GREGORY
LIBRARIAN
TERMINAL ANNEX PO BOX 60860
LOS ANGELES CA 90060

077001007067 1 99999
BECHTEL POWER CORP
CHARLES HALLIGAN
PO BOX 2166
HOUSTON TX 77001

077252064612 1 99999
BECHTEL POWER CORP
V L HICKMAN
RECORDS MANAGER
PO BOX 2166
HOUSTON TX 77252

020877064676 1 99999
BECHTEL POWER CORP
F P JONES
SUPERVISOR REC MGMT
15740 SHADY GROVE ROAD
GAITHERSBURG MD 20877

094119006457 1 99999
BECHTEL POWER CORP
G B JONES
PROJECT MANAGER
50 BEALE STREET - PO BOX 3965
SAN FRANCISCO CA 94119

020877006748 1 99999
BECHTEL POWER CORP
QA DEPT
D C KANSAL
15740 SHADY GROVE ROAD
1A1
GAITHERSBURG MD 20877

094119064719 1 99999
BECHTEL POWER CORP
D H LANIER
PROJECT ADMINISTRATOR-SUPP GRP
PO BOX 3965
SAN FRANCISCO CA 94119

090650064620 1 99999
BECHTEL POWER CORP
C Y LOPINSKY
STAFF ENGINEER
12400 E IMPERIAL HWY
NORWALK CA 90650

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BECHTEL POWER CORP

077252064631 1 99999
BECHTEL POWER CORP

002110006058   1   99999
BERMAN & LEWENBERG
FRANCIS S WRIGHT
211 CONGRESS STREET
BOSTON   MA   02110

018016001445   1   99999
BETHLEHEM STEEL CORP
MEREDITH HEMPHILL
ASSISTANT GENERAL COUNSEL
701 EAST THIRD STREET
BETHLEHEM   PA   18016

049720005383   1   99999
BIG ROCK POINT NUC POWER PLANT
PLANT SUPERINTENDENT
CHARLEVOIX   MI   49720

097210052948   1   99999
BINGHAM WILLAMETTE CO
GUY F ATKINSON CO
C I ROVL
QA MANAGER
2800 N W FRONT AVENUE
PORTLAND   OR   97210

002125052956   1   99999
BIW CABLE SYSTEMS INC
E MANCHESTER
PRESIDENT
65 BAY STREET
BOSTON   MA   02125

064114001128   1   99999
BLACK & VEATCH
LEO HACK
LIBRARIAN
1500 MEADOW LAKE PKWY
KANSAS CITY   MO   64114

064114005181   1   99999
BLACK & VEATCH
M H ROBINSON
PO BOX 8405
KANSAS CITY   MO   64114

064114021969   1   99999
BLACK & VEATCH
POWER
M J ROBINSON
P O BOX 8405
KANSAS CITY   MO   64114

011801001412   1   99999
BLAU & COHN
HOWARD L BLAU
ESQUIRE
217 NEWBRIDGE RD
HICKSVILLE   NY   11801

0WGERN005318   1   99999
BMFT
REACTOR SAFETY
HEINZ SEIPEL
ADMINISTRATOR
HEINEMANNSTRASSE 2
5300 BONN 2   WGERMANY

098124052953   1   99999
BOEING CO
ENGR & CONSTRUCTION
GEORGE I COULBOURN
PO BOX 3707
MS 9A-73
SEATTLE   WA   98124

022180064704   1   99999
BOEING COMPUTER SERVICES
ENERGY TECH APPL
S G LINEKER
7980 GALLOWS COURT
CV-31
VIENNA   VA   22180

097208029759   1   99999
BONNEVILLE POWER ADMIN
PO BOX 3621
PORTLAND   OR   97208

0CANAD052964   1   99999
BORG WARNER CORP
BYRON JACKSON DIV
JACK ANDERSON
SUPVR-REM ENGINEERING
PO BOX 180 STATION H - TORONTO
ONTARIO M4C 5J1   CANADA

090749078990   1   99999
BORG WARNER CORP
BYRON JACKSON PUMP DIVISION
MERRITT HUGINS
PO BOX 4966
M/S 79
CARSON   CA   90749

091409053079   1   99999
BORG WARNER CORP
NUCLEAR VALVE DIVISION
P L MILINAZZO
MANAGER QUALITY ASSURANCE
7500 TYRONE AVENUE
VAN NUYS   CA   91409

090749077817   1   99999
BORG WARNER CORP
BYRON JACKSON PUMP DIVISION
JOHN MILLAN
ENERGY SYSTEMS DEVELOPMT CTR
PO BOX 4966
CARSON   CA   90749

090058052965   1   99999
BORG WARNER CORP
BYRON JACKSON PUMP DIV
R L ROMERO
2300 EAST VERNON AVENUE
06
LOS ANGELES   CA   90058

002199064706   1   99999
BOSTON EDISON CO
W L ANDERSON
NUCLEAR ENGINEER
800 BOLYSTON STREET
BOSTON   MA   02199

002199001541   1   99999
BOSTON EDISON CO
NUCLEAR PROJECTS
R M BUTLER
MANAGER
800 BOYLSTON STREET
BOSTON   MA   02199

002184006262   1   99999
BOSTON EDISON CO
NUCLEAR OPERATIONS
JOHN F FRASER
25 BRAINTREE HILL PARK
P 260
BRAINTREE   MA   02184

002199064607   1   99999
BOSTON EDISON CO
B A GORDON
ENGR PERMANENT FILE SYS
800 BOYLSTON STREET
BOSTON   MA   02199

002199080929   1   99999

002199001437   1   99999

PO [illegible]
M/S 79
CARSON CA 90749

090749077817 1 99999
BORG WARNER CORP
BYRON JACKSON PUMP DIVISION
JOHN MILLAN
ENERGY SYSTEMS DEVELOPMT CTR
PO BOX 4966
CARSON CA 90749

002199064706 1 99999
BOSTON EDISON CO
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NUCLEAR ENGINEER
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BOSTON MA 02199

002184006262 1 99999
BOSTON EDISON CO
NUCLEAR OPERATIONS
JOHN F FRASER
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P 260
BRAINTREE MA 02184

002199080929 1 99999
BOSTON EDISON CO
M/C NUCLEAR
WILLIAM D HARRINGTON
SENIOR V PRESIDENT OF NUCLEAR
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BOSTON MA 02199

002360005625 1 99999
BOSTON EDISON CO
RICHARD D MACHON
PILGRIM STATION MANAGER
RFD 1 ROCKY HILL ROAD
PLYMOUTH MA 02360

002184029316 1 99999
BOSTON EDISON CO
M/C NUCLEAR
A V MORISI
MANAGER
25 BRAINTREE HILL PARK
BRAINTREE MA 02184

002199005493 1 99999
BOSTON EDISON CO
W M SIDES
QUALITY ASSURANCE MANAGER
800 BOYLSTON ST
BOSTON MA 02199

094601063657 1 99999
BOSTROM-BERGEN METAL PRODUCTS
D RIGGE
QA MANAGER
4700 COLISEUM WAY
OAKLAND CA 94601

006226052955 1 99999
BRAND REX CO
WIRE & CABLE DIVISIONS
P PETRUCHIK
PRESIDENT W & C DIVS
PO BOX 498
WILLIMANTIC CT 06226

020036006105 1 99999
BRICH MORTON BITTNER & MONROE
ERIC A EISEN
1140 CONNECTICUT AVE NW
SUITE 1100
WASHINGTON DC 20036

011973005037 1 99999
BROOKHAVEN NATIONAL LABORATORY
NUCLEAR SAFETY LIBRARY
LIBRARIAN
BLDG 130
UPTON NY 11973

[illegible]
VAN NUYS CA [illegible]

090058052965 1 99999
BORG WARNER CORP
BYRON JACKSON PUMP DIV
R L ROMERO
2300 EAST VERNON AVENUE
06
LOS ANGELES CA 90058

002199001541 1 99999
BOSTON EDISON CO
NUCLEAR PROJECTS
R M BUTLER
MANAGER
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BOSTON MA 02199

002199064607 1 99999
BOSTON EDISON CO
B A GORDON
ENGR PERMANENT FILE SYS
800 BOYLSTON STREET
BOSTON MA 02199

002199001437 1 99999
BOSTON EDISON CO
PILGRIM NUCLEAR STATION
WILLIAM F HICKEY
LICENSING MANAGER
800 BOYLSTON STREET
BOSTON MA 02199

002360081566 1 99999
BOSTON EDISON CO
C J MATHIS
STATION MANAGER
RFD 1 ROCKY HILL RD
PLYMOUTH MA 02360

002184005624 1 99999
BOSTON EDISON CO
NUCLEAR
A V MORISI
25 BRAINTREE HILL PARK
BOSTON MA 02184

002199006073 1 99999
BOSTON EDISON CO
WILLIAM S STOWE
800 BOYLSTON STREET
BOSTON MA 02199

002141062526 1 99999
BP TECHNICAL SERVICES
18 HURLEY STREET
CAMBRIDGE MA 02141

075081063099 1 99999
BRANT & ASSOCIATES
WILLIAM A BRANT
1513 BARCLAY
RICHARDSON TX 75081

024201052957 1 99999
BRISTOL STEEL & IRON WORKS
PLANT #3
G W ROBERTS
QA MANAGER
BRISTOL/WASH COUNTY INDUST PK
BRISTOL VA 24201

011973066851 1 99999
BROOKHAVEN NATIONAL LABORATORY
REACTOR DIVISION
W J ERYNDA
UPTON
BLDG 703
LONG ISLAND NY 11973

99999

BROOKHAVEN ... INC
ASSOC UNIVERSITIES INC
LEONARD C EMMA
ASSISTANT TO THE DIRECTOR NY 11973
UPTON

077001064762 1 99999
BROWN & ROOT INC
P JORDAN
PO BOX 3 TX 77001
HOUSTON

077001064728 1 99999
BROWN & ROOT INC
J E PADEN
MGR ENG DOCUMENTATION
PO BOX 3 BLDG 41 TX 77001
HOUSTON

019477053734 1 99999
BROWN BOVERI ELECTRIC INC
SWITCHGEAR SYSTEMS DIV
D D DUVALL
VICE PRESIDENT
ROUTE 309 & NORRISTOWN ROAD PA 19477
SPRING HOUSE

040202000508 1 99999
BROWN TODD & HEYBURN
JOSEPH B HELM
ESQUIRE
16TH FLOOR CITIZENS PLAZA KY 40202
LOUISVILLE

007649006086 1 99999
BURNS & ROE INC
S BARON
700 KINDERKAMACK ROAD NJ 07649
ORADELL

007649064616 1 99999
BURNS & ROE INC
S A JORDAN
STAFF ADMINISTRATOR
800 KINDERKAMACK ROAD NJ 07649
ORADELL

007649006828 1 99999
BURNS & ROE INC
KENNETH LISH
700 KINDERKAMACK RD NJ 07649
ORADELL

007649001160 1 99999
BURNS & ROE INC
NUCLEAR LICENSING
STANLEY S ROSEN
800 KINDERKAMACK RD NJ 07649
ORADELL

007645038055 1 99999
BUTLER SERVICE GROUP
ALEXANDER J MACDOUGALL
VICE PRESIDENT-SALES
110 SUMMIT AVE P O BOX 460 NJ 07645
MONTVALE

091802052970 1 99999
C F BRAUN & CO
A H WHITAKER
VICE PRES-POWER DIVISION CA 91802
ALHAMBRA

095814006335 1 99999
CALIFORNIA DEPT OF CONSERV
DIV OF MINES & GEOLOGY
PERRY AMIMOTO
... 9TH STREET

... MANAGER RECORDS ...
PO BOX 3 01-362 TX 77001
HOUSTON

077001064659 1 99999
BROWN & ROOT INC
RICHARD P NEGRI
ASSISTANT QA MANAGER
PO BOX 3 TX 77001
HOUSTON

077001052958 1 99999
BROWN & ROOT INC
POWER
W M RICE
GROUP VICE PRESIDENT
POST OFFICE BOX 3 TX 77001
HOUSTON

0WGERM053329 1 99999
BROWN BOVERI REAKTOR GMBH
ABT TPE 14
FACHBUCHEREI
DUDENSTR 44
6800 MANNHEIM 1 WGERMANY

006856057905 1 99999
BURNDY CORP
POWER
M R MONASHKIN
CONN AND RICHARDS AVE
203 CT 06856
NORWALK

099352006467 1 99999
BURNS & ROE INC
W G CONN
N/M GROUP SUPERVISOR
601 WILLIAMS BLVD WA 99352
RICHLAND

007649006459 1 99999
BURNS & ROE INC
MATERIALS QC STANDARDS SUPV
H C JUNG
700 KINDERKAMACK RD NJ 07649
ORADELL

007649064639 1 99999
BURNS & ROE INC
H L RIESS
DOCUMENT CONTROL SUPVR
700 KINDERKAMACK RD NJ 07649
ORADELL

015077005708 1 99999
BURROUGH OF SHIPPINGPORT
THOMAS J CZERPAK
MAYOR
PO BOX 26
SHIPPINGPORT PA 15077

091802008724 1 99999
C F BRAUN & CO
WILLIAM A JACK
VICE PRESIDENT
1000 SOUTH FREMONT CA 91802
ALHAMBRA

006450020109 1 99999
C N FLAGG CO
GLENN R LACEY
MANAGER-QUALITY ASSURANCE
450 MURDOCK AVENUE CT 06450
MERIDEN

095814006285 1 99999
CALIFORNIA DEPT OF HEALTH SVCS
RADIOLOGICAL HEALTH SECTION
CHIEF ENVIR RAD CONTROL UNIT
714 P STREET
ROOM 498 CA 95814
SACRAMENTO

007649006828 1 99999
BURNS & ROE INC
KENNETH LISH
700 KINDERKAMACK RD
ORADELL NJ 07649

007649001160 1 99999
BURNS & ROE INC
NUCLEAR LICENSING
STANLEY S ROSEN
800 KINDERKAMACK RD
ORADELL NJ 07649

007645038055 1 99999
BUTLER SERVICE GROUP
ALEXANDER J MACDOUGALL
VICE PRESIDENT-SALES
110 SUMMIT AVE P O BOX 460
MONTVALE NJ 07645

091802052970 1 99999
C F BRAUN & CO
A H WHITAKER
VICE PRES-POWER DIVISION
ALHAMBRA CA 91802

095814006335 1 99999
CALIFORNIA DEPT OF CONSERV
DIV OF MINES & GEOLOGY
PERRY AMIMOTO
1416 9TH STREET
ROOM 1341
SACRAMENTO CA 95814

095814012056 2 99999
CALIFORNIA DEPT OF HEALTH SVCS
RADIOLOGIC HEALTH BRANCH
JOE WARD
CHIEF
714 P STREET
SACRAMENTO CA 95814

095825006094 1 99999
CALIFORNIA ENERGY COMMISSION
CHRISTOPH ELLISON
1111 HOWE AVENUE
SACRAMENTO CA 95825

095814057457 1 99999
CALIFORNIA OFFICE OF GOVERNOR
ENERGY & ENVIRONMENT
PHILLIP A GREENBERG
1600 9TH ST
ROOM 134
SACRAMENTO CA 95814

094102007066 1 99999
CALIFORNIA PUBLIC UTILITY COM
JANICE E KERR
350 MCALLISTER ST
SAN FRANCISCO CA 94102

020003005766 1 99999
CALIFORNIA STATE OFFICE
ELEANOR SCHWARTZ
600 PENNSYLVANIA AVE SE
RM 201
WASHINGTON DC 20003

010016001415 1 99999
CAMMER & SHAPIRO
RALPH SHAPIRO
ESQUIRE
9 EAST 40TH STREET
NEW YORK NY 10016

019004052967 1 99999
CAPITAL PIPE & STEEL PRODUCT
EDWARD [illegible]

007649064639 1 99999
BURNS & ROE INC
H L RIESS
DOCUMENT CONTROL SUPVR
700 KINDERKAMACK RD
ORADELL NJ 07649

015077005708 1 99999
BURROUGH OF SHIPPINGPORT
THOMAS J CZERPAH
MAYOR
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SHIPPINGPORT PA 15077

091802008724 1 99999
C F BRAUN & CO
WILLIAM A JACK
VICE PRESIDENT
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ALHAMBRA CA 91802

006450020109 1 99999
C N FLAGG CO
GLENN R LACEY
MANAGER-QUALITY ASSURANCE
450 MURDOCK AVENUE
MERIDEN CT 06450

095814006285 1 99999
CALIFORNIA DEPT OF HEALTH SVCS
RADIOLOGICAL HEALTH SECTION
CHIEF ENVIR RAD CONTROL UNIT
714 P STREET
ROOM 498
SACRAMENTO CA 95814

095825080939 1 99999
CALIFORNIA ENERGY COMMISSION
SITING & ENVIRONMENTAL
E R DETTER
DIV CHIEF
1111 HOWE AVENUE
SACRAMENTO CA 95825

095814005986 1 99999
CALIFORNIA LEGAL AFFAIRS SECY
BRYON S GEORGIOU
STATE CAPITAL-GOVERNORS OFC
SACRAMENTO CA 95814

094102006009 1 99999
CALIFORNIA PUBLIC UTILITY COM
LAWRENCE Q GARCIA
5066 STATE BLDG
SAN FRANCISCO CA 94102

095814080938 1 99999
CALIFORNIA RESOURCES AGENCY
HUEY D JOHNSON
SECRETARY FOR RESOURCES
1416 9TH STREET
ROOM 1311
SACRAMENTO CA 95814

095630078956 1 99999
CALIFORNIANS FOR SAFE ENERGY
ELLIS S TRAUBER
CHAIRMAN
3673 MESA VERDE AVE
EL DORADO HILLS CA 95630

091361079030 1 99999
CANYON RESEARCH GROUP INC
ERNEST J CONWAY
741 LAKEFIELD ROAD
SUITE B
WESTLAKE VILLAGE CA 91361

028208000669 1 99999
CAROLINA ENVIRON STUDY GROUP

027602006083 1 99999
CAROLINA POWER & LIGHT CO
POWER SUPPLY CONSTRUCT & ENGN
E E UTLEY
EXEC VICE PRESIDENT
PO BOX 1551
RALEIGH NC 27602

019603052968 1 99999
CARPENTER TECHNOLOGY CORP
W R KEGERISE
GEN MGR NUCLEAR PRODUCTS
101 WEST BERN STREET
READING PA 19603

060521079443 1 99999
CATALYTIC INC
2301 SPRING RD
SUITE 530
OAK BROOK IL 60521

019102028466 1 99999
CATALYTIC INC
PAUL REICHART
1500 MARKET STREET
PHILADELPHIA PA 19102

090064001221 1 99999
CENTER FOR LAW PUBLIC INTEREST
JOHN R PHILLIPS
ESQUIRE
10951 WEST PICO BLVD
LOS ANGELES CA 90064

0SPAIN023959 1 99999
CENTRAL NUCLEAR DE ALMARAZ
JUAN M BLANCO
PRINCESA 3
MADRID - 8 SPAIN

078403001440 1 99999
CENTRAL POWER & LIGHT CO
M L BORCHELT
PO BOX 2121
CORPUS CHRISTI TX 78403

0BELGI057562 1 99999
CENTRE D'ETUDE DE L'ENER NUCL
POWER PLANT BX 3
FRANCOIS MOTTE
ADMINISTRATOR
BOERETANG 200
B-2400 MOL BELGIUM

0FRANC003524 1 99999
CENTRE D'ETUDES NUCLEAIRES
DEPT DE SURETE NUCLEAIRE
GABRIELLE KATZ
BP NO 6
92.260
FONTENAY-AUX-ROSES FRANCE

063102006104 1 99999
CHACKES & HOARE
KENNETH M CHACKES
ATNY FOR JOINT INTERVENORS
314 N BROADWAY
ST LOUIS MO 63102

005404053041 1 99999
CHAMPLAIN CABLE CORP
WIRE & CABLE DIVISION
C L MARTIN
P O BOX 7
WINOOSKI VT 05404

027514007042 1 99999
CHANGE
DANIEL F READ
PRESIDENT
PO BOX 524
CHAPEL HILL NC 27514

002199067190 1 99999
CHARLES T MAIN INC
THERMAL POWER DIVISION
J H NOBLE
PRUDENTIAL CENTER
BOSTON MA 02199

025304063988 1 99999
CHARLESTON AREA MEDICAL CENTER
MEMORIAL DIV
RIDGELY CONANT
DIRECTOR OF NUC MEDICINE
3200 MACCORKLE AVE SE
CHARLESTON WV 25304

028207007020 1 99999
CHARLOTTE MECKLENBURG ENV COAL
HENRY PRESLER
CHAIRMAN
943 HENLY PLACE
CHARLOTTE NC 28207

029210066210 1 99999
CHEM NUCLEAR SYSTEMS INC
INFORMATION SPECIALIST
240 STONERIDGE DRIVE
SUITE 100
COLUMBIA SC 29210

060602005200 1 99999
CHERRY & FLYNN
MYRON CHERRY
THREE FIRST NATIONAL PLAZA
SUITE 3700
CHICAGO IL 60602

021218006421 1 99999
CHESAPEAKE ENERGY ALLIANCE
ROBERT Q POLLARD
609 MONTPELIER STREET
BALTIMORE MD 21218

038113052969 1 99999
CHICAGO BRIDGE & IRON CO
2700 CHANNEL AVE P O BOX 13308
MEMPHIS TN 38113

035201052971 1 99999
CHICAGO BRIDGE & IRON CO
WELDING & QA DEPT
PO BOX 277
BIRMINGHAM AL 35201

060611005731 1 99999
CHICAGO METRO SANITARY DIST
CECIL LUE HING
DIRECTOR OF RESEARCH & DEV
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045219032023 1 99999
CINCINNATI ALLI FOR RESPON ENG
TOM CARPENTER
DIRECTOR
2270 VINE STREET
CINCINNATI OH 45219

038113052969 1 99999
CHICAGO BRIDGE & IRON CO
2700 CHANNEL AVE P O BOX 13308
MEMPHIS TN 38113

035201052971 1 99999
CHICAGO BRIDGE & IRON CO
WELDING & QA DEPT
PO BOX 277
BIRMINGHAM AL 35201

060611005731 1 99999
CHICAGO METRO SANITARY DIST
CECIL LUE HING
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CHICAGO IL 60611

045219032023 1 99999
CINCINNATI ALLI FOR RESPON ENG
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045201079120 1 99999
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045153080925 1 99999
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074107005184 1 99999
CITIZENS ACT FOR SAFE ENER INC
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045220081495 1 99999
CITIZENS AGAINST A RADIO ENVIR
2699 CLIFTON AVE
CINCINNATI OH 45220

018501001498 1 99999
CITIZENS AGAINST NUC DANGERS
THOMAS J HALLIGAN
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PO BOX 5
SCRANTON PA 18501

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CITIZENS AGAINST NUCLEAR POWER
CO-COORDINATORS
PATRICIA T NEWMAN
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CITIZENS ASSOC FOR SOUND ENGY
JUANITA ELLIS
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1426 SOUTH POLK
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CITIZENS CONCERNED ABT NUC PWR
LANNY SINKIN
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CITIZENS FOR A BETTER ENVIRON
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CITIZENS FOR A SAFE ENVIRON
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077422005309 1 99999
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078767007039 1 99999
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078767001481 1 99999
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092672005376 1 99999
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044101004553 1 99999
CLEVELAND ELEC ILLUMINATING CO
SYSTEMS ENGINEERING & CONSTR

BELOIT WI 53511

006095038343 1 99999
COMBUSTION ENGINEERING INC
PSG LIBRARY-DEPT 6435-405
LIBRARIAN-ATTN R&D DOCUMENTS
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020014005904 1 99999
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006095064587 1 99999
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6110-1901
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006905008721 1 99999
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037402052975 1 99999
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006095001299 1 99999
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CE POWER SYSTEMS
J W VEIRS
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0MEXIC066478 1 99999
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APDO POSTAL NO 5-214
MEXICO 5-DF MEXICO

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COMISION NACIONAL SEGURIDAD
NUCLEAR Y SALVAGUARDIAS
INSURGENTES SUR NO 1806
MEXICO DF 01030 MEXICO

0BRAZI008064 1 99999
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060690006231 1 99999
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061341081556 1 99999
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060690079389 1 99999
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049201001421 1 99999
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049201064735 1 99999
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PROJ RECORDS MGMT SUPVR
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049201006276 1 99999
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049201058220 1 99999
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049201003026 1 99999
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049043005337 1 99999
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049201005391 1 99999
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060450081512 1 99999
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048640066519 1 99999
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092714052938 1 99999
CONTROL COMPONENTS INC
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020852064749 1 99999
CONTROL DATA CORP
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036250005796 1 99999
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065066006489 1 99999
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060450005402 1 99999
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052406005646 1 99999
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054601006426 1 99999
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065251006106 1 99999
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049720005390 1 99999
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097501006345 1 99999
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BD OF COUNTY COMMISSIONERS
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017101005434 1 99999
COUNTY OF DAUPHIN
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EARL B HOF[illegible]N
COMMISSIO[illegible]
FRONT & MARKET STREET
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055066005941 1 99999
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044077006128 1 99999
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[illegible] CA [illegible]

068305005855 1 99999
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043452005779 1 99999
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061201005850 1 99999
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092101005377 1 99999
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023683005668 1 99999
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080631006315 1 99999
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007094037904 1 99999
COURTER & CO INC
QA/QC
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001151053737 1 99999
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002093052986 1 99999
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019468056822 1 99999
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REGULATION & LIC OF NUC REACT
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FERMI 2 NPS
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048226079139 1 99999
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048084006500 1 99999
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048226006228 1 99999
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048226004989 3 99999
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DONCHUE JONES BROWN & CLIFFORD
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048640081494 1 99999
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048640081492 1 99999
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RADIATION SAFETY CMTE
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060450005418 1 99999
DRESDEN NUCLEAR POWER STATION
PLANT SUPERINTENDENT
RURAL ROUTE #1
MORRIS IL 60450

060450006422 1 99999
DRESDEN NUCLEAR POWER STATION
PLANT SUPERINTENDENT
RURAL ROUTE #1
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029342080967 1 99999
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029678026043 1 99999
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028242007094 1 99999
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028242081033 1 99999
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028242001547 1 99999
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029710080962 1 99999
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028242065207 1 99999
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028031080896 1 99999
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028242006327 1 99999
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029678003050 1 99999
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EDS NUCLEAR INC
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094104064768 1 99999
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011375000979 1 99999
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019355008722 1 99999
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094598057030 1 99999
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030092066906 1 99999
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083401064744 1 99999
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083401062537 1 99999
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083415062748 1 99999
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015222062525 1 99999
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032301005802 1 99999
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006340020097 2 99999
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GENERAL ELECTRIC CO
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019142053020 1 99999
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095125029555 1 99999
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019142055290 1 99999
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052601053024 1 99999
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095125064718 1 99999
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020878064606   1   99999
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033515006269   1   99999
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068102080934 1 99999
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0CANAD064709 1 99999
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097310056502 1 99999
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SALEM OR 97310

097207002513 1 99999
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077701000770 1 99999
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094111006011 1 99999
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094111066113 1 99999
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020006031123 1 99999
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094106064688 1 99999
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094120005996 1 99999
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093424003219 1 99999
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095501057489 1 99999
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090255053081 1 99999
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032020052945 1 99999
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094106005747 1 99999
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097403007015 1 99999
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092803053082 1 99999
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090807053083 1 99999
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007094064697 1 99999
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035810081026 1 99999
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018101000626 1 99999
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018101008705 1 99999
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018101064665 1 99999
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018603009311 1 99999
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018101081560 1 99999
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018101064722 1 99999
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018101079441 1 99999

016103005706 1 99999

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019101007104 1 99999
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017314006576 1 99999
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PEACH BOTTOM APS
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0PHIL1057567 1 99999
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077471005143 1 [illegible]99
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097204005960 1 99999
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097204004657 1 99999
PORTLAND GENERAL ELECTRIC CO
BART D WITHERS
VICE PRESIDENT NUCLEAR
121 S W SALMON STREET
PORTLAND OR 97204

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PORTLAND GENERAL ELECTRIC CO
BART D WITHERS
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097204081459 1 99999
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006359026457 1 99999
POST SEAL INTERNATIONAL INC
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006001032077 1 99999
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061820007030 1 99999
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061820081497 1 99999
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022102064742 1 99999
PRC IMAGE DATA SYSTEMS CO
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063147053069 1 99999
PROGRESSIVE FABRICATORS INC
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095128064738 1 99999
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037830064617 1 99999
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006340081116 1 99999
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070802001283 1 99999
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087158003271 1 99999
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080651052349 1 99999
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080201006312 1 99999
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080120005753 1 99999
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080651004652 1 99999
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046320001522 1 99999
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074102007686 1 99999
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002210053092 1 99999
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015108079693 1 99999
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048124077486 1 99999
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CAMPBELL CA 95008

095030067156 1 99999
R F REEDY INC
RICK SWAYNE
236 N SANTA CRUZ AVE
SUITE 247
LOS GATOS CA 95030

094025026771 1 99999
RAYCHEM CORP
F E LAFETRA
300 CONSTITUTION DRIVE
MENLO PARK CA 94025

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RAYTHEON CO
P L EDSALL
TECHNICAL INFO SPECIALIST
2 WAYSIDE ROAD
BURLINGTON MA 01803

095128057907 1 99999
REACTOR CONTROLS INC
JACK C MILLETT
1245 SOUTH WINCHESTER BLVD
SUITE 222
SAN JOSE CA 95128

012181003805 3 99999
RENSSELAER POLYTECHNIC INST
SERIALS DEPT
FOLSOM LIBRARY
TROY NY 12181

044128012807 1 99999
REUTER STOKES INC
MANAGER QUALITY ASSURANCE
ROBERT L SIMON
18530 SOUTH MILES PARKWAY
WARRENSVILLE HEIGHTS OH 44128

002908023165 1 99999
RHODE ISLAND DEPT OF HEALTH
DIV OCCUPATNL HLTH & RAD CNTL
JAMES E HICKEY
CHIEF
DAVIS STREET-CANNON BLDG
PROVIDENCE RI 02908

0FRANC053096 1 99999

WALLINGFORD CT 06492

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ROCKWELL INTERNATIONAL
ENERGY SYSTEMS GROUP
A CARNAZZOLA
CODES & STANDARDS
8900 DE SOTO AVENUE
CANOGA PARK CA 91304

035801081550 1 99999
ROCKWELL INTERNATIONAL
NEILL LANGFORD
3322 S MEMORIAL PARKWAY
SUITE 39
HUNTSVILLE AL 35801

002110000449 1 99999
ROPES & GRAY
JOHN A RITSHER
ESQUIRE
225 FRANKLIN STREET
BOSTON MA 02110

030339064286 1 99999
ROSEMOUNT INC
RANDY HALEY
290 INTERSTATE NORTH
SUITE 150
ATLANTA GA 30339

092701000876 1 99999
ROURKE & WOODRUFF
ALAN R WATTS
ESQUIRE
1055 NORTH MAIN STREET
SUITE 1020
SANTA ANA CA 92701

064012064461 1 99999
RUSKIN MANUFACTURING CO
QUALITY ASSURANCE DEPT
QA MANAGER
PO BOX 129
GRANDVIEW MO 64012

029201009340 1 99999
S CAROLINA DEPT HLTH ENVIR CTL
DIVISION RADIOLOGICAL HEALTH
HEYWARD SHEALY
CHIEF
2600 BULL ST-J MARION SIMS BLD
COLUMBIA SC 29201

029065002890 1 99999
S CAROLINA ELEC & GAS CO
SUMMER 1 NPS
O S BRADHAM
STATION MANAGER
PO BOX 8
JENKINSVILLE SC 29065

029218062856 1 99999
S CAROLINA ELEC & GAS CO
SUMMER 1 NPS
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029218004542 1 99999
S CAROLINA ELEC & GAS CO
NUCLEAR OPERATIONS
T C NICHOLS
PO BOX 764
F-28
COLUMBIA SC 29218

029065081353 1 99999
S CAROLINA ELEC & GAS CO
A A SMITH
SITE QA COORDINATOR
PO BOX 9
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027602053099 1 99999
ROCKWELL INTERNATIONAL
FLOW CONTROL DIV
J V GRASSO
PLANT MANAGER
P O BOX 1961
RALEIGH NC 27602

002110067268 1 99999
ROPES & GRAY
THOMAS DIGNAN
ESQUIRE
225 FRANKLIN STREET
BOSTON MA 02110

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ROSEMOUNT INC
NUCLEAR QUALITY ASSURANCE
GERALD D ANDERSON
12001 WEST 78TH STREET
A1
EDEN PRAIRIE MN 55344

055435065512 1 99999
ROSEMOUNT INC
LYLE LOFGREN
SENIOR ENGR
PO BOX 35129
A1
MINNEAPOLIS MN 55435

095201055291 1 99999
RTE DELTA CORP
705 NORTH CARLTON AVENUE
STOCKTON CA 95201

002181056882 1 99999
RV & ASSOCIATES
AVRUM SILVERMAN
69 EDGEMOOR AVENUE
WELLESLEY MA 02181

029201001392 1 99999
S CAROLINA DEPT HLTH ENVIR CTL
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029218006033 1 99999
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O W DIXON
VICE PRESIDENT-NUCLEAR OPER
PO BOX 764 MAIL CODE F-04
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029065080954 1 99999
S CAROLINA ELEC & GAS CO
SUMMER 1 NPS
D A LAVIGNE
SITE QA COORDINATOR
PO BOX 88
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029218080953 1 99999
S CAROLINA ELEC & GAS CO
SUMMER 1 NPS
C A PRICE
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PO BOX 764
COLUMBIA SC 29218

029218006034 1 99999
S CAROLINA ELEC & GAS CO
MARK WHITAKER
POST OFFICE BOX 764
COLUMBIA SC 29218

1 99999

057501064189 1 99999

PO BOX 764
F-28
COLUMBIA SC 29218

NUCLEAR ENGNG MANAGER
PO BOX 764
COLUMBIA SC 29218

029065081353 1 99999
S CAROLINA ELEC & GAS CO
A A SMITH
SITE QA COORDINATOR
PO BOX 9
JENKINSVILLE SC 29065

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S CAROLINA ELEC & GAS CO
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POST OFFICE BOX 764
COLUMBIA SC 29218

029461006234 1 99999
S CAROLINA PUB SERV COM
HENRY CYRUS
SENIOR VICE PRESIDENT
233 NORTH LIVE OAK DRIVE
MONCKS NC 29461

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S DAKOTA STATE PLANNING BUREAU
CRAIG MCINTYRE
EXEC POLICY ANALYST
CAPITOL BUILDING
PIERRE SD 57501

095008079603 1 99999
S LEVY INC
LINDA SMITH
LIBRARIAN
1999 S BASCOM AVE
SUITE 725
CAMPBELL CA 95008

080302031308 1 99999
S M STOLLER CORP
1919 14TH STREET
SUITE 500
BOULDER CO 80302

080302052913 1 99999
S M STOLLER CORP
ERIC OLSON
SUITE 500 COLORADO BLDG
BOULDER CO 80302

0BELGI081072 1 99999
SA INTERCOM NV
PLACE DU TRONE 1 BI 0607
BRUSSELS 1000 BELGIUM

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SACRAMENTO MUNICIPAL UTIL DIST
J J JEWETT
SENIOR ENGINEER
6201 S STREET
SACRAMENTO CA 95813

095813006095 1 99999
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095813006531 1 99999
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6201 S STREET
BOX 15830
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095813007423 1 99999
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SACRAMENTO CA 95813

095813064384 1 99999
SACRAMENTO MUNICIPAL UTIL DIST
RANCHO SECO NUCLEAR PWR PLANT
R J RODRIGUEZ
MANAGER
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SACRAMENTO CA 95813

095813079430 1 99999
SACRAMENTO MUNICIPAL UTIL DIST
L G SCHWIEGER
PO BOX 15830
SACRAMENTO CA 95813

049508063180 1 99999
SALEM ENGINEERING CORP
2186 44TH STREET SE
GRAND RAPIDS MI 49508

053226063184 1 99999
SALEM ENGINEERING CORP
2360 N 124TH STREET
WAUWATOSA WI 53226

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SALEM ENGINEERING CORP
SALEM EMPLOYMENT SERVICES
4540 W 34TH STREET
HOUSTON TX 77092

001803063181 1 99999
SALEM ENGINEERING CORP
SALEM TECHNICAL SERVICES
CORPORATE PLACE 1
99 S BEDFORD ST SUITE 207
BURLINGTON MA 01803

029360006278 1 99999
SALUDA RIVER ELEC CO-OP INC
207 SHERWOOD DR
LAURENS SC 29360

078296001482 1 99999
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SAN ANTONIO TX 78296

092112001533 1 99999
SAN DIEGO GAS & ELECTRIC CO
POWER SUPPLY
G D COTTON
L BERNATH
PO BOX 1831
SAN DIEGO CA 92112

092112008694 1 99999
SAN DIEGO GAS & ELECTRIC CO
POWER PLANT ENGR & CONSTR
GARY D COTTON
PO BOX 1831
SAN DIEGO CA 92112

4540 W 34TH STREET
HOUSTON TX 77092

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029363006278 1 99999
SALUDA RIVER ELEC CO-OP INC
207 SHERWOOD DR
LAURENS SC 29360

078296001482 1 99999
SAN ANTONIO PUBLIC SERVICE BD
OPERATIONS
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PO BOX 1771
SAN ANTONIO TX 78296

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SAN DIEGO GAS & ELECTRIC CO
POWER SUPPLY
G D COTTON
L BERNATH
PO BOX 1831
SAN DIEGO CA 92112

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SAN DIEGO GAS & ELECTRIC CO
POWER PLANT ENGR & CONSTR
GARY D COTTON
PO BOX 1831
SAN DIEGO CA 92112

092112006330 1 99999
SAN DIEGO GAS & ELECTRIC CO
HENRY PETERS
PO BOX 1831
SAN DIEGO CA 92112

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SAN DIEGO GAS & ELECTRIC CO
HARRY B STOEHR
PO BOX 1831
SAN DIEGO CA 92112

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SAN DIEGO GAS & ELECTRIC CO
JACK E THOMAS
PO BOX 1831
SAN DIEGO CA 92112

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SANDIA LABORATORIES
DIVISION 4446
L L BONZON
ALBUQUERQUE NM 87185

060603006509 1 99999
SARGENT & LUNDY
NUCLEAR SFGDS & LICENSING DIV
EDWARD R CRASS
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CHICAGO IL 60603

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SARGENT & LUNDY
P J FROEHLKE
DOCUMENTATION SECTION LDR
55 EAST MONROE STREET
CHICAGO IL 60603

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SARGENT & LUNDY
J C GILBERT
CORPORTE RECORDS MANAGER
1225 DARTMOUTH
WILMETTE IL 60091

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SARGENT & LUNDY
R C HEIDER
PROJECT MANAGER
55 EAST MONROE ST
CHICAGO IL 60603

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SARGENT & LUNDY
E KABOT
SENIOR QA CONTROLLER
55 EAST MONROE STREET
CHICAGO IL 60603

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SARGENT & LUNDY
DIV-NUCLEAR SAFEGUARDS & LIC
DAN KANE
ASST HEAD
55 EAST MONROE ST
CHICAGO IL 60603

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SARGENT & LUNDY
ROGELIO V PENASALES
6322 N AVERS AVE
CHICAGO IL 60659

060603064686 1 99999
SARGENT & LUNDY
G PRAPUOLENIS
DOCUMENTATION SUPERVISOR
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CHICAGO IL 60603

060603064662 1 99999
SARGENT & LUNDY
M ROSA JR
DOCUMENTATION SUPERVISOR
55 EAST MONROE STREET
CHICAGO IL 60603

060603065213 1 99999
SARGENT & LUNDY
MARTIN E SCHUSTER
HEAD-QUALITY CONTROL DIVISION
55 E MONROE ST
FLOOR 18
CHICAGO IL 60603

060603053105 1 99999
SARGENT & LUNDY
H S TAYLOR
HEAD-QUALITY ASSURANCE DIV
55 EAST MONROE ST
CHICAGO IL 60603

060603064671 1 99999
SARGENT & LUNDY
G J VANCINA
DOCUMENTATION SUPERVISOR
55 E MONROE STREET 21ST
CHICAGO IL 60603

060603006506 1 99999
SARGENT & LUNDY
P L WATTELET
55 EAST MONROE STREET
CHICAGO IL 60603

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SAVAGE WALKER SCHULT ASSOC INC
10880 INDECO DR
CINCINNATI OH 45241

CHICAGO IL 60603

020036356038 1 99999
SCANDINAVIAN DOCUMENT CENTER
F-1277-9-L
1120 19TH STREET NW
WASHINGTON DC 20036

060606007033 1 99999
SCHIFF HARDIN & WAITE
SHELDON ZABEL
7200 SEARS TOWER 233 WACKER DR
CHICAGO IL 60606

015222038031 1 99999
SCHNEIDER CONSULTING ENGINEERS
QA DEPT
E J RUSH
CORP QA MANAGER
125 SEVENTH STREET
PITTSBURGH PA 15222

037830023285 1 99999
SCIENCE APPLICATIONS INC
TECHNICAL INFORMATION
LIBRARY
800 OAK RIDGE TNPK SUITE 1000
OAK RIDGE TN 37830

0SPAIN003439 1 99999
SENER INGENIERIA Y SISTEMAS
DOCUMENTATION SERVICE
F ALBISU
APARTADO 8
LAS ARENAS(VIZCAYA) SPAIN

001867000863 1 99999
SERVICAIR CO
EDWARD W KELLY
47 WINTHROP AVE
READING MA 01867

020036005762 1 99999
SHAW PITTMAN POTTS TROWBRIDGE
THOMAS BAXTER
1800 M STREET NW
WASHINGTON DC 20036

020036007098 1 99999
SHAW PITTMAN POTTS TROWBRIDGE
GERALD CHARNOFF
1800 M STREET NW
WASHINGTON DC 20036

020036005863 1 99999
SHAW PITTMAN POTTS TROWBRIDGE
BRUCE CHURCHILL
1800 M STREET NW
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020036005190 1 99999
SHAW PITTMAN POTTS TROWBRIDGE
GEORGE F TROWBRIDGE
ESQ
1800 M STREET NW
WASHINGTON DC 20036

094115005653 1 99999
SIERRA CLUB
LEGAL DEFENSE FUND INC
MICHAEL R SHERWOOD
2044 FILLMORE STREET
SAN FRANCISCO CA 94115

093105000947 1 99999
SCENIC SHORELINE PRESERV CONF
FREDERICK EISSLER
4623 MORE MESA DRIVE
SANTA BARBARA CA 93105

015222067155 1 99999
SCHNEIDER CONSULTING ENGINEERS
RAY KATHIRESAN
908 PENN AVE
PITTSBURGH PA 15222

0JAPAN057564 1 99999
SCIENCE & TECHNOLOGY AGENCY
ATOMIC ENERGY BUREAU
ADMINISTRATOR
2-1 KASUMIGASEKI 2-CHOME
CHIYODA-KU TOKYO 100 JAPAN

0FRANC008069 1 99999
SCSIN-MINISTRY OF INDUSTRY
ANDRE MESSIAH
ADMINISTRATOR
99 RUE DE GRENELLE
75700 PARIS FRANCE

053066065537 1 99999
SENTRY EQUIPMENT CORP
OPERATIONS DEPT
R J SUSLICK
MANAGER OPERATIONS DEPARTMENT
856 EAST ARMOUR ROAD
OCONOMOWOC WI 53066

094108005987 1 99999
SEYMOUR & WILLIS
HARRY M WILLIS
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SUITE 2100
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020036000650 1 99999
SHAW PITTMAN POTTS TROWBRIDGE
ERNEST E BLAKE
1800 M STREET NW
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020036010518 1 99999
SHAW PITTMAN POTTS TROWBRIDGE
GERALD CHARNOFF ESQ
1800 M STREET NW
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SHAW PITTMAN POTTS TROWBRIDGE
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011792006326 1 99999
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020007051946 1 99999
SIT
3421 R ST NW
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094115005653 1 99999
SIERRA CLUB
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085073000937 1 99999
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020850021731 1 99999
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035202008695 1 99999
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033528039976 1 99999
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091770058231 1 99999
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091770079429 1 99999
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092672081541 1 99999
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SAN ONOFRE NPS
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035202005789 1 99999
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035202001236 1 99999
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030367003262 1 99999
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033528004318 1 99999
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0782H4064614 1 99999
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046222053726 1 99999
SPEEDWAY MACHINE & TOOL CO INC
QA MANAGER
1802 NORTH LUETT ST
INDIANAPOLIS IN 46222

046970053110 1 99999
SQUARE D CO
POWER EQUIP DIV
LARRY D WEST
QA MGR
252 N TIPPECANOE ST
PERU IN 46970

080217053111 1 99999
STEARNS ROGER INC
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077001066465 1 99999
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008034064668 1 99999
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002107064708 1 99999
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S A COMEFORD
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070775064727 1 99999
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002107038032 1 99999
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002107064726 1 99999
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002107064630 1 99999
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D M OMAN
ASST MGR REC MGMT DIV
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BOSTON MA 02107

078284023410 1 99999
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I C DIVISION
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ELGIN IL 60120

033131006038 1 99999
STEEL HECTOR & DAVIS
NORMAN A COLL
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020006001101 1 99999
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008034064667 1 99999
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002107064?30 1 99999
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002107005849 1 99999
STONE & WEBSTER ENGR CORP
NUCLEAR ENERGY DEVELOPMENT
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020906055441 1 99999
STRIDE INC
GAJANAN M SABNIS
13721 TOWN LINE ROAD
SILVER SPRING MD 20906

094583066632 1 99999
STRUCTURAL MECHANICS ASSOC INC
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SAN RAMON CA 94583

008071053724 1 99999
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F P MERKEL
QUALITY CONTROL ENGINEER
LAMBS ROAD
PITMAN NJ 08071

016365053112 1 99999
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0SWEDE003323 1 99999
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STUDSVIK LIBRARY
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611 82 NYKOPING SWEDEN

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SUGARMAN & DENWORTH
NORTH AMERICAN BUILDING
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0SWITZ003423 1 99999
SULZER BROTHERS INC
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018703001492 1 99999
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063101053115 1 99999
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007013053116 1 99999
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ONE CLIFTON BLVD
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0SWITZ066988 1 99999
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049801064460 1 99999
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037830065216 1 99999
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0TAIWA003568 1 99999
TAIWAN POWER CO

011735053118 1 99999
TARGET ROCK CORP

0TAIWA003568 1 99999
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011735053118 1 99999
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019355057363 1 99999
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037922081438 1 99999
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037873004636 1 99999
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037902001295 1 99999
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DALLAS TX 75201

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TOLEDO OH 43652

043652005787 1 99999
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043449081508 1 99999
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T D MURRAY
STATION SUPERINTENDENT
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043652007103 1 99999
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054216005830 1 99999
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CHAIRMAN
ROUTE 1
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TOWN OF GENOA
TOWN CHAIRMAN
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TOWN HALL
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014519005395 1 99999
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001367005412 1 99999
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013126006536 1 99999
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OSWEGO NY 13126

054241005865 1 99999
TOWN OF TWO CREEKS
GORDON BLAHA
TOWN CHAIRMAN
ROUTE 3
TWO RIVERS WI 54241

[illegible] ST
PLAINVILLE CT 06062

013126006536 1 99999
TOWN OF SCRIPA
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RD #4
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006385005408 1 99999
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HALL OF RECORDS
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008038005632 1 99999
TOWNSHIP OF LOWER ALLOWAYS
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HANCOCKS BRIDGE NJ 08038

017314005775 1 99999
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CHAIRMAN
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DELTA PA 17314

094621052988 1 99999
TRANSAMERICA DELAVAL INC
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VP & GENERAL MGR
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OAKLAND CA 94621

010601055062 1 99999
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094550056865 1 99999
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011237053122 1 99999
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011901005351 1 99999
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TOWN CHAIRMAN
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017057005430 1 99999
TOWNSHIP OF S LONDONDERRY
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010601053120 1 99999
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063141038029 1 99999
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P-530
WASHINGTON DC 20555

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US NRC
NRR-DIV OF ENGINEERING
SECTION LEADER-ENV QUAL
EQUIP QUALIF BRANCH
440
WASHINGTON DC 20555

EQUIP QUALIF BRANCH
538
WASHINGTON DC 20555

SECTION LEADER-CHEM TECH
CHEMICAL ENG BRANCH
P-802
WASHINGTON DC 20555

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US NRC
NRR-DIV OF ENGINEERING
CHIEF
MATERIALS ENG BRANCH
P 328
WASHINGTON DC 20555

020555056746 1 99999
US NRC
NRR-DIV OF ENGINEERING
ASSIST DIRECTOR
COMPONENTS & STRUCTURE ENG
P-530
WASHINGTON DC 20555

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US NRC
NRR-DIV OF ENGINEERING
DIRECTOR
P-202
WASHINGTON DC 20555

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US NRC
NRR-DIV OF ENGINEERING
SECTION LEADER-ENV QUAL
EQUIP QUALIF BRANCH
440
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CHIEF
CHEMICAL ENG BRANCH
P-302
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NRR-DIV OF ENGINEERING
CHIEF
STRUCTURAL ENG BRANCH
P-1022
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NRR-DIV OF ENGINEERING
CHIEF
QUALITY ASSURANCE BRANCH
P-320C
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
DEPUTY DIRECTOR
528
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
CHIEF
LICENSING BRANCH #2
328
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
CHIEF ORB #1
440
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
CHIEF ORB #2
416B
WASHINGTON DC 20555

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NRR-DIV OF LICENSING
CHIEF ORB #3
430
WASHINGTON DC 20555

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NRR-DIV OF LICENSING
CHIEF ORB #5
308
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
CHIEF
LICENSING BRANCH #1
316
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
CHIEF
LICENSING BRANCH #3
124
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
CHIEF
SYSTEMATIC EVALUATION PGM BR
516
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
CHIEF
OPERATING REACTORS ASSESS BR
540
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
CHIEF
STAN & SPECIAL PROJECTS BRANCH
340
WASHINGTON DC 20555

020555064843 1 99999
US NRC
NRR-DIV OF LICENSING
DIRECTOR
528
WASHINGTON DC 20555

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US NRC
NRR-DIV OF LICENSING
CHIEF ORB #4
416A
WASHINGTON DC 20555

020555056753 1 99999
US NRC
NRR-DIV SAFETY TECHNOLOGY
TECHNICAL ASSISTANT
278
WASHINGTON DC 20555

020555056782 1 99999
US NRC
NRR-DIV SAFETY TECHNOLOGY
DIRECTOR
278
WASHINGTON DC 20555

020555056753 1 99999
US NRC
NRR-DIV SAFETY TECHNOLOGY
TECHNICAL ASSISTANT
278
WASHINGTON DC 20555

020555056784 1 99999
US NRC
NRR-DIV SAFETY TECHNOLOGY
ASSISTANT DIRECTOR
GENERIC PROJECTS
278
WASHINGTON DC 20555

020555056788 1 99999
US NRC
NRR-DIV SAFETY TECHNOLOGY
SECTION LEADER-STSS
LICENSING GUIDANCE BRANCH
286
WASHINGTON DC 20555

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US NRC
NRR-DIV SAFETY TECHNOLOGY
CHIEF
SAFETY PROG EVALUATION BRANCH
216
WASHINGTON DC 20555

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US NRC
NRR-DIV SYSTEMS INTEGRATION
CHIEF
INSTRU & CONTROL SYS BR
P-1030C
WASHINGTON DC 20555

020555056772 1 99999
US NRC
NRR-DIV SYSTEMS INTEGRATION
SECTION LEADER/SYS ANALYSIS
EFFLUENT TREATMENT SYS BRANCH
P-730
WASHINGTON DC 20555

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US NRC
NRR-DIV SYSTEMS INTEGRATION
CHIEF
RADIOLOGICAL ASSESSMENT BRANCH
P-712
WASHINGTON DC 20555

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US NRC
NRR-PLAN & PROG ANALYSIS STAFF
CHIEF
TECHNICAL SUPPORT BRANCH
P-408
WASHINGTON DC 20555

020555056741 2 99999
US NRC
NRR-TMI PROGRAM OFFICE
SECTION LEADER
ENVIR REVIEW
AR-5031
WASHINGTON DC 20555

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US NRC
OCM
TECH ASST TO COM ASSELSTINE
1149-H
WASHINGTON DC 20555

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US NRC
OCM
TECH ASST TO COM GILINSKY
1149-H
WASHINGTON DC 20555

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US NRC
NRR-DIV SAFETY TECHNOLOGY
DIRECTOR
278
WASHINGTON DC 20555

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US NRC
NRR-DIV SAFETY TECHNOLOGY
CHIEF
LICENSING GUIDANCE BRANCH
286
WASHINGTON DC 20555

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US NRC
NRR-DIV SAFETY TECHNOLOGY
ASSIST DIRECTOR
TECHNOLOGY
266
WASHINGTON DC 20555

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US NRC
NRR-DIV SAFETY TECHNOLOGY
CHIEF
REL & RISK ASSESSMENT BRANCH
216
WASHINGTON DC 20555

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US NRC
NRR-DIV SYSTEMS INTEGRATION
SECTION LEADER/RAD PROTECTION
RADIOLOGICAL ASSESSMENT BRANCH
P-712
WASHINGTON DC 20555

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US NRC
NRR-DIV SYSTEMS INTEGRATION
SPEC ASST-POLICY DEVELOPMENT
P-802
WASHINGTON DC 20555

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US NRC
NRR-DIV SYSTEMS INTEGRATION
DIVISION DIRECTOR
P-802B
WASHINGTON DC 20555

017057056750 1 99999
US NRC
NRR-TMI PROGRAM OFFICE
DEPUTY PROGRAM DIRECTOR
TMI SITE - PO BOX 311
MIDDLETOWN PA 17057

020555056749 1 99999
US NRC
NRR-TMI PROGRAM OFFICE
SECTION LEADER
SAFETY REVIEW
AR-5031
WASHINGTON DC 20555

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OCM
TECH ASST TO COM AHEARNE
1149-H
WASHINGTON DC 20555

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US NRC
OCM
TEC ASST TO CHAIRMAN PALLADINO
1159-H
WASHINGTON DC 20555

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US NRC

TECH ASST [illegible]
1149-H
WASHINGTON DC 20555

[illegible] ASST TO [illegible]
1159-H
WASHINGTON DC 20555

029678051339 1 99999
US NRC
OCONEE NPS
RESIDENT INSPECTOR
RT 2 BOX 610
SENECA SC 29678

008731005328 1 99999
US NRC
OYSTER CREEK NPS
RESIDENT INSPECTOR
PO BOX 445
FORKED RIVER NJ 08731

020555003928 1 99999
US NRC
PA
OFFICE DIRECTOR
3709 MNBB
WASHINGTON DC 20555

049043005342 1 99999
US NRC
PALISADES NPS
RESIDENT INSPECTOR
ROUTE 2 PO BOX 155
COVERT MI 49043

085001005071 1 99999
US NRC
PALO VERDE NPS
RESIDENT INSPECTOR
PO BOX 21324
PHOENIX AZ 85001

020555003916 1 99999
US NRC
PE
OFFICE DIRECTOR
H 1013
WASHINGTON DC 20555

017314005776 1 99999
US NRC
PEACH BOTTOM APS
RESIDENT INSPECTOR
PO BOX 399
DELTA PA 17314

044081005311 1 99999
US NRC
PERRY NPS
RESIDENT INSPECTOR
PARMLY AT CENTER RD
PERRY OH 44081

002360005629 1 99999
US NRC
PILGRIM NPS
RESIDENT INSPECTOR
PO BOX 867
PLYMOUTH MA 02360

054241005867 1 99999
US NRC
POINT BEACH NPS
RESIDENT INSPECTOR
6612 NUCLEAR ROAD
TWO RIVERS WI 54241

055089005910 1 99999
US NRC
PRAIRIE ISLAND NPS
RESIDENT INSPECTOR
ROUTE #2 BOX 500A
WELCH MN 55089

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US NRC
QUAD CITIES NPS
RESIDENT INSPECTOR
22712 206TH AVENUE-NORTH
CORDOVA IL 61242

095638006533 1 99999
US NRC
RANCHO SECO NPS
RESIDENT INSPECTOR
14410 TWIN CITIES RD
HERALD CA 95638

020555066490 1 99999
US NRC
RES-DIV OF ACCIDENT EVALUATION
DIRECTOR
1130 SS
WASHINGTON DC 20555

020555063080 1 99999
US NRC
RES-DIV OF ENGINEERING TECH
DIVISION DIRECTOR
1130 SS
WASHINGTON DC 20555

020555063085 1 99999
US NRC
RES-DIV OF ENGINEERING TECH
SECTION LEADER-STRUCTURAL ENGR
MECHANICAL/STRUCTURAL ENGR BR
1130 SS
WASHINGTON DC 20555

020555066611 1 99999
US NRC
RES-DIV OF ENGINEERING TECH
BRANCH CHIEF
MECHANICAL/STRUCTURAL ENGR BR
1130 SS
WASHINGTON DC 20555

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US NRC
RES-DIV OF FACILITY OPERATIONS
CHIEF
INSTRUMENTATION & CONTROL BR
1130 SS
WASHINGTON DC 20555

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US NRC
RES-DIV OF FACILITY OPERATIONS
CHIEF
HUMAN FACTORS BRANCH
1130 SS
WASHINGTON DC 20555

070775006453 1 99999
US NRC
RIVER BEND NPS
RESIDENT INSPECTOR
PO BOX 1051
ST FRANCISVILLE LA 70775

020555003995 1 99999
US NRC
RM/DMI
BRANCH CHIEF
7602 MNBB
WASHINGTON DC 20555

029550005820 1 99999
US NRC
ROBINSON NPS
RESIDENT INSPECTOR
ROUTE 5
BOX 266 1A
[illegible]

SALEM NPS
RESIDENT INSPECTOR
DRAWER I
HANCOCKS BRIDGE NJ 08038

US NRC
SAN ONOFRE NPS
RESIDENT INSPECTOR
PO BOX 4329
SAN CLEMENTE CA 92672

003874005155 1 99999
US NRC
SEABROOK NPS
RESIDENT INSPECTOR
PO BOX 700
SEABROOK NH 03874

020555003921 1 99999
US NRC
SECY
SECRETARY
H1149
WASHINGTON DC 20555

037379006309 1 99999
US NRC
SEQUOYAH NPS
RESIDENT INSPECTOR
2600 IGOU FERRY ROAD
SODDY DAISY TN 37379

011778005092 1 99999
US NRC
SHOREHAM NPS
RESIDENT INSPECTOR
PO BOX B
ROCKY POINT NY 11778

077414005191 1 99999
US NRC
SOUTH TEXAS NPS
RESIDENT INSPECTOR
PO BOX 910
BAY CITY TX 77414

020555056728 1 99999
US NRC
SP STATE & LICENSEE RELATIONS
ASSISTANT DIRECTOR
AR-2016
WASHINGTON DC 20555

020555003992 1 99999
US NRC
SP STATE AGREEMENTS PROGRAM
ASSISTANT DIRECTOR
AR-5233
WASHINGTON DC 20555

033457005156 1 99999
US NRC
ST LUCIE NPS
RESIDENT INSPECTOR
7900 S A1A
JENSEN BEACH FL 33457

033457056920 1 99999
US NRC
ST LUCIE NPS
RESIDENT INSPECTOR
7900 S HWY A1A
JENSEN BEACH FL 33457

029065005011 1 99999
US NRC
SUMMER NPS
RESIDENT INSPECTOR
PO BOX 64 ROUTE 1
JENKINSVILLE SC 29065

023883005667 1 99999
US NRC
SURRY NPS
RESIDENT INSPECTOR
ROUTE 1
PO BOX 166
SURRY VA 23883

018655001590 1 99999
US NRC
SUSQUEHANNA NPS
RESIDENT INSPECTOR
PO BOX 52
SHICKSHINNY PA 18655

017057064245 1 99999
US NRC
TMI SITE
DEPUTY PROGRAM DIRECTOR
PO BOX 311
MIDDLETOWN PA 17057

017057007106 1 99999
US NRC
TMI-1
SR RESIDENT INSPECTOR
PO BOX 311
MIDDLETOWN PA 17057

020555064864 1 99999
US NRC
TRAINING & ADM STAFF
DIRECTOR
EW359
WASHINGTON DC 20555

097501006344 1 99999
US NRC
TROJAN NPS
RESIDENT INSPECTOR
PO BOX 0
RAINIER OR 97501

033030005678 1 99999
US NRC
TURKEY POINT NPS
RESIDENT INSPECTOR
PO BOX 1204
HOMESTED FL 33030

005453005898 1 99999
US NRC
VERMONT YANKEE NPS
RESIDENT INSPECTOR
PO BOX 176
VERNON VT 05453

030830080044 1 99999
US NRC
VOGTLE NPS
RESIDENT INSPECTOR
PO BOX 572
WAYNESBORO GA 30830

070066005312 1 99999
US NRC
WATERFORD NPS
RESIDENT INSPECTOR
PO BOX 822
KILLONA LA 70066

037361053194 1 99999
US NRC
WATTS BAR NPS
RESIDENT INSPECTOR
RT 2 BOX 300

037831005099 1 99999
US NRC
WATTS BAR NPS
RESIDENT INSPECTOR
ROUTE 2 BOX 300

US NRC
VOGTLE NPS
RESIDENT INSPECTOR
PO BOX 572
WAYNESBORO GA 30830

US NRC
WATERFORD NPS
RESIDENT INSPECTOR
PO BOX 822
KILLONA LA 70066

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US NRC
WATTS BAR NPS
RESIDENT INSPECTOR
RT 2 BOX 300
SPRING CITY TN 37381

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US NRC
WATTS BAR NPS
RESIDENT INSPECTOR
ROUTE 2 BOX 300
SPRING CITY TN 37831

098541006223 1 99999
US NRC
WPPS 3/5
RESIDENT INSPECTOR
PO BOX 545
ELMA WA 98541

099352006274 1 99999
US NRC
WPPSS NPS
RESIDENT INSPECTOR
PO BOX 69
RICHLAND WA 99352

099352006325 1 99999
US NRC
WPPSS-2 NPS
RESIDENT INSPECTOR
PO BOX 69
RICHLAND WA 99352

001350005371 1 99999
US NRC
YANKEE ROWE NPS
RESIDENT INSPECTOR
PO BOX 28
MONROE BRIDGE MA 01350

045153054715 1 99999
US NRC
ZIMMER NPS
RESIDENT INSPECTOR
BOX 2021 US RT 52
MOSCOW OH 45153

020555003884 1 99999
US NRC
NRR-CLINCH RIVER BREEDER REAC
CARDIS L ALLEN
AR5003
WASHINGTON DC 20555

020555066453 1 99999
US NRC
RES
MARKLEY L AU
HUMAN FACTORS BR
1130 SS
WASHINGTON DC 20555

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US NRC
ASLBP
CHARLES BECHHOEFER
EW439
WASHINGTON DC 20555

020555007027 1 99999
US NRC
ATOMIC SAFETY & LICENSING BD
CHARLES BECHHOEFER
ESQ
WASHINGTON DC 20555

020555006129 1 99999
US NRC
ASLBP
PETER B BLOCH
CHAIRMAN
WASHINGTON DC 20555

020555005768 1 99999
US NRC
ASLBP
ELIZABETH S BOWERS
CHAIRMAN
WASHINGTON DC 20555

020555000552 1 99999
US NRC
ASLBP
LAWRENCE BRENNER
E-W 450
WASHINGTON DC 20555

020555003845 1 99999
US NRC
ASLBP
GLENN O BRIGHT
EW439
WASHINGTON DC 20555

020555003839 1 99999
US NRC
ASLAP
JOHN H BUCK
VICE CHAIRMAN
EW529
WASHINGTON DC 20555

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US NRC
ASLAP
JOHN H BUCK
EW-529
WASHINGTON DC 20555

020555029097 1 99999
US NRC
ASLAP
JOHN CHO
COUNSEL
EW-529
WASHINGTON DC 20555

045153005318 1 99999
US NRC
ZIMMER NPS
WALDMAN CHRISTIANSON
US ROUTE 52 RFD 1
BOX 2021
MOSCOW OH 45153

020555000017 1 99999
US NRC
ASLBP
RICHARD F COLE
WASHINGTON DC 20555

020555007774 1 99999
US NRC
ASLBP
RICHARD COLE

020555065759 1 99999
US NRC
AEOD
JOHN L CROOKS

[illegible] 1 [illegible]
US NRC
ASLAP
GARY J EDLES
CHAIRMAN
EW-529
WASHINGTON DC 20555

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US NRC
BYRON NPS
WILLIAM FOURNEY
RESIDENT INSPECTOR
4448 GERMAN CHURCH ROAD
BYRON IL 61010

020555007110 1 99999
US NRC
ASLAP
REGINALD L GOTCHY
EW-529
WASHINGTON DC 20555

020555005341 1 99999
US NRC
ASLBP
HERBERT GROSSMAN
CHAIRMAN
WASHINGTON DC 20555

020555007028 1 99999
US NRC
ATOMIC SAFETY & LICENSING BD
JERRY HARBOUR
ESC
WASHINGTON 20555

047147005349 1 99999
US NRC
MARBLE HILL NPS
J J HARRISON
RESIDENT INSPECTOR
3738 MARBLE HILL ROAD
NABB IN 47147

037411066452 1 99999
US NRC
REACTOR TRAINING CENTER
IRENE HAYCRAFT
OSBORNE OFFICE CTR SUITE 200
RTC
CHATTANOOGA TN 37411

020555005819 1 99999
US NRC
ASLBP
W R JOHNSON
EW-529
WASHINGTON DC 20555

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US NRC
ADM DSB MICROGRAPHICS
PAUL JONES
042
WASHINGTON DC 20555

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US NRC
IP
WILLIAM S LAVINE
403A EWS
WASHINGTON DC 20555

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US NRC
ASLBP
G A LINENBERGER JR
EW439
WASHINGTON DC 20555

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ASLAP
[illegible]

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US NRC
ASLBP
CHARLES J FITTI
ASSISTANT EXECUTIVE SECY
EW439
WASHINGTON DC 20555

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US NRC
ASLBP
JOHN H FRYE
CHAIRMAN
WASHINGTON DC 20555

020555051908 1 99999
US NRC
IE/PSB
L GREHER
359-EW-W
WASHINGTON DC 20555

020555038234 1 99999
US NRC
ASLBP
HERBERT GROSSMAN
EW439
WASHINGTON DC 20555

020555056993 1 99999
US NRC
ASLBP
JAMES HARD
EW-439
WASHINGTON DC 20555

037411006528 1 99999
US NRC
REACTOR TRAINING CENTER
OLIVER HAVENS
OSBORNE OFFICE CENTER
SUITE 200
CHATTANOOGA TN 37411

020555006247 1 99999
US NRC
ACRS
ELPIDIO G IGNE
H1016
WASHINGTON DC 20555

020555009812 1 99999
US NRC
ASLAP
W REED JOHNSON
ADMINISTRATIVE JUDGE-TECH
EW529
WASHINGTON DC 20555

020555006035 1 99999
US NRC
ASLAP
CHRISTINE N KOHL
WASHINGTON DC 20555

020555007810 1 99999
US NRC
ASLBP
ROBERT M LAZO
EW439
WASHINGTON DC 20555

048166006227 1 99999
US NRC
FERMI NPS
BRUCE LITTLE
RESIDENT INSPECTOR
6450 W DIXIE HIGHWAY
NEWPORT MI 48166

020555000542 1 99999
US NRC
ATOMIC SAFETY & LICENSING [illegible]

PECTOR
HILL ROAD
IN 47147

1 99999

NING CENTER
AYCRAFT
CE CTR SUITE 200
TN 37411

1 99999

DC 20555

1 99999

ROGRAPHICS

DC 20555

0 1 99999

AVINE

DC 20555

9 1 99999

RGER JR

DC 20555

8 1 99999

EBKE

DC 20555

4 1 99999

CRELESS

DC 20555

1 1 99999

MILLER

DC 20555

SUITE 200
CHATTANOOGA TN 37411

020555006247 1 99999
US NRC
ACRS
ELPIDIO G IGNE
H1016
WASHINGTON DC 20555

020555009812 1 99999
US NRC
ASLAP
W REED JOHNSON
ADMINISTRATIVE JUDGE-TECH
EW529
WASHINGTON DC 20555

020555006035 1 99999
US NRC
ASLAP
CHRISTINE N KOHL
WASHINGTON DC 20555

020555007810 1 99999
US NRC
ASLBP
ROBERT M LAZO
EW439
WASHINGTON DC 20555

048166006227 1 99999
US NRC
FERMI NPS
BRUCE LITTLE
RESIDENT INSPECTOR
6450 W DIXIE HIGHWAY
NEWPORT MI 48166

020555000542 1 99999
US NRC
ATOMIC SAFETY & LICENSING BD
EMMETH A LUEBKE
WASHINGTON DC 20555

020555030757 1 99999
US NRC
ACRS SUBCOMM ON WASTE MANAGE
R E MCKINLEY
H-1016
WASHINGTON DC 20555

020555006399 1 99999
US NRC
ASLAP
THOMAS S MOORE
WASHINGTON DC 20555

[illegible]
EW439
WASHINGTON DC 20555

[illegible]
WASHINGTON DC 20555

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US NRC
ASLAP
THOMAS S MOORE
ADMINISTRATIVE JUDGE-LEGAL
EW-529
WASHINGTON DC 20555

020555007785 1 99999
US NRC
ASLBP
DORIS M MORAN
EW439
WASHINGTON DC 20555

065077005073 1 99999
US NRC
CALLAWAY NPS
JOHN NEISLER
RESIDENT INSPECTOR
RR # 1
STEEDMAN MO 65077

029678005805 1 99999
US NRC
WILLIAM T ORDERS
SR RESIDENT INSPECTOR
ROUTE 2
BOX 610
SENECA SC 29678

020555006548 1 99999
US NRC
ASLBP
OSCAR H PARIS
WASHINGTON DC 20555

017057006382 1 99999
US NRC
TMI SITE
LENNIE PROUGH
PO BOX 311
MIDDLETOWN PA 17057

020555007109 1 99999
US NRC
ASLAP
LAWRENCE P QUARLES
EW-529
WASHINGTON DC 20555

020555081463 2 99999
US NRC
SECY
JEAN RATHJE
H ST LOBBY
WASHINGTON DC 20555

031513005792 1 99999
US NRC
R F ROGERS
ROUTE 1
PO BOX 279
BAXLEY GA 31513

020555007091 1 99999
US NRC
ASLAP
ALAN S ROSENTHAL
EW-529
WASHINGTON DC 20555

020555007090 1 99999
US NRC
ASLAP
RICHARD S SALTZMAN
EW-529
WASHINGTON DC 20555

019406006409 1 99999
US NRC
G SANBORN
631 PARK AVENUE
KING OF PRUSSIA PA 19406

030830006492 1 99999
US NRC
VOGTLE NPS
WILLIAM S SANDERS
RESIDENT INSPECTOR
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WAYNESBORO GA 30830

020555029190 1 99999
US NRC
ACRS
RICHARD SAVIO
1016 HST
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020555081604 1 99999
US NRC
EDO
WALT SCHWINK
PROGRAM MANAGER
6106 MNBB
WASHINGTON DC 20555

020555063102 1 99999
US NRC
ADM TIDC
STEVE SCOTT
LAND-1210
WASHINGTON DC 20555

020555081107 1 99999
US NRC
AEOD
K V SEYFRIT
263A EWS
WASHINGTON DC 20555

020555000023 1 99999
US NRC
ASLBP
FREDERICK J SHON
E-W 450
WASHINGTON DC 20555

020555009718 1 99999
US NRC
ASLBP
FREDERICK SHON
EW439
WASHINGTON DC 20555

020555005114 1 99999
US NRC
ASLBP
IVAN W SMITH
ESQ
EW-450
WASHINGTON DC 20555

020555007804 1 99999
US NRC
ASLBP
IVAN W SMITH
EW439
WASHINGTON DC 20555

076043005008 1 99999
US NRC
COMANCHE PEAK NPS
ROBERT G TAYLOR
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020555007804 1 99999
US NRC
ASLBP
IVAN W SMITH
EW439
WASHINGTON DC 20555

066839005350 1 99999
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WOLF CREEK NPS
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020555028206 1 99999
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SP
SUE WEISSBERG
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AR5037
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020555000144 1 99999
US NRC
ASLBP
SHELDON J WOLFE
ESQUIRE
E-W 450
WASHINGTON DC 20555

019406003570 20 99999
US NRC REGION I
REGIONAL ADMINISTRATOR
631 PARK AVENUE
KING OF PRUSSIA PA 19406

019406006340 1 99999
US NRC REGION I
KARL ABRAHAM
PUBLIC AFFAIRS OFFICER
631 PARK AVE
KING OF PRUSSIA PA 19406

019406081630 1 99999
US NRC REGION I
NYU CAMPBELL
STATE AGREEMENTS OFFICER
631 PARK AVE
KING OF PRUSSIA PA 19406

030303056497 1 99999
US NRC REGION II
REGIONAL STATE LIAISON OFFICER
101 MARIETTA STREET
SUITE 3100
ATLANTA GA 30303

060137006385 1 99999
US NRC REGION III
REGIONAL ADMINISTRATOR
799 ROOSEVELT ROAD
GLEN ELLYN IL 60137

076011006384 1 99999
US NRC REGION IV
REGIONAL ADMINISTRATOR
611 RYAN PLAZA DRIVE
SUITE 1000
ARLINGTON TX 76011

076012003589 1 99999
US NRC REGION IV
REACTOR PROJECT BRANCH 1
CHIEF
611 RYAN PLAZA DR
STE 1000
ARLINGTON TX 76012

076043005008 1 99999
US NRC
COMANCHE PEAK NPS
ROBERT G TAYLOR
RESIDENT INSPECTOR
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GLEN ROSE TX 76043

015077006496 1 99999
US NRC
GLENN WALTON
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020555005402 1 99999
US NRC
ASLAP
CHRIS WILLIAMS
EW529
WASHINGTON DC 20555

020555007803 1 99999
US NRC
ASLBP
SHELDON J WOLFE
EW439
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019406006387 1 99999
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019406026175 1 99999
US NRC REGION I
FAITH BRENNEMAN
REGIONAL STATE LIAISON OFFICER
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030303006386 1 99999
US NRC REGION II
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101 MARIETTA STREET
SUITE 3100
ATLANTA GA 30303

030303081629 1 99999
US NRC REGION II
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101 MARIETTA ST SUITE 3100
ATLANTA GA 30303

060137064870 1 99999
US NRC REGION III
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799 ROOSEVELT RD
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076011056496 2 99999
US NRC REGION IV
REGIONAL STATE LIAISON OFFICER
611 RYAN PLAZA DRIVE
SUITE 1000
ARLINGTON TX 76011

076011007009 1 99999
US NRC REGION IV
JOHN T COLLINS
611 RYAN PLAZA
SUITE 1000
ARLINGTON TX 76011

[illegible] 1 99999

076012081628 1 99999

US NRC REGION III
REGIONAL ADMINISTRATOR
799 ROOSEVELT ROAD
GLEN ELLYN IL 60137

US NRC REGION III
REGIONAL STATE LIAISON OFFICER
799 ROOSEVELT RD
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US NRC REGION IV
REGIONAL ADMINISTRATOR
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SUITE 1000
ARLINGTON TX 76011

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US NRC REGION IV
REGIONAL STATE LIAISON OFFICER
611 RYAN PLAZA DRIVE
SUITE 1000
ARLINGTON TX 76011

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US NRC REGION IV
REACTOR PROJECT BRANCH 1
CHIEF
611 RYAN PLAZA DR
STE 1000
ARLINGTON TX 76012

076011007009 1 99999
US NRC REGION IV
JOHN T COLLINS
611 RYAN PLAZA
SUITE 1000
ARLINGTON TX 76011

076012081627 1 99999
US NRC REGION IV
ROBERT DODA
STATE AGREEMENTS OFFICER
611 RYAN PLAZA DR SUITE 1000
ARLINGTON TX 76012

076012081628 1 99999
US NRC REGION IV
RALPH S HEYEN
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094596006383 1 99999
US NRC REGION V
REGIONAL ADMINISTRATOR
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WALNUT CREEK CA 94596

094596056507 1 99999
US NRC REGION V
REGIONAL STATE LIAISON OFFICER
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094596006567 1 99999
US NRC REGION V
JAMES HANCHETT
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094596081626 1 99999
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068102079365 1 99999
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007081064448 1 99999
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008225005638 1 99999
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084663053128 1 99999
VALTEK INC
G C CLARK
QUALITY ASSURANCE MNGR
MOUNTAIN SPRINGS PARKWAY
SPRINGVILLE UT 84663

0CANAD053129 1 99999
VELAN INC
A K VELAN
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H4M-1T6
MONTREAL QUEBEC CANADA

005495053130 1 99999
VELAN VALVE CORP
P J STRELCZYK
PLANT MANAGER
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WILLISTON VT 05495

005602005891 1 99999
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005602006557 1 99999
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005602011804 1 99999
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005602005895 1 99999
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005301005896 1 99999
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001701005883 1 99999
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005354064682 1 99999
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005301005886 1 99999
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005301007004 1 99999
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044104053725 1 99999
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001950065238 1 99999
VIKEM INDUSTRIES INC
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010511005686 1 99999
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010520005722 1 99999
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023219005670 1 99999
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002173066783 1 99999
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020874066211 1 99999
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061820001585 1 99999
WAALER EVANS & GORDON
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010007057894 1 99999
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019481053133 1 99999
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T R STRONG
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RADIATION CONTROL SEC
TERRY STRONG
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098504005964 1 99999
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098504080853 1 99999
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099352006466 1 99999
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099352081578 1 99999
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098541006224 1 99999
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098541079424 1 99999
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099352006469 1 99999
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099352006225 1 99999
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099352008688 1 99999
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098541064602 1 99999
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099352079423 1 99999
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098541079422 1 99999
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098541064621 1 99999
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099352081530 1 99999
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099352006468 1 99999
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099352006407 1 99999
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099352081529 1 99999
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099352057188 1 99999
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099352064626 1 99999
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099219053134 1 99999
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025305005701 1 99999
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CLARK HANSBARGER
STATE DIRECTOR OF HEALTH
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025305004355 1 99999
WEST VIRGINIA OFC OF EMER SVC
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ROOM EB80
CHARLESTON WV 25305

025305005704 1 99999
WEST VIRGINIA OFFICE OF GOV
CHARLESTON WV 25305

026506081876 1 99999
WEST VIRGINIA PUB INT RES GRP
W VIRGINIA UNIVERSITY
MOUNTAINCLAIR SUW
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019103028068 1 99999
WESTEC SERVICES INC
APPLIED SCIENCES
IAN H SARGENT
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073005005182 1 99999
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094124053135 1 99999
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007662079039 1 99999
WESTERN STRESS INC
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014845053148 1 99999
WESTINGHOUSE ELECTRIC CORP
INDUST & GOVT TUBE DIV
QUALITY ASSURANCE MANAGER
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015230006610 1 99999
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015220001629 1 99999
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032576053722 1 99999
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060099081500 1 99999
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015230065223 1 99999
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014240053138 1 99999
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WESTINGHOUSE ELECTRIC CORP
ELECTRONIC COMPONENTS DIV
WALLACE GILLIES
WESTINGHOUSE CIRCLE
111
HORSEHEADS NY 14845

015230006498 1 99999
WESTINGHOUSE ELECTRIC CORP
POWER SYSTEMS
J C HOEBEL
PO BOX 355
PITTSBURGH PA 15230

014240053147 1 99999
WESTINGHOUSE ELECTRIC CORP
POWER ELECTRONICS & DRIVE SYS
R G HOLDBROOK
GENERAL MANAGER
PO BOX 225
BUFFALO NY 14240

014240053723 1 99999
WESTINGHOUSE ELECTRIC CORP
MEDIUM MOTOR & GEARING DIV
D L HOOVER
GENERAL MANAGER
PO BOX 225
BUFFALO NY 14240

037830081131 1 99999
WESTINGHOUSE ELECTRIC CORP
ANGUS KIMMINS
PO BOX W
OAK RIDGE TN 37830

015230006522 1 99999
WESTINGHOUSE ELECTRIC CORP
ATOMIC POWER DISTRIBUTION
WILLIAM KORTIER
POST OFFICE BOX 355
PITTSBURGH PA 15230

015230001379 1 99999
WESTINGHOUSE ELECTRIC CORP
W LUCE
PO BOX 355
PITTSBURGH PA 15230

015663053142 1 99999
WESTINGHOUSE ELECTRIC CORP
ARO-WALTZ MILL SITE
A G MARTINI
QUALITY ASSURANCE MANAGER

032596053144 1 99999
WESTINGHOUSE ELECTRIC CORP
PENSACOLA DIVISION
T D MILLER
QUALITY ASSURANCE MANAGER

1445 [illegible] AVE
SAN FRANCISCO CA 94124

014845053148 1 99999
WESTINGHOUSE ELECTRIC CORP
INDUST & GOVT TUBE DIV
QUALITY ASSURANCE MANAGER
WESTINGHOUSE CIRCLE
HORSEHEADS NY 14845

015220001629 1 99999
WESTINGHOUSE ELECTRIC CORP
WNI LIBRARY
5 PARKWAY CENTER
PITTSBURGH PA 15220

060099081500 1 99999
WESTINGHOUSE ELECTRIC CORP
INSTRUCT & TRAINING REACTOR
C BACH
MANAGER
505 SHILOH BLVD
ZION IL 60099

015230038027 1 99999
WESTINGHOUSE ELECTRIC CORP
NUCLEAR SAFETY DEPT
R D BURCH
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014240053138 1 99999
WESTINGHOUSE ELECTRIC CORP
LARGE MOTOR DIVISION
T W CONWAY
MANAGER QUALITY ASSURANCE
4454 GENESEE STREET
BUFFALO NY 14240

015230006311 1 99999
WESTINGHOUSE ELECTRIC CORP
BOB FAAS
PO BOX 355
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015230001222 1 99999
WESTINGHOUSE ELECTRIC CORP
W C GANGLOFF
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PITTSBURGH PA 15230

015230006498 1 99999
WESTINGHOUSE ELECTRIC CORP
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014240053723 1 99999
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015230006610 1 99999
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LIBRARY MANAGER
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032576053722 1 99999
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C R ADKINS
PROJECT MANAGER BRCP
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015230065223 1 99999
WESTINGHOUSE ELECTRIC CORP
W H BAMFORD
PO BOX 355
PITTSBURGH PA 15230

015230081612 1 99999
WESTINGHOUSE ELECTRIC CORP
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H A CLAWSON
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085282053140 1 99999
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TEMPE AZ 85282

015230006709 1 99999
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014845001535 1 99999
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032596053144 1 99999
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015663053142 1 99999
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020014056984 1 99999
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CHESWICK PA 15024

015230006516 1 99999
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021030081454 1 99999
WESTINGHOUSE ELECTRIC CORP
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W PATALON
GENERAL MANAGER
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HUNT VALLEY MD 21030

015230081407 1 99999
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015230066433 1 99999
WESTINGHOUSE ELECTRIC CORP
MONROEVILLE NUCLEAR CENTER

015230064694 1 99999
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INFO & RECORDS SYSTEMS

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015230006311 1 99999
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014240053723 1 99999
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015230006522 1 99999
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015146001625 1 99999
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015230006709 1 99999
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014240053147 1 99999
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WESTINGHOUSE ELECTRIC CORP
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PITTSBURGH PA 15230

032596053144 1 99999
WESTINGHOUSE ELECTRIC CORP
PENSACOLA DIVISION
T D MILLER
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015230079598 1 99999
WESTINGHOUSE ELECTRIC CORP
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PITTSBURGH PA 15230

015230006516 1 99999
WESTINGHOUSE ELECTRIC CORP
A T PARKER
PO BOX 355
PITTSBURGH PA 15230

015230081407 1 99999
WESTINGHOUSE ELECTRIC CORP
WATER REACTOR DIV
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MANAGER-NUCLEAR POWER DEPT
PO BOX 355
PITTSBURGH PA 15230

015230064694 1 99999
WESTINGHOUSE ELECTRIC CORP
INFO & RECORDS SYSTEMS
L M RICHMAN
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015230064758 1 99999
WESTINGHOUSE ELECTRIC CORP
NUCLEAR ENERGY
D E RUSSELL
MGR RECORDS & FILES OP
PO BOX 355

015024053139 1 99999
WESTINGHOUSE ELECTRIC CORP
ELECTRO MECHANICAL DIVISION
C E GREN
PRODUCT ASSURANCE MANAGER
CHESWICK AVENUE
CHESWICK PA 15024

015230036516 1 99999
WESTINGHOUSE ELECTRIC CORP
A T PARKER
PO BOX 355
PITTSBURGH PA 15230

021030081454 1 99999
WESTINGHOUSE ELECTRIC CORP
NUCLEAR INSTRUMENTATION & CONT
W PATALON
GENERAL MANAGER
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HUNT VALLEY MD 21030

015230081407 1 99999
WESTINGHOUSE ELECTRIC CORP
WATER REACTOR DIV
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MANAGER-NUCLEAR POWER DEPT
PO BOX 355
PITTSBURGH PA 15230

015230066433 1 99999
WESTINGHOUSE ELECTRIC CORP
MONROEVILLE NUCLEAR CENTER
D H RAWLINS
PO BOX 355
BAY 417A
PITTSBURGH PA 15230

015230064694 1 99999
WESTINGHOUSE ELECTRIC CORP
INFO & RECORDS SYSTEMS
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015146001625 1 99999
WESTINGHOUSE ELECTRIC CORP
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015230064758 1 99999
WESTINGHOUSE ELECTRIC CORP
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MGR RECORDS & FILES OP
PO BOX 355
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015230081499 1 99999
WESTINGHOUSE ELECTRIC CORP
LICENSING SAFEGUARDS & SAFETY
A T SABO
DIRECTOR
PO BOX 355
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033065055288 1 99999
WESTINGHOUSE ELECTRIC CORP
RELAY INSTRUMENT DIV
W J SCHMIDT
MANAGER QUALITY ASSURANCE
4300 CORAL RIDGE DRIVE
CORAL SPRINGS FL 33065

015238053145 1 99999
WESTINGHOUSE ELECTRIC CORP
INDUSTRY ELECTRONICS DIVISION
T E SMITH
NCEP MGR
200 BETA DRIVE
PITTSBURGH PA 15238

015112053150 1 99999
WESTINGHOUSE ELECTRIC CORP
POWER SYS SWTCHGEAR DIV
H I STAHR
MANAGER QUALITY ASSURANCE
700 BRADDOCK AVE
E PITTSBURGH PA 15112

015230001417 1 99999
WESTINGHOUSE ELECTRIC CORP
POWER SYSTEMS DIVISION
F J THOGOOD
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015230005491 1 99999
WESTINGHOUSE ELECTRIC CORP
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099352053149 1 99999
WESTINGHOUSE HANFORD CO
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RICHLAND WA 99352

099352077527 1 99999
WESTINGHOUSE HANFORD CO
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RICHLAND WA 99352

099352055108 1 99999
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W-A-128
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099352039954 1 99999
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045153026371 1 99999
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004578006480 1 99999
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WISCASSET ME 04578

004578007115 1 99999
WISCASSET PUBLIC LIBRARY
ASSOCIATION
HIGH STREET
WISCASSET ME 04578

053702036123 1 99999
WISCONSIN DEPT OF JUSTICE
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MADISON WI 53702

053707080976 1 99999
WISCONSIN DILHR SAFETY & BLDG

053702064176 1 99999
WISCONSIN DIV OF STATE ENERGY

015024053139 1 99999
WESTINGHOUSE ELECTRIC CORP
ELECTRO MECHANICAL DIVISION
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015230036516 1 99999
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021030081454 1 99999
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015230081407 1 99999
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015230064694 1 99999
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015230081499 1 99999
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033065055288 1 99999
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015238053145 1 99999
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T E SMITH
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015112053150 1 99999
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021030081454 1 99999
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NCCP MGR
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RICHLAND WA 99352

099352077527 1 99999
WESTINGHOUSE HANFORD CO
J H MACLAREN
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WESTINGHOUSE HANFORD CO
ROBERT L MEADOR
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RICHLAND WA 99352

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WESTINGHOUSE HANFORD CO
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A R SCHADE
PO BOX 1970
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WILLIAM ZIMMER NUCLEAR
TOM JOHNSON
US ROUTE 52 PO BOX 201
MOSCOW OH 45153

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WISCONSIN DEPT OF JUSTICE
PATRICK W WALSH
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WESTINGHOUSE HANFORD CO
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053707080976 1 99999
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053201006250 1 99999
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WISCONSIN ELECTRIC POWER CO
POINT BEACH NUCLEAR PWR STATN
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6610 NUCLEAR ROAD
TWO RIVERS WI 54241

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WISCONSIN ELECTRIC POWER CO
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J J ZACH
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POSTAL ROUTE 3
TWO RIVERS WI 54241

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053703005866 1 99999
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053702005365 1 99999
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053707081507 1 99999
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WISCONSIN PUBLIC SERVICE CORP
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GREEN BAY WI 54305

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PLANT MANAGER
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WORCESTER POLYTECHNIC INST
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SPENCER C RELYEA
2001 BRYAN TOWER
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RADIOLOGICAL HLTH SVC
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CHIEF
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YADKIN RIVER COMMITTEE
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YANKEE ATOMIC ELECTRIC CO
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1671 WORCESTER ST
FRAMINGHAM MA 01701

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H AUTIO
PLANT SUPT
STAR ROUTE
ROWE MA 01367

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FARMINGHAM MA 01701

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YANKEE ATOMIC ELECTRIC CO
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001701081046 1 99999
YANKEE ATOMIC ELECTRIC CO
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FRAMINGHAM MA 01701

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YANKEE ATOMIC ELECTRIC CO
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001581007024 1 99999
YANKEE ATOMIC ELECTRIC CO
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VICE PRESIDENT OF ENGINEERING
20 TURNPIKE ROAD
WESTBORO MA 01581

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YARWAY CORP
W A VOLGER
MANAGER QUALITY SYSTEMS
BLUE BELL PA 19422

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YORK ELECTRO PANEL CONTROL CO
J MYERS
QA MANAGER
YORK COUNTY INDUSTRIES PARK
POB 1702
PA 17405

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[illegible] QUALITY SYSTEMS                QA MANAGER
BLUE BELL               PA  19422     YORK COUNTY INDUSTRIES PARK
                                      POB 1702
                                      YORK                    PA  17405

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P O BOX 2401                          PO BOX 3158
YOUNGSTOWN              OH  44509     TULSA                   OK  74101


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50 STANIFORD STREET                   MAYOR
BOSTON                  MA  02114     ZION                    IL  60099


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L R MARQUARDT
MARKET MGR SERV HEALTHCARE
3M CENTER BUILDING 220-9E
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December 3, 1982

MEMORANDUM FOR: Chairman Palladino

FROM: William J. Dircks
Executive Director
for Operations

SUBJECT: ZIMMER MONTHLY STATUS REPORT
TO COMMISSION

Enclosed is the Zimmer monthly status report for October.

(Signed) William J. Dircks

William J. Dircks
Executive Director
for Operations

Enclosure:
As stated

cc: Commissioner Gilinsky
Commissioner Ahearne
Commissioner Roberts
Commissioner Asselstine
SECY
OGC
OPE
OIA
OI
OCA
PDR

Distribution:
Dircks
Roe
Rehm
Stello
Keppler
DeYoung
Denton
Res. Insp., RIII
EDO r/f

8212080595 821203
PDR ADOCK 05000358
R PDR

(Transmitted via 5520 from RIII to OEDO 12/1/82)
#COM31

OFFICE	RIII	RIII	EDO				
SURNAME	ESchweibinz	JKeppler	WJDircks				
DATE	12/1/82	12/1/82	12/3/82				

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

ZIMMER MONTHLY
STATUS REPORT

October 1982

Table of Contents

A. Summary of the Project for the Month

Meetings were held on October 27, 1982 with senior NRC staff representatives and on October 28, 1982 with the Commission to discuss the status of the Zimmer project and problems being identified. (These meetings precipitated followup Commission meetings in November and led to the Commission decision to issue an "Order to Show Cause and Order Immediately Suspending Construction" on November 12, 1982 (Attachment III).)

Region III inspection efforts at the Zimmer facility during the month of October were concentrated on the activities of Catalytic, Inc. (CI), the CG&E overview of CI, review of nonconformance reports, and the routine monitoring and inspection of other ongoing activities including the QCP.

The resident inspector conducted a tour of the Zimmer site for the U.S. Attorney, Southern District of Ohio, representatives of his staff, and a representative of the F.B.I.

A management meeting was conducted with the licensee on October 19, 1982, (open meeting) regarding the NRC inspection findings from a Catalytic, Inc. inspection. Details are provided in section C.1 of this report.

The National Board of Boiler and Pressure Vessel Inspectors is continuing inspection efforts onsite. The National Board issued interim reports on May 12, July 1, (August 6 supplement), and September 30, 1982. The findings of the National Board are consistent with and similar to NRC findings. The licensee responded on August 5 and 30, October 20 and 29, 1982, and plans to provide a bi-weekly status report.

In accordance with 10 CFR 50.55(e), the licensee reported the following potential construction deficiencies:

On October 13, 1982, design changes made to the fire protection system piping in the cable spreading room in 1979 to upgrade the piping to have seismic supports was done with no evidence of quality assurance of the modifications or of Sargent & Lundy calculations for the modifications.

On October 26, 1982, the licensee determined that Sargent & Lundy (S&L) dynamic analysis of small bore piping at Zimmer was questionable. The stress analyses are being reviewed by S&L and the licensee. A similar report was submitted concerning Clinton.

On October 27, 1982, the licensee reported a potential problem relating to division separation between non-essential cables and bundling with essential cables during cable pulling and termination.

On October 29, 1982, the licensee determined that fire dampers manufactured by Air-Balance, Inc. have a fusable link held by a "J Hook" and that one prevented the fire damper from closing when the damper was actuated.

On October 29, 1982, the licensee determined that pipe support installation procedures did not contain seismic clearance criteria between pipe supports and cable trays or conduit and associated supports as required by the specification. This may indicate a lack of proper interface on design.

Region III is continuing evaluation of the licensee's responses to the investigation report 81-13 and civil penalty, and some progress was made during the month because of a temporary personnel assignment to this project.

Since the July Monthly Status Report was prepared, two petitions, both dated August 20, 1982, have been received from the Government Accountability Project (GAP). The first is for reconsideration of the Commission's Order of July 30, 1982. The staff has considered this petition and made it's recommendations to the Commission. The second petition is to suspend construction of the Zimmer Station. On September 24, 1982, the RIII Regional Administrator issued a "Demand for Information" pursuant to the Commission's authority under Section 182 of the Atomic Energy Act and 10 CFR 50.54(f) of the Commission's regulations. This "Demand for Information" requires the licensee to admit or deny each of the allegations applicable to the licensee's and its principal contractor's or subcontractor's performance contained in paragraphs 19 through 273 of the petition. If the allegations are not admitted, they must explain the basis for not admitting the allegations. In addition, they must identify the manner in which the Quality Confirmation Program (QCP) addresses the type of existing or potential quality assurance or construction deficiencies and problems identified in each of the above allegations. If the QCP does not address such deficiencies or problems, CG&E must describe the manner in which they will ensure such deficiencies or problems are corrected. They have until December 31, 1982 to respond. On October 18, 1982, GAP issued a supplement to their August 20, 1982 petition. The staff is reviewing this supplement.

NRC Region III was advised of allegations that relevant documentation on the welders at the Zimmer site was prepared for, or reviewed at, a meeting between Cincinnati Gas and Electric Company (CG&E) and H. J. Kaiser (HJK) held on July 8, 1982, but that such documentation was not discussed with, or made available to, Region III at a meeting on this subject held on July 9, 1982, between CG&E, HJK and Region III. Substantial documentation was made available to Region III in connection with the July 9 meeting, but additional relevant documentation was allegedly not made available. The matter of the existence of such documentation was informally discussed on October 15 and 20, 1982, between Messrs. B. R. Sylvia of CG&E and R. F. Warnick and D. R. Hunter of NRC Region III. Further information was required to resolve these concerns and, if these concerns were found to be valid, to determine whether enforcement action, including modification, suspension, or revocation of CG&E license, is warranted.

Accordingly, pursuant to Section 182 of the Atomic Energy Act and 10 CFR 50.54(f) of the Commission's regulations, on October 27, 1982, CG&E was required by Region III to submit under oath and in writing, by November 16, 1982, a list of all reports or other documentation prepared by CG&E's or HJK's document reviewers or other personnel in preparation for the meetings of July 8 and/or 9, 1982, or reviewed at those meetings. Copies of any such reports were requested to be made available at the Zimmer site for inspection by the NRC. (The licensee responded to this 10 CFR 50.54(f) "Request for Information" on November 15, 1982 (Attachment IV). This response is currently being reviewed by Region III).

B. Zimmer Section Manpower Availability and Utilization

1. Assigned Manpower

Section Chief (Assigned to Zimmer full time. Onsite part-time.)

Project Manager (Assigned to Zimmer full-time. Onsite as needed.)

Resident Inspectors

Senior Resident (Full-time onsite)
Resident Inspector (Full-time onsite)
Resident Inspector (Full-time onsite)

Investigators (Office of Investigations (OI))

Five Investigators (Full-time)

2. Summary of Manpower Utilization

Onsite and in office professional effort from January 3, 1982 through October 30, 1982 was approximately 8,949.5 manhours, with 1068 of these manhours occurring between October 3 and October 30, 1982. (These hours do not include those of OI investigators).

C. High Visibility Issues

1. Inspection of Catalytic, Inc. Activities

Special inspection performed by Region III during weeks of August 10 and September 7 and 13, 1982, regarding the adequacy of licensee control of Catalytic, Inc. work activities and to determine if the licensee had violated the December 24, 1980 NRC Immediate Action Letter (IAL) and the December 30, 1980 CG&E Stop Work Order (SWO).

Effort by Licensee

During the NRC inspection a number of corrective actions were taken by the licensee, including the issuance of a Limited Stop Work Order (SWO) on September 10, 1982, on miscellaneous work by Catalytic, Inc.; and a Stop Work Order (SWO) on October 11, 1982, for all essential work activities performed by Catalytic, Inc., following the discussion of the inspection findings with Region III on October 8, 1982.

Immediately prior to the inspection, the licensee notified the NRC that a Stop Work Order (SWO) was initiated on August 5, 1982, regarding Catalytic, Inc. work associated with the removal of Control Rod Drive (CRD) hangers and supports.

NRC Effort/Action

The NRC inspection identified a number of apparent programmatic concerns as well as individual findings in the field concerning the Catalytic, Inc. safety-related activities.

A management meeting was conducted at the Greater Cincinnati Airport (Open Meeting) on October 19, 1982, to discuss the inspection findings.

Additional working meetings were held on November 2-4, 1982, to discuss the specific technical and programmatic issues in more detail.

Findings to Date

The licensee and the NRC have identified a number of concerns in the areas of organizational description and interfaces, assignment of responsibilities and authority, training, design control, procedures, document control, inspections, nonconformance control, corrective action, records, and audits. A number of concerns are considered repeat items. The inspection results indicate a breakdown in the CG&E management controls regarding Catalytic, Inc. The above items were considered prior to the issuance of the November 12, 1982 NRC ORDER.

2. H. J. Kaiser (HJK) Internal Investigation Report

Region III anonymously received a partial copy of a report of a HJK investigation conducted at Zimmer. The report was mailed from Cincinnati, Ohio on March 23, 1982, and received by Region III on March 26, 1982. The NRC is investigating to determine the safety significance of the matters described in the investigation report and whether or not NRC reporting requirements were met.

Effort by Licensee

The licensee is conducting a review of the report.

NRC Effort/Action

The NRC investigation into the HJK investigation report and its significance has been transferred to the Office of Investigations.

Findings to Date

The findings will be discussed after the investigation is complete.

D. Quality Confirmation Program (QCP)

A summary of the progress of the QCP task areas is provided as Attachment I. The CG&E QCP Status Report as of October 30, 1982, is included as Attachment II. This attachment includes more detail than the following summary presents.

1. Task I, Structural Steel

Review of structural beams, beam welds, re-entrant corners, procurement of beams, beam and steel plate heat number traceability, and inspection of cable tray foot connections.

Effort by Licensee

The licensee is continuing to inspect structural steel items including foot connections, drywell steel, control room steel, gallery steel, and switchgear steel.

The task involves 26 personnel and is reported to be 58% complete with an estimated completion date of December 1, 1982.

NRC Effort/Action

Reviewed 2,013 initial issue nonconformance reports (NRs) and 138 dispositioned NRs and inspected a selected number of those reviewed.

Resident and specialist inspectors performed field walkdowns of actual conditions and rework activities.

Findings to Date

The licensee has generated 977 NRs identifying about 9546 weld deficiencies. The majority of the 4963 deficiencies dispositioned have been dispositioned as "rework".

Three construction deficiency reports have been reported to the NRC concerning laminated angle iron, cable tray hangers/weld deficiencies, and cable tray hanger "Nelson stud" deficiencies.

2. Task II, Weld Quality

Review of welds performed, weld rod control, transfer of weld rod heat numbers, and deletion of weld inspection criteria.

Effort by Licensee

The licensee is continuing to review the areas of structural weld cards, welding procedures, welder qualifications, small-bore piping welds, and large-bore piping welds.

The task involves six people and is reported to be 78% complete with the completion date still to be determined. The licensee is reorganizing the task work with both HJK and CG&E performing the reviews to improve efficiency.

NRC Effort/Action

Reviewed 642 initial issue NRs and 74 dispositioned NRs and inspected a selected number of those reviewed.

Findings to Date

The licensee has identified nonconforming conditions regarding weld data sheets, heat numbers, welder qualifications, and welding procedures. Discrepancies being identified include lack of objective evidence, white-outs and cross-outs, signature differences, inconsistent data, and lack of adequate acceptance criteria. (Welder qualifications are being reviewed by a special HJK/CG&E task group.)

Due to the identification of the lack of adequate control of drawings, inadequate acceptance criteria, and restart of the review activity under a new program, the large-bore piping weld review has not progressed.

Three construction deficiency reports have been identified to the NRC concerning weld procedure deficiency, carbon weld rod in stainless steel weld, and lack of weld preheat or post weld heat treatment.

3. Task III, Heat Number Traceability

Review of installed large-bore and small-bore pipe heat numbers, heat numbers on isometric drawings, incorrect or marked-up heat numbers, and purchase orders.

Effort by Licensee

The licensee is continuing to review heat number traceability in the areas of large-bore piping, small-bore piping, and by purchase orders.

The task involves 5 people (the August-September monthly status report incorrectly showed 14 people on this task, the correct number for that time was 6 people) and is reported to be 36% complete with the estimated completion date still to be determined.

NRC Effort/Action

Reviewed 529 initial issue NRs and 36 dispositioned NRs.

Findings to Date

The licensee has identified a substantial number of discrepancies regarding small-bore piping, large-bore piping, and purchase orders.

The nonconforming conditions include documentation deficiencies, lack of design change control, unsigned reports, use of unapproved vendors, and upgrade of materials.

Three construction deficiencies have been identified to the NRC concerning 2400 ft. of SA-106 Grade B piping, some bolting material, and control of "gamma plugs."

4. Task IV, Socket Weld Fitup

Review of socket weld fitup to ensure adequate disengagement for small bore piping.

Effort by Licensee

The task is substantially complete with 29,821 fitups reviewed of a total of 32,000. The final reviews of the identified discrepancies and radiographs are being made.

The task involves one person, and is reported as 98% complete with an estimated completion date of December 1, 1982.

NRC Effort/Action

Reviewed 115 initial issue NRs and 84 dispositioned NRs and inspected a selected number of those reviewed.

Findings to Date

The licensee identified 695 weld joints which lacked evidence of disengagement and the joints have been radiographed. Of the 695 welds 111 have been rejected for lack of disengagement.

5. Task V, Radiographs

Review the radiographs which did not meet ASME Code requirements due to inadequately shimmed penetrameters.

Effort by Licensee

The licensee has completed the review of the radiographs and is preparing a code inquiry to ASME concerning the shimming of the penetrameters.

The selected, qualifying radiographs are being reviewed to assess the weld conditions. Of the 61 welds initially identified, 2 welds have been found to be duplicates, 1 weld had been replaced and was removed from the list, and 12 welds are inaccessible. This reduces the total to 46 welds identified. All 46 have been re-radiographed and accepted.

This activity is reported to be approximately 99% complete with a completion date of November 15, 1982.

NRC Effort/Action

The resident inspector reviewed 97 initial issue NRs and 4 dispositioned NR.

Findings to Date

A substantial number of the M. W. Kellogg radiographs were not shimmed adequately; however, the quality and sensitivity of the radiographs appears adequate.

Two construction deficiency reports have been identified to the NRC concerning radiographs.

6. Task VI, Cable Separation

Review cable separation regarding essential and associated cables.

Effort by Licensee

The licensee is continuing the review and evaluation of wall penetrations/sleeves, associated cables for Class 1E panels, all 1E panels, and responses to engineering evaluation requests (EERs).

The task involves nine persons and is reported to be 50% complete (scope expanded) with a completion date of June 1, 1983 (scope expanded).

NRC Effort/Action

Reviewed 879 initial issue NRs and 285 dispositioned NRs and inspected a selected number of those reviewed.

Findings to Date

Nonconforming conditions have been identified concerning cable separation, identification, and routing. The licensee has written 776 NRs and 345 NRs have been dispositioned (131 rework, 25 repair, and 189 accept-as-is) of which 217 NRs have been closed to date, with an additional 12 NRs cancelled because of duplication.

Three construction deficiency reports have been identified to the NRC concerning electrical cable separation.

7. Task VII, Nonconformance Reports

Review of noncomformances documented in surveillance reports, punch lists, and exception lists; and nonconformances not documented, not entered, and voided rather than adequately dispositioned. Review 300 closed NRs and solicit NRs not entered into the system.

Effort by Licensee

The licensee is continuing to review and evaluate nonconformances.

Approximately 200 letters to former QC inspectors soliciting nonconformances not entered into the system return receipt requested were mailed by the licensee. The responses were limited. One report has been received from an individual identifying four potential NRs.

Additional procedures, field walkdowns, and a material impound area are being provided.

The task involves three persons and is approximately 61% complete with an estimated completion date of January 30, 1983.

NRC Effort/ Action

Reviewed 367 initial issue NRs and 257 dispositioned NRs and inspected a selected number of those reviewed.

Findings to Date

Fifty eight NRs have been written due to reopening of previously voided NRs. Of these NRs, 43 have been dispositioned (15 rework, 1 repair, and 27 accept-as-is).

8. Task VIII, Design Control and Verification

Review procedures controlling design calculation completion, S&L program for controlling deviations from FSAR, correctness and consistency of FSAR, and design deviation identification and disposition.

Effort by Licensee

A programmatic audit of S&L design control was needed to complete this task. The audit was conducted on September 1-3, 1982 in Chicago. There were no findings. The audit will not be closed until the two recommendations given by the audit team have been resolved.

The task is 99% complete with an estimated completion date of December 15, 1982.

NRC Effort/Action

The activity is being monitored with a closeout inspection planned to verify completion of the QCP Task VIII items, to identify any generic implications, and to evaluate the licensee conclusions.

Findings to Date

No significant findings have been identified to date regarding the S&L work; although, the S&L system has been made more formal.

Four construction deficiency reports have been identified to the NRC regarding design control.

9. Task IX, Design Document Changes (DDC)

Establish an accurate and complete listing of DDCs, DDC records, and associated QC inspection records.

Effort by Licensee

This task was divided into five phases as follows:

Phase I Classification of CG&E, S&L, and HJK, DDCs as essential or nonessential.

Phase IA Classification of WY&B and other site contractor DDCs as essential or nonessential.

Phase II Review inspection documentation to determine if CG&E, S&L, and HJK DDCs have been incorporated and inspected.

Phase IIA Review inspection documentation to determine if WY&B and other site contractor DDCs have been incorporated and inspected.

Phase III Electrical inspections in the control room.

The original task (Phases I and II) is 40.8% complete. Phases IA and IIA are 9.2% complete and Phase III is 10% complete.

The task involves 14 persons for Phases I, II, IA, and IIA with an estimated completion date of January 1983. Phase III involves 6 persons with an estimated completion date of June 1983.

NRC Effort/Action

Reviewed 511 initial issue NRs and 3 dispositioned NRs and inspected a selected number of those reviewed.

Findings to Date

A number of deficiencies have been identified concerning missing documentation, misclassification of DDCs, inspection program deficiencies, failure to incorporate deficiencies, premature inspection, and incomplete inspection documentation.

10. Task X, Subcontractor QA Programs

Confirm the quality of the Bristol Steel work and review all subcontractor QA programs or safety-related work to ensure the safety-related activities performed were acceptable.

Effort by Licensee

Bristol field welds are being reviewed within Task I and 80 audits of subcontractors were identified encompassing 13 subcontractors. All 80 audits have been reviewed.

The task involves two persons and is 78% complete with an estimated completion date of December 1982.

NRC Effort/Action

None during this period.

Findings to Date

Six subcontractor audits may require audits of subsequent activities for confirmation of work that cannot be verified by document review or inspection.

Many of the 80 audits reviewed do not address all applicable criteria.

11. Task XI Audits

Review all past audits of HJK, S&L, GE, GED, EPD, EOTD, and GCD to determine depth and adequacy with respect to Appendix B to 10 CFR 50 and appropriate closeout of audit findings. Justify the acceptability of the areas not audited.

Effort by Licensee

Review of audits, audit summaries prepared, and all of the 296 past audits have been reviewed.

The task involves two persons and is 82% complete with estimated completion date of November 15, 1982.

NRC Effort/Action

None during this period.

Findings to Date

Coverage of the audits was not sufficient to verify adequate implementation of program requirements.

E. Ongoing Construction Activities

Major ongoing construction activities during October included installation and modification of pipe supports, drywell steel modifications, installation of drywell air coolers, seismic modification to switchgear, rework of control room structural steel, and installation of seismic columns.

As of November 12, 1982, all safety-related construction activities, including rework of identified deficient construction was suspended by an NRC ORDER.

F. Potential Plant Problems

The following is a list of areas or items which Region III considers as potential problems and are being monitored by the inspectors.

- Rust on the stainless steel liner plate in suppression pool
- Containment liner leak rate channel leakage (welds)
- Sacrificial shield weld inadequacies (records and actual weld conditions)
- Cable tray trapeze weld and support stud inadequacies
- Past personnel qualifications
- Past weld procedures
- Purchase of equipment
- Structural steel bolting
- Control panel and electrical switchgear mounting plug weld inadequacies
- Cable seperation inadequacies

G. Freedom of Information Act (FOIA) Requests

None during this period.

SUMMARY OF THE PROGRESS OF QUALITY CONFIRMATION PROGRAM (QCP) TASK AREAS
AS OF JUNE, JULY, AUGUST, SEPTEMBER, AND OCTOBER 1982

TASK AREA	PERCENT COMPLETE/EXPECTED COMPLETION AS OF				
	JUNE	JULY	AUGUST	SEPTEMBER	OCTOBER
I. STRUCTURAL STEEL*	31% 12/01/82	35% 12/01/82	50% 12/01/82	57% 12/01/82	58% 12/01/82
II. WELD QUALITY*	58% **	58% **	62% **	68% **	70% **
III. HEAT NUMBER TRACEABILITY*	30% **	30% **	33% **	33% **	36% **
IV. SOCKET WELD FITUP	98% 08/01/82	95% 10/01/82	96% 10/01/82	98% 12/01/82	98% 12/01/82
V. RADIOGRAPHS	97% 08/01/82	97% 09/15/82	98% 11/15/82	98% 11/15/82	99% 11/15/82
VI. CABLE SEPARATION*	54% 12/31/82	52% 12/31/82	35% 06/01/83	44% 06/01/83	50% 06/01/83
VII. NONCONFORMANCES	61% 12/31/82	40% 12/31/82	52% 12/31/82	61% 01/30/83	61% 01/30/83
VIII. DESIGN CONTROL AND VERIFICATION	97% 07/15/82	99% 08/15/82	99% **	99% **	99% 12/15/82
IX. DESIGN DOCUMENT CHANGES	34% 12/31/82	35% 12/31/82	<32% 04/15/83	<33% 04/15/83	<35% 06/01/83
X. SUBCONTRACTOR QA PROGRAMS	60% 08/13/82	65% 09/15/82	75% 10/15/82	75% 10/30/82	78% 12/31/82
XI. AUDITS	70% 10/08/82	72% 10/08/82	74% 11/15/82	80% 11/15/82	82% 11/15/82

*Areas viewed by Region III as potentially requiring a significant amount of rework.
**Estimated completion date to be determined.

Attachment I

QUALITY CONFIRMATION PROGRAM

STATUS

(AS OF OCTOBER 31, 1982)

BY

J. F. SHAFFER

DIRECTOR, QUALITY CONFIRMATION PROGRAM

Attachment II

QUALITY CONFIRMATION PROGRAM

TASK I: STRUCTURAL STEEL

ACTION BEING TAKEN

A. 100% VISUAL INSPECTION OF ACCESSIBLE STRUCTURAL STEEL BEAM FIELD WELDS, BRISTOL SHOP WELDS, RE-ENTRANT CORNERS AND WELDED CABLE TRAY FOOT CONNECTIONS.

B. 100% OF ALL BEAMS INSPECTED ARE COMPARED TO THE DESIGN DRAWINGS AND DESIGN DOCUMENT CHANGES (DDC'S).

C. 100% VISUAL INSPECTION OF HVAC SUPPORTS IN THE CONTROL ROOM.

SUMMARY OF TASK

AREA	TOTAL # OF ITEMS	ITEMS COMPLETED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPLETED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOURS REMAININ
1)DRYWELL LESS 525 ELV	322	0	322	0	100	0	0
2)REACTOR, AUX SRV WATER BLDG	*2992	23	1628	1	55	1452	8184
3)CONTROL RM HVAC SUPTS	105	0	49	0	47	0	336
TOTAL	3419	23	1999	1	58	1452	8520

CURRENT STATUS AND RESULTS

1. NONCONFORMANCE (NR) SUMMARY

AREA	NR'S WRITTEN	QUANTITY DEFICIENCIES IDENTIFIED	NUMBER DEFICIENCIES DISPOSITIONED	ACCEPT AS IS	REWORK	REJECT	REPAI
FOOT CONN.	253	975	242	139	88	0	15
DRYWELL	279	1825	314	10	306	4	3
CONTROL RM	173	2488	2331	17	2312	4	1
GALLERY STL	40	122	121	0	121	0	1
SWITCHGR 567'5"	34	1953	1955	0	1933	0	2
SWITCHGR 546'	66	869	0	8	7	0	1
AUX BLD ROOF 591'	1	119	0	0	0	0	0
CABLE SPRD 546'	82	78	-	-	-	-	-
HVAC	49	1117	-	-	-	-	-
TOTAL	977	9546	4963	174	4767	8	23

QUALITY CONFIRMATION PROGRAM

TASK I: (CONT'D)

NOTE: MAJORITY OF THE DEFICIENCIES ATTRIBUTED TO: OVERLAP, UNDERSIZE WELD, UNDERCUT, WELD PROFILE, LACK OF FUSION, AND INCORRECT INSTALLATION.

2. BEAMS BEING INSPECTED ARE COMPARED TO DESIGN DRAWINGS AND DESIGN DOCUMENT CHANGES AT PRESENT, THE BEAMS THAT HAVE BEEN INSPECTED, AND HAVE GONE THROUGH THE DESIGN, DOCUMENT REVIEW, APPEAR ON THE DESIGN DRAWING AND/OR A DESIGN DOCUMENT CHANGE (DDC).

3. COSMETIC REWORK HAS BEEN REQUIRED ON 70% OF THE BEAMS INSPECTED, TO OBTAIN RESULTS IN COMPLIANCE WITH AWS D1.1 1972 ACCEPTANCE CRITERIA.

COMMENTS

1. TASK I PROVIDED INSPECTORS TO PERFORM 152 INSPECTIONS FOR TASK IX ON ELECTRICAL CONDUIT SUPPORTS.

2. THE STOP WORK ORDER #82-02 STOPPING ALL WORK ON SCAFFOLDING, PAINT REMOVAL, ETC., HAS GREATLY IMPACTED THE TOTAL NUMBER OF INSPECTIONS COMPLETED THIS MONTH.

3. THE RANDOM SAMPLE INSPECTION OF 100 CONNECTIONS IN THE SERVICE WATER BUILDING IS IN PROGRESS. DEFICIENCIES NOTED THUS FAR ARE PRIMARILY COSMETIC IN NATURE. COMPLETION IS CONTINGENT UPON THE RESOLUTION OF ITEM 2 ABOVE.

PRESENT MANPOWER SUMMARY	ACTUAL
TASK COORDINATOR	1
QUALITY SPECIALISTS	3
INSPECTORS	15
DOCUMENT REVIEWERS	7
TOTAL	26

STATUS

THIS TASK IS APPROXIMATELY 58% COMPLETE.

ESTIMATED COMPLETION DATE

DECEMBER 1, 1982

QUALITY CONFIRMATION PROGRAM

TASK II: WELD QUALITY

ACTION BEING TAKEN

A. PERFORM A 100% REVIEW OF CODE PIPING KE-1 WELD DATA SHEETS TO DETERMINE WELD ROD HEAT NUMBERS, INSPECTION STAMPS AND DATES, IDENTIFY MISSED HOLD POINTS, AND MISSING OR ALTERED DOCUMENTATION.

B. VERIFY PROPER WELD PROCEDURE AND WELDER QUALIFICATION.

SUMMARY OF TASK

AREA	TOTAL # OF ITEMS	ITEMS COMPLETED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPLETED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOURS REMAINING
1) STRUCT. KE-1 REVIEW	11,000	0	11,000	0	98	0	416
2) SM BORE PIPE	32,000 (APPX)	0	25,841	0	79	*160	837
3) LG BORE PIPE	9,400 (380 PSK'S)	0	0	0	0	0	1373
4) WELD PROCEDURE REVIEW	91 (PROCEDURES) 121 REV'S	14	24	15	26	187	1413
5) WELDER QUAL. REVIEW	4600 (DOCUMENTS)	357	2456	7.5	53	80	1223
5) WELD ROD CONTROL		0	0	0	0	0	416
TOTAL	57,121	371	39,321	2	70	427	5678

1) *SEE ITEM #2 BELOW

CURRENT STATUS AND RESULTS

1. AREA #1 PRELIMINARY REPORT DRAFTED ON KE-1 REVIEW CER TO BE INITIATED ON AREAS OF CONCERNS.

2. AREA #2 *REWRITE OF PROCEDURE 19-QA-14 TO INCORPORATE A FINAL REVIEW OF DOCUMENTS GENERATED, *MANHOURS EXPENDED IN THE REVISION OF 69 NR'S TO THIS TASK FOR CLARITY AND PROCEDURE REWRITE.

3. AREA #3 PROCEDURE TO BE WRITTEN, SEE TO BE RESOLVED, ITEM #1.

TASK II: (CONT'D)

4. AREA #4 24 CER'S HAVE BEEN GENERATED ON PROCEDURE'S REVIEWED.

5. AREA #6 PROCEDURE HAS NOT BEEN WRITTEN DUE TO LACK OF PERSONNEL - INVESTIGATION TO BE STARTED NOVEMBER 1982 WITH PROCEDURE DEVELOPED AT THE SAME TIME FRAME.

TO BE RESOLVED

1. COORDINATION WITH HJK REVIEW FOR NPP-1 PROGRAM TO EFFECT A COORDINATED N-5 PROGRAM INCLUDING ITEM (1) IN COMMENTS SECTION.
2. CONFIRM ACCEPTABLE HEAT NUMBERS ESTABLISHED BY TASK III.
3. COMPLETION OF REWRITE OF 19-QA-14 TO SUPPLY DIRECTION FOR A SECOND REVIEW.

MANPOWER SUMMARY	ACTUAL
TASK COORDINATOR	1/2
INSPECTORS	1
DOCUMENT REVIEWERS	4
CLERKS	1/2
TOTAL	6

STATUS

THIS TASK IS APPROXIMATELY 70% COMPLETE.

ESTIMATED COMPLETION DATE

TO BE DETERMINED BY HJK SCHEDULE OF NPP-1/N-5 PROGRAMS (TO DATE, SCHEDULE NOT POSTED).

QUALITY CONFIRMATION PROGRAM

TASK III: HEAT NUMBER TRACEABILITY

ACTION BEING TAKEN

A. PERFORM A 100% REVIEW OF SMALL BORE PIPING DOCUMENTATION AND CONDUCT A FIELD WALKDOWN OF THE SYSTEMS IDENTIFIED ON THE QCP LIST OF SAFETY-RELATED AND IMPORTANT TO SAFETY SYSTEMS.

B. PERFORM A 100% INSPECTION AND/OR REVIEW OF DOCUMENTATION OF LARGE BORE PIPING FIELD MODIFICATIONS OF SYSTEMS ON THE SAFETY-RELATED AND IMPORTANT TO SAFETY SYSTEM LIST.

C. PERFORM A 100% REVIEW OF ALL CODE AND STRUCTURAL PURCHASE ORDERS TO ESTABLISH A LIST OF ACCEPTABLE HEAT NUMBERS.

D. PERFORM A REVIEW OF PURCHASE ORDERS FOR STRUCTURAL STEEL AND STEEL SHAPES TO DETERMINE IF PURCHASED ESSENTIAL OR NONESSENTIAL (THESE P.O.'S SHALL BE PART OF ACTION C). RESULTS OF THE REVIEW WILL BE USED TO ANSWER TASK 1 CONCERNS REGARDING ACCEPTABILITY OF STRUCTURAL STEEL.

E. PERFORM INSPECTION AND REVIEW OF GAMMA PLUGS TO GAIN ADDITIONAL INFORMATION FOR EVALUATION AND DISPOSITION OF 10CFR50.55(e) REPORT M-56.

SUMMARY OF TASK

AREA	TOTAL # OF ITEMS	ITEMS COMPLETED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPELTED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOU REMAIN
A) SM BORE DOC	2691 ISK	27	52	1	2	226	2540
SM BORE WLKDWN	2691 ISK	0	2691	0	100	0	0
B) LG BORE DOC	(APPX) 380 PSK	0	0	0	0	0	74,600
LG BORE WLKDWN	(APPX) 380 PSK	0	0	0	0	0	
C) PURCHASE ORDER REVW	2688	26	44	1.0	1.6	215	2951
D) HJK STRCTL STEEL PO'S ESS/NONESS	1900	0	1900	0	95	0	4C
E) GAMMA PLUG 50.55(e)M-56	404	179	219	44	54	223	20[illegible]
TOTAL	11,134	232	4906	3	36	664	80,34C

TASK III: (CONT'D)

1) *TOTAL MANHOURS SHOWN IS FOR BOTH LARGE BORE DOCUMENTATION AND WALKDOWN.

2) **INCLUDES STRUCTURAL STEEL PO'S IDENTIFIED IN ACTION D. TOTAL NUMBER OF ITEMS CHANGED DUE TO CHANGE IN METHOD OF ACCOUNTING, I.E. PURCHASE ORDERS RATHER THAN SHIPMENTS RECEIVED.

3) ***5% REMAINING TO COMPLETE IS TO PERFORM FINAL REVIEW.

CURRENT STATUS AND RESULTS

1. HJK IS VALIDATING CMTR'S FOR STRUCTURAL ITEMS RECEIVED ON NONESSENTIAL PO'S THROUGH USER TESTS OR QUALIFYING SUPPLIER. SIMILAR REVIEW AND ACTION BEING COMPLETED FOR ASME PIPING COMPONENTS.

2. 1900 STRUCTURAL STEEL PURCHASE ORDERS HAVE BEEN IDENTIFIED, OF THIS NUMBER, 1142 WERE ORDERED ESSENTIAL AND 758 WERE ORDERED NONESSENTIAL.

3. 219 OF 404 GAMMA PLUGS HAVE BEEN INSPECTED. THE DATA HAS BEEN SUBMITTED TO CG&E NUCLEAR ENGINEERING DEPARTMENT. ACTION ON THIS ITEM IS STOPPED PENDING NED EVALUATION.

TO BE RESOLVED

1. A METHOD OF SURVEILLING THE H.J. KAISER N-5 PROGRAM IS BEING RESEARCHED TO ADDRESS ACTION B.

COMMENTS

1. THE FIGURE SHOWN FOR LARGE BORE REFLECTS AN APPROXIMATE NUMBER OF MANHOURS NECESSARY FOR THE QCP TO PERFORM THE LARGE BORE WALKDOWN AND DOCUMENTATION REVIEW. THIS FIGURE WOULD BE MUCH LESS IF HJK PERFORMED THIS FUNCTION AND QCP PERFORMED A SURVEILLANCE OF THEIR ACTIVITY.

MANPOWER SUMMARY	ACTUAL
TASK COORDINATOR	1
TASK LEAD	1
INSPECTORS	1
DOCUMENT REVIEWERS	2
TOTAL	5

TASK III: (CONT'D)

STATUS

THIS TASK IS 36% COMPLETE.

ESTIMATED COMPLETION DATE

TO BE DETERMINED AFTER COORDINATION WITH HJK AND CG&E DOCUMENTATION REVIEW GROUP.

TASK IV: SOCKET WELD DISENGAGEMENT

ACTIONS BEING TAKEN

A. IDENTIFY SMALL BORE PIPING SOCKET WELDS FOR WHICH VERIFICATION FOR DISENGAGEMENT DOES NOT EXIST.

B. RADIOGRAPH 100% OF THE ACCESSIBLE WELD NOT HAVING VERIFICATION OF DISENGAGEMENT.

SUMMARY OF TASK

AREA	TOTAL # OF ITEMS	ITEMS COMPELTED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPLETED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOUR REMAINI
IV SOCKET WELD DISENGAGE-MENT	32,000	0	29,821	0	98	*90	240

CURRENT STATUS AND RESULTS

A. 348 RADIOGRAPHS TO DATE HAVE BEEN TRANSMITTED TO QADVG.

B. 111 SOCKET WELDS HAVE BEEN REJECTED FOR LACK OF DISENGAGEMENT.

C. 695 WELDS HAVE BEEN IDENTIFIED TO DATE AS LACKING EVIDENCE OF A QUALITY INSPECTION.

D. REJECT WELD HAVE BEEN IDENTIFIED ON NONCONFORMANCE REPORTS.

COMMENTS

1. MANHOURS EXPENDED THIS TASK DUE TO CER 82-240 WRITTEN ON 19-QA-02 FOR NON-COMPLIANCE OF PROCEDURE. RESULTS: A REWRITE OF 19-QA-02 FOR CLARITY, DIRECTION AND TO INITIATE A SECOND REVIEW OF DOCUMENTS GENERATED BY TASK IV. IMPACT OF SECOND REVIEW HAS NOT BEEN DETERMINED TO DATE, PROCEDURE WAS REVIEWED BY THE REVIEW BOARD AND COMMENTS ADDED 10-26-82.

MANPOWER SUMMARY	ACTUAL
TASK COORDINATOR	1/2
CLERKS	1/2
TOTAL	1

TASK IV: (CONT'D)

STATUS

THIS TASK IS 98% COMPLETE. INITIAL REVIEW/CER 82-240.

ESTIMATED COMPLETION DATE

DECEMBER 1, 1982 PENDING APPROVAL OF THE H. THIELSCH REPORT. AN ESTIMATED COMPLETION DATE FOR THE SECOND REVIEW WILL BE ADDRESSED AFTER IMPLIMENTATION OF 19-QA-02.

TASK V: RADIOGRAPHS

ACTION TO BE TAKEN

A. CONFIRM THAT THE EXISTING RADIOGRAPHS OF LARGE BORE PIPING SUPPLIED BY M. W. KELLOGG ARE ADEQUATE TO IDENTIFY WELD DEFICIENCIES.

SUMMARY OF TASK

AREA	TOTAL # OF ITEMS	ITEMS COMPLETED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPLETED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOURS REMAININ
RADIOGRAPHY REVIEW	4250	0	4250	0	100	0	0
RADIOGRAPHS PHASE II	46	4	46	27.3	100	40	40
TOTALS	4296	4	4296	1	99	40	40

CURRENT STATUS AND RESULTS

A. REVIEW OF 4250 WELD RADIOGRAPHS FOR SENSITIVITY IS COMPLETE.

B. 46 WELDS HAVE BEEN IDENTIFIED TO BE RE-RADIOGRAPHED, COVERING ALL VARIATION IN PIPE AND WALL THICKNESS.

C. PROGRAM TO CONFIRM RADIOGRAPHS HAS BEEN APPROVED BY THE NATIONAL BOARD OF BOILER PRESSURE VESSEL INSPECTORS AND THE STATE OF OHIO.

D. 46 OF THE 46 RADIOGRAPHS HAVE BEEN RE-RADIOGRAPHED, AND ACCEPTED.

E. MOCK-UPS OF THE 4 INACCESSABLE WELDS WERE FABRICATED AND RADIOGRAPHED.

TO BE RESOLVED

COMMENTS

A FINAL REPORT FOR TASK V IS BEING WRITTEN AND WILL BE PRESENTED TO ALL APPLICABLE PARTIES.

MANPOWER SUMMARY

PERSONNEL ASSIGNED TO TASK III WILL SUPPORT THIS ACTIVITY.

STATUS

OVERALL TASK IS APPROXIMATELY 99% COMPLETE.

ESTIMATED COMPLETION DATE

NOVEMBER 15, 1982

TASK VI: CABLE SEPARATION

ACTION BEING TAKEN

A. 100% INSPECTION OF SLEEVES AND CLASS IE FLOOR PENETRATIONS FOR SEPARATION IDENTIFICATION AND ROUTING.

B. INSPECT A MINIMUM OF 10% OF THE ASSOCIATED CABLES TO ARRIVE AT A 95% CONFIDENCE LEVEL THAT 95% OF THE ASSOCIATED CABLES ARE PROPERLY SEPARATED IN TRAYS AND CONDUITS. CONFIDENCE LEVEL WAS NOT MET. THIS INSPECTION IS SUPERSEDED BY THE INSPECTION OF CABLE TRAYS.

C. 100% INSPECTION OF CABLE TRAYS IN CATEGORY 1 STRUCTURES.

D. 100% INSPECTION OF CABLES REQUIRING SEPARATION INSIDE PANELS AND TO THE FIRST ASSIGNED RACEWAY TO VERIFY PROPER SEPARATION AND IDENTIFICATION. THIS INCLUDES ALL CABLES INSTALLED BETWEEN THE CABLE SPREADING ROOM AND THE MAIN CONTROL ROOM.

E. TRACK RESOLUTION OF SIX SAMPLES OF FAILURE TO MEET CABLE SEPARATION CRITERIA IDENTIFIED BY THE NRC.

SUMMARY OF TASK

AREA	TOTAL # OF ITEMS	ITEMS COMPLETED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPLETED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOURS REMAINING
WALL PENETRATION SLEEVES (SEE NOTE B)	182	0	182	0	100	0	32
ASSOC. CABLES	392	0	392	0	100	0	0
TRAY SYS. WLKDWN (SEE NOTE C)	1778(TR) (12446 UN)	1903(UN)	5589(UN)	15.3	44.9	538.5	8172
CABLES REQUIRING SEP. INSIDE PANELS (SEE NOTE D)	8951 CABLES (35,804 UN)	0	0	0	0	217	6357
REVIEW OF EER RESPONSES		0	37	0	100	0	0
ADMINISTRATIVE	-	-	-	-	-	589.5	1592
TOTAL	48,824	1903	6200	6	50	1345	16,153

(TR = TRAYS)
(UN = UNITS)

TASK VI: (CONT'D)

CURRENT STATUS AND RESULTS

A. A TOTAL OF 776 NONCONFORMANCE REPORTS HAVE BEEN WRITTEN FOR SEPARATION, IDENTIFICATION AND ROUTING DEFICIENCIES. 345 NR'S DISPOSITIONED (131 REWORK, 25 REPAIR, 189 ACCEPT-AS-IS). 217 OF THE 345 DISPOSITIONED NR'S HAVE BEEN CLOSED. 12 NR'S WERE CANCELLED BECAUSE OF DUPLICATION. 419 NR'S HAVE NOT BEEN DISPOSITIONED.

B. WALL PENETRATION AND SLEEVE INSPECTIONS COMPLETE. LEVEL II REVIEW OF INSPECTION RECORDS REMAINING.

C. THE DECISION HAS BEEN MADE TO INSPECT CABLE TRAYS IN PLACE OF CLOSING OUT ASSOCIATED CABLE INSPECTION RECORDS SINCE CONFIDENCE LEVEL HAS NOT BEEN MET. THE PROCEDURE FOR TRAY INSPECTION 19-QA-38 WAS APPROVED ON AUGUST 24, 1982. THE CABLE TRAY INSPECTIONS FOR THE CONTROL ROOM AND THE SERVICE WATER PUMP STRUCTURE ARE COMPLETE AND INSPECTIONS ARE BEING PERFORMED IN THE REACTOR BUILDING.

D. THERE ARE 4210 CABLES THAT REQUIRE .INSPECTION IN THE CONTROL ROOM AND FROM THE CABLE SPREADING ROOM TO THE CONTROL ROOM. INSPECTIONS CANNOT START UNITL THE SEPARATION REQUIREMENTS ARE FINALIZED.

E. STATUS OF THE SIX EXAMPLES OF LACK OF SEPARATION: (2) CORRECTED, (1) IN-PROCESS OF BEING REWORKED, (2) ACCEPTED BY ENGINEERING, AND (1) IDENTIFIED ON NR.

COMMENT

MEETINGS WERE HELD WITH S&L TO CLARIFY AND RESOLVE SEPARATION REQUIREMENTS.

MANPOWER SUMMARY	ACTUAL
TASK COORDINATOR	1
QUALITY ENGINEER	1
LEAD INSPECTOR	1
INSPECTORS	4 1/2
DOCUMENTATION (NR'S & SCOPING PKGS)	1 1/2
TOTAL	9

TASK VI: (CONT'D)

STATUS

50% COMPLETE (EXPANDED SCOPE MANHOURS).

ESTIMATED COMPLETION DATE

JUNE 1, 1983

QUALITY CONFIRMATION PROGRAM

TASK VII: NONCONFORMANCE REPORTS

ACTION BEING TAKEN

A. 100% REVIEW OF ALL VOIDED NONCONFORMANCE REPORTS (NR'S), SURVEILLANCE REPORTS (SR'S), PUNCHLIST AND EXCEPTION LIST ITEMS.

B. 100% SOLICITATION TO PAST AND PRESENT QC INSPECTORS, BY CERTIFIED MAIL, REQUESTING THEIR KNOWLEDGE OF ANY NONCONFORMANCE REPORTS NOT ENTERED INTO THE SYSTEM.

C. REVIEW 300 RANDOMLY SELECTED, CLOSED NR'S.

SUMMARY OF TASK

AREA	TOTAL # OF ITEMS	ITEMS COMPLETED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPLETED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOURS REMAINING
VOIDED NR REVIEW INITIAL	1318	100	1050	7.58	79.66	125	539
VOIDED REVIEW SECONDARY	1318	64	656	4.85	49.77	80	565
SURVEILL-ANCE RPT REVIEW	3500	0	3285	0	94	0	5430
PUNCHLIST REVIEW	(APPX) 25,000	0	0	0	0	10	1190
CLOSED NR REVIEW	300	0	50	0	16.66	0	145
NR'S FROM INSPECTORS	UNKNOWN	0	0	(NO NR'S RECEIVED THIS MONTH)			
TOTAL	31,436	0	5,041	0	61	215	7869

CURRENT STATUS AND RESULTS

A. OF THE 100 NR'S REVIEWED, 50 HAVE BEEN SENT TO HJK FOR ADDITIONAL INFORMATION/ BACK-UP DOCUMENTATION AND 50 ARE READY FOR TRANSMITTAL UPON RECEIPT OF ORIGINAL 50 (EXPECTED 10/26; LETTER FOR ADDITIONAL 50 FOR SIGNATURE).

B. NO NO COMMITTEE HELD THIS MONTH.

TASK VII: CONT'D.

C. TWENTY MAN HOURS WERE EXPENDED THIS PERIOD FOR DISCUSSION AND PREPARATION OF PROCEDURES TO CLOSE OUT PUNCHLIST ITEMS.

D. PREPARATION OF PROCEDURE (19-QA-20 REVIEW OF PUNCHLIST) IN PROGRESS AS OF THIS DATE. PROCEDURE AND FINAL REVIEW CYCLE TO BE COMPLETED BY NOVEMBER 15, 1982.

E. 58 NR'S HAVE BEEN GENERATED FROM THE REVIEW OF VOIDED HJK NR'S AND SURVEILLANCE REPORTS.

MANPOWER SUMMARY	ACTUAL
TASK COORDINATOR	1
DOCUMENT REVIEWERS	2
TOTAL	3

STATUS

THIS TASK IS APPROXIMATELY 61% COMPLETE.

ESTIMATED COMPLETION DATE

JANUARY 30, 1983

TASK VIII: DESIGN CONTROL AND VERIFICATION

ACTION BEING TAKEN

VERIFY ADEQUACY OF S&L DESIGN CONTROL AND VERIFICATION PROGRAM. PROCEDURES WERE IN PLACE AND WERE CLARIFIED.

CURRENT STATUS AND RESULTS

THE FINAL REPORT FOR TASK VIII IS BEING WRITTEN.

STATUS

THIS TASK IS 99% COMPLETE TO DATE.

ESTIMATED COMPLETION DATE

DECEMBER 15, 1982

QUALITY CONFIRMATION PROGRAM

TASK IX: DESIGN DOCUMENT CHANGES

ACTION BEING TAKEN

1. COMPILE A COMPUTER LISTING OF ALL DDC'S.
2. REVIEW CG&E, HJK, S&L AND OTHER SITE CONTRACTORS' DDC'S TO DETERMINE PROPER CLASSIFICATION.
3. REVIEW CG&E, HJK, S&L AND OTHER SITE CONTRACTOR'S ESSENTIAL DDC'S TO DETERMINE IF DDC'S WERE INCORPORATED IN INSPECTION DOCUMENTATION.
4. PERFORM ELECTRICAL INSPECTIONS IN THE CONTROL ROOM FOR CABLE TRAY HANGERS, CONDUIT, AND CONDUIT SUPPORTS. (ADDITION TO ORIGINAL QCP WORK SCOPE)

SUMMARY OF TASK

AREA	TOTAL # OF ITEMS	ITEMS COMPLETED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPLETED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOURS REMAINING
PHASE I	19,711	0	18,725	0	95	0	80
PHASE II	9,111	227	3,303	2.5	36.3	551.5	13,858.5
PHASE IA	2,600	190	754	7.3	29	20	146
PHASE IIA	2,200	68	167	3.1	7.6	228	5,030.5
PHASE III							
DOCUMENTATION	714	24	124*	3.3	17.3	45.0	-
INSPECTION	714	79	98*	11	13.7	491.5	-
OTHER	-	-	-	-	-	1017.0	-
TOTAL	35,050	-	23,171	2	35	2353	19,115

* EQUIVALENT ITEMS

** OTHER TIME SPENT FOR TRAINING AND CERTIFICATION OF PERSONNEL, PROBLEM RESEARCH AND RESOLUTION, PROCEDURE PREPARATION, CLERICAL, PHASE II ADMINISTRATION, AND SETTING UP A DDC REVIEW AND HANDLING SYSTEM. (386.5 HRS. FOR DDC SYSTEM)

PHASE I - CLASSIFICATION OF CG&E, S&L AND HJK, DDC'S AS ESSENTIAL OR NON-ESSENTIAL.

PHASE II - REVIEW INSPECTION DOCUMENTATION TO DETERMINE IF CG&E, S&L AND HJK DDC'S HAVE BEEN INCORPORATED AND INSPECTED.

PHASE IA - CLASSIFICATION OF WY&B AND OTHER SITE CONTRACTOR DDC'S AS ESSENTIAL OR NON-ESSENTIAL.

PHASE IIA- REVIEW INSPECTION DOCUMENTATION TO DETERMINE IF WY&B AND OTHER SITE CONTRACTOR DDC'S HAVE BEEN INCORPORATED AND INSPECTED.

PHASE III- ELECTRICAL INSPECTIONS IN THE CONTROL ROOM.

QUALITY CONFIRMATION PROGRAM

TASK IX: (CONT'D)

RESULTS

1. IN PHASES II & IIA, 760 DEFICIENCIES IDENTIFIED TO DATE FOR REPORTING VIA DOCUMENT DEFICIENCY NOTICES OR NONCONFORMANCE REPORTS AS APPLICABLE.
2. IN PHASE III, 95 DEFICIENCIES HAVE BEEN IDENTIFIED TO DATE FOR REPORTING VIA NONCONFORMANCE REPORTS. A MAJORITY OF THESE DEFICIENCIES ARE DUE TO WELD DEFICIENCIES.
3. TASK IX GENERATED THE FOLLOWING CER'S DURING THE PAST MONTH:
   i) DOCUMENTATION FOR MISCELLANEOUS HVAC INSTALLATION
   ii) RESPONSE TIME FOR CER'S
   iii) DOCUMENTATION FOR INSPECTION OF GE MECHANICAL FDDR'S AND FDI'S
   iv) CABLE TRAY HANGER WELDS (VENDORS SHOP VS. FIELD)
   v) CONDUIT HANGER DEADWEIGHT CALCULATIONS
   vi) CONDUIT HANGER DIMENSION TOLERANCES
   vii) NR'S WITH CONDITIONAL DISPOSITIONS

PROMBLEM/DEFICIENCY	ELECT.	MECH.	STRUCT.	TOTAL
1. MISSING DOCUMENTATION	32		203	235
2. DDC MIS-CLASSIFICATION	13		3	16
3. INSPECTION PROGRAM DEFICIENCIES & MISC.	26		32	58
4. DDC NOT INCORPORATED IN DOCUMENTATION	101	10	307	418
5. INSPECTION PRIOR TO DDC BEING WRITTEN (AS-BUILT)	3			3
6. INCOMPLETE INSPECTION	8		22	30
TOTAL	183	10	567	760

TO BE RESOLVED

A. TASK IX IS STILL WAITING FOR HJK TO RESPOND TO CAR'S IN ORDER TO COMPLETE SOME DDC REVIEWS.
B. THE METHOD USED TO CALCULATE CONDUIT DEAD LOADS, AS CONTAINED IN THE E-189 SERIES GENERAL NOTES, MAY BE INADEQUATE. THIS IS BEING ADDRESSED BY CG&E QUALITY ENGINEERING.

TASK IX: (CONT'D)

C. THE WELD INSPECTORS HAVE STATED THAT THEY CANNOT DETERMINE WHICH WELDS ON TRAY SUPPORTS WERE MADE BY THE SMAW PROCESS AND WHICH WERE MADE BY THE MIG PROCESS AFTER THE WELDS HAVE BEEN HOT-DIPPED GALVANIZED. THERE IS PRESENTLY NO WAY TO DETERMINE WHICH SUPPORTS WERE FABRICATED BY THE VENDOR AND WHICH WERE FABRICATED HERE ON SITE. THE LATTER REQUIRE INSPECTION OF THE "SHOP" WELDS. THIS PROBLEM IS BEING ADDRESSED BY CG&E QUALITY ENGINEERING, PER CER WRITTEN BY TASK IX.

D. THERE ARE SEVERAL HUNDRED DDC'S OPEN AGAINST THE E-189 DRAWINGS, OF WHICH MORE THAN 150 APPLY TO THE CONTROL ROOM AREA. A SYSTEM IS CURRENTLY BEING DEVELOPED TO ASSURE THAT ALL APPLICABLE DDC'S ARE INCLUDED IN THE INSPECTION PACKAGES AND PROVIDE FOR VERIFICATION OF THAT FACT.

E. A LARGE NUMBER OF CONDUIT SUPPORTS IN THE MAIN CONTROL ROOM STILL HAVE OPEN WORK TICKETS FOR LINE REVISIONS. THE PRESENT HJK PROGRAM ONLY REQUIRES INSPECTION OF THE ACTUAL REVISION, AND NOT THE ENTIRE SUPPORT. THEREFORE, SINCE THESE SUPPORTS HAVE BEEN PREVIOUSLY ACCEPTED, THEY STILL FALL UNDER 19-QA-35. THIS PROBLEM HAS NOT YET BEEN ADDRESSED, BUT WILL REQUIRE RESOLUTION BEFORE ALL OF THE SUPPORTS CAN BE COMPLETED.

F. TASK IX IS GOING TO PARTICIPATE ON A STRUCTURAL DDC NR REVIEW BOARD WHICH IS GOING TO BE FORMED TO FACILITATE THE DISPOSITIONING OF DOCUMENTATION DEFICIENCIES. THIS REVIEW BOARD NEEDS TO BE FORMED.

G. WAITING FOR RESOLUTION OF PROBLEMS WITH X & Y DIMENSION TOLERANCES ON CONDUIT HANGERS.

H. WAITING FOR RESOLUTION/CLARIFICATION OF HANGER SEPARATION CRITERIA.

COMMENTS

1. PHASE III OF TASK IX REQUIRES MANPOWER ELECTRICAL INSPECTORS IN ORDER TO ACHIEVE COMPLETION BY THE DATE INDICATED BELOW. A TOTAL OF 14 PEOPLE WILL BE REQUIRED BY NOVEMBER 1, 1982 IN ORDER TO MEET THE COMPLETION DATE.

MANPOWER SUMMARY	PHASES I, II, IA, IIA
ENGINEERS (INCLUDING TASK COORDINATOR)	4
DOCUMENT REVIEWERS	2
CLERKS	1
TOTAL	7
	PHASE III
DOCUMENT REVIEWERS	4 **
INSPECTORS	3
TOTAL	7 (DOES NOT INCLUDE WELD INS

** NUMBER INCLUDES PHASE COORDINATOR & LEAD INSPECTOR. TWO DOCUMENT REVIEWERS ARE A
CERTIFIED AS ELECTRICAL INSPECTORS.

QUALITY CONFIRMATION PROGRAM

TASK IX: (CONT'D)

STATUS

PHASE I 95% COMPLETE

PHASE II 36.3% COMPLETE

PHASE IA 29.0% COMPLETE

PHASE IIA 7.6% COMPLETE

PHASE III 10% COMPLETE (ITEMS COMPLETED)

THE ORIGINAL TASK (PHASES I & II) IS APPROXIMATELY 40.8% COMPLETE.

PHASES IA AND IIA ARE APPROXIMATELY 9.2% COMPLETE.

THE OVERALL TASK IS 35% COMPLETE (DOES NOT INCLUDE PHASE III)

ESTIMATED COMPLETION DATE

PHASES I, II, IA, IIA - JUNE 1983

PHASE III - JANUARY 1983 *

* BASED ON RESOLUTION OF ITEMS "B", "C", "D", "E", "G" & "H" BY NOVEMBER 15, 1982.

TASK X: SUBCONTRACTOR/QA PROGRAM

ACTION BEING TAKEN:

A. ALL CG&E QA AUDITS OF SUBCONTRACTORS/VENDORS HAVE BEEN IDENTIFIED.

B. MATRICES HAVE BEEN DEVELOPED SHOWING WHICH 10CFR50 APPENDIX B CRITERIAS WERE VERIFIED DURING THESE AUDITS.

C. AUDIT REPORTS ARE BEING REVIEWED AND SUMMARIZED ON INDIVIDUAL FORMS.

D. EVALUATIONS ARE MADE OF THE SCOPE AND DEPTH OF THE AUDITS TO DETERMINE WHETHER THESE AUDITS COVERED ALL APPLICABLE 10CFR50, APPENDIX B CRITERIAS IN SUFFICIENT DETAIL.

1. IF EVALUATIONS PROVE SUBCONTRACTORS PROGRAMS ACCEPTABLE THESE CAN BE CLOSED.
2. IF EVALUATIONS CANNOT PROVIDE EVIDENCE THAT SUBCONTRACTOR PROGRAMS WERE SATISFACTORY, ALTERNATE MEASURE TO VERIFY HARDWARE INTEGRITY, SUCH AS REVIEW OF INTERNAL AUDITS PERFORMED BY SUBCONTRACTORS, REVIEW OF QUALITY DOCUMENTATION (INSPECTION RESULTS) TO PROVE ADEQUATE PROGRAM COVERAGE, COVERAGE IN OTHER QCP TASKS, RE-AUDIT OR HARDWARE INSPECTION AND/OR TESTING, WILL BE PURSUED..

SUMMARY OF TASK

AREA	TOTAL # OF ITEMS	ITEMS COMPLETED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPLETED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOURS REMAINING
SUBCONTRACTOR QA PROGRAMS	80	0	80	3	78*	20	130

* REMAINING 22% TO COMPLETE EVALUATION

CURRENT STATUS AND RESULTS

A. PROCEDURE FOR INTERFACE WITH OTHER QCP TASKS TO VERIFY ADEQUACY OF WORK HAS BEEN DEVELOPED. THIS PROCEDURE IS BEING REWRITTEN AND WILL BE SUMITTED FOR REVIEW.

B. EIGHTY (80) AUDITS OF SUBCONTRACTORS WERE IDENTIFIED AS BEING PERFORMED PRIOR TO APRIL 8, 1981. THESE AUDITS ENCOMPASSED THIRTEEN(13) DIFFERENT SUBCONTRACTORS. OF THE THIRTEEN SUBCONTRACTORS, SIX (6) MAY REQUIRE AUDITS OF SUBSEQUENT ACTIVITIES FOR CONFIRMATION OF THE WORK WHICH CANNOT BE ACCOMPLISHED BY DOCUMENT REVIEW OR INSPECTION. ALL AUDITS HAVE BEEN REVIEWED. THE SCOPE OF MANY AUDITS DO NOT ADDRESS APPLICABLE CRITERIA.

TASK X: CONT'D.

C. EVALUATIONS OF THE AUDITS WILL BEGIN WHEN THE PROCEDURE FOR THIS TASK IS APPROVED.

TO BE RESOLVED

COMMENTS

A. THE ESTIMATED TIME REQUIRED FOR COMPLETION IS 15 MAN DAYS WHICH INCLUDES EVALUATION OF PAST AUDITS OF SIX SUBCONTRACTORS.

B. IF THE SCOPE AT THIS TASK IS EXPANDED TO INCLUDE VENDORS WHICH SUPPLIED MATERIAL BUT HAVE NOT BEEN AUDITED, IT IS ESTIMATED THAT A MINIMUM OF 10 MAN DAYS PER VENDOR WILL BE REQUIRED TO COMPLETE AN AUDIT.

STATUS

THIS TASK IS 78% COMPLETE.

ESTIMATED COMPLETION DATE

DECEMBER 1982 FINAL REPORT AND EVALUATION.

QUALITY CONFIRMATION PROGRAM

TASK XI: AUDITS

ACTION BEING TAKEN

A. REVIEW ALL PAST CG&E QA AUDITS OF HJK, S&L, GE, NPD, NED, EOTD, AND GCD.

B. DEVELOP MATRICES SHOWING WHICH 10CFR50 APPENDIX B CRITERIA WERE IDENTIFIED DURING AUDITS.

SUMMARY

AREA	TOTAL # OF ITEMS	ITEMS COMPLETED THIS MONTH	ITEMS COMPLETED TO DATE	PERCENT COMPLETED THIS MONTH	PERCENT COMPLETED TO DATE	MANHOURS EXPENDED THIS MONTH	MANHOURS REMAINING
AUDITS REVIEWED	296	0	296	2	82	40	210

CURRENT STATUS AND RESULTS

A. PRELIMINARY EVALUATION INDICATES THAT MOST AUDITS WERE OF LIMITED SCOPE, HOWEVER, COLLECTIVELY DUE TO THE LARGE NUMBER OF AUDITS PERFORMED, THE APPLICABLE 18 CRITERIA WERE COVERED FOR HJK, EOTD, NED, GCD, AND NPD. THERE WERE AREAS NOT COVERED IN AUDITS OF S&L AND GE SUBSEQUENT TO APRIL 8, 1981. HOWEVER, CURRENT AUDITS OF S&L AND GE HAVE COVERED THE MAJORITY OF APPLICABLE CRITERIA.

B. PROCEDURE FOR INTERFACE WITH THE QCP TASKS TO VERIFY ADEQUACY OF WORK HAS BEEN DEVELOPED. THIS PROCEDURE IS BEING REWRITTEN AND WILL BE SUBMITTED FOR REVIEW.

COMMENTS

A. IT IS ESTIMATED THAT 30 MAN DAYS WILL BE REQUIRED FOR THE EVALUATION OF THESE AUDITS.

STATUS

THIS TASK IS 82% COMPLETE

ESTIMATED COMPLETION DATE

OCTOBER 15, 1982 - REVIEWS COMPLETE

NOVEMBER 15, 1982 FINAL REPORT AND EVALUATIONS.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Nunzio J. Palladino, Chairman
Victor Gilinsky
John F. Ahearne
Thomas M. Roberts
James K. Asselstine

'82 NOV 12 P4:50

SERVED NOV 12 1982

In the Matter of CINCINNATI GAS & ELECTRIC COMPANY (William H. Zimmer Nuclear Power Station)	Docket No. 50-358 Construction Permit ~~No.~~ CPPR-88 EA 82-129

ORDER TO SHOW CAUSE AND
ORDER IMMEDIATELY SUSPENDING CONSTRUCTION
(CLI-82-33)

I.

The Cincinnati Gas and Electric Company (CG&E) holds Construction Permit No. CPPR-88 which was issued by the Commission in 1972. The permit authorizes the construction of the William H. Zimmer Nuclear Power Station Unit 1, a boiling water reactor to be used for the commercial generation of electric power. The Zimmer plant is located on the licensee's site in Moscow, Ohio.

II.

A. Initial Identification of QA Problems

In early 1981 the NRC conducted an investigation into allegations made by present and former Zimmer site employees and by the Government Accountability Project. The NRC investigation revealed a widespread breakdown in CG&E's management of the Zimmer project as evidenced by numerous examples of non-compliance with twelve of the eighteen quality assurance Criteria of

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Attachment III

NOV 22 1982

DS02

Appendix B to 10 CFR Part 50. Consequently, CG&E paid a civil penalty of $200,000 for the failure to implement an acceptable quality assurance program, false quality assurance documents, and intimidation and harassment of quality control inspectors. (See Notice of Violation and Proposed Imposition of Civil Penalties, dated November 24, 1981 and Investigation Report No. 50-358/81-13.) In addition CG&E agreed to take actions to correct identified QA failures and prevent their recurrence and to determine quality of completed construction work.

1. <u>Actions to Correct Identified QA Failures and Prevent Recurrence</u>

A meeting was conducted by Region III on March 31, 1981, and the utility agreed to implement ten actions to correct quality assurance failures identified during the January - March 1981 investigation and to preclude their recurrence. These actions included: (1) increasing the size and technical expertise of the CG&E QA organization; (2) taking action to assure independence and separation of the QA/QC function performed by Kaiser from the construction function; (3) conducting 100% reinspections of the quality control (QC) inspections performed after that date by Kaiser and other contractors; (4) reviewing for adequacy, and revising as appropriate, all QC inspection procedures; (5) training QA/QC personnel on new and revised procedures; (6) reviewing for adequacy, and revising as appropriate, the procedures governing the identification, reporting, and resolution of deviations from codes and Final Safety Analysis Report (FSAR) statements; (7) reviewing for adequacy the procedures governing nonconformance reporting and justifying the disposition of each voided nonconformance report; (8) establishing an adequate program for control of QA and QC records; (9) performing a 100% review of all future surveillance and nonconformance reports written by contractor personnel; and (10) reviewing and revising the CG&E audit program so that it included technical audits of construction work and more comprehensive and effective programmatic audits. These commitments were confirmed in an Immediate Action Letter to the licensee on April 8, 1981.

2. Actions to Determine Quality of Completed Construction Work

Following the identification in 1981 of significant quality assurance problems and related management breakdowns, CG&E agreed to establish a comprehensive program to determine the quality of the completed construction work. The Quality Confirmation Program (QCP) was submitted to the NRC by the licensee on August 21, 1981. The QCP addressed problems identified by the investigation in the following areas: (1) structural steel; (2) weld quality; (3) traceability of heat numbers on piping; (4) socket weld fitup; (5) radiographs; (6) electrical cable separation; (7) nonconformance reports; (8) design control and verification; (9) design document changes; (10) subcontractor QA programs; and (11) audits.

3. Results of Actions Taken by the Licensee to Determine the Quality of Completed Construction Work

Many construction deficiencies have been identified by the licensee during the conduct of the QCP and other quality reviews and reported to the NRC pursuant to 10 CFR 50.55(e) which could have been prevented or identified in a timely manner by the licensee and its contractors had there been a properly managed QA program. Major construction deficiencies identified to date by the quality reviews are listed in order of identification and include the following:

- Welds performed using an unqualified welding procedure for welds greater than 0.864 inches.
- Unauthorized stamping of fittings and use of "high-stress" stamps.
- ASME structural weld and welder qualification deficiencies.
- Welds performed and welders not qualified for weld thickness range per ASME requirements.
- Approximately 2400 feet of small bore piping identified with questionable heat treatment.
- Welder qualifications with a substantial number of documentation discrepancies.
- Carbon steel weld rod may have been used for a portion of several stainless steel recirculation line welds.

- Electrical cable tray installation and inspection deficiencies.

- Hangers installed for the control rod drive system are of indeterminate quality.

- Both weld and radiograph quality deficiencies for sacrificial shield welds and radiograph deficiencies identified for the containment monorail and the ventilation stack.

- Deficiencies in the H. J. Kaiser procurement program for structural steel and other materials.

- Inadequate design control by Sargent & Lundy (architect engineer) for electrical separation.

- Inadequate weld preparation prior to radiography (ripples not removed) which caused masking of discontinuities in some welds.

- Reactor control, reactor protection, and neutron monitoring panels, including field installed wiring do not, in some cases, conform to design drawings with regard to cable separation.

- Inadequate engagement of "gamma plugs" in large-bore piping and lack of heat number traceability of the "gamma plugs." (During radiography of a pipe weld, a gamma source is sometimes inserted through a small hole in the side of the pipe. After radiography the hole is plugged to provide a pressure boundary.)

- Inadequate inspection program and installation procedures for "Nelson stud" installation for cable tray hangers.

- Concrete and steel coating program not in accordance with the QA Program and the Sargent & Lundy specification requirements.

- Design changes made to the Fire Protection System piping in the cable spreading room in 1979 were inadequately controlled.

- The Sargent & Lundy (architect engineer) dynamic stress analysis of small bore piping is questionable.

- Cable separation problem with regard to division separation between non-essential cables being bundled with essential cables of different divisions.

- Pipe support installation procedures did not contain seismic clearance criteria between pipe supports and cable trays or conduit and associated supports as required by the specification.

These deficiencies represent those which the staff considers most significant. There were additional 10 CFR 50.55(e) reports made by the licensee and the licensee has identified a large number of nonconformances (which could reflect construction or other types of deficiencies). As of September 30, 1982 the licensee's continuing quality confirmation program reviews had identified approximately 4,200 nonconformances of which about 800 have been "dispositioned", i.e., the licensee had made a determination as to resolution. (Inspection Report No. 50-358/82-12, report pending.) The large number of noncomformance reports and the significance of the matters being identified corroborate the staff's 1981 finding of significant breakdown in the licensee's quality assurance program.

B. Findings Subsequent to Licensee Actions Taken to Correct QA Failures and Prevent Recurrence

Since the Immediate Action Letter was issued on April 8, 1981 and quality assurance and management deficiencies were brought to the attention of the licensee, hardware and programmatic QA/QC problems have been identified by the NRC and the National Board of Boiler and Pressure Vessel Inspectors. These problems are discussed in the following paragraphs and indicate the licensee and the constructor are still having difficulty implementing satisfactory QA/QC programs:

During an inspection conducted the latter part of 1981 and the early part of 1982 (Inspection Report No. 50-358/82-01, issued on June 24, 1982) three items of noncompliance were identified. The findings concerned (1) the failure to clearly establish and document the authorities and duties of all QA Department personnel, (2) the failure to provide adequate certification of qualifications of all QA Department personnel, and (3) the failure to provide adequate procedures. The licensee failed to adequately address the provisions of Regulatory Guide 1.58 (ANSI N45.2.6-1978) concerning personnel in the QA Department. Additionally, inadequately qualified personnel were reviewing and approving quality procedures controlling electrical activities, which contained deficiencies.

Furthermore, as a result of the licensee reviews it was revealed that some weld inspectors involved in the QCP Task I, Structural Steel, were not adequately certified and the task was stopped. The task was restarted following upgrade of the inspectors through training provided by additional certified weld inspectors.

During an inspection conducted in March and April 1982 (Inspection Report No. 50-358/82-05, issued on July 1, 1982) two items of noncompliance were identified. The findings concerned the lack of implementation and timeliness of corrective actions and the failure to adequately review and document potentially reportable matters.

During an inspection conducted in April, May, and June of 1982 (Inspection Report No. 50-358/82-06, issued on November 2, 1982) two items of noncompliance were identified. The findings concerned (1) the performance of quality activities required of the welding engineers by inadequately qualified clerks and (2) the failure to perform required calibrations during a critical quality activity, Induction Heating Stress Improvement (IHSI) program.

A recent inspection conducted during June and July of 1982 (Inspection Report No. 50-358/82-10, report pending) identified a number of significant concerns. These concerns were discussed with the licensee on July 9, July 15, August 15, and October 19, 1982. Four significant items of concern (potential items of noncompliance) were identified: (1) the inadequate control and documentation of welder qualifications; (2) the failure to take corrective actions following the identification of inadequate records to support welder qualifications; (3) the unauthorized correction, supplementation, and alteration of quality records; and (4) the failure to follow procedures controlling weld filler metal control, logging and control of requests for information/evaluation, and imposition of reporting requirements on contractors. The NRC findings concerning welder qualifications resulted in the requalification of approximately 100 active onsite welders and the need for the licensee to develop a program to evaluate the previous work of the welders whose qualifications were not adequately documented.

An inspection was conducted following notification of the Region III Office that a CG&E Stop Work Order (SWO) had been initiated on August 5, 1982, pertaining to Catalytic, Inc. (CI) activities in the area of the control rod drive system hangers and supports. CI is a contractor of the licensee performing construction work including rework activities identified by the QCP program. During this inspection conducted during August and September of 1982 (Inspection Report No. 50-358/82-13, report pending), significant concerns were identified regarding the implementation of CG&E's quality assurance program and its management program established to control and monitor the activities of Catalytic, Inc. (CI). The concerns involved the areas of (1) the description of organization and functional interfaces, (2) training of CI personnel, (3) design control measures, (4) procedure content and implementation, (5) document control, (6) inspection and surveillance activities, (7) nonconforming conditions, (8) corrective actions, (9) records, and (10) audits. The findings were discussed with the licensee on August 12, September 10 and 17, and October 19, 1982.

As a result of the inspection findings and subsequent discussions with the licensee, Stop Work Orders were issued by the licensee, stopping all essential work by CI on October 11, 1982, pending resolution of the programmatic problems identified by the NRC and licensee reviews.

The licensee has initiated Stop Work Orders in addition to those affecting CI due to inadequate quality assurance in the areas of application of coatings (October 12, 1982), electrical cable installation (October 12, 1982), and special process procedures (November 1, 1982). The Stop Work Orders involve ongoing activities. The November 1, 1982 Stop Work Order involved procedures not meeting requirements notwithstanding that the procedures had been specifically reviewed by CG&E for adequacy subsequent to the issuance of the April 8, 1981 Immediate Action Letter.

Additionally, during the week of October 10, 1982, the Authorized Nuclear Inspector (ANI) for the N-stamp holder (H. J. Kaiser) recalled ASME work packages then being used in the field because of the performance of ASME code work (hanger attachment removal and piping cutouts) was outside the approved QA Program procedures. The ASME code work was being controlled and performed utilizing an H. J. Kaiser administrative memo which bypassed the ANI's required involvement in the code activities. The NRC was apprised of the required corrective actions during a meeting involving CG&E and H. J. Kaiser on October 15, 1982. The corrective actions taken and planned were considered acceptable by the Authorized Nuclear Inspector.

The National Board of Boiler and Pressure Vessel Inspectors, at the request of the State of Ohio, have been onsite since March 1, 1982. The National Board has issued three interim reports documenting findings regarding ASME code activities. The National Board findings include deficiencies in the following areas regarding on-going ASME code activities: design control, procurement, procedures, special processes, nonconforming conditions, and corrective actions. The findings are generally consistent with past and present NRC findings.

C. Rework Activities

As a result of the information obtained from the licensee's reviews of plant quality, the licensee is proceeding, prior to completion of the relevant QCP tasks, to initiate rework activities. A major example of rework activities is the area of structural steel welding. The reinspection and rework of structural steel welds located in a number of areas of the plant have been in process for a number of months. Approximately 70 percent of the structural welds are being reworked to make the welds acceptable. In the case of these welds, rework is being undertaken prior to the completion of the quality reviews to determine the acceptability of all structural steel welds and beam/hanger materials. The rework of these welds prematurely may result in the addition of new weld material over unacceptable weld material or beam/hanger materials. Following completion of the quality reviews unacceptable areas may require additional rework activities. This approach to rework activities indicates a lack of a comprehensive management program to address rework activities and the safety impact of those activities on the facility.

III.

The foregoing information indicates that: 1) the Zimmer facility has been constructed without an adequate quality assurance (QA) program to govern construction and to monitor its quality, resulting in the construction of a facility which currently is of indeterminate quality; 2) substantial efforts are underway to determine the quality of past construction activities and numerous construction deficiencies have been identified and are continuing to be identified such that both reanalysis and rework will be required to bring the facility into conformance with the application and regulatory standards on the basis of which the construction permit was originally issued; and 3) rework of deficiencies identified by the Quality Confirmation Program (QCP) has been undertaken prior to completion of other relevant QCP tasks and other reviews, resulting in the potential for additional reworking of the same item if further deficiencies are found, as has been the case, by the quality reviews. Consequently, the NRC presently lacks reasonable assurance that the Zimmer plant is being constructed in conformance with the terms of its construction permit and 10 CFR Part 50, Appendix B, and that there is adequate management control over the Zimmer project to ensure that NRC requirements are being met.

The verification of the facility's quality and appropriate actions to correct deficiencies in construction are of utmost importance to the public health and safety should the licensee receive a license to operate the facility. Moreover, the licensee must be in a position to assure that its construction activities have been properly carried out in accordance with Commission requirements, as the Commission inspectors are not able to personally verify every individual aspect of construction that may impact on safety. In view of the importance to safety of construction verification and corrective actions and the past pattern of quality assurance deficiencies, the Commission has concluded that safety-related construction, including rework activities, should be suspended until there is reasonable assurance that future construction activities will be appropriately managed to assure that rework activities and all other construction activities will be conducted in accordance with 10 CFR Part 50, Appendix B, and other Commission requirements. The Commission has further determined that in light of the foregoing considerations the public health, safety and interest require suspension of construction, effective immediately pending further authorization.

IV.

Accordingly, pursuant to sections 103, 161i, 182 and 186 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT:

A. Effective immediately, safety-related construction activities, including rework of identified deficient construction, shall be suspended.

B. The licensee shall show cause why safety-related construction activities, including reworking activities, should not remain suspended until the licensee:

(1) Has obtained an independent review of its management of the Zimmer project, including its quality assurance program and its quality verification program, to determine measures needed to ensure that construction of the Zimmer plant can be completed in conformance with the Commission's regulations and construction permit.

(a) The independent organization conducting this review shall be knowledgeable in QA/QC matters and nuclear plant construction and shall be acceptable to the Regional Administrator. The independent organization shall make recommendations to the licensee regarding necessary steps to ensure that the construction of the facility can be completed in conformance with the Commission's regulations and the construction permit. A copy of the independent organization's recommendations and all exchanges of correspondence, including drafts, between the independent organization and CG&E shall be submitted to the Regional Administrator at the same time as they are submitted to the licensee. In making recommendations, the independent organization shall consider at a minimum the following alternatives for management of the Zimmer project and shall weigh the advantages and disadvantages of each alternative:

1. Strengthening the present CG&E organization.
2. Creation of an organizational structure where the construction management of the project is conducted by an experienced outside organization reporting to the chief executive officer of CG&E.
3. Creation of an organizational structure where the quality assurance program is conducted by an experienced outside organization reporting to the chief executive officer of CG&E.
4. Creation of an organizational structure with both quality assurance and construction project management conducted by an experienced outside

organization reporting to the chief executive officer of CG&E.

(b) The licensee shall submit to the Regional Administrator the licensee's recommended course of action on the basis of this independent review. In evaluating the recommendations of the independent organization, the licensee shall address why it selected particular alternatives and rejected others. The licensee's recommendations and its schedule for implementation of those recommendations shall be subject to approval by the Regional Administrator.

(2) Following the Regional Administrator's approval in accordance with section IV B(1)(b),

(a) Has submitted to the Regional Administrator an updated comprehensive plan to verify the quality of construction of the Zimmer facility and the Regional Administrator of NRC Region III has approved such plan. In preparing this updated comprehensive plan, the licensee shall review the ongoing Quality Confirmation Program to determine whether its scope and depth should be expanded in light of the hardware and programmatic problems identified to date. The updated plan shall include an audit by a qualified outside organization, which did not perform the activities being audited, to verify the adequacy of the quality of construction; and

(b) Has submitted to the Regional Administrator a comprehensive plan, based on the results of the verification program, for the continuation of construction, including reworking activities, and the Regional Administrator has confirmed in writing that there is reasonable assurance that construction will proceed in an orderly manner and will be conducted in accordance with the requirements of the Commission's regulations and the Construction Permit No. CPPR-88.

(3) The Regional Administrator may relax all or part of the conditions of section IV.B for resumption of specified construction activities, provided such activities can be conducted in accordance with the Commission's regulations and the provisions of the construction permit.

V.

Within 25 days of the date of this order, the licensee may show cause why the actions described in section IV should not be ordered by filing a written answer under oath or affirmation that sets forth the matters of fact and law on which the licensee relies. As provided in 10 CFR 2.202(d), the licensee may answer by consenting to the order proposed in section IV of this order to show cause. Upon the licensee's consent, the terms of section IV.B of this order will become effective. Alternatively, the licensee may request a hearing on this order within 25 days after the issuance of this order. Any request for a hearing or answer to this order shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. A copy of the request or answer shall also be sent to the Director, Office of Inspection and Enforcement, and to the Executive Legal Director at the same address, and to the Regional Administrator, NRC Region III, 799 Roosevelt Road, Glen Ellyn, Illinois 60137. A request for a hearing shall not stay the immediate effectiveness of section IV.A of this Order.

If the licensee requests a hearing on this order, the Commission will issue an order designating the time and place of hearing. If a hearing is held, the issues to be considered at such a hearing shall be whether the facts set forth in sections II and III of this order are true and whether this order should be sustained.

Commissioners Ahearne and Roberts dissent from this decision. Their dissenting views are attached.

It is so ORDERED.

For the Commission

John C. Hoyle
Acting Secretary of the Commission

Dated at Washington, D.C.
this 12th day of November, 1982.

DISSENTING VIEWS OF COMMISSIONER AHEARNE

I agree with both the substance and the direction for change described in this order. However, I would have simply issued a Show Cause Order and would not have made it immediately effective.

DISSENTING VIEW OF COMMISSIONER ROBERTS

I disagree with the action taken by the Commission majority on several grounds. First, I believe the Commission's action in immediately suspending construction at the Zimmer facility is precipitous. Earlier this year, Cincinnati Gas and Electric Company (CG&E) made substantial changes in its management structure in order to manage more effectively construction activities and to monitor more carefully quality assurance programs. Despite the fact that this new organizational structure is relatively untested, the Commission is now suspending effective immediately all construction and corrective actions at the site. Additionally, the NRC Staff admits that CG&E's enhanced Quality Confirmation Program (QCP) and large quality control staff is effectively identifying existing construction problems. Moreover, to the extent that actual construction deficiencies have been found, CG&E's management has demonstrated its willingness to take strong remedial actions by issuing stop work orders in those areas where construction deficiencies have been found. In a plant that is approximately 98 percent complete, the Commission is requiring the relatively few remaining construction activities and the ongoing corrective actions necessitated by the QCP to stop immediately while additional organizational changes are implemented.

Second, I believe the Commission's action does not comport with its own practice. In Licensees Authorized to Possess . . . Special Nuclear Materials, CLI-77-3, 5 NRC 16, 20 (1977), the Commission said that "[a]vailable information must demonstrate the need for [such] emergency actions and the insufficiency of less drastic measures" (emphasis added). See also Consumers Power Co. (Midland Plant, Units 1 & 2), CLI-73-38, 6 AEC 1082, 1083 (1973). I believe that, in this case, some of the less drastic alternatives proposed by the Staff would be adequate to resolve the problems at this facility. For example, the Commission could send CG&E a letter indicating that at this time the Commission does not have sufficient information to conclude that Zimmer has been constructed in substantial conformance with the construction permit. The Commission could request the provision of information on the part of CG&E which, if available, would provide the Commission with the necessary assurance. See 10 CFR 50.54(f).

Third, in the absence of willfulness, the Commission may suspend construction effective immediately in accordance with Section 9b of the Administrative Procedures Act and the Commission's regulations only if the Commission finds that the public health, safety, or interest requires such action. I do not believe that the concerns listed in the Commission's Order show that the public health and safety requires immediate suspension of all construction and corrective actions at the Zimmer site. Indeed, Mr. James Keppler, the Region III Administrator, has stated that CG&E's QCP has been successful in identifying existing construction problems. Transcript of Public Meeting on the Status of Zimmer, October 28, 1982 at 5. Additionally, most of the NRC inspection findings arising out of the QCP point to administrative or procedural deficiencies, rather than to actual material or construction errors. While the NRC's level of confidence in the adequacy of the plant construction has been reduced, it has not been shown by the NRC that problems exist which require immediate resolution to protect the public health and safety. Moreover, I do not believe this action is in the public interest.

I am also concerned that the Order has been approved without consideration for the Applicant's proposal to correct management and construction problems. That proposal, outlined in a letter to the Commissioners dated November 10, 1982, contained all of the essential elements approved by this Order. Specifically, the proposal calls for obtaining new project management, stopping all rework on quality confirmation matters, and an independent third party review to confirm the acceptability of selected safety systems. In view of the voluntary agreement by CG&E to such drastic measures, I feel that this Order is primarily punitive in nature and does little to correct problems in the interest of public health and safety.

Finally, I disagree with the Commission's Order because of the potential for delay inherent in this procedure. CG&E has an absolute right to a hearing on the Commission's Order. If CG&E avails itself of this right, then other "interested persons" will be entitled to demand a hearing. Once started, the hearing would be difficult to bring to an expeditious close. Even if the Staff and CG&E were to reach agreement on the corrective actions to be taken, litigation of the requirements imposed by the Commission Order would continue. Consumers Power Co. (Midland Plant, Units 1 & 2), ALAB-315, 3 NRC 101 (1976); Dairyland Power Cooperative (LaCrosse Boiling Water Reactor), LBP-81-7, 13 NRC 257, 264-65 (1981).

THE CINCINNATI GAS & ELECTRIC COMPANY

CINCINNATI, OHIO 45201

B. R. SYLVIA
VICE PRESIDENT
NUCLEAR OPERATIONS

November 15, 1982

U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Attention: Mr. J.G. Keppler
Regional Administrator

RE: WM. H. ZIMMER NUCLEAR POWER STATION - UNIT 1 - NRC ALLEGATIONS - WELDER QUALIFICATION - DOCKET NO. 50-358, CONSTRUCTION PERMIT NO. CPPR-88, W.O. #57300, JOB E-5590 - FILE NO. NRC-19 & I.E. INSPECTION REPORT NO. ___

Gentlemen:

By letter dated October 27, 1982, CG&E was notified by the NRC that "NRC Region III has been advised of allegations that relevant documentation on the welders at the Zimmer Site was prepared for, or reviewed at, a meeting between Cincinnati Gas & Electric Company (CG&E) and H.J. Kaiser (HJK) held on July 8, 1982, but that such documentation was not discussed with, or made available to, Region III at the meeting on this subject held on July 9, 1982 among CG&E, HJK and Region III. Substantial documentation was made available to Region III in connection with the July 9 meeting, but additional relevant documentation was allegedly not made available."

The July 8, 1982 meeting between CG&E and HJK was a routine Zimmer Site management meeting. One of the topics scheduled for discussion near the end of the meeting was the resolution of current welder performance qualification. These concerns over welder performance qualification records were identified earlier in 10CFR50.55(e) Report M-45 and in numerous HJK and CG&E nonconformance and Corrective Action Reports which were available to the NRC. Generic categories of document deficiencies identified in these CARs and NRs were discussed at the meeting for the purpose of

Attachment IV

NOV 18 1982

determining resolutions acceptable to CG&E and HJK management. The results of this discussion were presented to the NRC in a meeting on July 9, 1982. Additional supporting documentation was subsequently made available to the NRC.

CG&E's approach to formulating a response to the aforementioned allegation was to seek information from each individual who attended the July 8, 1982 meeting. Everyone attending the meeting was asked to identify documentation or reports made available or discussed at the meeting, or used in preparing for the meeting.

On the basis of the information given to me by the meeting participants, and my personal knowledge, I have determined that no reports or documentation other than that already made available to Region III were prepared for use at either the July 8, or July 9, 1982 meeting. A list of documents or reports either used at the July 8 and/or July 9, 1982 meetings or used in preparation for these meetings is provided as Attachment 1. This list is a compilation of the information provided by each of the July 8, 1982 meeting attendees. This list, of course, does not include all information generated as part of the general review of welder performance qualification records not prepared specifically for or discussed at the July 8, or July 9, 1982 meetings. These records have been and are available for NRC review.

Very truly yours,

By B.R. Sylvia
B.R. Sylvia
Vice President - Nuclear Operations

BRS/bcf
Attachments
cc: D. Hunter
W.F. Christianson

State of Ohio)
County of Clermont) ss

Sworn to and subscribed before me this
15th day of November, 1982

[illegible signature]
Notary Public
My commission expires [illegible]

ATTACHMENT 1

HJK QA Documentation Group

A) Current Welder Qualification Status Report (July 9, 1982)
B) Current Welders with Gladstone Laboratories Test
C) Welder Status Index
D) Current Welders Status
E) QA Records Review Welder Status Checklist

HJK Welding Department

A) Welder Qualification List

CG&E Quality Confirmation Program

A) QCP Welder Qualification Log generated by 19-QA-21
B) Welder Qualification Record: #50-238
C) Welder Qualification Record: #9-3174
D) Welder Qualification Record: Number not recalled

Other documents referenced during the July 8, 1982 meeting

A) HJK Procedure: WCP-2
B) HJK Procedure: SPPM 3.2, R.4
C) HJK Procedure: SPPM 3.2, R.7
D) Various CARs and NRs referenced in the above documents.

C F BRAUN & CO Engineering and Construction Subsidiary of Santa Fe International Corporation

November 24, 1982

B R Shelton
Commonwealth Edison Co
SNED-35 FNB
P O Box 767
Chicago, IL 60690

BL-26

Dear Mr. Shelton

ADDITIONAL MATERIAL TESTS
LA SALLE STATION
ADVANCE PURCHASE ORDER 805-023
BRAUN PROJECT 6356-N

On November 19, we received additional material sampling data from the Nuclear Regulatory Commission Region III office to supplement the material addressed in Section 5.3 MATERIAL of the October 27, 1982, Independent HVAC Review Final Report. The data has been reviewed by our welding and material specialists. His comments, and a copy of the information received from the NRC Region III office, are attached.

Our specialist agrees with, or has no comment on, the NRC data. For attachments 4, 5 & 6 "No substantive comments" means that the information in the letters is technically correct but there are some editorial corrections. For example, 163° pyramid indenter should be 136° per ASTM E92.

Sincerely yours

AJK df

Andrew J Kempiak
Project Manager

Dr. A Bournia
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NR C Docket Nos. 50-373/374

James G Keppler, Regional Administrator -3 copies
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Daniel L Shamblin -2 copies
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La Salle County Station
Section 1 through 5 only

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C F Braun & Co
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Alhambra, Ca 91802

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Gerald J Diederich
Superintendent
La Salle County Station
R.R. #1
Marseilles, IL 61341

CF BRAUN & CO A Subsidiary of Santa Fe International Corporation

Date NOVEMBER 23, 1982

To ANDY KEMPIAK
POWER DIVISION

From LEONARD BOYD
RESEARCH

On 6356-N, COMMONWEALTH EDISON, LA SALLE 1, MATERIAL PROPERTIES

On November 19 the U S Nuclear Regulatory Commission's (USNRC) Region III office telecopied data from their material sampling program that was not included in their transmittal to C F Braun on October 7. Copies of the six documents are attached. The identification of the documents and my comments are listed below.

ATTACHMENT 1
1 Page

Letter dated November 19 from US NRC, C E Cornelius, to C F Braun, W L Stiebe, transmittal letter for other attachments.

No comment.

ATTACHMENT 2
6 pages

Sargent and Lundy Responses to NRC Region III "Questions on HVAC System"

Question 1 I agree with S & L's discussion of the effects of increased hardness of bolting material. Although the reported hardness of 287 Brinell is well above the maximum of 241 allowed for A307 Grade A bolts, it is near the middle of the range of 248 to 331 Brinell for A325, High-Strength Bolts for Structural Steel Joints. A325 is a commonly specified bolting material when higher strength is required.

Questions 2 and 3. No comments.

ATTACHMENT 3
4 pages

Results of Sample Analysis (56 samples)

My comments on the first 48 samples are contained in my pink letter to you dated October 13.*

Samples 49 and 50 conform to the chemical requirements for A575, Grade M1015, and are acceptable for use in accordance with ASTM A29, paragraph 4.3.1, because no misapplication is indicated.

Samples 51 through 56 are for A307 bolts and A563 nuts. The results shown are within the specification requirements.

*The first 48 samples are discussed in Section 5.3 MATERIAL and Appendix L of the final report.

File

21/3-82

Attachment 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

November 19, 1982

C. F. Braun & Company
ATTN: Mr. Wayne L. Stiebe
Assistant Vice-President
1000 South Fremont Avenue
Alhambra, California 91802

Gentlemen:

Please find attached additional data we obtained as part of our material sampling program at LaSalle County Nuclear Station which is an addition to the data provided to you on October 7, 1982.

If you have any questions on the above, please contact Nick Jackiw or Roger Lanksbury of my staff at (312) 932-2500.

Sincerely,

C. E. Norelius

C. E. Norelius, Director
Division of Engineering
and Technical Programs

Attachments: As Stated

cc: T. Novak, NRR
A. Bournia, NRR
L. Spessard

SARGENT & LUNDY
ENGINEERS
CHICAGO

Attachment 2

NRC REGION III

QUESTIONS ON HVAC SYSTEMS

Question 1: The NRC tested 4 bolts which were 3/8 inch A307 Grade A. The ASTM hardness requirement is between 121 and 241 Brinell. One of the 4 bolts tested had a hardness of 287 Brinell. The chemical analysis was OK. The NRC's lab could not conduct elongation or tensile tests on a 3/8 inch bolt. What does the 287 Brinell hardness mean?

S&L Response: With increasing hardness yield and fracture strength increases and ductility decreases. Increased hardness is advantageous from a strength point of view.

Increased hardness also decreases ductility of the bolt material. However, as no impact loads are expected on HVAC ductwork, this decreased ductility does not affect the safety of the bolts.

Hence, increased hardness is beneficial.

Attached is a list of stresses on the bolts obtained by conservative analysis of typical HVAC ductwork containing components (dampers, registers, grills, etc.).

Attachment 2

SARGENT & LUNDY
ENGINEERS
CHICAGO

Summary of findings for A307 bolts used in HVAC Duct. Companion flange analysis.

Note: 1) The analysis was conservatively based on highest duct and duct component weights.

2) Analysis was based on 3/8 inch bolts.

3) Yield stress for A307 is minimum 36000 psi.

Building - Reactor 1 and 2 Service Level: Emergency

Duct Size (W X H inches)	Calculated Stress (ksi)	Duct Size (W X H inches)	Calculated Stress (ksi)
10 X 6	8.0	30 X 14	7.544
10 X 6	10.0	30 X 20	8.323
12 X 8	9.6	32 X 20	4.633
12 X 10	10.7	36 X 30	9.92
12 X 12	9.6	40 X 20	9.102
14 X 10	10.6	40 X 36	7.995
16 X 16	10.33	42 X 18	7.831
18 X 8	9.76	42 X 36	5.95
18 X 12	9.35	48 X 16	9.76
18 X 14	8.94	48 X 32	5.002
18 X 18	9.92	60 X 40	2.624
24 X 24	10.05	72 X 60	5.054
24 X 18	8.99	96 X 40	6.44
24 X 20	7.18		
26 x 12	8.16		
26 X 14	9.184		
26 X 20	9.963		
28 X 14	9.512		
28 X 20	9.061		

SARGENT & LUNDY
ENGINEERS
CHICAGO

Building - Auxiliary

Service Level: Emergency

Duct Size (W X H inches)	Calculated Stress (ksi)
10 X 6	10.414
18 X 8	10.91
20 X 6	6.81
12 X 16	10.54
12 X 20	10.91
12 X 30	10.62
14 X 40	6.57
18 X 44	9.31
20 X 16	8.57
12 X 36	9.594
36 X 18	8.41
26 X 20	10.62
24 X 54	8.57
48 X 36	8.82
30 X 28	7.75
30 X 38	9.23
22 X 18	7.71
28 X 70	9.512
72 X 72	5.54
40 Ø	9.8
70 Ø	9.23

SARGENT & LUNDY
ENGINEERS
CHICAGO

Attachment 2

4

Building - Containment

Service Level: Emergency

Duct Size (W X H inches)	Calculated Stress (ksi)
12 X 24	8.16
32 X 10	8.1
30 X 32	8.364
18 Ø	9.27

Attachment 2

5

SARGENT & LUNDY
ENGINEERS
CHICAGO

Question 2: What are S&L's design factors for companion flanges and stiffeners?
flanges and stiffeners?

S&L Response: Between companion flanges and duct sheet metal, the sheet metal is the weaker member.

Stiffeners are not load carrying members. They are installed to prevent warping of the duct sheet metal. The stiffeners are installed to industry practice.

SARGENT & LUNDY
ENGINEERS
CHICAGO

Question 3: Was A500 structural tubing ever added to HVAC Spec?

S&L Response: A500 tubing was approved for use by Field Change Request (FCR) 34,307. It will soon be incorporated into the HVAC Spec.

Attachment 3

RESULTS OF SAMPLE ANALYSIS

Sample Number	ASTM* Material Type	System Removed From	System Component	Weight % C	Weight % S	Weight % P	Weight % Mn	Tensile Strength, KSI Calculated Min.	Tensile Strength, KSI ASTM Min.
1	A36	VE	Hanger	0.19	0.035	0.003	0.68	-	58
2	A527	VC	Duct	0.07	0.027	0.005	0.38	N/A	N/A
3	A36	VC	Hanger	0.15	0.033	0.006	0.69	63	58
4	A575	VC	Stiffner	0.18	0.030	0.009	0.49	N/A	N/A
5	A36	VE	Hanger	0.23	0.038	0.012	0.68	-	58
6	A36	VE	Hanger	0.21	0.026	0.006	0.68	-	58
7	A527	VE	Duct	0.07	0.031	0.006	0.35	N/A	N/A
8	A575	VE	Stiffner	0.16	0.036	0.018	0.56	N/A	N/A
9	A36	VC	Hanger	0.18	0.029	0.006	0.75	69	58
10	A36	VC	Hanger	0.21	0.032 0.033	0.001	0.86	-	58
11	A36	VC	Hanger	0.20	0.041	0.027	0.38	-	58
12	A527	VX	Duct	0.05	0.019 0.018	0.005	0.41	N/A	N/A
13	A36	VX	Hanger	0.18	0.035	0.012	0.62	-	58
14	A36	VX	Hanger	0.16	0.035	0.006	0.65	65	58
15	A36	VX	Hanger	0.15	0.035	0.008	0.54	-	58
16	A36	VE	Hanger	0.19	0.046	0.023	0.73	65	58
17	A36	VX	Hanger	0.19	0.032	0.005	0.61	-	58
18	A575	VX	Companion Flange	0.22	0.031 0.032	0.027	0.38	N/A	N/A
19	A575	VD	Hanger	0.20	0.025	0.007	0.50	N/A	N/A

*All ASTM A575 Samples are specified to be Grade M1020, except sample [illegible] which was M[illegible]

8

RESULTS OF SAMPLE ANALYSIS

Sample Number	ASTM* Material Type	System Removed From	System Component	Weight % C	Weight % S	Weight % P	Weight % Mn	Tensile Strength, KSI Calculated Min.	Tensile Strength, KSI ASTM Min.
20	A527	VD	Duct	0.05	0.025 0.024	0.008	0.35	N/A	N/A
21	A575	VD	Companion Flange	0.23	0.026	0.005	0.52	N/A	N/A
[illegible]	A36	VD	Hanger	0.19	0.038	0.008	0.60	-	58
23	A36	VD	Hanger	0.19	0.036	0.009	0.68	64	58
24	A575	VD	Stiffner	0.20 0.19	0.026	0.008	0.61	N/A	N/A
25	A36	VY	Hanger	0.20	0.042	0.009	0.64	-	58
26	A575	VY	Stiffner	0.22	0.031	0.008	0.61	N/A	N/A
27	A36	VY	Hanger	0.17 0.18	0.037	0.007	0.56	-	58
28	A575	VY	Stiffner	0.21	0.029	0.008	0.58	N/A	N/A
29	A36	VY	Hanger	0.15	0.048	0.035 0.035	0.72	-	58
30	A36	VY	Hanger	0.21	0.030	0.009	0.72	-	58
31	A527	VD	Duct	0.08	0.02	0.006	0.42	N/A	N/A
32	A527	VD	Duct	0.05	0.02	0.009	0.30	N/A	N/A
33	A527	VD	Duct	0.06	0.02	0.006	0.43	N/A	N/A
[illegible]	A527	VY	Duct	0.06	0.03	0.007	0.28	N/A	N/A
35	A527	VY	Duct	0.06	0.02	0.007	0.28	N/A	N/A
36	A36	VY	Hanger	0.17 0.18	0.04	0.006	0.71	66	58

*All ASTM A575 Samples are specified to be Grade M1020, except sample 41 which was M1015.

RESULTS OF SAMPLE ANALYSIS

This is the bolt that is too hard

Sample Number	ASTM* Material Type	System Removed From	System Component	Weight % C	Weight % S	Weight % P	Weight % Mn	Tensile Strength, KSI Calculated Min.	Tensile Strength, KSI ASTM Min.
37	A527	VX	Duct	0.08	0.03	0.008	0.32	N/A	N/A
38	A307	VX	Bolt	-	0.03	0.011	0.69	143	60
39	A563	VX	Nut	0.05	-	0.009	0.39	97**	68**
40	A575	VX	Hanger	0.21	0.03	0.013	0.65 0.62	N/A	N/A
41	A575	VX	Stiffner	0.16	0.03	0.010	0.42	N/A	N/A
42	A575	VC	Hanger	0.19	0.04	0.018	0.64	N/A	N/A
43	A575	VC	Stiffner	0.19	0.04	0.012	0.38	N/A	N/A
44	A575	VC	Companion Flange	0.13 0.12	0.03	0.016	0.45	N/A	N/A
45	A527	VC	Duct	0.04	0.05 0.04	0.009	0.32	N/A	N/A
46	A527	VC	Companion Flange	0.14	0.04 0.03	0.016	0.42	N/A	N/A
47	A527	VE	Duct	0.09	0.02	0.006	0.40	N/A	N/A
48	A527	VC	Duct	0.07	0.03	0.011	0.33	N/A	N/A
49	A575	VC	Companion Flange	0.14	0.03	0.017	0.45	N/A	N/A
50	A575	VC	Companion Flange	0.13	0.03	0.017	0.46	N/A	N/A
51	A307	VX	Bolt	-	[illegible]	[illegible]	-	[illegible]	60
52	A563	VX	Nut	[illegible]	[illegible]	0.016	0.37	[illegible]**	[illegible]**

*All ASTM A575 samples are specified to be Grade M1020, except sample 41 which was [illegible].

**Proof load stress.

RESULTS OF SAMPLE ANALYSIS

Sample Number	ASTM* Material Type	System Removed From	System Component	Weight % C	Weight % S	Weight % P	Weight % MN	Tensile Strength, KSI Calculated Min.	Tensile Strength, KSI ASTM Min.
53	A307	VX	Bolt	-	0.02	0.014	-	102	60
54	A563	VX	Nut	0.09	0.02	0.009	0.50	87**	68**
55	A307	VR	Bolt	-	0.02	0.031	-	96	60
56	A563	VR	Nut	0.11	0.01	0.026	0.34	107**	68**

*All ASTM A575 Samples are specified to be Grade M[illegible], except sample [illegible] which was M[illegible].

**Proof load stress.

Attachment 4

ARGONNE NATIONAL LABORATORY

9700 South Cass Avenue, Argonne, Illinois 60439

Telephone 312-972-5137

August 6, 1982

Roger Lanksbury
USNRC
799 Roosevelt Rd.
Glen Ellyn, IL 60137

<u>SUBJECT</u>: Hardness Tests on A36 Specimens

Dear Roger:

Hardness tests were performed to determine whether the material samples which meet the chemical requirement for A36 material also meet the tensile requirements for this material. The requirements include a minimum and maximum tensile strength, a minimum yield strength, and minimum elongation requirements. These should be determined by uniaxial tensile tests. However, because of limitations on the amount of material available hardness tests offered the only means to estimate these properties. The relationship between hardness and flow stress is well known (see, e.g., F. McClintock and A. Argon, <u>Mechanical Behavior of Materials</u>). Because this material strain hardens substantially and because these specimens were probably subjected to significant deformation when they were removed, the flow stress is best interpreted as a lower bound on the tensile strength.

Five specimens of the nominal A36 were judged suitable for testing. These specimens were mounted and polished to removed ~20 mils of the most severely deformed material from the surface. Vickers hardness tests were then performed using a 163° pyramid indenter with a loading force of 30 kg. Some exploratory tests showed that the particular choice of the loading force had little effect on the measured hardness values. The mean diagonal of the impression d_1 was measured and the Vickers hardness V computed from

$$V = \frac{P}{A_C} = \frac{1.854P}{d_1^2} \quad (P\ \text{kg},\ d_1\ \text{mm})$$

where P is the loading force and A_C the contact area. The Meyer-Vickers hardness M_V which is based on the projected area A_P instead of the contact area A_C was computed from

$$M_V = \frac{2P}{d_1^2} = \frac{V}{0.927}\ \text{kg/mm}^2$$

Since the Meyer-Vickers hardness generally is a better representation of the average pressure on the area of contact it was used to compute the flow stress S_y from

$H_V = 3.2\ S_T$

The results from the five specimens are tabulated below, with S_T expressed in the usual American engineering units of psi (1 kg/mm^2 = 1422 psi).

Specimen	V (kg/mm^2)	H_V (kg/mm^2)	S_T (psi)
A36-3	136	137	65,000
A36-9	145	156	69,000
A36-14	136	147	65,000
A36-16	136	147	65,000
A36-23	134	145	64,000

Since the minimum tensile strength for A36 material is 58,000 psi, these results indicate that all these specimens do meet this specification.

The measurements and calculations presented here were actually performed by J. Y. Park and D. Perkins. Since I will be away for the next two weeks, if you have additional questions please call Jang Yul Park at 972-5030.

Sincerely,

William J. Shack

William J. Shack
Materials Science Division

WJS:dkm

cc: J. Y. Park
D. Perkins
R. W. Weeks

Attachment 5

ARGONNE NATIONAL LABORATORY

9700 South Cass Avenue, Argonne, Illinois 60439

Telephone 312/972-[illegible]

September 22, 1982

Mr. Roger Lanksbury
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

<u>SUBJECT</u>: Hardness Tests on Assorted Specimens

Dear Roger:

Hardness tests were performed to determine whether the material samples met the mechanical requirements of the appropriate ASTM specification. In some cases the requirements include a minimum and maximum tensile strength, a minimum yield strength, and minimum elongation requirements. These should be determined by uniaxial tensile tests. However, because of limitations on the amount of material available hardness tests offered the only means to estimate these properties. The relationship between hardness and flow stress is well known (see, e.g., F. McClintock and A. Argon, <u>Mechanical Behavior of Materials</u>).

Specimens were mounted and polished to remove ~30 mils of the most severely deformed material from the surface. Vickers hardness tests were then performed using a 163° pyramid indenter with a loading force of 30 kg. Some exploratory tests showed that the particular choice of the loading force had little effect on the measured hardness values. The mean diagonal of the impression d_i was measured and the Vickers hardness V computed from

$$V = \frac{P}{A_c} = \frac{1.854P}{d_i^2} \text{ (P kg, } d_i \text{ mm)} \qquad (1)$$

where P is the loading force and A_c the contact area. The Meyer-Vickers hardness M_v which is based on the projected area A_p instead of the contact area A_c was computed from

$$M_v = \frac{2P}{d_i^2} = \frac{V}{0.927} \text{ kg/mm}^2 \qquad (2)$$

Since the Meyer-Vickers hardness generally is a better representation of the average pressure on the area of contact it was used to compute the flow stress S_F from

$$M_V = 3.2\ S_F \tag{3}$$

The results are tabulated below with S_F expressed in the usual American engineering units of ksi (1 kg/mm² = 1.422 ksi).

Item	ASTM Specifications*	Measured Hardness V**	Measured Hardness M_V^{+}	Flow Stress S_F^{++}
Plate ASTM A36 Sample # 36	Tensile Strength 58-80 ksi Minimum Yield Point 36 ksi	137	148	66 ksi
Bolt ASTM A307 Grade A This is the bolt that is too hard. Sample # 38	Tensile Requirement ~~60~~ 42 ksi Brinell Hardness 121-241 min max Rockwell Hardness B69-B100 min max	303	327	145 ksi
Nut ASTM A563 Grade A Sample # 39	Proof Load Stress 68 ksi Brinell Hardness 116-302 min max Rockwell Hardness B69-C32 min max	203	219	97 ksi
Washer Unknown	This data not used since no spec. for washers.	99	107	48 ksi

*Parts of the required specifications. See "Annual Book of ASTM Standards" for full specifications.

**Vickers hardness, Eq. (1).

+Meyers hardness, Eq. (2). ++Eq. (3).

The flow stress obtained for the A36 plate is consistent with the tensile strength requirements for this material. The requirements for the bolt and nut are given in terms of Brinnell hardness rather than Vickers hardness. However, the two are approximately equal; a more accurate comparison of the two hardness measurements can be obtained from the ASM Metals Handbook, Vol. I, p. 1234. According to these results:

Vickers	is equivalent to	Brinnell	or	Rockwell
303		287		B106
203		193		B92
99		94		B56

Thus the nut seems to meet the hardness specification and the calculated flow stress is consistent with the proof load stress. However, the bolt appears to exceed the hardness specification which would tend to reduce the toughness on the bolt. We have made measurements on the washer, but it is not clear what specifications it should meet.

I hope this information will be of help to you. As before the measurements and calculations presented here were actually performed by J. Y. Park and D. Perkins.

Sincerely,

William J. Shack

William J. Shack
Materials Science and
Technology Division

WJS:dkm

cc: J. Y. Park
D. Perkins
R. W. Weeks

Attachment 6

ARGONNE NATIONAL LABORATORY

9700 South Cass Avenue, Argonne, Illinois 60439

Telephone 312/972-5137

October 7, 1982

Mr. Roger Lanksbury
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

SUBJECT: Hardness Test on Bolts and Nuts

Dear Roger:

Hardness tests were performed on the three bolts (1/2, 3/8, 1/4-in.) and the corresponding nuts to see if they conformed to the appropriate ASTM specification (A307 for the bolts, A563 for the nuts). The results of the tests are summarized below:

Our sample's	Specimen			Hardness Vickers	Hardness Brinell
51	No. 22	3/8-in.	Bolt	169	152
52			Nut	193	182
53	No. 23	1/4-in.	Bolt	211	201
54			Nut	190	181
55	No. 24	1/2-in.	Bolt	187	178
56			Nut	222	211

The tests were actually performed using a Vickers 163° pyramid indenter with a loading force of 30 kg. The correspondence between the Vickers and Brinell hardness follows the correlation given in the ASM Metals Handbook, Vol. 1, p. 1234.

The specimens do meet the required specifications, and in most cases are roughly in the middle of the range. The well known correspondence between hardness and flow indicate that they almost certainly meet the tensile requirements, but we have not checked them directly.

Sincerely,

William J. Shack
Materials Science & Technology Division

WJS:dkm

OCT 12 1982

PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 6 1982

NOTE TO: File (50-373)

FROM: Anthony Bournia, Project Manager
Licensing Branch No. 2, DL

SUBJECT: PROCESS DOCUMENT FROM C. F. BRAUN

Please process this document thru DCS. This is additional information to be included in the C. F. Braun HVAC Independent Review Report for La Salle Unit 1.

A. Bournia

Anthony Bournia, Project Manager
Licensing Branch No. 2, DL

cc: Jeff Bartlett

DEC 6 1982

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LB#2 File
ABournia

NOTE TO: File [redacted]

FROM: Anthony Bournia, Project Manager
Licensing Branch No. 2, DL

SUBJECT: PROCESS DOCUMENT FROM C. F. BRAUN

Please process this document thru DCS. This is additional information to be included in the C. F. Braun HVAC Independent Review Report for La Salle Unit 1.

/S/

Anthony Bournia, Project Manager
Licensing Branch No. 2, DL

cc: Jeff Bartlett

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Docket No.: 50-275

Philip A. Crane, Jr., Esq.
Pacific Gas and Electric Company
P. O. Box 7442
San Francisco, California 94120

Dear Mr. Crane:

At the October 20, 1982 Commission meeting on Diablo Canyon the staff indicated that their consultant, Brookhaven National Laboratory (BNL), would be requested to perform a limited number of additional independent analyses. In this regard, a discussion of the BNL effort was provided in Enclosure 2 of SECY Paper 82-414 dated October 13, 1982.

The purpose of the BNL analyses are to provide the staff with additional insight as to the character of results obtainable by use of state-of-the-art analytical techniques without regard to methods or procedures previously approved in the licensing process for Diablo Canyon. These analyses are therefore not intended as a substitute for the design and evaluation efforts now underway for the Diablo Canyon project; nor are they a substitute for the analytical efforts being performed by the Independent Design Verification Program (IDVP). Our experience has been, however, that such analyses often provide the bases for judgements that expedite the review process.

Attached is a list of the drawings and data that have been identified as necessary to perform the intended analyses. We request that you provide us with this information as soon as possible. Please advise us promptly if any of the requested information can not be provided within 2 weeks after receipt of this letter.

Sincerely,

Original signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

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P PDR

Enclosure:
As stated

cc: See next page

See previous

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HSchierling	
GKnighton	

Docket No.: 50-275

Philip A. Crane, Jr., Esq.
Pacific Gas and Electric Company
P. O. Box 7442
San Francisco, California 94120

Dear Mr. Crane:

The participation of Brookhaven National Laboratory (BNL) in the Diablo Canyon design verification effort as a staff consultant was discussed at the Commission meeting on October 20, 1982. Details of the BNL effort provided in Enclosure 2 of SECY Paper 82-414 of October 13, 1982.

Attached is a letter from BNL to the NRC which includes a request for structural drawings, for other material needed for structural analysis and items needed for piping analysis. We request that you provide us with the information identified as soon as possible. Please advise us promptly if any of the information cannot be submitted within two weeks after receipt of this letter.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: See next page

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cc: Mr. Richard Hubbard
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San Francisco, California 94120

Dr. Jose Rosset
3506 DuVal Road
Austin, Texas 78751

Dr. Morris Reich
Structural Analysis Division
Brookhaven National Laboratory
Upton, Long Island, New York 11973

Diablo Canyon Nuclear Power Plant

Structural Drawings Request

		*
438232	Interior Concrete Outline Plans at El 74'-8" & 91'0" Containment Structure	Rev. 10
438233	Interior Concrete Outline Plans at El 124'-0" & 140'-0" Containment Structure	Rev. 12
438234	Interior Concrete Outline-Main Sections Containment Structure	Rev. 12
438235	Interior Concrete Outline Miscellaneous Sections Containment Structure	Rev. 6
438239	Interior Concrete Reinforcing Slab at El. 91'-0" North Containment Structure Area F	Rev. 5
438240	Interior Concrete Reinforcing Slab at El. 91'-0" South Containment Structure Area G	Rev. 6
438274	Interior Concrete Details Reactor Nozzles Area Containment Structure	Rev. 4
438276	Detailed Plans at El 111 of Steam Generator Supports Containment Structure	Rev. 7
438278	Reactor Coolant Pump Supports at El 106 Containment Structure	Rev. 5
438280	Pressurizer Support Containment Structure Area F	Rev. 4
439571	Equipment Supports Plan Below El 113 Containment Structure Areas F & G	Rev. 8
439572	Layout of Lateral Support for Steam Generators at El 139'-0" Containment Structure Areas F & G	Rev. 2
439573	Steam Generator Support at El 139'-0" Containment Structure Area F & G	Rev. 6
438135	Requirement for 40,000 Gal. Diesel Fuel Oil Storage Tanks	Rev. 6

*It is understood the revision shown is the latest. If not, provide the latest revision.

Request for Other Material Needed for Structural Analysis

1. Equipment mass data similar to that attached to PG&E letter to J. Blume dated 9/13/82. Ref. Diablo Canyon Unit 1 Containment Structure.

2. Geotechnical Studies Intake Structure, Water Storage Tanks Diesel Fuel Oil Storage Tanks Diablo Canyon Nuclear Power Plant San Luis Obispo County Conf., HLA Job No. 569,031.04, Harding-Lawson Assoc., April 12, 1978.

3. Letter report from Harding Lawson dated October 11, 1982 updating the above (2) report.

4. 7175360 51 series Vertical Steam Generator Outline
   Sheets 1 of 2 and 2 of 2
   Bechtel #'s DC-663206-27-1
   and DC-663206-1-10

5. The following sections are required for the polar crane:
   (a) through the horizontal girders
   (b) at the top of the columns
   (c) at the bottom of the columns

6. The horizontal Newmark HOSGRI 7.5 M digitized time history for the Reactor Containment Building (cards plus printed copy of cards).

7. Horizontal surface HOSGRI 7.5 M Newmark digitized time history (cards plus card listing) for the buried diesel oil tank.

8. Latest sketches and/or drawings of modifications to the connections of the annulus structural steel.

9. Total weight and weight distribution of the steam generators.

Items Needed for Piping Analysis

A PG&E piping document package or its equivalent and computer input-output listings are requested for each of the following Diablo Canyon Nuclear Power Plant, Unit 1 piping problems;

A) PG&E No. 4A-122 (4A-26)
B) 6-102 (6-11)
C) 8-116, 8-117, 8-118
D) Westinghouse Problem
RHR Loop 4 (6-4, 6-7)

The document packages to include at least;

1. walkdown isometric
2. Hosgri envelope spectra
3. thermal and pressure operating modes from DCM-M-46
4. building displacements from DCM C-28
5. pipe specification and material data
6. support descriptions
7. equipment movements if appropriate
8. applied spring forces if appropriate
9. list of data preliminary in nature and and subject to change

Please note the data requests for Problem No. 4A-122 and 6-102 will be used to update the BNL calculations for Problem No. 4A-26 and 6-11. The data requests for Problem 1-10 and the RHR Loop 4 will be used to perform two new confirmatory piping evaluations as per SECY-82-414.

In addition the following general information is requested:

1. Piping Reverification Analysis Log
2. Westinghouse Piping Analysis List
3. Stress Combination and Criteria DCM M-42
4. ME 101 Computer Code Manual

November 23, 1982

DMB 016

Docket No. 50-346

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Mr. Richard P. Crouse
Vice President, Nuclear
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Crouse:

SUBJECT: APPENDIX R TO 10 CFR 50 - EXEMPTION FROM CERTAIN TECHNICAL REQUIREMENTS

By letter dated April 29, 1982 (No. 815), Toledo Edison Company submitted a request for exemption from certain technical requirements of Section III.G of Appendix R to 10 CFR 50. The technical requirements from which exemption is requested are: (1) the requirement for a fixed fire suppression system in the control room, and (2) the requirement for one-hour-rated fire barriers where less than 20 feet of separation exists between redundant trains of equipment in the component cooling water heat exchanger and pump room.

We have completed our evaluation of your request for exemption, and we conclude (1) that the installation of a fixed fire suppression system will not increase significantly the level of fire protection safety in the control room, and (2) that the installation of one-hour-rated fire barriers between the component cooling water pumps will not increase significantly the level of fire protection in the component cooling water heat exchanger and pump room. Therefore, the Commission hereby grants your requested Exemption. A copy of our Safety Evaluation is enclosed.

In accordance with 10 CFR 50.12, we have determined that this exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. We have also determined that this exemption does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the exemption involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with this action.

OFFICE							
SURNAME							
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We have concluded that: (1) because the exemption does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the exemption does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will not be inimical to the common defense and security or to the health and safety of the public.

A Notice of Exemption, which is being forwarded to the Office of the Federal Register for publication, is also enclosed.

Sincerely,

Original signed by

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
1. Safety Evaluation
2. Notice of Exemption

cc w/enclosures:
See next page

CEB
RFerguson
8/10/82

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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Docket File
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RIngram

November 23, 1982

Docket No. 50-346

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Appendix R to 10 CFR 50 - Exemption from Certain Technical Requirements.
  Referenced documents have been provided PDR.

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE	ORB#4:DL					
SURNAME	RIngram/cf					
DATE	11/30/82					

NRC FORM 102 7—79

Toledo Edison Company

cc w/enclosure(s):

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U.S. Nuclear Regulatory Commission
Resident Inspector's Office
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Oak Harbor, Ohio 43449

Mrs. Julia Baldwin, Librarian
Government Documents Collection
William Carlson Library
University of Toledo
2801 W. Bancroft Avenue
Toledo, Ohio 43606

Regional Radiation Representative
EPA Region V
230 South Dearborn Street
Chicago, Illinois 60604

cc w/enclosure(s) and incoming dtd.:
4/29/82

Ohio Department of Health
ATTN: Radiological Health
Program Director
P. O. Box 118
Columbus, Ohio 43216

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING EXEMPTION FROM CERTAIN REQUIREMENTS

OF APPENDIX R TO 10 CFR 50

THE TOLEDO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 Introduction

By letter dated April 29, 1982 (No. 815), the licensees requested an exemption from certain technical requirements of Section III.G of Appendix R to 10 CFR 50. Specifically, the licensees request exemption from the requirement for the installation of a fixed fire suppression system in the control room and from the requirement for one-hour-fire-rated barriers where less than 20 feet of separation exists between redundant trains of equipment in the component cooling water heat exchanger and pump room (Fire Zone T-1).

2.0 Discussion and Evaluation

Toledo Edison Company has indicated in its April 29, 1982 letter, that the fire protection features currently installed in the control room/cabinet room and the continuous manning of the control room provide adequate defense-in-depth fire fighting capability for these areas. The licensees have stated that the control room/cabinet room is equipped with area fire detectors and internal cabinet fire detectors for safety related control panels. The control room/cabinet room is provided with both a hose station and fire extinguishers for manual fire fighting, and fire load in the area is low.

In addition, an alternate shutdown system is available which provides remote control capabilities for those systems necessary to maintain safe-shutdown capability from outside the main control room.

Plant Technical Specifications require continuous occupancy of the control room by the operators. Because the operators constitute a continuous fire watch, manual fire suppression in event of a fire would be prompt and effective and, thus, a fixed suppression system is not necessary to achieve adequate fire protection in this area.

The component cooling water heat exchanger and pump room is an L-shaped room. The approximate length of the room is 67'-6"; the width of the room in the area of the heat exchanger and the crossover valves at the north end is approximately 26'-3", and at the south end of the room, the approximate width increases to 35'-6" to accommodate the CCW pumps.

The walls, floor, and ceiling slabs of the CCW heat exchanger and pump room are three-hour-fire-rated barriers. Access door 332 leading into the area is a Class "A" three-hour-fire-rated door assembly. The piping and electrical penetrations in the CCW room boundary are filled with silicone foam fire barrier sealant material which provides a seal equivalent to the wall in which it is installed. Where the HVAC ducting penetrates the CCW room enclosure, the duct opening is protected by three-hour-fire-rated dampers installed in accordance with the manufacturer's recommendations.

The fixed combustibles associated with this area consist of 6 gallons of lubricating oil. Each CCW pump and motor contains 2 gallons of oil. The lube oil in the CCW pumps and pump motors is enclosed in a self-contained non-pressurized lubricating system. The lube oil utilized has a flash point of 450°F and an ignition temperature of approximately 700°F. All the power and instrumentation cabling associated with the equipment located in the room is routed in Schedule 40 conduit. There are no cable trays routed in or through the room. The fire load based on the amount of fixed combustibles located in the CCW heat exchanger and pump room is 392 BTU/FT^2.

The following equipment and its associated cabling is located in this room:

a. CCW pumps
b. CCW valving
c. CCW flow switches for pump discharge header
d. CCW temperature indicators
e. Service water valves serving the CCW heat exchangers
f. CCW pump room ventilation fans C75-1 and C75-2, associated dampers motorized inlet louvers and temperature interlocks.

The three CCW pumps are located at the south end of the room. Pumps 1 and 2, which are normally used during plant operations, are separated from one another, pump center line to center line, by 22 feet. Pump 3 is a swing pump and is located between CCW pumps 1 and 2. The center line of pump 3 is 11 feet from the center line of pump 1 and 2. One CCW pump is needed for safe shutdown.

The CCW heat exchanger and pump room is protected by an automatic sprinkler system. Each of the CCW pump motors is baffled to protect it from the impingement of water. The motor is protected from water impinging vertically by its drip proof design. The sprinkler system also covers the floor area of the room for protection from an exposure fire. This includes sprinklers under the mezzanine floor grating and under CCW crossover header valves at the opposite end of the room from the pumps.

Conduits and valves which are required for safe shutdown are protected by a one-hour-fire-rated barrier. The barrier consists of two 1-inch thick Kaowool blankets wrapped around and banded to the conduit and valves with 1/2-inch wide type 316 stainless steel bands and buckles.

Additionally, the floor around each of the CCW pumps is curbed to confine oil leaking from any one pump or motor to the floor area directly around the affected pump. The curbing and the diked floor area around each CCW pump and motor is sized to contain the entire oil content of the pump and motor plus an additional 90% by volume for sprinkler flow.

An automatic smoke-detection system is installed to provide early warning detection in the area. Portable fire extinguishers are located on the north wall of the room. Additional 20-lb dry chemical extinguishers in the stairway and the turbine building are directly accessible to the area. A manual hose station is accessible to the CCW heat exchanger and pump room.

Section III.G.2 of Appendix R to 10 CFR 50 requires that one train of cables and equipment and associated non-safety circuits necessary to achieve and maintain hot shutdown conditions must be maintained free of fire damage by one of the following means:

a. Separation of cables and equipment and associated non-safetv circuits of redundant trains by a fire barrier having a three-hour rating. Structural steel forming a part of or supporting such fire barriers shall be protected to provide fire resistance equivalent to that required of the barrier;

b. Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area; or

c. Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a one-hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area:

We have evaluated the licensee's request on the basis of equivalent protection provided by the specific features of this fire area. The following features were identified as providing passive fire protection equivalent to a one hour fire rated enclosure or the 20 foot separation free of intervening combustibles for one of the redundant CCW pumps:

1. The in-situ combustible loading is significantly less than that needed for a fire of one hour duration;

2. The number 1 and number 2 CCW pumps are horizontally separated by 22 feet. The third pump, which is an installed spare pump for either of the other two and contains only 2 gallons of lubricating oil enclosed in a self-contained, non-pressurized lubricating system, comprises the only significant intervening combustible. If CCW pump number 3 is used for one of the other two, there is eleven feet of separation with no intervening combustibles. This condition would exist only a small fraction of time;

3. A curb is provided around each pump to contain any potential leakage of oil; and

4 A one hour fire rated barrier is provided for the cables and valves in the area.

We have concluded that, based on the above features, a one hour fire rated enclosure for one CCW pump will not enhance the fire protection features for accomplishing safe shutdown and is not required. We further conclude that in the event of a fire in this room, that the above features will provide ample time for the installed detection and automatic suppression system to detect and extinguish the fire prior to damaging both redundant trains of CCW equipment.

3.C Conclusion

Based on our evaluation, we conclude that the licensees' fire protection features for the control room meet the objectives of Section III.G, Fire Protection of Safe Shutdown Capability, of Appendix R to 10 CFR 50, and that the installation of a fixed fire suppression system will not increase, significantly, the level of fire protection in the control room/cabinet room. Therefore, the licensees' request for exemption from the requirement to provide a fixed fire suppression system in the control room should be granted.

Based on our evaluation, we conclude that the existing arrangements in the component cooling water heat exchanger pump room provide a level of fire protection equivalent to that required by Appendix R to 10 CFR 50, and that the addition of a one-hour-fire-rated barrier around one of the component cooling water pumps will not increase, significantly, overall facility safety. Therefore, the licensees' request for exemption from the requirement for a one-hour-rated fire barrier around one of the component cooling water pumps should be granted.

The Commission has determined that pursuant to 10 CFR 50.12, an exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest.

We have determined that the exemption does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the exemption involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this action.

We have concluded, based on the considerations discussed above, that: (1) because the exemption does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the exemption does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this exemption will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 23, 1982

The following NRC personnel have contributed to this Safety Evaluation: R. Eberly, A. De Agazio.

UNITED STATES NUCLEAR REGULATORY COMMISSION

THE TOLEDO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

NOTICE OF EXEMPTION

The Nuclear Regulatory Commission (the Commission) has granted an Exemption to The Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees) for the Davis-Besse Nuclear Power Station, Unit 1 (located in Ottawa County, Ohio), from the following technical requirements set forth in Section III.G of Appendix R to 10 CFR 50: (1) the requirement for a fixed fire suppression system in the control room, and (2) the requirement for one-hour-rated fire barriers where less than 20 feet of separation exists between redundant trains of equipment in the component cooling water heat exchanger and pump room. The Exemption is effective as of its date of issuance.

In granting the Exemption, the Commission determined that it is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest. The Commission also determined that granting the Exemption will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this action.

For further details, see (1) Toledo Edison's request by letter dated April 29, 1982, and (2) the Commission's letter to Toledo Edison dated November 23, 1982. These items can be reviewed at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 and at the William Carlson Library, University of Toledo, 2801 Bancroft Avenue, Toledo, Ohio 43606.

A copy of item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 23rd day of November 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing