POLICY ISSUE
(Notation Vote)

July 24, 1991 SECY-91-220

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: YANKEE ROWE PRESSURE VESSEL EMBRITTLEMENT ISSUES

Purpose: To provide the Commission with the proposed Decision, information requested in SRM-M910711A and other relevant information concerning the subject matter.

Discussion: Following the July 11, 1991, Commission meeting and public meeting held in Rowe, Massachusetts, on July 22 and 23, 1991, on the subject of "Yankee Rowe Pressure Vessel Embrittlement Issues," the Staff has prepared a proposed Decision on this matter. In addition, the Staff has prepared various responses to questions from the Commissioners and a technical response to a letter received by the Chairman from Dr. Pryor N. Randall. Yankee Atomic Electric Company has also submitted responses to two questions from the Commissioners. These responses are in enclosures attached to this paper and are as follows:

Enclosure 1: Proposed Decision Under 10 CFR Section 2.206 including Federal Register Notice

Enclosure 2: Responses to Staff Requirements Memorandum (M910711A)

Enclosure 3: Responses to Commissioner Curtiss' Questions (EDO 6770)

Enclosure 4: Response to technical issues raised in Dr. Pryor N. Randall's July 15, 1991 Letter to the Chairman

Enclosure 5: Response to Commissioner Rogers (EDO 6761) and response to a question from the Staff by Yankee Atomic Electric Company

NOTE: PAPER HAS BEEN MADE PUBLICLY AVAILABLE

Following coordination with the Secretary, this paper and its enclosures have been sent to the Union of Concerned Scientists (UCS) and the New England Coalition on Nuclear Pollution (NECNP) and placed in the Public Document Room.

Recommendation: Based upon the enclosed proposed Decision, the other enclosures, and the entire record of this matter, the Staff recommends that the Commission deny the UCS/NECNP Petition and authorize the Director of NRR to issue the Decision in this matter in the form proposed.

James M. Taylor,
Executive Director
for Operations

Enclosures:
As stated

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Commissioners' comments or consent should be provided directly to the Office of the Secretary ASAP .

Commission Staff Office comments, if any, should be submitted to the Commissioners ASAP , with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

DD-91-

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
YANKEE ATOMIC ELECTRIC COMPANY	)	Docket No. 50-029
	)	
(Yankee Rowe Nuclear Power Station)	)	(10 CFR Section 2.206)

PROPOSED DECISION UNDER 10 CFR SECTION 2.206

I. INTRODUCTION

On June 4, 1991, Ms. Diane Curran submitted a "Petition for Emergency Enforcement Action and Request for Public Hearing" (Petition) on behalf of the Union of Concerned Scientists and the New England Coalition on Nuclear Pollution (Petitioners) to the Commissioners of the Nuclear Regulatory Commission (NRC). The Petition sought the immediate shutdown of the Yankee Nuclear Power Station (Yankee Rowe) of the Yankee Atomic Electric Company (YAEC or Licensee) based upon allegations that the continued operation of the Yankee Rowe facility poses a serious threat to public health and safety. The Petition further requested that the Yankee Rowe facility remain shut down until it complies with regulatory requirements and that the Commission provide a public hearing, with rights of discovery and cross examination, to determine regulatory compliance before permitting the facility to resume operation.

Many letters have been received by the NRC from members of the public, interested public groups and governmental entities. The NRC will communicate separately with each individual or group and acknowledge their inquiry. This Decision is the NRC Staff's formal response to the issues raised by the Petition regarding the Yankee Rowe pressure vessel and involved with the continued operation of the Yankee Rowe facility. No further Decisions pursuant to 10 CFR Section 2.206 are contemplated.

The Petition was filed pursuant to 10 CFR Section 2.206 and thus should have been filed with the Executive Director for Operations. However, Petitioners sought relief directly from the Commissioners because they believe that the NRC Staff had failed to properly execute its responsibilities in permitting the Yankee Rowe facility to continue operating through Cycle 21 (until approximately April 1992). The Petition was referred by the Commission to the Office of Nuclear Reactor Regulation (NRR) for response as specified in 10 CFR Section 2.206.

The Petition alleged specifically that the Yankee Rowe reactor pressure vessel failed to meet NRC requirements. The Petition argues that the Yankee Rowe facility does not comply with the requirements of 10 CFR Section 50.61 regarding the reference temperature for the reactor vessel material, the requirements in Appendix G to 10 CFR Part 50 regarding fracture toughness, and the requirements in Appendix H to 10 CFR Part 50 regarding a surveillance program for reactor vessel material.

The Petition also made specific allegations regarding the adequacy of the "Safety Assessment of Yankee Rowe Vessel" (Safety Assessment) issued by the NRC NRC Staff on August 31, 1990 (Reference 18), which concluded that the Yankee Rowe facility could be operated safely through Cycle 21. The Petition alleged that the Safety Assessment contained errors and insufficient information in the assumptions underlying the calculations regarding the amount of neutron irradiation absorbed by the reactor vessel, the temperature of the metal during the time it was exposed to neutron irradiation and the chemical composition of the metal.

The Petition also alleged that information regarding the presence of material defects, which may result from inservice nondestructive examination, is not available. In addition, the Petition alleged that the Safety Assessment is inconsistent with the NRC policy on Safety Goals and failed to take into account the explicit recommendation of an NRC Staff expert on reactor pressure vessel integrity that the Yankee Rowe facility not be permitted to operate.

In a letter to Ms. Curran dated June 25, 1991, the NRR Director acknowledged receipt of the Petition and informed her that it would be handled under 10 CFR Section 2.206 of the Commission's regulations. Ms. Curran was informed in the letter that the Petition failed to present any new information in regard to the integrity of the Yankee Rowe reactor vessel. The letter discussed each of the issues raised in the Petition and concluded that emergency relief, i.e., immediate shutdown of the Yankee Rowe facility, was not warranted. The letter stated that the NRC Staff had carefully evaluated the Yankee Rowe reactor vessel issues and had concluded that the vessel condition continues to provide adequate protection of public health and safety. The letter stated that the NRC would issue a Decision pursuant to 10 CFR Section 2.206 within a reasonable time to address the specific issues raised in the Petition.

Subsequently, the Commission decided to hold a Public Meeting to hear the views of the Petitioners, the Licensee and the NRC Staff. The Public Meeting was held in Rockville, Maryland on July 11, 1991, in the Commission Meeting Room. Also on July 11, 1991, Petitioners submitted their "Renewed and Supplemented Petition for Emergency Enforcement Action and Request for Public Hearing" (Supplement) to the Commission. The Supplement urged the Commission to take jurisdiction over the matter and to cease _ex parte_ contacts with the Licensee and the NRC Staff in this matter. The Commission stated at the Public Meeting that it would review and vote on this matter and also ruled that the

Commission's ex parte rule, 10 CFR Section 2.780, did not apply to matters being considered pursuant to 10 CFR Section 2.206. See 10 CFR Section 2.206 (c)(1). In the Supplement, Petitioners also restated their concerns in regard to continued operation of the Yankee Rowe facility and renewed their requests for an immediate shutdown of the Yankee Rowe facility and for an adjudicatory hearing on whether it should be allowed to resume operation. NRC conclusions with regard to the specific issues raised in the Petition and its Supplement follow.

II. BACKGROUND

The Yankee Nuclear Power Station is located in Rowe, Massachusetts, and is owned and operated by the Yankee Atomic Electric Company. The Nuclear Steam Supply System (NSSS) was designed by the Westinghouse Electric Corporation and the Architect/Engineer for the plant was Stone and Webster. The plant utilizes a pressurized water reactor to generate 600 megawatts-thermal with an average electrical output of approximately 185 megawatts-electrical. Operating License No. DPR-3 was issued by the NRC (then the AEC) on July 9, 1960. Commercial operation of the plant began on July 1, 1961. The facility is presently in its 21st Cycle of operation and has therefore been refueled twenty times. The plant is licensed to operate for 40 years.

The reactor vessel at Yankee Rowe is constructed of Type SA-302, Grade B carbon steel approximately 8 inches thick with internal cladding approximately 0.1 inches thick fabricated from Type 308-L stainless steel. The vessel was designed and fabricated by Babcock and Wilcox in accordance with Section VIII of the ASME Code. The approximate dimensions of the reactor vessel are 9 feet in diameter and 33 feet high. The NSSS has four coolant loops, each with a steam generator and loop isolation valves. The vessel, steam generators and other principal components of the NSSS are enclosed in a spherical steel containment shell 125 feet in diameter and designed to withstand about 35 psig internal pressure. The reactor coolant system normally operates at an average temperature of 530 degrees Fahrenheit (°F) and a nominal pressure of 2000 psig. The coolant enters the reactor vessel at 509°F and is heated to an average temperature of 551°F by the time it leaves the vessel.

In the 30 years that Yankee Rowe has been operating, the industry's standards, predominantly the ASME Boiler and Pressure Vessel Code, and the NRC's regulations concerning material quality assurance (i.e., quality control during fabrication, surveillance testing, and inservice nondestructive inspection) have evolved to increase the reliability of reactor vessels and their resistance to phenomena such as pressurized thermal shock (PTS). As new regulations were implemented, plants already having an operating license may not have been able to comply with newer requirements, in which case licensees should have requested exemptions. Four regulations that have been applied to Yankee Rowe long after issuance of its operating license, and that involve the Yankee Rowe reactor vessel, are: (1) limits on Charpy upper shelf energy (USE), required by 10 CFR Part 50, Appendix G, (2) a material surveillance program required by 10 CFR Part 50, Appendix H, (3) screening limits on PTS reference temperature (RT_{PTS}) required by 10 CFR 50.61, and (4) inservice inspection (ISI) requirements of 10 CFR Section 50.55a. Appendices G and H to 10 CFR Part 50 were originally made part of the Code of Federal Regulations in 1974, 14 years after Yankee Rowe was licensed. Both appendices were amended and published in the FEDERAL REGISTER (FR) on May 2, 1983 (48 FR 24008) to "update them after seven years of use and to make them more consistent with current technology and pertinent National Standards." It was also at this time that 10 CFR Section 50.60 was added to the regulations (See Section III.B of this Decision). As discussed below, the Yankee Rowe vessel was evaluated with regard to the technical intent of Appendices G and H requirements in 1979 and were found to be acceptable. The PTS Rule, 10 CFR Section 50.61, was incorporated into the NRC's regulations on July 23, 1985 (50 FR 29944). That rule required all Licensees to supply projected values for RT_{PTS} for the duration of plant life to the NRC Staff by January 23, 1986. The Yankee Rowe Licensee responded to this requirement by letter dated January 22, 1986, with an update on February 4, 1987. Both responses indicated that the screening criteria would not be reached by the expiration date of the operating license. The ISI requirements (10 CFR Section 50.55a(g)) were amended in July 1970. For reactors of Yankee Rowe's vintage, the regulation takes the approach of applying ISI requirements to the degree practical but requires no backfits.

In its Systematic Evaluation Program (SEP) in the late 1970's, the NRC Staff evaluated the Yankee Rowe reactor vessel, as well as those at ten other plants in the United States. In its 1979 SEP report on this subject, NUREG-0569 (Reference 1), the NRC Staff compared the reactor vessel condition of these 11 plants to the intent of the requirements of Appendixes G and H then in effect. The report noted that the Yankee Rowe surveillance program generally conformed to the requirements of Appendix H but did not conform to certain aspects of Appendix H. For example, the number of irradiated test samples available for the Yankee Rowe reactor vessel was smaller than for most reactor vessels. The small number of irradiated test samples from the Yankee Rowe reactor vessel resulted from the fact that the materials surveillance program was initiated prior to the existence of 10 CFR Part 50, Appendix H. The test samples did not include any weld material specimens or samples from the heat affected zone. The test specimens were not oriented in the most conservative direction. The surveillance program was abandoned in 1965, after 4 years of plant operation, due to a failure of the surveillance sample holders. However, the NRC concluded in NUREG-0569 that the discontinued Yankee Rowe program had gathered sufficient information on the vessel plate and that irradiated properties for the welds could be conservatively calculated.

The NRC Staff also concluded in the SEP report that the Yankee Rowe reactor vessel had been fabricated to quality requirements beyond those required by ASME Code Section VIII, and these quality measures were comparable to those required by ASME Code Section III for vessels constructed under then current 1979 requirements. The NRC Staff's prediction was that end-of-life USE would be 42 ft-lbs; that is, below the regulatory requirements of Appendix G to 10 CFR Part 50. However, the SEP Discussion Section indicated that the below-regulatory value was offset by the fact that primary stresses in the vessel beltline region are low, about 70% of that allowed by ASME Code Section III. The SEP report recommended a) sampling of several vessel welds made by the same technique as the vessel beltline weld to determine material composition, b) using acoustic emission testing to verify continued integrity of those welds that cannot be inspected by normal methods, and c) submitting a report to the NRC at some later date following resolution of Unresolved Safety Issue (USI) A-11 by the NRC Staff. The Licensee has committed to obtain samples of the actual Yankee Rowe vessel beltline welds and to perform a volumetric examination of the welds and plate material in the beltline region during the next refueling outage. As a result of NRC concerns regarding Yankee Rowe reactor vessel embrittlement, the Licensee submitted a report on July 5, 1990 (Reference 25) addressing the PTS issues. Thus, the SEP report recommendations regarding PTS will be fully addressed prior to plant start up after the next refueling outage. NRC Staff action on USI A-11 was completed in October 1982, with issuance of Generic Letter 82-26 in October 1982. Generic Letter 82-26 did not impose any new requirements on Licensees but did provide analytical methodology to Licensees that can be used to demonstrate compliance with Appendices G and H to 10 CFR Part 50 (See Section III).

The Licensee believes it has compensated for lack of a material surveillance program that fully complies with Appendix H, as discussed below in Section III.E of this document, by irradiating Yankee Rowe specimens in a Belgian reactor (BR3) at accelerated irradiation rates and through a special on-going test program at the University of Michigan.

From 1983 to 1990, both the NRC Staff and the Licensee believed that Yankee Rowe was complying with applicable regulations as they were published, amended, or interpreted through generic correspondence. Moreover, no problems were identified during refueling or operation that would indicate that the Licensee was not pursuing alternatives, as appropriate, to the inspection requirements of the regulations. In the late 1980's and early in 1990, the NRC Staff began reviewing its records to prepare for the anticipated action by plants such as Yankee Rowe that would be the ones most likely to apply for license renewal under 10 CFR Part 52. In that review, concerns were raised about the exact status of Yankee Rowe with respect to the fracture toughness, reactor vessel embrittlement, and PTS. The NRC Staff contacted the Licensee and requested clarification of the vessel's status with respect to compliance with the regulations. Further discussions with the Licensee and further NRC Staff review resulted in the August 1990 Safety Assessment.

Although the NRC Staff had concluded in NUREG-0569 that adequate margins existed in the Yankee Rowe reactor vessel, generic concern regarding the Charpy USE of several reactor vessels, including Yankee Rowe, continued during the 1980's. Periodic guidance was issued to the Licensees of these plants regarding the Charpy USE calculations. Also, NRC Regulatory Guide (RG) 1.99 was revised in May 1988.

As part of NRC Staff discussions on license renewal, the NRC Staff met with the Licensee in early 1990 to discuss concerns with the Yankee Rowe reactor vessel and with operating license renewal. As discussed below, this meeting initiated a renewed dialogue between the NRC Staff and the Licensee, resulting in submittal of a special report on July 5, 1990 (Reference 2).

From 1979 until 1990, the only reactor vessel integrity analysis specific to Yankee Rowe was an evaluation of its compliance to the PTS rule, (10 CFR 50.61). This analysis was completed in March 10, 1987 (see Section III.A). During this period, the staff developed generic analysis methods and pursued, through the American Society of Mechanical Engineers (ASME), the development of criteria for evaluation of reactor vessel materials with Charpy USE less than 50 ft-lb.

In the 1979 SEP evaluation (NUREG-0569), the NRC Staff determined that, at its end-of-life neutron irradiation (or fluence), the Yankee Rowe reactor vessel would have a Charpy USE of 42 ft-lb but would have acceptable fracture toughness properties. The Charpy USE for the Yankee Rowe reactor vessel was estimated from data obtained from the Yankee Rowe material surveillance program and other similar materials in research programs. The NRC staff in its SEP evaluation recommended that the Yankee Rowe fracture toughness issue be re-reviewed upon completion of USI A-11. This technical activity was completed with the issuance of Generic Letter 82-26, which included publication of NUREG-0744, Revision 1(17). Licensee's were informed of this resolution through issuance of Generic Letter 82-26 as discussed above. This guidance contained a methodology for evaluating the fracture toughness of material but did not contain acceptance criteria. In a letter dated April 20, 1982, the NRC requested that the ASME provide assistance in developing criteria to determine the safety margins needed to demonstrate the capability for continued safe operation of a reactor vessel with Charpy USE less than 50 ft-lb.

The NRC Staff received criteria in a letter dated November 20, 1989 (Reference 21) from the Chairman of the ASME Subgroup on Evaluation Standards for Section XI of the ASME Code. In a letter dated May 1, 1990 (Reference 22), the NRC Staff sent the Licensee the proposed criteria, but included a modification that will appear in a future edition of the Code. This letter to the Licensee states that the Charpy USE for upper plate shelf energy was estimated to be 35 ft-lb from the reported unirradiated Charpy USE of the Yankee Rowe upper plate and using Branch Technical Position MTEB 5-2 in Standard Review Plan 5.3.2 and Regulatory Guide (RG) 1.99, Revision 2 (Reference 23), to account for plate directionality and neutron irradiation, respectively.

The NRC Staff concluded in its Safety Assessment that it is acceptable to operate the Yankee Rowe reactor vessel until the end of the current fuel cycle (Cycle 21). The NRC Staff also identified a number of long-term actions that should be completed prior to the NRC Staff approving subsequent operation. These actions included development of inspection methods for the beltline welds and each beltline plate to determine if the metal contains flaws, testing of typical Yankee Rowe vessel base material to determine the effect of irradiation, austentizing temperature and nickel composition, determining composition of the weld metal in the beltline by removing samples from the weld, and installing surveillance capsules in the Yankee Rowe vessel in positions for accelerated irradiation.

III. DISCUSSION

Each of the topics addressed in the Petition are discussed below. The format is a summary of what the regulations require, when appropriate, a summary of the Petitioners' contention, followed by the NRC Staff's response.

A. Pressurized Thermal Shock (PTS)

Requirements of 10 CFR Section 50.61 - Fracture toughness requirements for protection against pressurized thermal shock events.

The RT_{PTS} screening criteria are defined by 10 CFR Section 50.61 as 270°F for plates, forgings and axial weld materials and 300°F for circumferntial welds.

10 CFR Section 50.61 requires that:

1. Licensees for each operating pressurized water reactor shall submit an assessment by January 23, 1986, with projected values of reference temperatures calculated in accordance with the method prescribed in this section. Section 50.61(b)(1).
2. If the projected reference temperatures exceed the screening criterion established by this section before the expiration of the operating license, the Licensee shall submit by April 23, 1986, an analysis and schedule for such flux reduction programs as are reasonably practicable to avoid exceeding the screening criterion. Section 50.61(b)(3).
3. If no reasonably practicable flux reduction programs will avoid exceeding the screening criterion, the Licensee shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criterion is allowed. The analysis must be submitted at least three years before it is projected that the reference temperature will exceed the screening criterion, or by one year after the

issuance of the Commission Guidance and Acceptance Criteria for these analyses, whichever is later. Section 50.61(b)(4).

4. The Commission may, on a case-by-case basis after consideration of the Licensee's analysis, approve operation of the facility at a reference temperature in excess of the screening criterion. Section 50.61(b)(5).

5. If the Commission concludes that operation of the facility at a reference temperature in excess of the screening criterion cannot be approved on the basis of the Licensee's analysis, the Licensee must request and receive Commission approval prior to any operation beyond the screening criteria based upon modifications to equipment, systems, and operation of the facility in addition to those previously proposed, or upon further analyses based upon new information or improved methodology. Section 50.61(b)(6).

Petitioners' Contentions

Petitioners contend that (1) the reference temperatures for the upper plate, lower plate, and the circumferential weld of the Yankee Rowe reactor vessel exceed the screening criteria for PTS set forth in 10 CFR Section 50.61(b)(2), (2) the Licensee's safety analysis was not provided at least three years before the criteria were exceeded, and (3) since the reference temperatures exceed the screening criteria, the risk of vessel failure under the conditions of pressurized thermal shock is greater than permitted by the Commission's regulations and the plant should therefore be shut down. Further, in its Supplement, Petitioners contend that the NRC Staff has inappropriately used probabilities in determining the adequacy of the Licensee's analysis to show compliance with the rule. Petitioners contend that the NRC Staff cannot justify noncompliance with the screening criteria based on essentially the same probability estimate used to develop the screening criteria and claimed, therefore, that the NRC Staff's analysis was flawed and not appropriate.

NRC Staff Response

In a letter to the NRC dated January 22, 1986 (Reference 6), the Licensee provided an assessment of the projected value of reference temperature for all beltline materials. The values were calculated in accordance with the methods prescribed at that time in Section 50.61(b)(2). Additional information was provided in letters dated August 12, 1989 (Reference 7), October 28, 1986 (Reference 8) and February 4, 1987 (Reference 9). The NRC Staff reviewed this information and, in a letter dated March 10, 1987 (Reference 19), concluded that the RT_{PTS} for the Yankee Rowe reactor vessel were below the screening criteria at 32 effective full power years, which is beyond the expiration date of the license. Therefore, the NRC Staff concluded that the Yankee Rowe facility met the requirements of the original PTS rule.

The NRC Staff began implementing a revised method of calculating the increase in reference temperature resulting from neutron irradiation through issuance of RG 1.99, Revision 2. The formulae and Tables in RG 1.99 were empirically derived from surveillance data from U.S. commercially operated nuclear power plants that have a nominal irradiation temperature of 550°F. The RG indicates that its procedures are valid for a nominal irradiation temperature of 550°F and that irradiation below 525°F should be considered to produce greater embrittlement.

As a result of meetings with the Licensee in March through May, 1990, the NRC Staff determined that it was necessary to evaluate the amount of embrittlement in the Yankee Rowe reactor vessel in accordance with methods other than those prescribed in the then effective Section 50.61(b)(2) and RG 1.99, Revision 1. A revised method of analysis was considered necessary because of Yankee Rowe's unique operating characteristics, its limited surveillance data and uncertainty in its weld chemical composition. For example, the Licensee operated the reactor vessel with an average cold leg temperature during normal operation of 500°F to 512°F and with coastdown to temperatures below 500°F. As a result of this method of operation, the reactor vessel is estimated to have accumulated 15 percent of its neutron irradiation at temperatures below 500°F, and 85 percent of its neutron irradiation at temperatures between 500°F to 520°F.

The reference temperature for the Yankee Rowe vessel, making conservative assumptions regarding weld chemistry, lower operating temperature and limited surveillance data, was estimated to exceed the mean value for the formulae in 10 CFR Section 50.61(b)(2) and RG 1.99, Revision 2, by a substantial amount. Since the Yankee Rowe reactor vessel operated at temperatures below 525°F, the NRC Staff decided that the procedures specified in 10 CFR 50.61(b)(2) could be nonconservative in predicting the amount of embrittlement to the Yankee Rowe beltline materials. Also, 10 CFR Section 50.61 requires that the amount of embrittlement be calculated using best estimates of the amount of copper and nickel in the beltline material. Greater amounts of copper and nickel produce greater amounts of embrittlement. The Licensee's best estimate of copper in its welds was the same as the value reported for the beltline weld in a Belgian reactor (BR-3) reactor vessel, which was fabricated by Babcock & Wilcox in the same time frame as the Yankee Rowe reactor vessel. However, the Licensee could not determine that the heat number of the weld wire used to fabricate its beltline welds was the same as that used in fabricating the BR-3 beltline weld. Since the amount of copper in a weld depends upon the heat of the weld wire and several different heats of wire can be used to fabricate a reactor vessel, the NRC Staff was concerned that the Licensee's best estimate of copper content in the weld could be nonconservative. Thus, although the Licensee had earlier demonstrated compliance with 10 CFR Section 50.61, unique aspects of Yankee Rowe operation and uncertainty in weld chemistry lead the NRC Staff to conclude that additional analyses of the adequacy of the Yankee Rowe vessel to PTS were required.

The NRC Staff provided to the Licensee conservative values of reference temperature for use in their probabilistic fracture mechanics analysis. These values were based on conservative values for vessel weld chemistry and the effects of lower irradiated temperature. Each of these conservative assumptions is discussed in more detail in later sections of this Decision. The NRC Staff reviewed the Licensee's results from the probabilistic fracture mechanics analysis which indicated a conditional vessel failure probability of $2x10^{-3}$. Based upon the Licensee's analysis and considering the uncertainties resulting from low USE of the vessel materials, the lack of beltline inspection, and the reactor vessel's unique spot-welded cladding, the NRC Staff concluded a conservative conditional probability of reactor pressure vessel failure to be in the range of 10^{-1} to 10^{-2}. The NRC Staff estimated the frequency of the limiting PTS transient to be 10^{-3} per reactor year. Thus, the NRC Staff concluded that a conservative estimate of the probability of a vessel through-wall crack due to a PTS event was in the range of 10^{-4} to 10^{-5} per reactor year. Based on this conservative assessment, the NRC Staff authorized the Licensee to operate the Yankee Rowe reactor vessel until the end of fuel Cycle 21.

An amended PTS Rule was published in the Federal Register on May 15, 1991. The amended rule requires that reference temperatures be calculated in a more rigorous manner, including a calculation of the reference temperature in accordance with the formulae and tables in RG 1.99, Revision 2, and an assessment of the effect of irradiation temperature and surveillance data. The rule became effective on June 14, 1991. The amended rule allows Licensees to submit an analysis one year after the effective date of the rule. The results of the Licensee's analysis submitted in August 1990 indicate probability of vessel failure to be less than 5 in one-billion. Since Yankee Rowe's or YAEC's analysis was submitted in August 1990, and that analysis meets the requirements of Section 50.61 now in effect, the Licensee has complied with the amended rule for continued operation through the end of fuel cycle 21. However, the NRC Staff has challenged the assumptions that were used in this analysis and requested the Licensee to redo the analysis using the conservative values provided by the NRC Staff.

B. Compliance with the Requirements of Appendix G to 10 CFR Part 50 - Fracture Toughness Requirements

10 CFR Section 50.60 requires that:

1. All light water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to this Part, except as provided in Paragraph (b) of this Section. Section 50.60(a).
2. Proposed alternatives to the requirements described in Appendices G and H of this Part or portions thereof may be used when an exemption is granted by the Commission under Section 50.12. Section 50.60(b).

Thus Appendix G to 10 CFR Part 50 applies to the Yankee Rowe facility. The requirements of Appendix G are:

1. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code supplemented as follows for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences:

a. Reactor vessel beltline materials must have Charpy upper-shelf energy (USE) of no less than 75 ft-lb initially and must maintain USE throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, NRR, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. Section IV.A.1.

b. When the core is not critical, pressure-temperature limits for the reactor vessel must be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code supplemented by the requirements of Section V of this Appendix. In addition, when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests, unless a lower temperature can be justified by showing that the margins of safety for those regions when they are controlling are equivalent to those required for the beltline when they are controlling. Section IV.A.2.

c. When the core is critical (other than for the purpose of low-level physics tests), the temperature of the reactor vessel must not be lower than 40°F above the minimum permissible temperature of Paragraph 2 of this Section, nor lower than the minimum permissible temperature for the in-service system hydrostatic pressure test. Section IV.A.3.

d. If there is no fuel in the reactor during system hydrostatic pressure tests or leak tests, the minimum permissible test temperature must be 60°F above the adjusted reference temperature of the reactor vessel material in the region that is controlling (as specified in Paragraph IV.A.2 of the Appendix). Section IV.A.4.

e. If there is fuel in the reactor during system hydrostatic pressure tests or leak tests, the requirements of Paragraphs 2 and 3 of this Section apply, depending on whether the core is critical during the test. Section IV.A.5.

2. Reactor vessels for which the predicted value of USE at the end of life is below 50 ft-lb or for which the predicted value of adjusted reference temperature at the end of life exceeds 200°F must be designed to permit a thermal annealing treatment at a sufficiently high temperature to recover material toughness properties of ferritic materials of the reactor vessel beltline. Section IV.B.

3. The effects of neutron radiation on the reference temperature and USE of reactor vessel beltline materials, including welds, are to be predicted from the results of pertinent radiation effects studies in addition to the results of the surveillance program of Appendix H to this Part. Section V.A.

4. Reactor vessels may continue to operate only for that service period within which the requirements of Section IV of Appendix G are satisfied using the predicted value of the USE at the end of the service

period to account for the effects of radiation on the fracture toughness of the beltline materials. Section V.B.

5. In the event that the requirements of Section V.B. of the Appendix cannot be satisfied, reactor vessels may continue to be operated provided all of the following requirements are satisfied:

   a. A volumetric examination of 100 percent of beltline materials that do not satisfy the requirements of Section V.B. of the Appendix is made and any flaws characterized according to Section XI of the ASME Code and as otherwise specified by the Director, NRR.

   b. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation is to be obtained from results of supplemental fracture toughness tests.

   c. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation. Section V.C.

6. The proposed program for satisfying the requirements of Section V.C. of the Appendix must be submitted, as specified in Section 50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of Section V.B of the Appendix. Section V.E.

Petitioners' Contentions

The Petitioners contend that Sections IV and V.A of Appendix G to 10 CFR Part 50 forbid the operation of a nuclear power plant unless its pressure vessel has a Charpy USE of no less than 50 ft-lb. They also contend that

Section V.C provides that, if this 50 ft-lb requirement is not met, the plant may continue to operate only provided all of the following requirements are satisfied:

1. A volumetric examination of 100 percent of the beltline materials that do not satisfy the requirements of Section V.B of this Section is made and any flaws characterized according to Section XI of the ASME Code and as otherwise specified by the Director, Office of Nuclear Reactor Regulation (NRR).
2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron radiation is to be obtained from the results of supplemental fracture toughness tests.
3. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation.

The Petitioners contend that not one of these requirements has been met. They contend that neither the NRC Staff nor the Licensee could even pretend that the first two of these requirements have been met, because they are still engaged in planning to meet them at the next refueling outage, at the earliest. They also contend that no valid analysis has been performed that shows, let alone alleges, that "equivalent margins of safety" have been achieved, thus justifying continued operation. They contend that the NRC Staff's August 1990 Safety Assessment, which purports to justify continued operation, pointedly fails to issue any finding that Yankee Rowe facility achieves margins of safety "equivalent" to compliance with the regulations. Instead, they contend, it obscurely states that continued operation of the plant for another fuel cycle is "acceptable."

NRC Staff Response

The Petitioners contend that the Licensee has failed to satisfy the requirements of Section V.C to Appendix G to 10 CFR Part 50. It is the NRC Staff's position that Section V.C does not apply to the Yankee Rowe facility.

Section V.C applies only if the requirements of Section V.B cannot be met. Section V.B states that reactor vessels may continue to be operated only for that service period within which the requirements of Section IV are satisfied. Section IV.A.1 allows operation if it is demonstrated in a manner approved by the Director, NRR, that lower values of USE (less than 50 ft-lb) will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. Since the Licensee's analysis satisfies Section IV.A.1. of Appendix G as is further discussed below, the requirements of Section IV are satisfied and the requirements of Sections V.C.1., 2. and 3. do not apply. In addition, the reporting requirements of Section V.E of Appendix G do not apply because reports are required only when Sections V.C and V.D need to be met.

The analyses that the Licensee has submitted have been reviewed by the NRC Staff and have been found adequate to meet the requirements of Section IV.A.1. As discussed in NUREG-0569, an analysis of the adequacy of the Yankee Rowe vessel was performed using technology that was current at the time (i.e., 1979). That analysis demonstrated that a USE of 42 ft-lb was acceptable. NUREG-0569 recommended, however, that this analysis be reviewed once Unresolved Safety Issue "Reactor Vessel Materials Toughness," A-11 was resolved. While the NRC Staff published an evaluation methodology in NUREG-0744, Revision 1 (October 1982) acceptance criteria were not available until November 1989. These criteria (with one addition) were sent to the Licensee via an NRC letter dated May 1, 1990 (Reference 22).

The Licensee submitted Report No. YAEC 1735 in July 1990 (Reference 2), that included an analysis demonstrating that the acceptance criteria developed by members of the ASME Code Subgroup on Evaluation Standards were satisfied at Yankee Rowe for Charpy USE as low as 35 ft-lb. These criteria were sent to the Director of the Division of Engineering Technology, NRR, on November 20, 1989 via a letter (Reference 26) from W. Bamford, Chairman of the Subgroup on Evaluation Standards. The NRC Staff was closely involved in developing the acceptance criteria and has recommended their use as a means to demonstrate margins of safety equivalent to those in ASME Code Section III, Appendix G.

Using the ASME criteria, the Licensee provided a fracture mechanics analysis in YAEC Report No. 1735 (Reference 2), to demonstrate that the Yankee Rowe reactor vessel would have equivalent margins of fracture toughness to those required by Appendix G of the ASME Code with a Charpy USE of 35 ft-lbs. In its August 31, 1990 Safety Assessment, the NRC Staff accepted the Licensee's analysis and approved the operation of the Yankee Rowe reactor vessel at levels of Charpy USE less than the limits in Section IV.A.1 of Appendix G to 10 CFR Part 50. Page 21 of the NRC Staff's Safety Assessment states that "...it appears that the Licensee's analysis satisfies the ASME Code criteria for Service Levels A and B and provides margins of safety against fracture equivalent to those required by Appendix G of the ASME Code." Thus, the NRC Staff concluded in its Safety Assessment that the Licensee's analysis demonstrated the required equivalent margins of safety. Consequently, the NRC Staff concluded that Yankee Rowe complies with Section IV.A.1. of Appendix G. The Safety Assessment documents the NRC Staff's review of this issue and concludes that the Licensee has demonstrated margins of safety equivalent to those in the ASME Code, Section III, Appendix G, for the Yankee Rowe reactor vessel. The Yankee Rowe facility also complies with the other applicable requirements of Appendix G. Specifically, the facility meets the requirements of Section IV.A.2 through 5 and Section IV.B.

With regard to the capability for thermal annealing, the Yankee Rowe reactor vessel was designed and constructed many years before the requirement for thermal annealing capability was imposed. However, the Licensee has reported to the NRC Staff that it is developing contingency plans for annealing the pressure vessel at 650°F. While this temperature is not as high as might be desired, Appendix G does not specify annealing conditions or methods. The Belgian BR3 reactor vessel, which was fabricated by Babcock and Wilcox the same time frame as the Yankee Rowe reactor vessel and by the same fabrication techniques, was successfully annealed at 650°F in 1984. Therefore, the BR3 annealing experience and the Licensee's planning efforts indicate the capability to anneal the Yankee Rowe reactor vessel.

In conclusion, the NRC Staff considers that the Yankee Rowe facility complies with Appendix G and the safety issues have been, and continue to be, adequately addressed.

C. <u>Nonconservative Assumptions in the NRC Staff's Safety Assessment</u>

1. <u>Neutron irradiation</u>

<u>Petitioners' Contention</u>

The Petitioners contend that the Safety Assessment significantly underestimated the degree to which the Yankee Rowe reactor vessel has been exposed to neutron irradiation. Subsequent to the issuance of the Safety Assessment, the Licensee determined that previously calculated neutron irradiation values for Yankee Rowe were about 13 percent lower than the actual irradiation received by the vessel.

NRC Staff Response

In its August 31, 1990 Safety Assessment, the NRC Staff assumed a fluence value of 2.6×10^{19} neutrons/cm^2. This value was used in anticipation of receiving revised fluence estimates from the Licensee in response to questions raised in the course of the review of the initial Licensee submittal of July 5, 1990. The Licensee submittal on July 5, 1990 assumed a value of 2.16×10^{19} neutrons/cm.

In a letter dated September 28, 1990 (i.e., subsequent to issuance of the NRC Staff's Safety Assessment), the Licensee reported a preliminary revised fluence estimate for the inside surface of the pressure vessel of 2.6×10^{19} neutron/cm^2 for the end of Cycle 21. In a letter dated February 20, 1991, the Licensee provided its final estimate of fluence based upon calculational assumptions reviewed and accepted by the NRC Staff. The final value of fluence was estimated as 2.58×10^{19} neutrons/cm^2 . This final value included a bias correction (increase) of about 13% as well as an additional increase of 7% to account for the presence of some structural materials (both increases relative to the July 5, 1990 value of 2.16×10^{19} neutrons/cm^2). This value is slightly less than the preliminary value reported on September 28, 1990, and slightly less than the value assumed by the NRC Staff in its Safety Assessment.

Since the fluence value assumed by the NRC Staff (2.6×10^{19} neutrons/cm^2) for its assessment is greater than or equal to the initial (2.16×10^{19} neutrons/cm^2), preliminary revised (2.6×10^{19} neutrons/cm^2), and final revised (2.58×10^{19} neutrons/cm^2) values submitted by the Licensee in July, 1990, September 1990, and February 1991, respectively, the effect of the revised values is enveloped by the NRC staff's August 31, 1990 Safety Assessment.

2. Operating Temperature

Petitioners' Contentions

Yankee Rowe has been operated with a cold leg temperature below the cold leg temperature range specified in Regulatory Guide 1.99. Thus, in calculating RT_{NDT}, the NRC Staff's Safety Assessment made a 50°F correction for the temperature difference. However, NRC RG 1.99 requires that the use of such a correction figure "should be justified by reference to actual data." The Petitioners contend that the NRC Staff's Safety Assessment references no "actual data" from testing of materials identical to those used in the vessel, and which have been irradiated to the same degree and at the same temperature as the vessel.

NRC Staff Response

The RG 1.99, Revision 2, guidance to justify a temperature correction by reference to actual data does not impose a restriction to use materials identical to those used in the vessel. The NRC Staff made use of irradiation temperature effect data obtained from an assessment of literature regarding typical reactor pressure vessel steels. These data were used to develop a conservative temperature correction of 1°F for each 1°F below the nominal 550°F irradiation temperature -- for a correction of 50°F for weld metal. The NRC Staff's literature survey is discussed on page 10 of the August 31, 1990, Safety Assessment.

The temperature effects data used by the NRC Staff were specific to a typical pressure vessel plate material (A533-B) and to the so-called "Linde 80" welds, referring to the welding flux used for the Yankee Rowe reactor vessel. Although the plate material used in the Yankee Rowe reactor pressure vessel is A302-B, the A533-B data are judged to reflect the temperature effects trend for A302-B plate. However, the A533-B data do not reflect the absolute degree of irradiation damage. Moreover, the Linde 80 weld data are judged to correctly reflect the trends and degree of irradiation damage for the welds used in fabricating the Yankee Rowe reactor pressure vessel. Since the irradiation temperatures were applicable to the Yankee Rowe operating conditions, the NRC Staff concludes that this temperature correction is based on appropriate conditions and materials, and that it represents a conservative temperature correction.

3. Vessel Chemical Composition

Petitioners' Contentions

The chemistry of the pressure vessel beltline materials is important to the estimates of the RT_{NDT}. The chemistry factor at Yankee Rowe is, to a critical extent, simply unknown. Although the chemical composition and heat numbers for the Yankee Rowe upper plate and lower plate are known, no data exist regarding the chemical composition of axial welds and the circumferential weld. Because this information was missing, in order to calculate the reference temperature for Yankee Rowe, the NRC Staff turned to "bounding values" recommended by RG 1.99 for instances in which data are unavailable. The Petitioners, however, contend that the NRC Staff's use of bounding values in this case was both inaccurate and inappropriate. First, they contend that the NRC Staff failed to use the RG recommended value of 1.0% for nickel, and instead, without justification, used a less conservative figure of 0.70%. This error could have a significant impact on the NRC Staff's reference temperature calculations. Moreover, what little data are available from Yankee Rowe demonstrates that the RG recommendations cannot be relied upon. Data from early test specimens, which were removed from the pressure vessel years ago, indicate that the chemistry factor based on the copper and nickel content is substantially greater than the chemistry factors in RG 1.99. Therefore, as the NRC Staff has observed, the increase in reference temperature resulting from neutron irradiation, may be greater for the Yankee Rowe vessel beltline materials than predicted by RG 1.99. The Petitioners contend that this observation was not taken into account in the Safety Assessment when the NRC Staff made its reference temperature calculations.

NRC Staff Response

The chemistry of the welds in the Yankee Rowe pressure vessel is unknown. However, the welds in the Yankee Rowe vessel were fabricated by the Babcock & Wilcox Company (B&W) using materials and processes that were very similar to those used in fabricating the reactor pressure vessel for Belgian Reactor No. 3 (BR-3). Copper and nickel content for BR-3 has been determined to be 0.18 wt.% Cu and 0.7 wt.% Ni.

Further, B&W fabricated several other pressure vessels using materials and processes similar to those used in fabricating the Yankee Rowe reactor pressure vessel. Chemistry values for those materials are known and have been reported in Topical Report BAW-1799, dated July 1983 (Reference 13). Based on those results, the population average nickel content is 0.60 wt.%, and the population average copper content is 0.27 wt.% and the range of mean values of copper was 0.18 to 0.35%. B&W also reports that the mean copper concentration in the range of 0.27-0.31 wt.% reflects a 95% confidence interval and the largest standard deviation for any heat of weld wire was 0.07wt.%. In view of this information, the NRC Staff used a copper content of 0.35 wt.% and a nickel content of 0.7 wt.% as conservative estimates of the Yankee Rowe chemistry.

The NRC Staff's Safety Assessment made use of the historical data from the Yankee Rowe reactor pressure vessel manufacturer. Given the similarity between materials and processes among the several pressure vessels fabricated by B&W, the population averages were considered to be generally representative of the

Yankee Rowe pressure vessel chemistry. However, the upper values of 0.35 wt.% Cu and 0.70 wt.% Ni were used as conservative values.

With regard to the contention that copper and nickel content were substantially greater than the chemistry factors in RG 1.99, the surveillance data is from samples removed from the upper shell plate. RG 1.99, Revision 2 predicts a chemistry factor of 90.1 for the upper shell plate. The NRC staff estimate of the reference temperature for the Yankee Rowe upper and lower shell plates is based on the analysis of the surveillance data performed by Odette. This resulted in a chemistry factor of 184. The NRC Staff used this chemistry factor in determining the amount of embrittlement in the upper and lower shell plates. Hence, the NRC Staff assessment took into account the fact that the amount of embrittlement from the surveillance data exceeded the values predicted by the RG.

The NRC Staff's evaluation of irradiation damage trends for the plate materials is based on a limited data base for coarse grain materials. The RT_{NDT} estimates for the plate materials conservatively account for the microstructure, the chemistry, and the irradiation temperature.

D. Safety Goals for the Operation of Nuclear Power Plants (Policy Statement)

On August 4, 1986, the Commission published its Policy Statement (Reference 14) on Safety Goals for the Operation of Nuclear Power Plants. The Commission emphasized that current regulatory practices are believed to ensure that the basic statutory requirement, adequate protection, is met. However, the Commission's Policy Statement expressed the Commission's views "on the level of risks to public health and safety that the industry should strive for in its nuclear power plants." Therefore, the Commission adopted two qualitative safety goals that are supported by quantitative health effects objectives for use in the regulatory decision making process. The two qualitative safety goals are as follows:

a. Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

b. Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The Commission adopted the following two health effects as the quantitative objectives concerning mortality risks to be used in determining achievement of the qualitative safety goals:

a. The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

b. The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from other causes.

In adopting these quantitative objectives, the Commission emphasized that they establish guidance that nuclear power plant designers and operators should strive to meet, but that because of limitations in the state of the art of quantitatively estimating risks, these quantitative health effects objectives "are not a substitute for existing regulations." Finally, the Commission placed in context the intended use of the safety goals:

> "To provide adequate protection of the public health and safety, current NRC regulations require conservatism in design, construction, testing, operation and maintenance of nuclear power plants. A defense-in-depth approach has been mandated in order to prevent accidents from happening and to mitigate their consequences. Siting in less populated areas is emphasized. Furthermore, emergency response capabilities are mandated to provide additional defense-in-depth protection to the surrounding population. These safety goals and these implementation guidelines are not meant as a substitute for NRC's regulations and do not relieve nuclear power plant permittees and Licensees from complying with regulations. Nor are the safety goals and these implementation guidelines in and of themselves meant to serve as a sole basis for licensing decisions. However, if pursuant to these guidelines, information is developed that is applicable to a particular licensing decision, it may be considered as one factor in the licensing decision."

The Commission Safety Goal contains no requirements. The Safety Goal Policy Statement describes Qualitative Safety Goals, Quantitative Objectives, and Guidelines for Regulatory Implementation. Among the Guidelines for Regulatory Implementation is the following which was proposed by the Commission for further NRC Staff consideration:

> "Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance for containment systems, the overall mean frequency of a large release of

radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

Petitioners' Contentions

Petitioners contend that the NRC Staff's decision to allow the Licensee to continue to operate Yankee Rowe is "flatly inconsistent with the Commission's "Safety Goal" Policy that the risk of a severe accident should be kept to less than one chance in a million." Petitioners contend that this conclusion is based on the NRC Staff's own calculation that the risk of pressure vessel rupture is between $5x10^{-5}$ and $5x10^{-6}$* and is thus greater than the Commission's large release guidance of $1x10^{-6}$ per reactor year (that is, one in a million reactor years).

NRC Staff Response

The Safety Goal is not, and was never intended to be, a measure of adequate protection of public health and safety. Rather, the Safety Goal is a higher level of safety that the Commission believes the industry should strive to achieve. As noted above, the Commission's Policy Statement on Safety Goals (Reference 17) states the following:

"Current regulatory practices are believed to ensure that the basic statutory requirement, adequate protection of the public, is met."

*The NRC Staff conservatively concluded that the probability of a PTS event causing a reactor vessel through-wall crack is in the range $1x10^{-5}$ to $1x10^{-4}$ per reactor year. The NRC Staff did not calculate the probability of a large release.

Also:

"This statement of NRC safety policy expresses the Commission's views on the level of risks to public health and safety that the industry should strive for in its nuclear power plants."

The NRC Staff's decisions regarding plant operation are based upon adequate protection of the public health and safety, not the Commission's Safety Goal Policy.

E. <u>Compliance with the Requirements of Appendix H to 10 CFR Part 50 - Reactor Vessel Material Surveillance Program Requirements</u>

10 CFR Section 50.60 requires that:

1. All lightwater nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to this Part, except as provided in Paragraph (b) of this Section. Section 50.60(a).
2. Proposed alternatives to the requirements described in Appendices G and H of this Part or portions thereof may be used when an exemption is granted by the Commission under Section 50.12. Section 50.60(b).

Thus, Appendix H to 10 CFR Part 50 applies to the Yankee Rowe facility. The requirements of Appendix H are:

1. Except for certain reactor vessels receiving a limited neutron fluence (the Yankee Rowe vessel does not meet this exception), all reactor vessels must have their beltline materials monitored. Section II.B.
2. That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirements of the edition of American Society for Testing Materials (ASTM) Standard No. E 185 that

is current on the issue date of the ASME Code to which the reactor vessel was purchased. For each capsule withdrawal after July 26, 1983, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practical for the configuration of the specimens in the capsule. For each capsule withdrawal prior to July 26, 1983, either the 1973, the 1979, or the 1982 edition of ASTM E 185 may be used. Section II.B.1.

3. Surveillance specimen capsules must be located near the inside vessel wall in the beltline region. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds must be done according to the requirements for permanent structural attachments to reactor vessels given in Sections III and XI of the ASME Code. The design and location of the capsule holders shall permit insertion of replacement capsules. Section II.B.2.

4. A proposed withdrawal schedule must be submitted with a technical justification as specified in Section 50.4. The proposed schedule must be approved prior to implementation. Section II.B.3.

5. Each capsule withdrawal and the test results must be the subject of a summary technical report to be submitted within one year after capsule withdrawal, unless an extension is granted by the Director of NRR. Section III.A.

6. The report must include the data required by ASTM E 185, as specified in Paragraph II.B.1 of the Appendix, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions. Section III.B.

7. If a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures to meet the limits, the expected date for submittal of the revised Technical Specifications must be provided with the report. Section III.C.

Petitioners' Contentions

Petitioners contend that Appendix H to 10 CFR Part 50 requires installation in the reactor vessel of specimens of materials (including welds) used in fabricating the reactor pressure vessel. These specimens are to be removed periodically and tested in order to determine the extent to which the vessel beltline materials have been damaged by the neutron radiation. Fracture toughness test data from these specimens are used to determine whether the requirements of 10 CFR Part 50, Appendix G, are met.

Petitioners further contend that the Licensee has no material surveillance program for Yankee Rowe. A few surveillance specimens were placed in the vessel when it began operating, but the last ones were removed many years ago. Thus, the Licensee and NRC Staff have no test data on which to base their calculations of RT_{NDT} and USE to assure the integrity of the Yankee Rowe pressure vessel, and the Licensee is in flagrant violation of Commission regulations.

NRC Staff Response

The Licensee had a surveillance program that started with the second cycle. It did meet the requirements of ASTM E 185, 1961. Failure of the surveillance specimen holders resulted in the program being terminated in 1965. The materials were exposed to fluences ranging from $2x10^{18}$ to $9x10^{19}$ neutrons/cm^2 and five capsules were tested. The expected fluence at end-of-life is less than $9x10^{19}$ neutrons/cm^2. The NRC Staff approved pressure-temperature limits in July of 1976 based on that surveillance program and concluded that there were sufficient data to cover the operating life of the reactor vessel.

The Yankee Rowe surveillance program was reviewed again in 1979. That review is documented in NUREG-0569 (Reference 1), which concluded that the Licensee met the purposes of Appendix H with the surveillance data that were available.

The safety question not addressed by the Yankee Rowe program is the effect of neutron irradiation on the fracture behavior of the axial and circumferential welds. However, the NRC Staff has developed RG 1.99 to provide a method for estimating irradiation damage when surveillance data are not available. While this RG was developed for irradiation temperatures greater than 525°F, the NRC Staff has described a method to conservatively account for the lower irradiation temperature at Yankee Rowe. Although these methods are approximate, they do provide a conservative bound to the irradiation damage for the Yankee Rowe materials. Thus, the safety issue consideration is satisfied.

On July 26, 1983, the provisions of 10 CFR Section 50.60 became applicable to the Yankee Rowe facility and the Licensee was required to either have a surveillance program that met the requirements of 10 CFR Part 50, Appendix H, or to request an exemption. The Licensee has not requested an exemption and believes it is in compliance with Appendix H based upon the prior in-vessel surveillance program, the BR3 Belgian surveillance program and the accelerated testing program at the University of Michigan. The staff believes the Licensee should have requested an exemption or documented how it intended to comply with Appendix H. The NRC Staff will continue to consider this matter. If a determination is made that a violation exists, the NRC Staff will consider appropriate enforcement action in accordance with the NRC Enforcement Policy.

F. Requirements of 10 CFR Section 50.55a(g) - Compliance with the Inservice Inspection Requirements

All light water reactors whose construction permit was issued prior to January 1, 1971, shall meet the following requirements of 10 CFR Section 50.55a.

1. Components that are part of the reactor coolant pressure boundary and their supports shall meet the requirements applicable to components classified as ASME Code Class 1. Section 50.55a(g)(i).
2. Section XI of the ASME Boiler and Pressure Vessel Code (and effective Addenda) applies (to the extent practical) to all ASME Code Class 1, 2, and 3 components and their supports. Exceptions to this are that design and installation for inspection access provisions are not required and preservice examination requirements do not apply. Section 50.55a(g)(4).
3. Inspection programs will be developed to cover 120-month intervals. The inservice inspection (ISI) program, inservice testing (IST) program (of pumps and valves) and system pressure tests established to meet the above requirements shall comply with the edition of the Code in effect 12 months prior to the beginning of the 120-month inspection interval. Section 50.55a(g)(4)(i) and (ii).
4. Subject to Commission approval, Licensees may use later editions of the code provided that all provisions of the later edition or addenda are followed. Section 50.55a(g)(4)(iv).
5. If a revised ISI program conflicts with the facility Technical Specifications (TS), the Licensee shall request a change to the TS at least six months before the ISI program becomes effective. Section 50.55(a)(g)(5)(ii).

6. If an inspection or test requirement required by the Code is determined to be impractical by the Licensee, the Licensee shall demonstrate the basis for this determination to later than 12 months after the end of the respective 120-month inspection interval. The Commission may grant relief from the requirements deemed to be impractical or may impose alternate tests or inspections. Section 50.55(a)(g)(6).

Petitioners' Contentions

The Petitioners contend that NRC regulations require the reactor vessel wall and welds to be inspected approximately once every ten years using ultrasonic testing. The purpose of such inspections is to determine the size and orientation of any flaws, such as cracks, within vessel wall materials and whether any such flaws have increased in size since the last inspection. The Petitioners contend that the Yankee Rowe pressure vessel has never been inspected ultrasonically for cracks. The Petitioners contend that periodic inspection of the Yankee Rowe vessel by ultrasonic testing is not possible for most of the vessel because of its unique cladding, which is spot welded rather than bonded to the vessel. The space between the vessel and the cladding thwarts ultrasonic testing, because the ultrasonic waves reflect back from the vessel wall rather than penetrating it.

The Petitioners contend that Yankee Rowe is in clear violation of NRC requirements for beltline inspection and, in the absence of any assurance that cracks or flaws in the Yankee Rowe vessel are minor or nonexistent, the operation of the Yankee Rowe facility poses a serious threat to public health and safety.

NRC Staff Response

The Licensee discussed its ultrasonic inspection program in YAEC Report No. 1735, dated July 1990 (Reference 2). The beltline welds in the Yankee Rowe reactor vessel were volumetrically examined by radiography as a part of its fabrication quality control. All flaws detected that exceeded the acceptance criteria were removed and repaired. Although the Licensee has not ultrasonically examined the beltline welds since the plant has been in service, it has examined other welds in the reactor vessel and observed no unacceptable indications.

During the course of its operating period, accessible portions of the Yankee Rowe reactor vessel and highly stressed regions of the pressurizer near the surge nozzle were inspected ultrasonically. Cracks in the cladding were identified. However, none of these flaws have penetrated into the ferritic steel suggesting that service induced growth of cracks in the ferritic steel will be minimal or non-existent.

The low operating stresses in the reactor vessel beltline are unlikely to initiate flaws in service. The reactor vessel was designed to Section VIII of the ASME Boiler and Pressure Vessel Code, 1956 edition plus Winter 1957 Addenda. That code has an allowable membrane stress of only 20 Ksi as opposed to the current applicable ASME Code, Section III, which allows 26.7 Ksi. The actual stress in the beltline region due to pressure is about 16 Ksi.

In summary, (1) the volumetric nondestructive examinations of the beltline welds during fabrication did not reveal any unacceptable flaws, (2) volumetric nondestructive examinations of other welds in the vessel during service have shown no unacceptable indications and (3) low beltline stresses make the probability of service induced flaw growth unlikely. Although the possibility of initiation of a flaw in service cannot be eliminated, the NRC Staff believes that flaws which could be of concern are not present in the Yankee Rowe reactor vessel.

With regard to the Yankee Rowe facility's compliance with ISI requirements, specifically 10 CFR Section 50.55a, the Yankee Rowe facility received its construction permit prior to January 1, 1971. The plant was not required to be designed to provide access provisions for inservice inspection. The revisions to 10 CFR Section 50.55a published in February 1976 required that ISI Programs be updated to meet the requirements (to the extent practical) of the Edition and Addenda of Section XI of the ASME Code incorporated in the regulation by reference in Section 50.55a(b). As specified in the February 1976 revision, for plants with Operating Licenses issued prior to March 1, 1976, the regulations became effective after September 1, 1976, at the start of the next regular 40-month inspection period.

The Licensee sought relief in 1982 from the requirement to volumetrically examine the reactor vessel shell welds in the core region and outside the core region. The basis for this request was that the welds were totally inaccessible due to the Neutron Shield Tank outside the vessel, and the presence of a non-removable thermal shield inside the vessel.

The Licensee's request for relief from the ISI requirements was granted (Reference 24) based on the impracticality of performing the inspection, supported by the Systematic Evaluation Program evaluation of the pressure vessel integrity, principally the low stresses and the determination of adequate Charpy USE.

The NRC Staff decision to grant relief is further supported by the following considerations:

a. The pressure vessel received a preservice inspection which did not reveal any flaw-like indications.

b. The low operating stresses are unlikely to initiate flaws.

c. Inspection of the vessel head and the pressurizer have identified flaws in the cladding. However, none of these flaws have penetrated into the ferritic steel, suggesting that service induced growth in the ferritic steel will be minimal, if any.

Recent advances in manipulation tooling and UT transducer design have resulted in relatively small transducers that can be used in the narrow annular gap between the thermal shield and the vessel wall. The Licensee has committed to perform a volumetric examination of these welds during the next outage. However, in the early 1980's, this technology did not exist.

The reactor vessel beltline has not been examined since installation. Therefore, the possibility of the initiation of a flaw in service cannot be eliminated in the beltline region. A probabilistic fracture mechanics analysis by the NRC Staff assumed that there were flaws in the vessel and used a distribution developed by Dr. W. Marshall of the United Kingdom (Reference 16), based on both nuclear and non-nuclear vessels. The NRC Staff believes that the flaw density distribution prescribed by Marshall is appropriate, but that the Marshall flaw size distribution may be very conservative compared to results of inspections conducted on other vessels.

The Marshall distribution assumes a random distribution of flaws throughout the volume of the material. The probabilistic fracture mechanics analysis performed for the Yankee Rowe vessel assumed that the flaws were all surface flaws and all oriented in the worst direction. This is a very conservative assumption. Consequently, the flaw distribution assumptions that went into the probabilistic fracture mechanics analyses performed are conservative.

In conclusion, the Yankee Rowe facility meets the requirements for in-service inspection of 10 CFR Section 50.55a based upon relief from the requirements for ISI granted by the NRC Staff as permitted by 10 CFR Section 50.55a

The lack of precise data regarding reactor vessel flaws at the Yankee Rowe facility was taken into account by the NRC Staff in the use of suitably conservative assumptions in its probabilistic fracture mechanics analysis.

G. Consideration Given of the views of an NRC Staff Expert

Petitioners' Contentions

Petitioners contend that the NRC Staff ignored the explicit recommendation of its own expert on pressure vessel integrity, Dr. P. N. Randall, that Yankee Rowe should not be allowed to operate under current conditions. Petitioners also contend that Dr. Randall, who has participated extensively in the development of NRC regulations and regulatory guidance for pressure vessels, advised responsible NRC Staff members on August 13, 1990, that "if restart of Yankee Rowe is allowed at all, it should be conditional on the results of the complete review of the probabilistic safety analysis now underway at Oak Ridge."

NRC Staff Response

In determining to authorize operation for the current cycle, the NRC Staff thoroughly considered the views of Dr. P. N. Randall, an NRC technical NRC Staff member who disagreed with the NRC Staff's August 31, 1990, Safety Assessment. As reflected in four memoranda (Reference 4) between Dr. P. N. Randall and members of the NRC Staff, the NRC Staff was aware of and considered Dr. Randall's views. Subsequently, the NRC's Advisory Committee on Reactor Safeguards (ACRS) reviewed the Yankee Rowe reactor vessel issues, taking into account the views of Dr. Randall. By letter dated September 12, 1990 (Reference 5), the ACRS concurred in the decision of the NRC Staff to permit the Licensee to operate the facility for the additional operating cycle.

In a letter dated July 15, 1991 (Reference 20), Dr. Randall provided additional comments concerning the NRC Staff's Safety Assessment of August 31, 1990. A response to the technical issues raised by Dr. Randall are presented in Reference 26. However, no new information is contained in Dr. Randall's letter.

IV. CONCLUSION

The NRC Staff has reviewed each of the allegations in the Petition to determine if there are significant issues that would warrant immediate shutdown of the plant. The principal safety issue is the likelihood of a plant overcooling transient causing fracture of the Yankee Rowe reactor vessel, thereby releasing radioactivity into the reactor containment and possibly to the outside environment. The ability to predict the likelihood of such an accident requires plant-specific information regarding reactor vessel chemical composition, reactor vessel material irradiation testing, and results of inservice nondestructive examinations. Some of this information is not presently available for the Yankee Rowe vessel, which makes precise estimates of the likelihood of vessel failure difficult. Unavailable information includes results of reactor vessel beltline weld and plate inservice nondestructive examinations, test results to determine the effect of radiation on steel plate samples, and determination of the chemical composition of the actual reactor vessel weld material, specifically the copper and nickel content.

The NRC Staff decision to permit continued operation of Yankee Rowe was reviewed by a systematic process to evaluate the level of protection provided to the public by continued operation of the facility. The NRC Staff first performed a deterministic assessment of Yankee Rowe design features important to PTS response. This review covered primary and secondary system design, component and system flow capacities, system operating and trip logic, system transient response, and normal and emergency procedures. The NRC staff concluded that Yankee Rowe plant specific features are such that severe overcooling transients are less likely at Yankee Rowe than for a typical large

PWR. The NRC Staff also reviewed the Yankee Rowe operating history relative to cooldown or LTOP events and found no significant challenges to the reactor vessel in the plant's 31-year operating history.

The NRC Staff then reviewed the Licensee's claim that all PTS requirements were satisfied. This review found that important uncertainties remain relative to materials properties and that additional information should be gathered to narrow these uncertainties. Therefore, to assist in the decision on restart of the plant following the 1992 outage, the NRC Staff contracted with Oak Ridge National Laboratory for a study to be completed in the fall of 1991 to develop probabilistic insights relative to vessel embrittlement concerns. The NRC Staff also reviewed a probabilistic study provided by the Licensee and modified the Licensee's assumptions as necessary in order to conservatively estimate the frequency of vessel failure due to PTS.

The Licensee's thermal hydraulic response estimates were accepted as conservative because of the low decay heat levels used (500 kw), the assumptions of early and prolonged stagnation in the primary system for the small break loss of coolant (SBLOCA) case and no credit for operator actions. Assumption of low decay heat rate minimizes natural circulation mixing of the warm primary system water with the cold ECCS water in the downcomer region and therefore gives a colder downcomer temperature. Assumption of stagnation for the SBLOCA further decreases mixing by eliminating even the natural circulation effects. This significantly accelerates the cooldown rate and leads to a lower final downcomer temperature. Operator actions in accordance with existing facility emergency procedures would mitigate PTS event severity.

In the fracture mechanics analyses, the NRC Staff made substantial changes to the Licensee's assumptions. Most notable was the assumption regarding copper content in the weld material. The larger the copper content, the more sensitive the vessel is to neutron fluence embrittlement. Based upon data from the vessel manufacturer, the NRC Staff calculations assumed that the weld material contains a large copper content. The vessel failure probability is also particularly sensitive to the presence of pre-existing flaws in the inside surface of the vessel. Again, the NRC Staff made conservative assumptions regarding critical flaw location, size distribution and density. Furthermore, the NRC Staff did not allow credit for a saturation effect due to coarse grain size as claimed by the Licensee.

Using these assumptions, the NRC Staff estimated the frequency of vessel failure due to PTS challenges to conservatively be in the range of 10^{-4} to 10^{-5} per reactor year. Although not every vessel failure event would lead to core damage, the NRC Staff has not attempted to determine the conditional core damage probability. The NRC Staff believes, however, that core damage frequencies in the range of 10^{-4} to 10^{-5} per reactor year (when conservatively estimated), and considered in conjunction with results from a comprehensive deterministic assessment, provide an adequate basis for the conclusion that, for interim operation, there is reasonable assurance of no undue risk to public health and safety. The NRC Staff also concluded, however, that the Licensee should undertake a specific program to narrow the range of uncertainties in materials properties and establish a vessel surveillance program. The NRC Staff judged that such a program should be effected during the next scheduled outage. Interim operation of the plant until the end of Cycle 21 was, therefore, approved by the Director, Office of Nuclear Reactor Regulation.

The NRC Staff has also undertaken studies to gain a better understanding of the sensitivity of vessel failure probability to the various parameters which affect vessel integrity. The sensitivity study addresses both the thermal hydraulic response of the reactor system given a limiting small break LOCA, and the vessel response for the cooldown transient. The effect of thermal hydraulic analysis assumptions were examined as well as the effect of variations in weld copper content, irradiation temperature, flaw size distribution and other parameters which could effect the conditional vessel failure probability. The results of these studies quantitatively support the previous NRC Staff conclusion that the estimates used for the frequency of core damage from PTS events are conservative.

The decision to permit interim operation of Yankee Rowe is based on the fact that the NRC Staff conclusion that operation of Yankee Rowe through the end of the fuel cycle poses no undue risk to the public health and safety. The staff has not concluded that operation of Yankee Rowe beyond Cycle 21 (approximately April 1992) would result in a lack of adequate protection of public health and safety. Rather, the NRC Staff has judged it prudent to draw the line that operation beyond the present cycle requires submission of the information discussed above and a subsequent Director's Decision pursuant to 10 CFR 50.60 and 10 CFR 50.61.

The institution of proceedings pursuant to 10 CFR Section 2.206, as requested by Petitioners, is appropriate only where substantial health and safety issues have been raised. _See_ _Consolidated Edison Company of New York_ (Indian Point, Units 1, 2 and 3), CLI-75-8, 2 NRC 173, 175 (1975), and _Washington Public Power System_ (WPPS Nuclear Project No. 2), DD-84-7, 19 NRC 899, 923 (1984). As discussed above, there is reasonable assurance that the Yankee Rowe facility may continue to operate through Cycle 21 with adequate protection of the public health and safety. Based on the foregoing, the Commission has found that institution of a proceeding pursuant to 10 CFR Section 2.202 to modify, suspend, or revoke the NRC license held by Yankee Atomic Power Company is not warranted. As provided in 10 CFR Section 2.206(c), a copy of the Decision will be filed with the Secretary of the Commission for its review.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

REFERENCES

1. NUREG-0569 - Evaluation of the Integrity SEP Reactor Vessels - December 1979.

2. YAEC 1735 - Yankee Reactor Pressure Vessel Evaluation - July 1990.

3. Letter from Thomas E. Murley (NRC) to Andrew C. Kodak (Yankee) - "Yankee Rowe Reactor Vessel" - August 31, 1990.

4. Four Memorandums Between Dr. Pryor N. Randall and Members of NRC Staff:

    a. Memorandum for James E. Richardson from Dr. Pryor N. Randall dated: July 31, 1990.

    b. Memorandum for Dr. Pryor N. Randall from James E. Richardson dated: September 10, 1990.

    c. Memorandum for Thomas E. Murley from Dr. Pryor N. Randall dated: September 11, 1990.

    d. Memorandum for Dr. Pryor N. Randall from Thomas E. Murley dated: September 18, 1990.

5. Letter from Carlyle Michelson (ACRS) to Kenneth Carr (NRC) - "Yankee Rowe Reactor Pressure Vessel Integrity" - September 12, 1990.

6. Letter from George Papanic (Yankee) to George Lear (NRC) - "Pressurized Thermal Shock Rule - 10 CFR 50.61(b)" - January 22, 1986.

7. Letter from George Papanic (Yankee) to Eileen McKenna (NRC) - "Pressurized Thermal Shock Rule" - August 12, 1989.

8. Letter from George Papanic (Yankee) to Eileen McKenna (NRC) - "Fast Neutron Fluence for the Reactor Vessel - 10 CFR 50.61" - October 28, 1986.

9. Letter from George Papanic (Yankee) to Eileen McKenna (NRC) - "Pressurized Thermal Shock Rule" - February 4, 1987.

10. Letter from John Haseltine (Yankee) to Mr. William Russell (NRC) - "Reactor Vessel Fluence Assessment" - September 28, 1990.

11. Letter from John Haseltine (Yankee) to Patrick Sears (NRC) - "Reactor Pressure Vessel Fluence Uncertainty" - February 20, 1991.

12. Memorandum from Thomas E. Murley (NRR) to Raymond F. Fraley (ACRS) - "Yankee Rowe Reactor Vessel Integrity" - October 9, 1990.

13. BAW-1799 B&W 177-FA - Reactor Vessel Beltline Weld Chemistry Study - July 1983.

14. Policy Statement - Federal Register - Vol. 51, No. 162/Thursday, August 21, 1986.

15. Letter from Eileen McKenna (NRC) to George Papanic (Yankee) - "Yankee Nuclear Power Station - Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" - March 10, 1987.

16. "An Assessment of the Integrity of PWR Pressure Vessels" - Dr. W. Marshall, C.B.E., F.R.S. - March 1982.

17. NUREG-0744, Revision 1, - Pressure Vessel Material Fracture Toughness," - October 1982.

18. "Safety Assessment of Yankee Rowe Vessel," - transmitted by letter, T. Murley to A. Kadak, dated August 31, 1990.

19. NRC Safety Evaluation transmitted by letter dated March 10, 1987, - from Eileen McKenna (NRC) to YAEC.

20. Letter from P. N. Randall to NRC Chairman I. Selin dated July 15, 1991.

21. Letter from W. Bamford (ASME) to NRC dated November 20, 1989.

22. Letter to YR 5/1/90.

23. Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials" - dated May 1988.

24. Letter entitled "Inservice Inspection (ISI) Technical Specifications and Relief" - dated January 3, 1984, D. Crutchfield (NRC) to J. Kay (YAEC).

25. Letter from YAEC to NRC dated July 5, 1990.

26. Response to technical issues raised in Dr. Pryor N. Randall's July 15, 1991 Letter to the Chairman, Enclosure 4 to Commission Paper dated July 24, 1991.

UNITED STATES NUCLEAR REGULATORY COMMISSION

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE NUCLEAR POWER STATION (ROWE)

DOCKET NO. 50-029

ISSUANCE OF DECISION

UNDER 10 CFR SECTION 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has issued a Decision concerning a request filed pursuant to 10 CFR 2.206 by Ms. Diane Curran on behalf of the Union of Concerned Scientists and the New England Coalition on Nuclear Pollution. The Petition sought the immediate shutdown of the Yankee Nuclear Power Station (Yankee Rowe) of the Yankee Atomic Electric Company (Licensee) based upon allegations that the continued operation of the Yankee Rowe facility poses a serious threat to public health and safety. The Petition further requested that the Yankee Rowe facility remain shut down until it complies with regulatory requirements and that the Commission provide a public hearing, with rights of discovery and cross examination, to determine regulatory compliance before permitting the facility to resume operation.

The Commission has determined that the Petition should be denied. The reasons for this determination are explained in the "Decision under 10 CFR Section 2.206," (DD 91-), which is available for public inspection in the Commission's Public Document Room, the Gelman Building, Lower Level, 2120 L Street, NW, Washington, DC 20555 and at the Local Public Document Room for the Yankee Rowe facility located at the Greenfield Community College Library, 1 College Drive, Greenfield, Massachusetts 01301.

A copy of the Decision will be filed with the Office of the Secretary.

Dated at Rockville, Maryland, this day of July, 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

ENCLOSURE 2

RESPONSES TO STAFF REQUIREMENTS MEMORANDUM (M910711A)

On July 19, 1991, the Commission issued a staff requirements memorandum (SRM) related to the briefing on Yankee Rowe Pressure Vessel Embrittlement Issues held on July 11, 1991. The responses to the SRM are as follows:

SRM QUESTION 1

The Commission requested the staff to provide an analysis of the uncertainties associated with the factors that make up the calculations of the probability of a PTS event (i.e. the uncertainty in the RT_{NDT} values, the upper shelf energy, the probability of various initiating events, and the conditional vessel failure probability). The staff should also indicate what factors are the most sensitive in causing large variations in the final calculated values of a PTS vessel failure.

Staff Response

See Attachment R1

SRM QUESTION 2

Relative to the $5X10^{-6}$ probability value which was used as a base for the screening temperatures in 10 CFR 50.61; the Chairman requested the staff to indicate what portion of the value was attributable to the various events causing the transient and what portion was attributable to the conditional failure probability of the vessel.

Staff Response

See Attachment R2

SRM QUESTION 3

Commissioner Rogers requested the licensee to submit a set of curves for the RT_{NDT} that shows the error bands of the data used to calculate the temperature values.

Licensee Response

See Attachment R3

SRM QUESTION 4

Commissioner Curtiss asked UCS/NECNP to comment on the assumptions used by the staff in evaluating the integrity of the reactor vessel (UCS/NECNP, as well as the licensee, were subsequently provided with a copy of the staff's memo setting forth bases for the various assumptions.)

UCS/NECNP Response

The UCS/NECNP has not yet submitted the information requested.

SRM QUESTION 5

Commissioner Curtiss requested the staff to provide information on whether there are facilities other than Yankee Rowe where the Upper-Shelf Energy (USE) requirements of 10 CFR 50, Appendix G, Section IV.A.1 are not met, and whether, if such cases exist, not only has the analysis authorized under IV.1 been done by the Director of NRR but the licensee has also been required to meet V.C.1 as well.

Staff Response

See Attachment R5

SRM QUESTION 6

Commissioner Curtiss requested the staff to provide for the record their positions on whether the requirements of V.C. in Appendix G to 10 CFR 50 apply if the Upper Shelf Energy Values specified in IV.A.1 are not met.

Staff Response

See Attachment R5

SRM QUESTION 7

The staff agreed to provide the Commission with a list of any other plants which have required an exemption to the requirement of 10 CFR 50, Appendix H.

Staff Response

See Attachment R6

SRM QUESTION 8

Commissioner Curtiss requested the staff to provide the Commission with an evaluation of the impact of raising the temperatures of the ECC injection water as specified by the licensee.

Staff Response

See Attachment R7

SRM RESPONSE 1

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

OFFICE OF THE
EXECUTIVE DIRECTOR
FOR OPERATIONS

July 24, 1991

TO: D. Rathbun, OCM/IS
J. Scarborough, OCM/KR
J. Gray, OCM/JC
J. Guttman, OCM/FR

FROM: James L. Blaha, AO/OEDO

SUBJECT: YANKEE ROWE INFORMATION

Enclosed is the staff's PTS sensitivity analysis for Yankee Rowe. This analysis will be provided again in the overall staff proposal later today.

Jim

James L. Blaha, AO/OEDO

Enclosure:
As stated

cc: J. Taylor, EDO
J. Sniezek, DEDR
SECY
OGC

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

July 23, 1991

MEMORANDUM FOR: James L. Blaha, Assistant for Operations
Office of the Executive Director for Operations

FROM: William T. Russell, Associate Director
for Inspection and Technical Assessment
Office of Nuclear Reactor Regulation

SUBJECT: STAFF'S PTS SENSITIVITY ANALYSIS FOR
THE YANKEE ROWE NUCLEAR POWER STATION

The enclosed sensitivity analysis is provided in accordance with my July 16, 1991, memorandum on the same subject.

It is important to note in using the study results that the interaction of the variables is highly nonlinear. As a result it is inappropriate to attempt to estimate combined effects from plots of sensitivity to single parameter variation.

The Staff has also attempted to perform a best estimate calculation of the probability of reactor pressure vessel failure due to thermal shock. The thermal-hydraulic transient response for a limiting small break LOCA was modified to incorporate the effect of expected operator actions. The expected action is to throttle ECCS flow about 25 minutes after the start of the event. The effect of this action is a more rapid reduction in primary system pressure. All other inputs being the same as the limiting case SBLOCA (i.e., high copper, Marshall flaw distribution, etc.) this action results in a reduction in conditional vessel failure probability by a factor of two.

With regard to the materials aspects of the problem, the staff determined that the models currently at its disposal are conservatively oriented. In order to evaluate the degree of conservatism used in estimating the conditional probability of vessel failure, the NRC staff decided it would be desirable to perform a "best estimate" calculation of this failure probability. However, after considerable study and debate, it was not possible to perform a "best estimate" calculation at this time. The two computer codes (OCA-P and VISA-II) used for these calculations need to be modified to include the current embrittlement estimates described in Regulatory Guide 1.99, Revision 2. In addition, both codes conservatively assume that all flaws are located at the inner surface of the vessel. A "best estimate" calculation should treat the flaws as distributed throughout the volume of the vessel. An informal survey of expert opinion performed to support this study suggests that the Marshall distribution that was used is believed to be a conservative representation for weld material, but there is no opinion on a "best estimate" distribution for plate materials. Finally, the amount of copper in the welds and the effects of lower irradiation temperature are simply unknown and "best estimates" cannot be made. Therefore a "best estimate" calculation cannot be performed until the computer codes are modified and more is known regarding flaw distribution, weld chemistry and the effects of low irradiation temperature.

Please forward this information to the Commission. If you receive further guidance on the Commission's request, please let me know as soon as possible.

W T Russell

William T. Russell, Associate Director
for Inspection and Technical Assessment
Office of Nuclear Reactor Regulation

Systems Assessment

Uncertainties are associated with the initiating events both in the quantification of the frequency of an event as well as the thermal-hydraulic analysis used to calculate the final downcomer temperature and system pressure. Yankee Atomic's analyses for the Yankee Rowe plant addressed a broad spectrum of initiating events including steam line breaks, small break Loss-of-Coolant-Accidents, and overcooling transients. The Yankee analyses accounted for the plant-specific features of Yankee Rowe. The following table summarizes the Yankee Atomic estimates of uncertainty for important initiating events.

Yankee Estimates

Category	Specific Initiating	Mean Frequency	5%	95%
SBLOCA(1)	1-2 inch	5.2E-4	2.3E-5	1.6E-3
SLB(2)	Inside VC(3) greater than 6"	3.1E-5	1.1E-6	1.1E-4
SLB	Outside VC Upstream NRV(4) greater than 6"	9.2E-5	3.3E-6	3.3E-4
SLB	Downstream of NRV greater than 6"	1.5E-4	5.6E-6	5.6E-4
Feedline Rupture	Inside VC greater than 6"	3.6E-5	1.5E-6	1.5E-4
All Others	Not significant for PTS			

(1) Small Break Loss of Coolant Accident
(2) Steam Line Break
(3) Vapor Containment
(4) Non-Return Valves

As discussed in the Staff's August 31, 1990 safety assessment, the small break LOCA (1.31 inch diameter) was found to be the most limiting event relative to PTS concerns. For comparison with the licensee uncertainty values given in the above Table, we note that NUREG/CR-4550, Vol.3, Rev.1, Part 1, "Analysis of Core Damage Frequency for Surry, Unit 1 Internal Events" lists a small break LOCA (diameters from 0.5 inches to 2 inches) frequency range of 1E-2 to 1E-4 per reactor year. The mean value given is 1E-3 per reactor year.

However, it should be noted that if piping smaller than 1 inch diameter is excluded (as is appropriate for consideration of PTS) the SB-LOCA frequency would be reduced.

With regard to the probability of a large main steamline break (upstream of the non-return valves) we note that even for an increase in probability by a factor of 10 (i.e. to a mean value of 1E-03 per reactor year) the conditional vessel failure probability is so low (7E-04) that the overall likelihood of core damage (7E-07 per reactor year) is clearly bounded by the SB-LOCA scenario. The low conditional vessel failure probability for this main steamline case is due to the relatively high final downcomer temperature (307 degrees Fahrenheit) experienced.

With regard to the uncertainties in the thermal hydraulic response, the parameters of importance to PTS are downcomer temperature and system pressure. The thermal hydraulic response analyses for the important events were estimated by the licensee using the RETRAN code for conditions when forced or natural circulation is present, supplemented by the REMIX code for conditions when flow stagnation is assumed or predicted to occur. The licensee's analyses were judged to be conservative because of the low decay heat levels used, the assumption of early and prolonged stagnation in the primary system, and no credit taken for operator action to mitigate PTS challenges. The assumption of low decay heat rate minimizes natural circulation mixing of the warm primary system water with the cold ECCS water in the downcomer region and therefore gives a colder downcomer temperature. Assumption of stagnation for the SB-LOCA further decreases mixing by eliminating even the natural circulation effects. This significantly accelerates the cooldown rate and leads to a lower final downcomer temperature. The limiting case SB-LOCA reached a minimum downcomer temperature of 151 degrees F at a pressure of 670 psia. Even with consideration of additional failures following steamline/feedline breaks and other transients, the most limiting PTS challenge is from the SB-LOCA case. Steamline break cases led to downcomer temperatures in the range of 200 to 300 degrees F with a maximum pressure of 1550 psia.

Sensitivity Assessment

As noted above, the thermal hydraulic analyses have been performed using conservative assumptions. The most significant conservatism for the limiting case is the effect of stagnation and lack of credit for operator actions. The low decay heat rate is more important under conditions when natural circulation is maintained or restored.

The effect of potential procedural actions is greatest if they result in maintenance or restoration of forced or natural circulation. This will greatly enhance mixing in the downcomer and keep temperatures at a higher level.

Actions which maintain the final downcomer temperature above 200 degrees F are estimated to reduce the conditional vessel failure probability for the limiting SB-LOCA by a factor of 6.* Maintaining the downcomer temperature above 300 degrees F would reduce the failure probability to a negligible value.

Examples of procedural actions are: 1) restarting or maintaining in operation, the main reactor coolant pumps; and 2) throttling of ECCS injection flow. If the main coolant pumps are restarted within about 20 minutes from the initiation of the limiting SB-LOCA (assuming the pumps have been tripped), stagnated conditions would be terminated and the downcomer temperature should remain above approximately 240 degrees F. The conditional vessel failure probability for limiting case SB-LOCA would be reduced from approximately 3E-2 to well below 1E-3.

Throttling of ECCS flow would also reduce the likelihood of vessel failure. For the limiting SB-LOCA, if ECCS flow is reduced by isolating LPI flow after 25 minutes of injection, the conditional vessel failure probability is estimated to be reduced by a factor of 2. Since Yankee Rowe procedures direct operators to reduce flow given signals of both adequate subcooling and pressurizer level, the operators would be expected to take this action in a realistic response to a SB-LOCA.

Potential Design Modifications

There are also possible design feature or hardware modifications which can be made to soften (with regard to PTS) the system response to initiating events. Examples are increasing ECCS water temperatures, or changing the ECCS injection location to the primary system hot legs rather than the cold legs.

Increasing the ECCS water temperature to 160 degrees F (from 120) is estimated to decrease the conditional vessel failure probability for the limiting SB-LOCA by about a factor of 3 to 4. Changing the ECCS injection location is likely to greatly reduce PTS concerns because mixing of the cold ECCS water with warm primary system water would be assured for all events involving the ECCS injection.

It is important to note, however, that there are other considerations which must be addressed prior to any system or procedure modifications. Most importantly, the effect on core cooling response must be evaluated in parallel with PTS concerns to assure that overall risk is reduced.

*Note: Thermal hydraulic sensitivity analyses are all performed using the conservative values of Cu (0.35%).

Summary

The overall probability of a vessel failure from PTS also depends on the initiating event frequency and the transient thermal hydraulic response. Most important in the thermal hydraulic response estimate is the rate of coolant circulation which controls fluid mixing in the vessel downcomer. If actions are taken by the operators to re-establish or maintain forced or natural circulation, a significant reduction in vessel failure probability can be realized. In a realistic scenario it would also be expected that operators would reduce ECCS flow on signals of adequate subcooling and pressurizer level and thereby reduce the conditional vessel failure probability.

Vessel Assessment

During the July 11, 1991, Commission meeting addressing the Yankee Nuclear Power Station, the Commission requested that the staff perform an analysis examining the sensitivity of the calculated probability of failure due to a Pressurized Thermal Shock (PTS) event to changes in the key variables. The results of that analysis are provided herein.

OBJECTIVE

The objective of this study was to illustrate the sensitivity of the calculated probability of pressure vessel failure to changes and uncertainty in the key input variables. Key variables in four areas were examined: material type and properties; fracture mechanics analyses; flaw size distribution and flaw density; and PTS event pressure-temperature-time history.

BASIS OF THE ANALYSIS

The staff's sensitivity study was performed using two computer codes -- the OCA-P code written by Oak Ridge National Laboratory (ORNL), and the VISA-II code written by Pacific Northwest Laboratory (PNL). The OCA-P code was written in the early 1980's and reflects the state-of-the-art in embrittlement estimates and fracture technology at that time. Various modifications to OCA-P have been examined by ORNL since that time, and the code currently is being modified extensively to perform a detailed sensitivity study examining material chemistry and flaw type. However, the original version of OCA-P was used in the present study. There are some limitations in this version of the code, but the staff chose to use it because it is the only publicly available version and the only one with any documentation.

The VISA-II computer code is an adaptation of the VISA code written by the staff. The VISA-II code was written in the mid-1980's and includes embrittlement estimates that are consistent with current regulatory guidance. The VISA-II computer code also is undergoing revision and updating. Again, however, the original version of the code was used in this study because of the public availability and documentation issues.

The OCA-P and VISA-II codes have been compared in the past and again as part of this study. The probability of failure calculated by the two computer codes generally is within one order of magnitude, and for many cases the agreement is within a factor of 2. However, because of different equations used to evaluate the effect of neutron irradiation on the RT_{NDT} (the reference nil-ductility temperature), the sensitivity of the variations in chemistry appears different in the results of the two codes. The embrittlement estimate is the most important difference in the two computer codes, and the on-going modifications to each code include using the current embrittlement estimates described in Regulatory Guide 1.99, Revision 2.

There are three other important differences in OCA-P and VISA-II: stress analysis methods; fracture mechanics equations; and method for defining the flaw size distribution.

Stress Analysis Methods. The OCA-P computer code uses a finite element analysis to perform a one-dimensional radial heat transfer analysis for the PTS temperature-time history. This permits evaluating complex temperature-time histories. Pressure values for each time step are used, in conjunction with the thermal analysis, to determine a stress-time history as a function of position through the pressure vessel wall. This stress-time history is used in the fracture mechanics analysis to determine probability of failure. The thermal analysis has been benchmarked against larger finite element analyses and against experimental data, and the agreement is excellent.

The VISA-II computer code uses an analysis for heat transfer through an infinite flat plate. The results of this analysis for specific test cases agree very well with the OCA-P thermal stress results. However, the VISA-II code can only accommodate five time intervals and uses either a polynomial or an exponential curve fitting scheme to interpolate between the time intervals. Pressure values are provided for each of the five time intervals. Thus, complex pressure-temperature-time histories cannot be analyzed accurately. However, for most PTS transients the VISA-II and OCA-P methods provide similar stress-time histories.

Fracture Mechanics Equations. The fracture mechanics analysis methods in both OCA-P and VISA-II are based on linear-elastic fracture mechanics. The analysis evaluates a stress intensity factor, K_I, for the applied loading and crack shape and size. This value is compared with a material property, K_{Ic}, that varies as a function of temperature. If K_I exceeds K_{Ic} the crack extends in size, termed crack initiation. As the crack extends in size, the K_I decreases because the thermal stress decreases with depth through the wall. If the K_I falls below another material property, K_{Ia}, the crack will stop extending, termed arrest. The K_{Ia} is a function temperature, and increases with position through the wall because the temperature is increasing. Thus, calculating the applied stress intensity factor, K_I, is a central part of the analysis.

The methods for computing K_I in OCA-P and VISA-II are substantially different, with the OCA-P methods being more elegant but much more difficult to use. The calculated K_I values agree reasonably well, with much better agreement for very long flaws. However, for shorter flaws the differences can be important. For the purposes of this sensitivity study, long flaws have been used and the results from the two computer codes are comparable.

Flaw Size Distributions. Both OCA-P and VISA-II use the Marshall flaw size distribution. However, in OCA-P this distribution is embedded in the code, while in VISA-II it is provided in the input data file. Therefore, to examine the effect of flaw size distribution, this study makes use of VISA-II for reasons of convenience and to avoid code changes to OCA-P.

Both OCA-P and VISA-II estimate the probability of failure, given a pressure-temperature-time history, using a Monte Carlo simulation technique. The simulation techniques used in the two codes provide essentially identical results. However, neither code permits including a distribution on the pressure or temperature values. Thus, evaluation of the sensitivity of the calculated values to uncertainty in the pressure-temperature-time history must be performed "outside" the Monte Carlo simulation.

ANALYSIS CONDITIONS AND ASSUMPTION

This sensitivity study examines four areas: material type and properties; fracture mechanics analyses; flaw size distribution and flaw density; and PTS event pressure-temperature-time history.

Material Type and Properties.

The material types examined are weld and plate. The properties of primary concern are the initiation fracture toughness, K_{Ic}, and the arrest fracture toughness, K_{Ia}. The mean values of these properties are estimated by multiplying the lower-bound curves provided in Section XI of the ASME Code by factors of 1.43 for K_{Ic} and 1.25 for K_{Ia}. The resulting curves are conservative estimates of the mean fracture toughness, but the data base is not adequate to improve on this estimate. This approach generally is accepted by the technical community, but the NRC does have an effort underway to examine the current fracture toughness data in an attempt to improve the mean fracture toughness curves.

Changes in fracture toughness are estimated by using the reference nil-ductility temperature, RT_{NDT}. The fracture toughness is plotted as function of (T- RT_{NDT}), providing an index between the generic fracture toughness curves and specific materials and material conditions. The initial RT_{NDT} is determined by material acceptance tests, and for the Yankee Rowe materials was in the range 10-30 °F. With irradiation, the RT_{NDT} increases, effectively lowering the fracture toughness for a given temperature. Thus, any factor that affects RT_{NDT} will affect the fracture toughness estimate and the probability of failure.

The sensitivity study was structured to evaluate the effects on RT_{NDT} of the initial RT_{NDT}, the material chemistry, the irradiation temperature, and the neutron fluence. These variables are examined directly in the Monte Carlo simulations for the welds using the correlations between the variables and RT_{NDT} based on generic materials. However, because the behavior of the plate materials is not believed to be consistent with the generic models, the staff's method for estimating RT_{NDT} is used with fluence and upper plate versus lower plate as the primary variables. This is included in the OCA-P analyses directly as a variation in RT_{NDT}. For the VISA-II analyses, the material chemistry values are varied to achieve the desired RT_{NDT}. However, for this case the chemistry values are taken as point values and the distribution in RT_{NDT} is addressed by a distribution on the initial RT_{NDT}.

Fracture Mechanics Analyses.

The sensitivity study examines the effects of residual stress and flaw orientation -- axial versus circumferential -- because these variables have a significant impact on K_I. The residual stress effect is approximated by an increase in the pressure to provide the desired value of stress on the inner surface of the pressure vessel. This is taken as a point value and is not treated in the Monte Carlo simulations. Because of the relatively large wall thickness in the Yankee Rowe pressure vessel, relatively large increases in pressure are required to achieve the residual stress -- a 6 ksi residual stress requires a pressure of 863 psi.

Flaw orientation effects arise because of stress differences and material property differences. The longitudinal pressure stress is one-half the hoop pressure stress but the thermal stress is approximately equal in these two directions. Thus, the K_I for circumferentially oriented flaws is not simply one-half that for axially oriented flaws. Further, if there were a significant temperature variation around the circumference of the pressure vessel, there would be an associated axial stress distribution that is not addressed by the current calculations. This issue is being considered in other work and it appears that it is not particularly significant. Therefore, it has not been addressed in this study.

Flaw Size Distribution, Density, and Shape.

The flaw size distribution is used in the Monte Carlo simulations to provide a flaw size for each simulation. The distribution is based on the distribution of manufacturing defects, modified by the efficacy of the preservice inspection and repair procedures in detecting and repairing these manufacturing defects. Thus, the flaw size distribution is intended to reflect the state of the pressure vessel as it enters service.

Two flaw size distributions were considered in the sensitivity study: the Marshall distribution, and the Dufrense and Lucia distribution. The Marshall distribution was developed as part of the U.K. assessment of PWR pressure vessel integrity. The specific data base used in developing the distribution is not available, although there have been several reports and papers describing the distribution. The Marshall distribution emphasizes relatively larger flaws -- greater than 1 inch. This distribution has been conservatively extrapolated to smaller flaw sizes important in PTS analyses. It generally is accepted that the Marshall distribution is very conservative for smaller flaws.

The Dufrense and Lucia distribution was developed as part of the French pressure vessel integrity program. It uses a different data base of initial fabrication defects, and offers three assessments of the efficacy of preservice inspection techniques in detecting these defects: pessimistic, average, and optimistic. It is assumed that detected defects are repaired.

This distribution is much less conservative than the Marshall distribution, even for the pessimistic case. While it probably is an improvement over the Marshall distribution, there is no clear consensus on the appropriate flaw size distribution.

Both the Marshall and the Dufrense and Lucia distributions implicitly assume defects in weld material. There is very little information specific to appropriate flaw size distributions for plate materials. An informal survey of expert opinion performed to support this study suggests that the Marshall distribution is believed to be a conservative representation, but there is no opinion on a "best estimate" distribution for plate materials.

For the sensitivity study, both the Marshall distribution and the pessimistic Dufrense and Lucia distribution are used. However, based on the informal survey of expert opinion, the Marshall distribution is taken by the staff to be a "best estimate" for both weld and plate materials.

The flaw density is simply the number of flaws of all sizes located anywhere in the material. Based on the Marshall distribution and engineering judgement, the staff has adopted 45 flaws/m^3 as a mean flaw density value for weld materials. This value is supported by recent work by PNL examining weld removed from the abandoned Midland Unit 1 reactor pressure vessel. That work found approximately 40 flaws/m^3.

The mean flaw density value for plate is taken to be 4.5 flaws/m^3, based on the informal survey of expert opinion. There is very little information to support any value, and the choice of 4.5 flaws/m^3 probably is conservative.

The flaw size distribution and flaw density for welds are based on flaws located throughout the weld volume. However, the PTS analyses make the assumption that all flaws are oriented in the worst direction and that they are surface flaws. This assumption introduces a significant, but currently unquantified, degree of conservatism. The staff has asked ORNL to modify the OCA-P code to explicitly treat different flaw types, treating flaw location through the vessel wall as a random variable. The modified OCA-P code will be used in a sensitivity study specific to the Yankee Rowe pressure vessel. However, for the present study, all flaws have been treated as surface flaws.

The PTS analyses typically have treated all flaws as infinitely long. The basis for this assumption is that the material at the pressure vessel inner surface is highly embrittled and if a flaw initiates it will propagate along the surface before it extends in depth. This behavior has been observed in the thermal shock validation tests performed at ORNL for the NRC. Thus, the long-flaw assumption is reasonable. However, the calculated probability of failure is sensitive to this assumption. Consequently, the sensitivity study has examined this assumption for both plate and weld.

Pressure-Temperature-Time History.

The computed probability of failure is sensitive to the rate of cooldown, the asymptotic temperature reached during the cooldown, and the pressure history. A series of pressure-temperature-time histories were examined in this sensitivity study. These histories reflect the range of transients considered in the Yankee Rowe analysis. However, for the purposes of reporting the analysis results, the licensee's 1.31-inch diameter small break LOCA transient, with a conservative decay heat, was used.

Tables 1-4 describe the staff's sensitivity analysis base cases and ranges for the variables considered.

QUALITY ASSURANCE

The staff's efforts in performing this sensitivity study were performed over a short time period. Thus, detailed quality assurance reviews were not possible. However, the input files were reviewed by an independent engineer, providing a first level of quality assurance for the specific calculations. Further, the approach used for plates versus welds provides a check on "internal" consistency -- similar trends are observed for changes in variables such as flaw distribution for both plate and weld regions. This consistency check provides a second level of review for the calculations.

The computer codes used are the publicly available versions of OCA-P and VISA-II. The codes were written by two different laboratories at different times. While they use similar approaches, there are significant differences in them. These two codes were compared in earlier analyses and generally provide estimates that agree within one order of magnitude. The staff performed limited comparison analyses as part of this study and the agreement for the cases analyzed was within a factor or 5, with OCA-P tending to provide higher estimates. Thus, the two codes have been benchmarked against one another. Since these two independent codes provide comparable results, it is reasonable to conclude that they are functioning as intended.

ANALYSIS RESULTS

The results of the sensitivity study are discussed for welds and plates in separate sections because of the significantly different methods for evaluating RT_{NDT} for the two materials. However, the sensitivities and trends discussed for welds are appropriate for typical materials.

Welds

Figures 1 and 2 demonstrate the sensitivity of the conditional failure probability -- (P(F|E)) is the probability of failure given that the event occurs -- to the copper and nickel content, respectively.

- o For a nickel content of 0.7 wt.%, varying the copper content from 0.18 to 0.35 wt.% changes P(F|E) by a factor of 85.

- o The sensitivity to nickel content is dependent on the copper content. For 0.18 wt.% Cu, varying Ni from 0.6 to 1.0 wt.% changes P(F|E) by a factor of 6. However, for 0.35 wt.% Cu, P(F|E) essentially is insensitive to Ni content.

Figure 3 demonstrates the sensitivity of P(F|E) to fluence for three copper concentrations, all at 0.7 wt.% Ni. For 0.18 wt.% Cu, varying fluence by a factor of 2 varies P(F|E) by a factor of 5. However, for 0.35 wt.% Cu, varying the fluence by a factor of 2 only changes P(F|E) by a factor of 2.

Figure 4 demonstrates the effect of changes in initial RT_{NDT} for two copper concentrations, at 0.7 wt.% Ni. At 0.18 wt.% Cu, changing the initial RT_{NDT} from 0°F to 20°F increases P(F|E) by a factor of 5. However, at 0.35 wt.% Cu, this same increase in initial RT_{NDT} only increases P(F|E) by a factor of 1.6.

Figure 5 demonstrates the effect of irradiation temperature for three copper concentrations, all at 0.7 wt.% Ni. Irradiation temperature is included in the analysis by increasing the initial RT_{NDT} by 1°F/°F for irradiation below 550°F. For Yankee Rowe, that is a 50°F increase. From Figure 5, it is clear that for low Cu, irradiation at 550°F has a very significant effect -- more than 2 orders of magnitude reduction in P(F|E). However, for higher copper concentrations, irradiation temperature has a much smaller effect. Obviously, this is the same trend observed for changes in the initial RT_{NDT}.

The sensitivity of P(F|E) to various levels of residual stress is illustrated in Figure 1, which examines the effect of copper content at three residual stress levels. As with the other variables, residual stress has a relatively large effect at low copper contents, but a much smaller effect at high copper levels.

Figure 6 demonstrates the effect on P(F|E) of the Marshall flaw size distribution versus the Dufrense and Lucia distribution. Figure 6(a) shows the two flaw distributions, and Figure 6(b) shows their effect on P(F|E). This comparison was made for 0.35 wt.% Cu and 0.7 wt.% Ni. For these conditions, changing from the Marshall distribution to the Dufrense and Lucia distribution reduces P(F|E) by a factor of 75. A slightly greater reduction -- a factor of 84 -- was observed for 0.26 wt.% Cu and 0.62 wt.% Ni. The reduced sensitivity to changes in chemistry simply is reflecting the fact that the flaw size distribution does not impact the level of embrittlement.

Figure 7 demonstrates the effect of P(F|E) of changes in the flaw density. This simply relates to the number of flaws in the pressure vessel and, below approximately 100 flaws/m^3 the relationship is linear. Above that value the relationship is nonlinear and was not treated by the staff in this analysis. It has been treated in an approximate manner in earlier work examining other reactor pressure vessels -- see NUREG/CR-4183. However, for the purposes of the present study, the limitation is not significant.

Figure 8 illustrates the effect on P(F|E) of changes in flaw orientation. These calculations were performed for 0.35 wt.% Cu and 0.7 wt.% Ni, with the same volume of material and the Marshall flaw distribution. Changing the flaw orientation from axial to circumferential reduces P(F|E) by a factor of 100.

Finally, Figure 9 examines the effect on P(F|E) of the long-flaw assumption for the conditions of a circumferentially oriented flaw with 0.35 wt.% Cu and 0.7 wt.% Ni. The two conditions examined are for an infinitely long flaw and a finite-length flaw that is 6 times as long as deep. Changing the flaw length from infinite to 6 times its depth reduces P(F|E) by a factor of 2.5.

Plates

Unlike the weld materials, the embrittlement trends of the plates are not adequately defined by Regulatory Guide 1.99, Rev. 2. For the upper plate, the specific trend of embrittlement as a function of fluence (Figure 10(a)) incorporates the Yankee Rowe surveillance data and surveillance data from the Belgian BR-3 reactor, with a correction to match the irradiation temperature of the BR-3 surveillance data with the Yankee Rowe operating temperature. For the lower plate, a nickel adjustment is made to the upper plate trend, as shown in Figure 10(b).

The impact of adjusted reference temperature (ART) on P(F|E) is illustrated in Figure 11. At adjusted reference temperature values of 330°F and higher, the P(F|E) reaches a nearly constant level, indicating that the failure probability will not increase at higher ART levels. At an ART of 230°F, the P(F|E) is less than 1E-6 (i.e., no failures in 1,000,000 simulations).

For the plates, the sensitivity study is performed with ART as the parameter to define the material fracture toughness. While the embrittlement trends illustrated in Figures 10(a) and (b) are used to evaluate ART of the Yankee Rowe plates as a function of the fluence to the plates, results of this sensitivity study are appropriate (as a function of ART in Figure 11 for example) for any embrittlement estimation scheme. As an example, the Reg. Guide 1.99 Rev. 2 estimates of the shifts for the two plates (assuming that Yankee Rowe operated at 550°F and the plates were fine-grain) are 123°F for the upper plate and 214°F for the lower plate; hence, P(F|E) would be below 1E-6 if the Reg. Guide equations were appropriate for the Yankee Rowe plates.

The impact of initial RT_{NDT} on P(F|E) is illustrated for a high shift in ART (285°F) and a low shift in ART (355°F) in Figure 12. Hence, the ART was allowed to change. As illustrated in Figure 12, the two ART levels exhibit different behaviors. The higher of the two exhibits no substantial effect of varying the initial RT_{NDT}, whereas the lower of the two exhibits a distinct difference, of about two-orders of magnitude change in P(F|E) between the lowest and the highest initial RT_{NDT} values. This difference in behavior can be rationalized in Figure 11. Specifically, the higher of the ART shifts is on the plateau portion of the curve in Figure 11, where the P(F|E) is relatively insensitive to the ART level. In contrast, the higher of the two ART shifts is on the sloped portion of the curve, which is highly sensitive to the ART.

Residual stresses in the plate arise from the forming process and welding. The stress relief heat treatment is believed to be successfull in relieving the residual stresses due to forming, leaving only the residual stresses due to welding. These would apply only to a small portion of each plate near the welds. The residual stresses were modeled by supplementing the pressure term in the analyses, as described in the section on welds. As illustrated in Figure 13, each of the ART levels indicates a sensitivity to residual stress, with the lowest ART more sensitive than the highest ART.

The effect of modeling uncertainty in the initial RT_{NDT} and the embrittlement shift were also evaluated (Figure 14). In one case (indicated as "1-sig" in the figure), one standard deviation levels were set as 17°F on shift and 10°F for initial RT_{NDT} (the value for plates in Reg. Guide 1.99 Rev. 2 for shift and a typical value for initial RT_{NDT}); these values were doubled for a second case (indicated as "2-sig" in the figure). As illustrated in Fig. 14, the higher ART level is relatively insensitive, whereas the lower ART level is moderately sensitive (about a factor of two as the uncertainty level is increased).

The effect of flaw distribution was evaluated with the VISA-II code, in a manner similar to that used for the welds. As illustrated in Figure 15, the Dufrense and Lucia distribution results in P(F|E) values which are about two orders of magnitude (about a factor of 100) less than those for the Marshall distribution. This is consistent with the results for welds, as expected.

The effect of flaw aspect ratio was also evaluated. The flaw ratios used were the same as those for the welds, specifically a 6:1 ratio and a long flaw assumption. As illustrated in Figure 16, the 6:1 flaw case results in a P(F|E) which is within an order of magnitude below that for the long flaw case.

The interaction of the variables in these analyses is very nonlinear and it is inappropriate to attempt to estimate combined effects from these individual figures. To illustrate these interactions, a "summary analysis" was performed and is discussed below.

SUMMARY ANALYSIS

The probability of through wall cracking, defined herein as pressure vessel failure, given a PTS event (P(F|E)) is computed by evaluating P(F|E) for each region of the pressure vessel and then summing the individual values. The staff attempted to use "best estimate" values for the input parameters where a best estimate could be defined. However, it simply is not possible to determine a "best estimate" value for the copper content. Similarly, a "best estimate" flaw distribution has not been determined. Further, the assumption in the computer programs that all flaws are to be treated as surface flaws, oriented in the worst possible direction, introduces an unquantified degree of conservatism, but one that is thought to be significant.

Work on-going at ORNL will resolve the problems in how flaws are treated. Further, the licensee's plans to experimentally determine the weld chemistry will provide a basis for resolving the copper content issue. Also, the work being performed by PNL to determine the flaw distribution and density for a typical Linde 80 weld, coupled with the licensee's planned inspection of the beltline materials, will contribute to the determination of a realistic flaw size distribution and flaw density for the YNPS reactor pressure vessel.

While this work will be useful in the future, the staff was unable to determine "best estimate" values for the copper content and for the flaw size distribution. Further, the limitations in the computer codes could not be removed for the purposes of this study. Therefore, the staff selected "best estimate" values when possible, but performed the summary calculations for two copper levels and using the Marshall flaw size distirubtion.

The staff's analysis used the VISA-II code because it uses current embrittlement estimates for the weld materials. The embrittlement estimates for the plate materials were made using the staff's method, and the chemistry values input to the computer program were adjusted to provide the specified mean value of RT_{NDT}.

Five regions were considered: the upper axial weld, the lower axial weld, the circumferential weld, the upper plate, and the lower plate. The input data are described in Tables 5 through 9.

The variation in neutron fluence was included by subdividing the regions of interest based on the licensee's table of fluence values. These values have been reviewed by the staff and represent a "best estimate" of the fluence. The P(F|E) for each subregion was computed and summed to provide the P(F|E) for the entire region.

The fluence distribution provided by the licensee is given in Table 10. The fluence calculations provide a relatively fine distribution of fluence. To simplify the calculations, the staff used a more coarse distribution based on the peak fluence in the plate regions. However, the licensee's distribution was used in evaluating the welds. The staff's fluence "map" is provided in Table 11.

It should be noted that the summary analysis for plates includes circumferentially oriented flaws rather than axially oriented flaws. The basis for using circumferentially oriented flaws is two-fold. First, manufacturing defects in the plate would tend to be oriented along the rolling direction of the plate, which is the circumferential direction of the pressure vessel. Second, the inspection results for the Midland Unit 1 materials show 5 base metal flaws, all oriented in the circumferential direction. Thus, for the purposes of this analysis, the staff chose to consider only circumferentially oriented flaws in the plate material.

Because of the potential for "orientation effects" on the fracture toughness of the plate, the fracture toughness of the plates was adjusted by adding an additional 20°F to the initial RT_{NDT} of these materials. This adjustment was based on orientation studies for other plate materials, and on the engineering judgement of the staff. Fracture toughness values for the weld materials were not adjusted because the generic values correctly reflect the toughness behavior of the Linde 80 welds.

As discussed above, it is not possible to provide a reasonable "best estimate" for the copper content for the weld materials. The staff has used a value determined by averaging copper measurements from the overall population of Linde 80 welds, and this value is 0.26 wt.%. However, the licensee has offered a rationale for why 0.18 wt.% is more reasonable. This cannot be resolved until the licennsee actually measures the chemistry of the pressure vessel welds. Therefore, the results are presented for each copper concentration.

For the analyses performed, the upper plate did not contribute to the probability of failure. The lower plate has a conditional probability of failure of 2.6×10^{-3} per reactor year. For the 0.26 wt.% copper concentration, the conditional failure probability for the upper axial weld is 1.8×10^{-2}, 1.5×10^{-3} for the lower axial weld, and 5.5×10^{-3} for the circumferential weld. This gives a combined conditional failure probability of 2.8×10^{-2} per reactor year. However, for the 0.18 wt.% copper concentration, the conditional failure probability per reactor year for the upper axial weld is 1.9×10^{-3}, and 5×10^{-5} for the circumferential weld. The lower axial weld did not contribute to the conditional failure probability at this low copper content. The combined conditional failure probability for the 0.18 wt.% copper content is 4.6×10^{-3} per reactor year.

This study clearly demonstrates the need for reliable estimates of chemical content and embrittlement trends for the YNPS materials, and for reliable inspection of the pressure vessel beltline materials. Both of these issues will be addressed by the licensee during the next scheduled outage.

TABLE 1 CONDITIONS FOR UPPER AXIAL WELD BASE CASE

Volume	1088.64 in^3
Flaw Density	0.00074 flaws/in^3 (45 flaws/m^3)
Flaw Size Dist.	Marshall
Residual Stress	6 ksi
Copper Content	0.35 wt. %
Nickel Content	0.70 wt. %
Fluence	1.06 x 10^{19} n/cm^2 (E > 1 MeV)
Initial RT_{NDT}	10 °F
Irradiation Temp.	500 °F (treated by adding 50 °F to initial RT_{NDT})
Transient	SBLOCA 7 per YAEC 1735

TABLE 2 RANGES AND VARIABLES CONSIDERED IN WELD SENSITIVITY STUDY

Flaw Density	1 to 45 flaws/m^3 -- linear effect on P_f
Flaw Size Dist.	2 distributions: (Marshall, Dufrense and Lucia)
Residual Stress	3 -- 0 ksi, 6 ksi, and 10 ksi
Copper Content (wt. %)	5 -- 0.18, 0.20, 0.25, 0.29, 0.35
Nickel Content (wt. %)	4 -- 0.60, 0.70, 0.80, 1.00
Fluence (x 10^{19})	5 -- 1.0, 1.24, 1.5, 2.0, region avg.
Initial RT_{NDT} (°F)	3 -- 0, 10, 20
Irradiation Temp. (°F)	2 -- 500 and 550
Transients	12 -- examine asymptotic temperature and cooldown rate, and pressure
Regions	3 -- Upper axial weld, lower axial weld, circumferential weld

NOTE: This was not a "full factorial" study. Combinations of variables were chosen to illustrate relative sensitivity of the calculated probability of failure to the inputs.

TABLE 3 CONDITIONS FOR UPPER PLATE BASE CASE

Volume	214272 in^3 (124 ft^3)
Flaw Density	0.000074 flaws/in^3 (4.5 flaws/m^3)
Flaw Size Dist.	Marshall
Residual Stress	0 ksi
Irradiated RT_{NDT}	285 °F (Ni = 0.18 wt. % and Cu = 0.18 wt. %)
Fluence	2.58 x 10^{19} n/cm^2 (E > 1 MeV)
Initial RT_{NDT}	+30 °F
Irradiation Temp.	500 °F (treated in embrittlement estimate for irradiated RT_{NDT})

TABLE 4 RANGES AND VARIABLES CONSIDERED IN PLATE SENSITIVITY STUDY

Flaw Density	0.1 to 45 flaws/m^3 -- linear effect on P_f
Flaw Size Dist.	2 distributions: (Marshall, Dufrense and Lucia)
Residual Stress	4 -- 0 ,3, 6, and 10 ksi
Irradiated RT_{NDT}	Upper Plate: 5 -- 235, 247, 260, 285, and 310°F, Fluence of 2.58 x 10^{19} Lower Plate: 6 -- 255, 280, 305, 330, 355, and 380°F, Fluence of 2.30 x 10^{19}
Initial RT_{NDT} (°F)	3 -- Upper Plate: -10, 10, 30 Lower Plate: 10, 30, 50
Irradiation Temp. (°F)	2 -- 500 and 550 (500 handled as above; 550 handled by using R.G. 1.99, Rev. 2, and specified chemistries)
Transients	2 -- builds on transient study for welds
Regions	2 -- Upper plate and lower plate

NOTE: This was not a "full factorial" study. Combinations of variables were chosen to illustrate relative sensitivity of the calculated probability of failure to the inputs.

TABLE 5

INPUT DATA FOR SUMMARY ANALYSIS OF THE UPPER AXIAL WELD

- o Computer code: VISA-II
- o Transient: 1.31-inch diameter SBLOCA with a realistic decay heat
- o Flaw Characterization:

  Distribution: Marshall distribution
  Flaw Density: 45 flaws/m^3
  Orientation: Axial, surface flaws

- o Materials Data:
  - Nickel = 0.7 wt %
  - Copper = 0.26 wt %, and 0.18 wt %
  - Fluence = See Table 11
  - Initial RT_{NDT} = 60°F (includes 50°F adjustment for irradiation temperature)
  - Irradiation embrittlement predicted from R.G. 1.99, Rev. 2
  - Toughness: ASME K_{Ic} x 1.43 for mean initiation and K_{Ia} x 1.25 for mean arrest
- o Monte Carlo Simulations:
  - 100,000 simulations
  - Standard Deviations

    Copper -- 15% of the Cu level
    Nickel -- 0.0 wt %
    Fluence -- 10%
    Initial RT_{NDT} -- 0°F
    Shift in RT_{NDT} -- 28°F

TABLE 6

INPUT DATA FOR SUMMARY ANALYSIS OF THE LOWER AXIAL WELD

- o Computer code: VISA-II
- o Transient: 1.31-inch diameter SBLOCA with a realistic decay heat
- o Flaw Characterization:

  Distribution: Marshall distribution
  Flaw Density: 45 flaws/m^3
  Orientation: Axial, surface flaws

- o Materials Data:

  - Nickel = 0.7 wt %
  - Copper = 0.26 wt %, and 0.18 wt %
  - Fluence = See Table 11
  - Initial RT_{NDT} = 60°F (includes 50°F adjustment for irradiation temperature)
  - Irradiation embrittlement predicted from R.G. 1.99, Rev. 2
  - Toughness: ASME K_{Ic} x 1.43 for mean initiation and K_{Ia} x 1.25 for mean arrest

- o Monte Carlo Simulations:

  - 100,000 simulations
  - Standard Deviations
    Copper -- 15% of the Cu level
    Nickel -- 0.0 wt %
    Fluence -- 10%
    Initial RT_{NDT} -- 0°F
    Shift in RT_{NDT} -- 28°F

TABLE 7

INPUT DATA FOR SUMMARY ANALYSIS OF THE CIRCUMFERENTIAL WELD

- o Computer code: VISA-II
- o Transient: 1.31-inch diameter SBLOCA with a realistic decay heat
- o Flaw Characterization:

  Distribution: Marshall distribution
  Flaw Density: 45 flaws/m^3
  Orientation: Circumferential, surface flaws

- o Materials Data:
  - Nickel = 0.7 wt %
  - Copper = 0.26 wt %, and 0.18 wt %
  - Fluence = See Table 11
  - Initial RT_{NDT} = 60°F (includes 50°F adjustment for irradiation temperature)
  - Irradiation embrittlement predicted from R.G. 1.99, Rev. 2
  - Toughness: ASME K_{Ic} x 1.43 for mean initiation and K_{Ia} x 1.25 for mean arrest
- o Monte Carlo Simulations:
  - 100,000 simulations
  - Standard Deviations

    Copper -- 15% of the Cu level
    Nickel -- 0.0 wt %
    Fluence -- 10%
    Initial RT_{NDT} -- 0°F
    Shift in RT_{NDT} -- 28°F

TABLE 8

INPUT DATA FOR SUMMARY ANALYSIS OF THE UPPER PLATE

- o Computer code: VISA-II
- o Transient: 1.31-inch diameter SBLOCA with a realistic decay heat
- o Flaw Characterization:

  Distribution: Marshall distribution
  Flaw Density: 4.5 flaws/m^3
  Orientation: Circumferential, surface flaws

- o Materials Data:

  - Nickel = 0.18 wt %
  - Copper = 0.18 wt %
  - Fluence = See Table 11
  - Initial RT_{NDT} = 30°F
  - Irradiation embrittlement predicted from Figure 10
  - Toughness: ASME K_{Ic} x 1.43 for mean initiation and K_{Ia} x 1.25 for mean arrest

- o Monte Carlo Simulations:

  - 100,000 simulations
  - Standard Deviations
    Copper -- 0.0 wt %
    Nickel -- 0.0 wt %
    Fluence -- 10%
    Initial RT_{NDT} -- 0°F
    Shift in RT_{NDT} -- 17°F

TABLE 9

INPUT DATA FOR SUMMARY ANALYSIS OF THE LOWER PLATE

- Computer code: VISA-II
- Transient: 1.31-inch diameter SBLOCA with a realistic decay heat
- Flaw Characterization:

  Distribution: Marshall distribution
  Flaw Density: 4.5 flaws/m^3
  Orientation: Circumferential, surface flaws

- Materials Data:

  - Nickel = 0.63 wt %
  - Copper = 0.20 wt %
  - Fluence = See Table 11
  - Initial RT_{NDT} = 30°F
  - Irradiation embrittlement predicted from Figure 10
  - Toughness: ASME K_{Ic} x 1.43 for mean initiation and K_{Ia} x 1.25 for mean arrest

- Monte Carlo Simulations:

  - 100,000 simulations
  - Standard Deviations
    Copper -- 0.0 wt %
    Nickel -- 0.0 wt %
    Fluence -- 10%
    Initial RT_{NDT} -- 0°F
    Shift in RT_{NDT} -- 17°F

TABLE 10

FLUENCE DISTRIBUTION FOR YANKEE ROWE PROVIDED BY THE LICENSEE

Fluence Distribution for Beltline Materials

Peak Fluence at End of Cycle 21 2.58 e19 n/cm2

		AZIMUTHAL VARIATION									Axial Welds
		0 to 5	5 to 10	10 to 15	15 to 20	20 to 25	25 to 30	30 to 35	35 to 40	40 to 45	40 to 45
Upper Plate											
	10 to 20	0.362	0.378	0.387	0.365	0.323	0.272	0.221	0.179	0.159	0.159
	20 to 30	1.242	1.298	1.329	1.252	1.108	0.934	0.757	0.615	0.546	0.546
	30 to 40	1.959	2.047	2.095	1.973	1.747	1.473	1.194	0.970	0.861	0.861
	40 to 50	2.234	2.334	2.389	2.251	1.992	1.680	1.362	1.106	0.982	0.982
% of Height	50 to 60	2.347	2.453	2.510	2.365	2.094	1.765	1.431	1.162	1.032	1.032
	60 to 70	2.347	2.453	2.510	2.365	2.094	1.765	1.431	1.162	1.032	1.032
	70 to 80	2.412	2.521	2.580	2.430	2.152	1.814	1.471	1.195	1.060	1.060
	80 to 90	2.369	2.475	2.534	2.387	2.113	1.781	1.444	1.173	1.041	1.041
	90 to 100	2.347	2.453	2.510	2.365	2.094	1.765	1.431	1.162	1.032	1.032
Circ Weld		2.147	2.243	2.296	2.163	1.915	1.614	1.309	1.063	0.944	
Lower Plate											35 to 40
	0 to 10	2.147	2.243	2.296	2.163	1.915	1.614	1.309	1.063	0.944	1.063
% of Height	10 to 20	1.689	1.764	1.806	1.701	1.506	1.270	1.029	0.836	0.742	0.836
	20 to 30	0.975	1.018	1.042	0.982	0.869	0.733	0.594	0.483	0.428	0.483
	30 to 40	0.169	0.176	0.181	0.170	0.151	0.127	0.103	0.084	0.074	0.084

TABLE 11

FLUENCE MAP USED BY THE STAFF FOR YANKEE ROWE

		AZIMUTHAL VARIATION									Axial Welds
		0 to 5	5 to 10	10 to 15	15 to 20	20 to 25	25 to 30	30 to 35	35 to 40	40 to 45	40 to 45
Upper Plate											
	10 to 20	U9	U9	U9	U9	U9	U9	U9	U9	U9	0.159
	20 to 30	U5	U5	U5	U5	U6	U7	U9	U9	U9	0.546
	30 to 40	U3	U2	U2	U3	U3	U4	U6	U7	U8	0.861
	40 to 50	U2	U1	U1	U2	U3	U4	U5	U6	U7	0.982
% of Height	50 to 60	U1	U1	U1	U1	U2	U3	U5	U6	U7	1.032
	60 to 70	U1	U1	U1	U1	U2	U3	U5	U6	U7	1.032
	70 to 80	U1	U1	U1	U1	U2	U3	U4	U6	U7	1.060
	80 to 90	U1	U1	U1	U1	U2	U3	U4	U6	U7	1.041
	90 to 100	U1	U1	U1	U1	U2	U3	U5	U6	U7	1.032
Circ Weld		2.147	2.243	2.296	2.163	1.915	1.614	1.309	1.063	0.944	
Lower Plate											35 to 40
	0 to 10	L1	L1	L1	L1	L2	L3	L4	L5	L5	1.065
% of Height	10 to 20	L3	L2	L2	L2	L3	L4	L5	L6	L7	0.836
	20 to 30	L5	L5	L5	L5	L6	L7	L8	L9	L9	0.483
	30 to 40	L9	L9	L9	L9	L9	L9	L9	L9	L9	0.084

Upper Plate

Region	Fluence	Shift
U1	2.429	250
U2	2.156	240
U3	1.903	230
U4	1.671	220
U5	1.459	210
U6	1.265	200
U7	1.089	190
U8	0.929	180
U9	0.244	114

Lower Plate

Region	Fluence	Shift
L1	2.156	320
L2	1.903	310
L3	1.671	300
L4	1.459	290
L5	1.089	270
L6	0.929	260
L7	0.786	250
L8	0.545	230
L9	0.087	160

TABLE 12

SUMMARY ANALYSIS RESULTS

	P(F|E) per Reactor Year	
	0.26 wt% Cu	0.18 wt % Cu
Upper Axial Weld	1.8×10^{-2}	1.9×10^{-3}
Lower Axial Weld	1.5×10^{-3}	$< 10^{-6}$
Circumferential Weld	5.5×10^{-3}	5×10^{-5}
Lower Plate	<----------2.6E-3---------->	
Upper Plate	<---------(< 1E-5)--------->	
Total	2.8×10^{-2}	4.6×10^{-3}

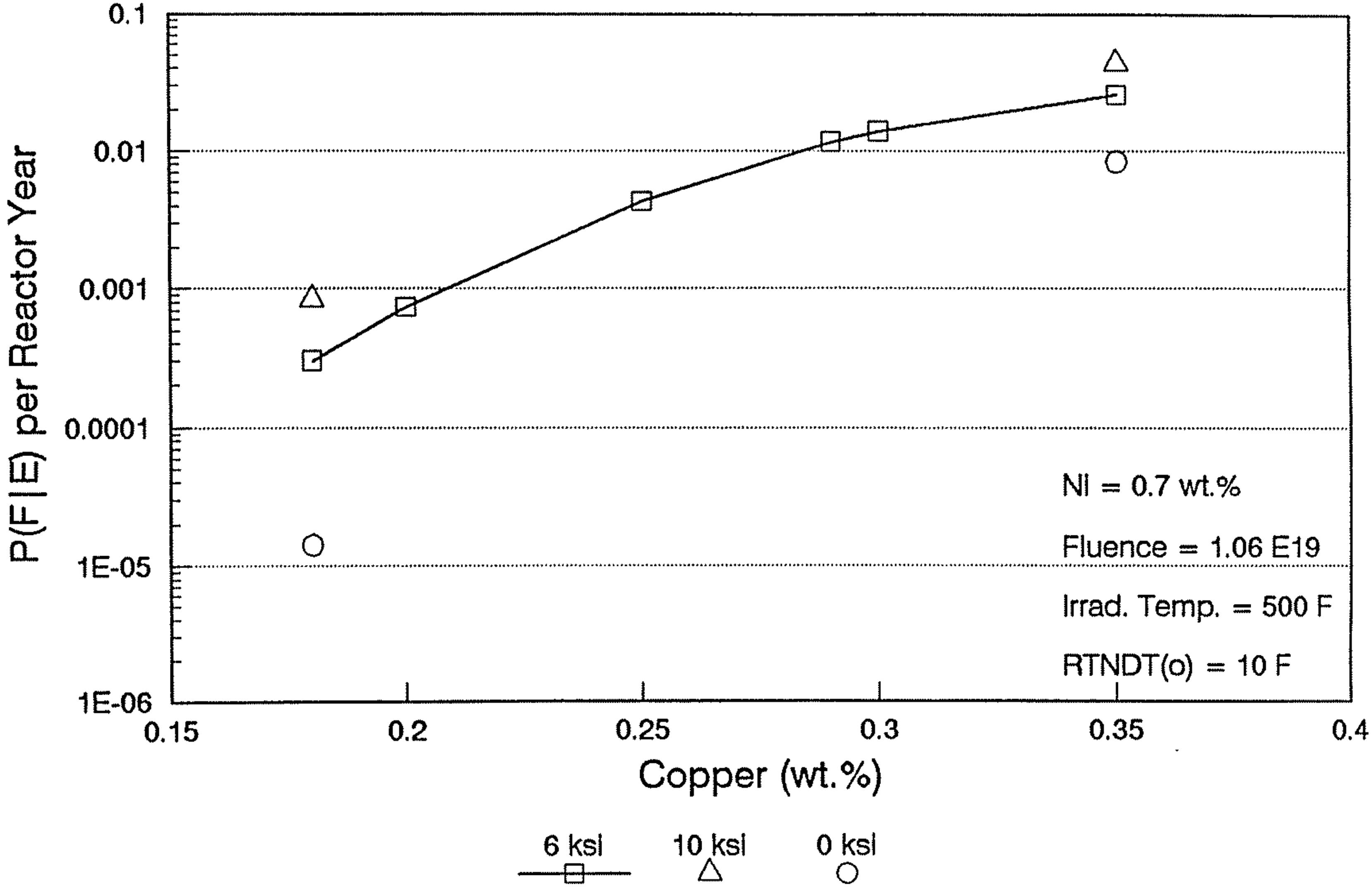


Fig. 1 P(F|E) per reactor year as a function of copper content for residual stress levels of 0 ksi, 6 ksi and 10 ksi.

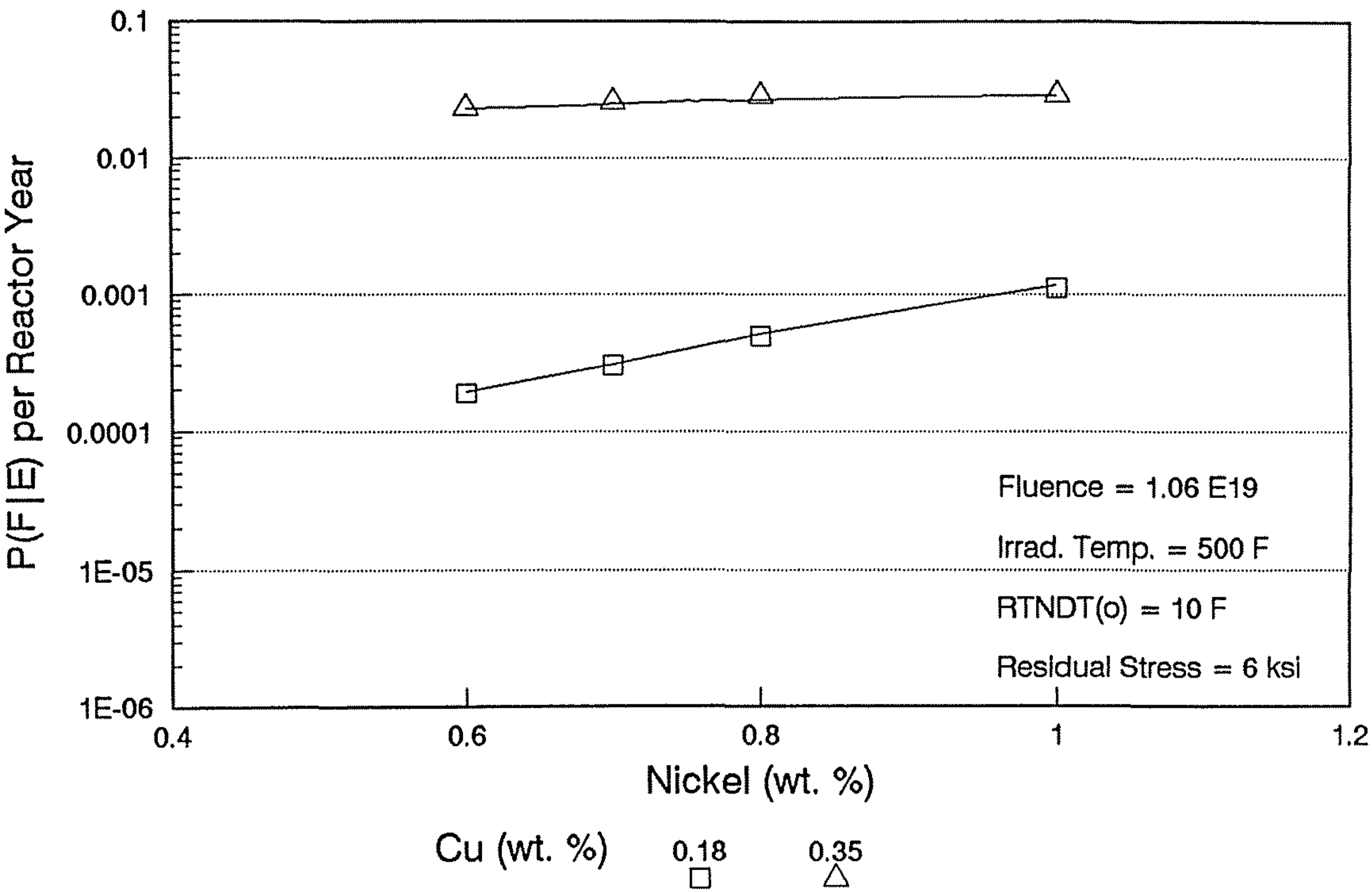


Fig. 2 P(F|E) per reactor year as a function of nickel content for copper levels of 0.18 and 0.35 wt.%.

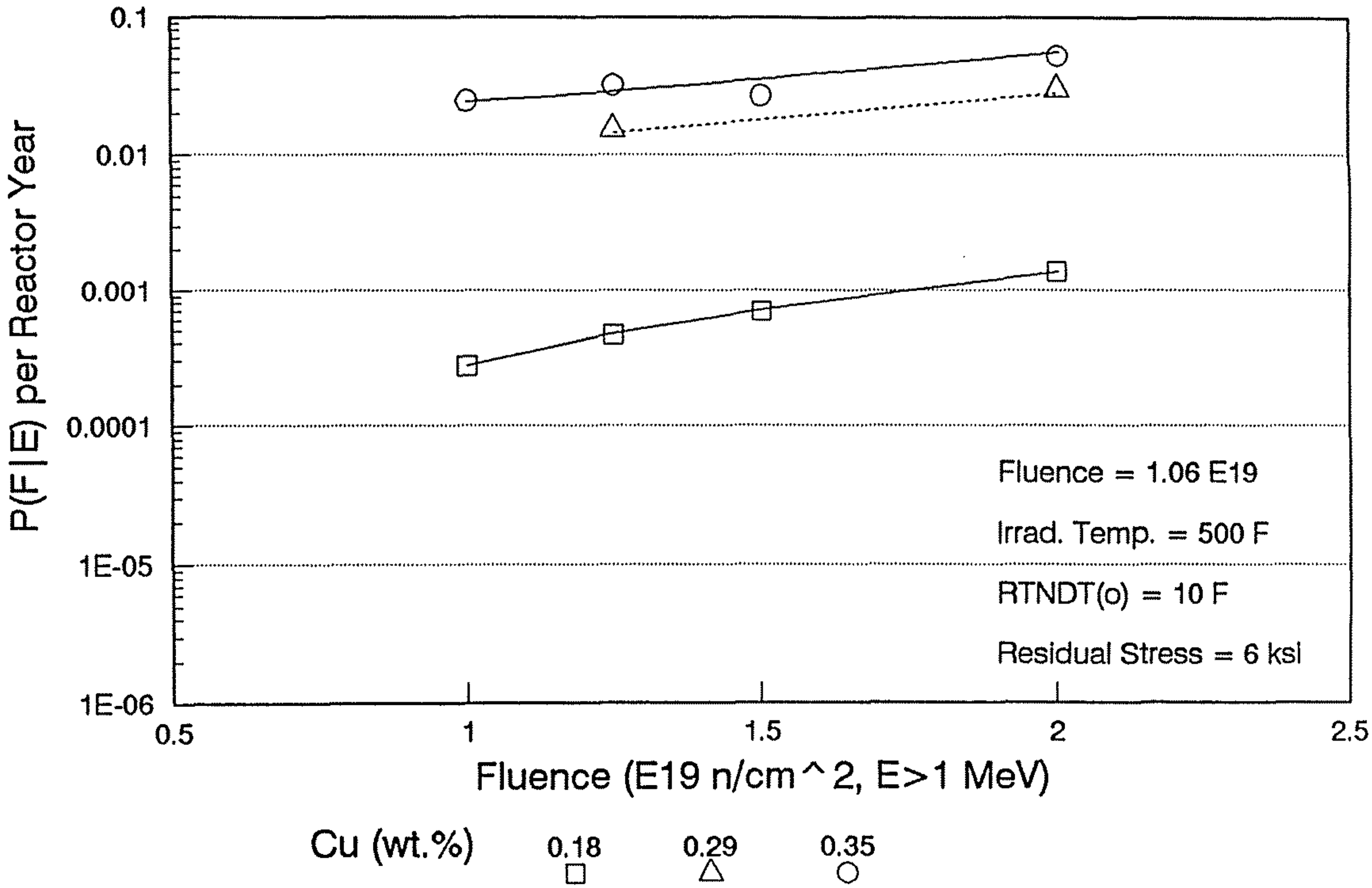


Fig. 3 P(F|E) per reactor year as a function of neutron fluence for copper levels of 0.18, 0.29, and 0.35 wt.%.

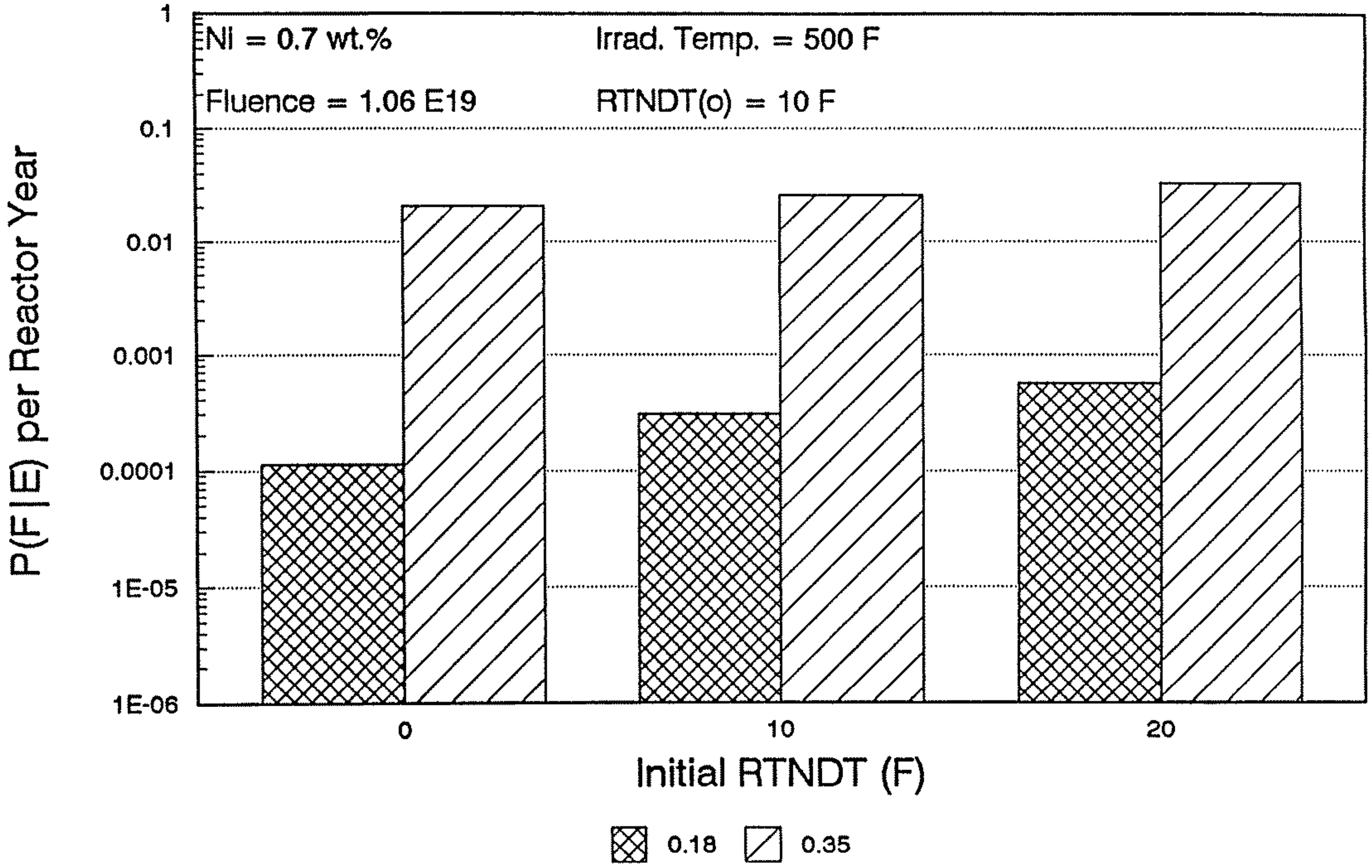


Fig. 4 P(F|E) per reactor year for initial RT_{NDT} values of 0, 10, and 20°F at copper levels of 0.18 and 0.25 wt.%.

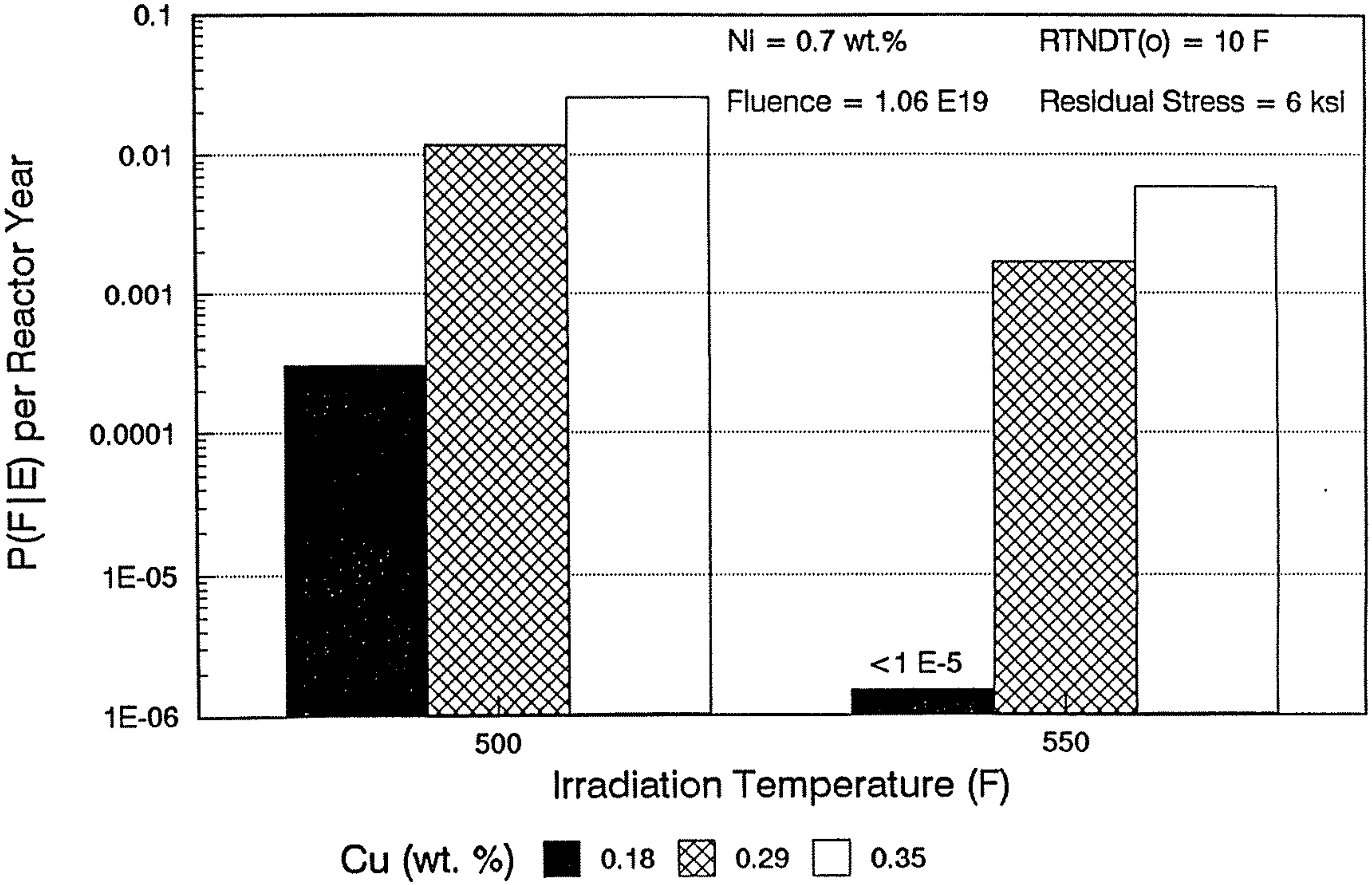


Fig. 5 P(F|E) per reactor year for irradiation temperatures of 500 and 550°F at copper levels of 0.18, 0.29, and 0.35 wt.%.

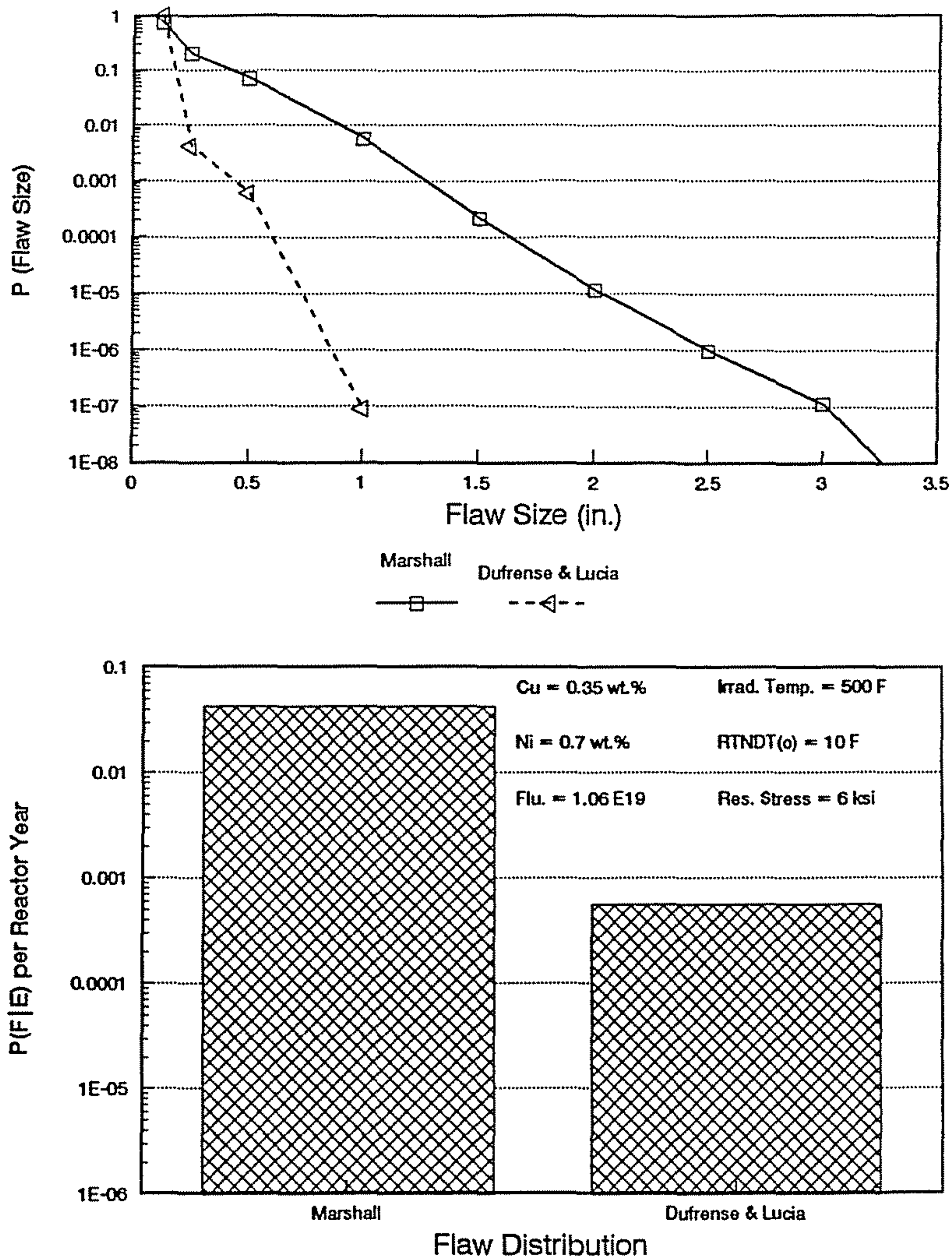


Fig. 6 P(F|E) per reactor year for the Marshall flaw size distribution and the Dufrense and Lucia flaw size distribution.

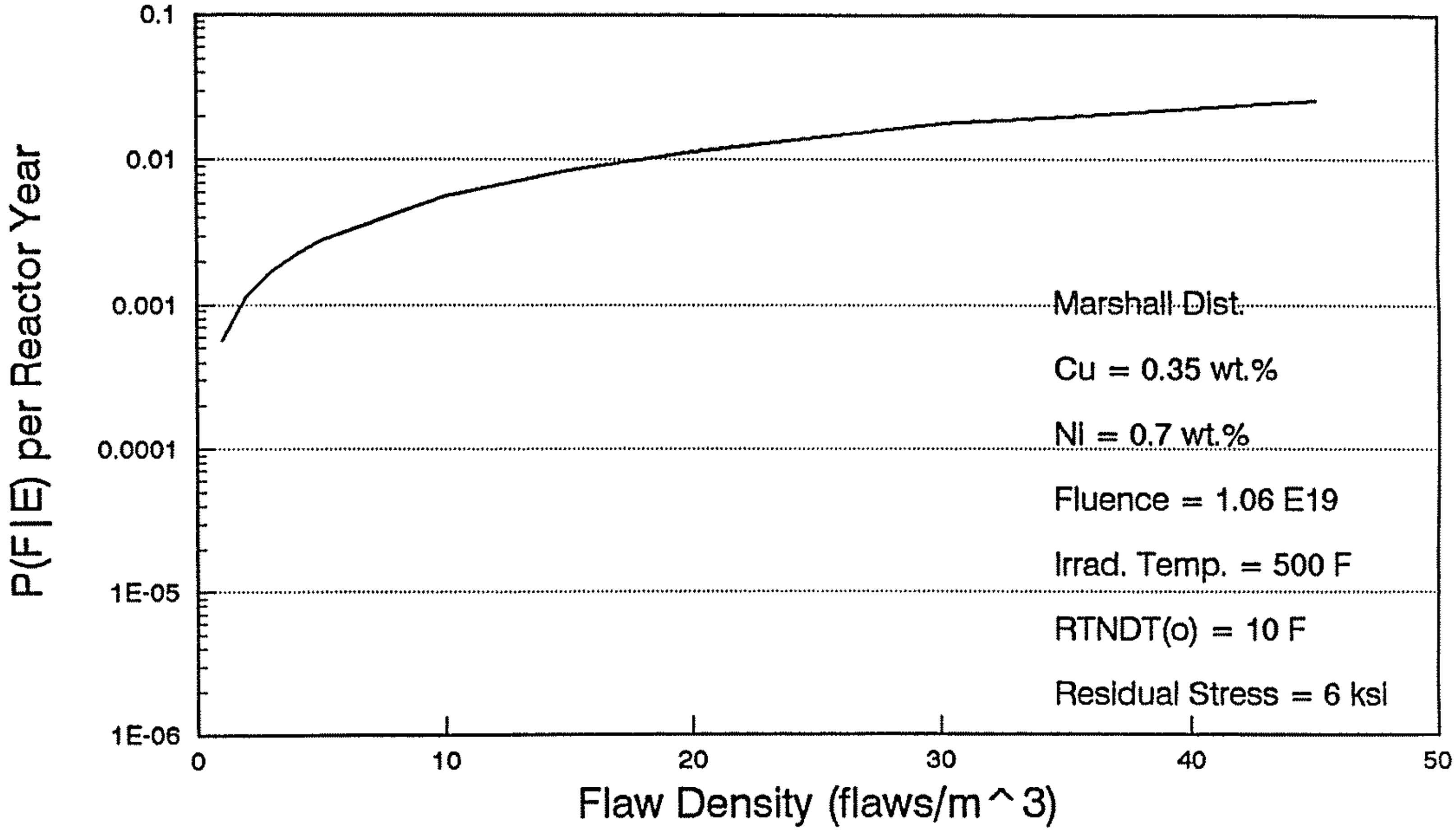


Fig. 7 P(F|E) per reactor year as a function of flaw density.

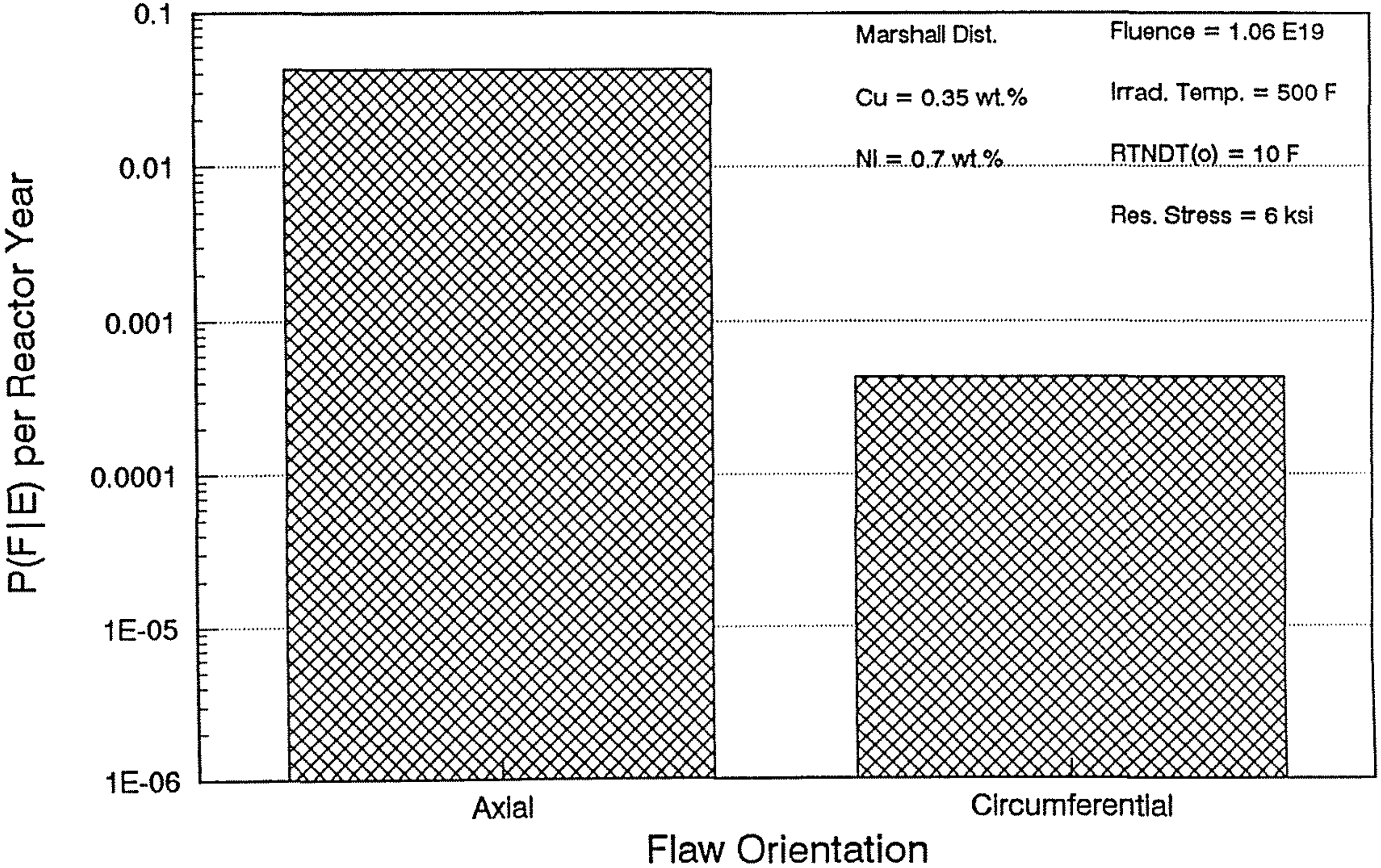


Fig. 8 P(F|E) per reactor year as a function of flaw orientation at RT_{NDT} = 286°F.

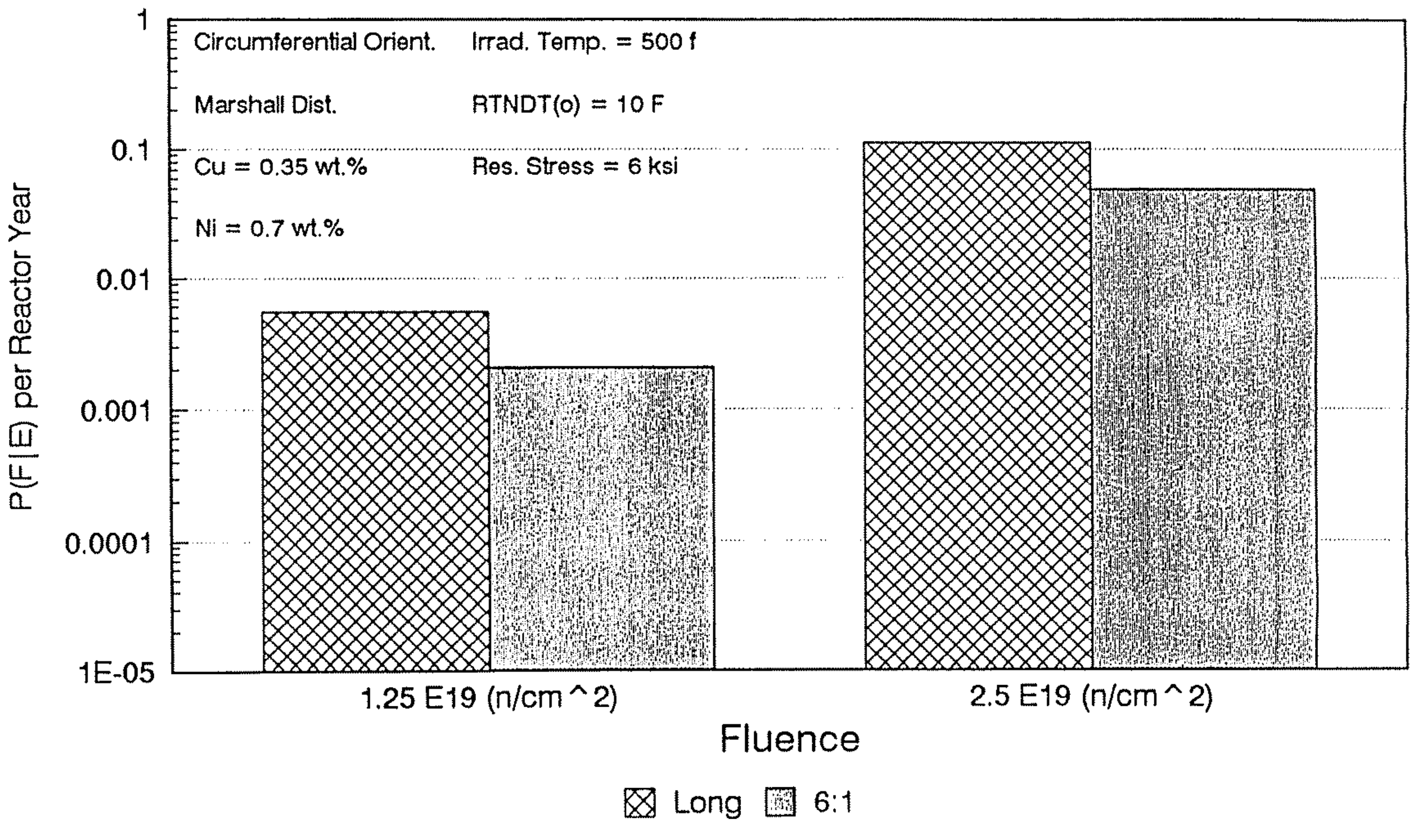
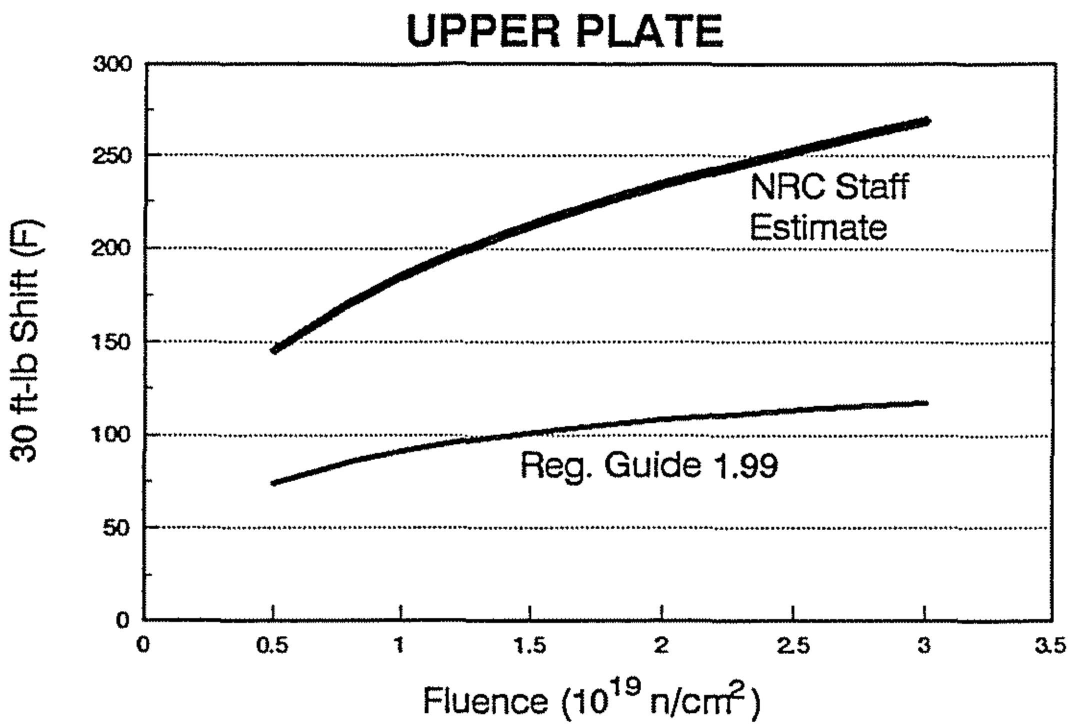


Fig. 9 P(F|E) per reactor year for long flaws and finite length (length is 6 times depth) flaws, at fluence levels of 1.25 and 2.5 x 10^{19} n/cm^2, E > 1 MeV.



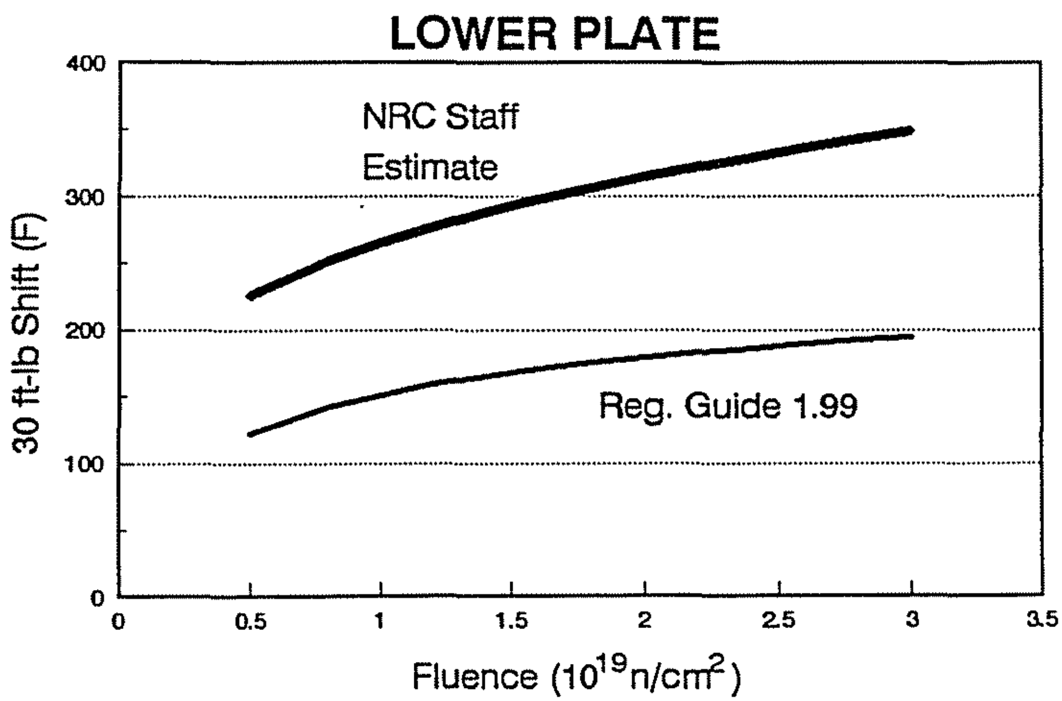


Fig. 10 Embrittlement trends for the upper plate and the lower plate, using Regulatory Guide 1.99 Rev. 2 and the NRC staff estimates.

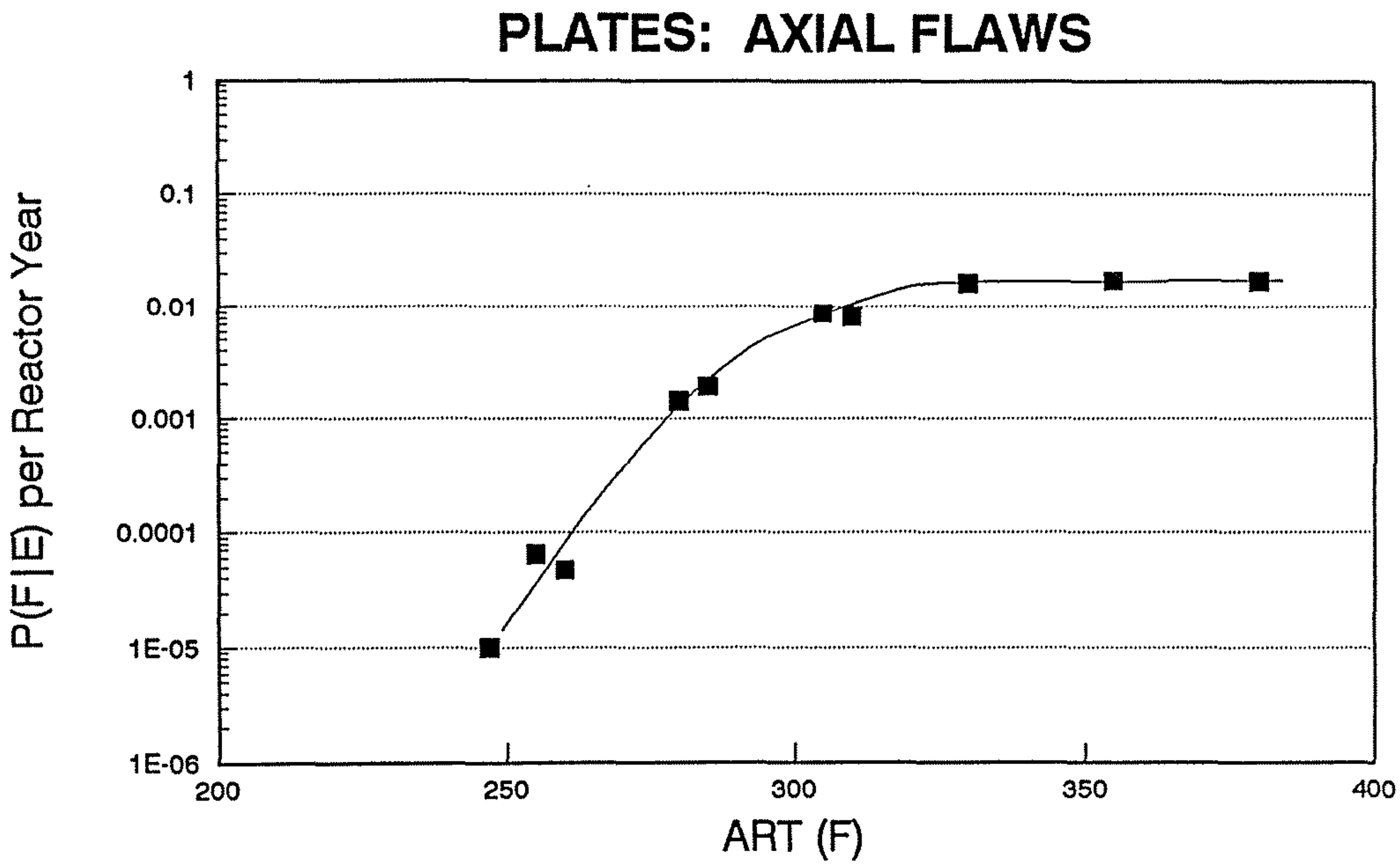


Fig. 11 P(F|E) per reactor year as a function of irradiated reference temperature.

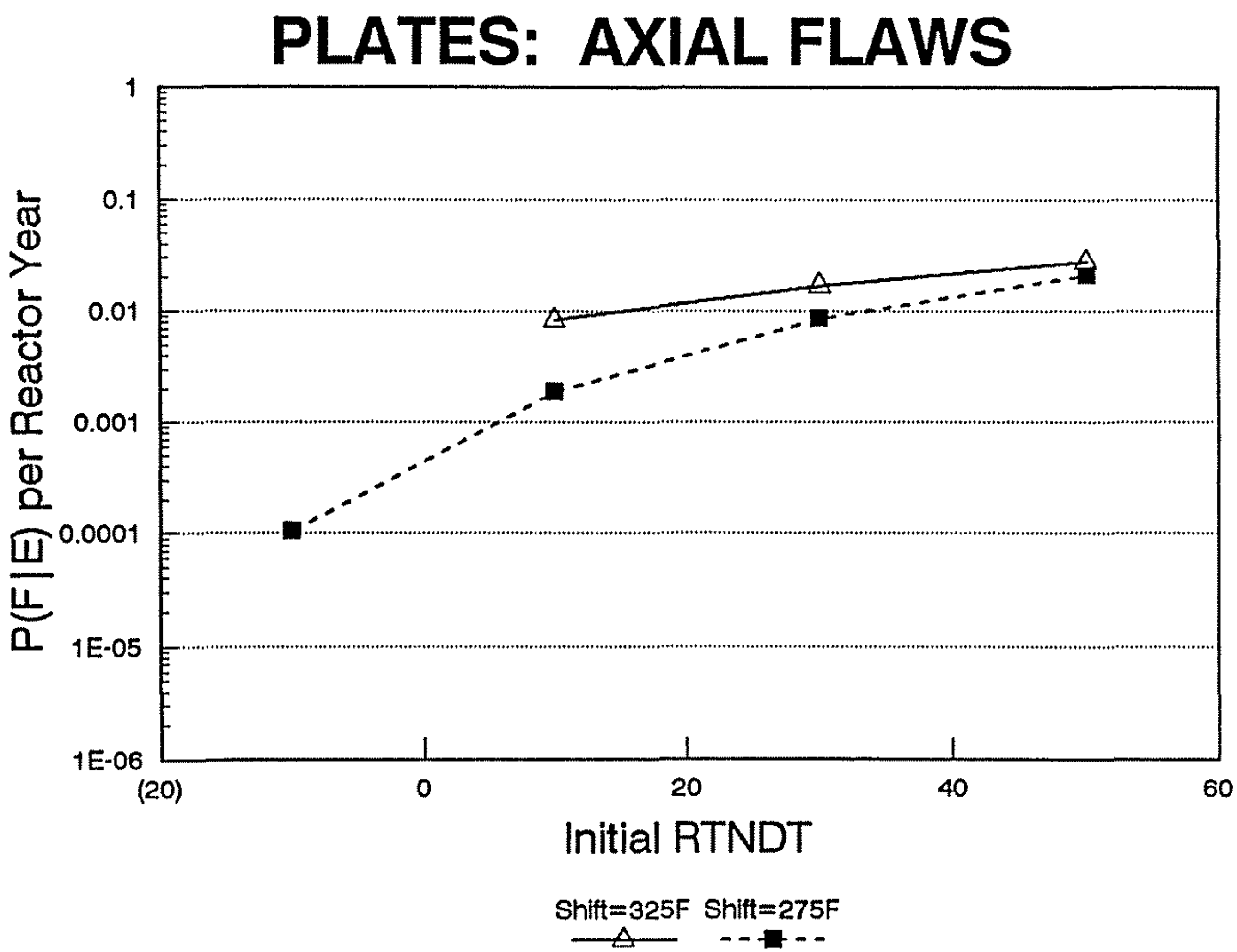


Fig. 12 P(F|E) per reactor year as a function of initial RT_{NDT}, for shifts of 275°F and 325°F.

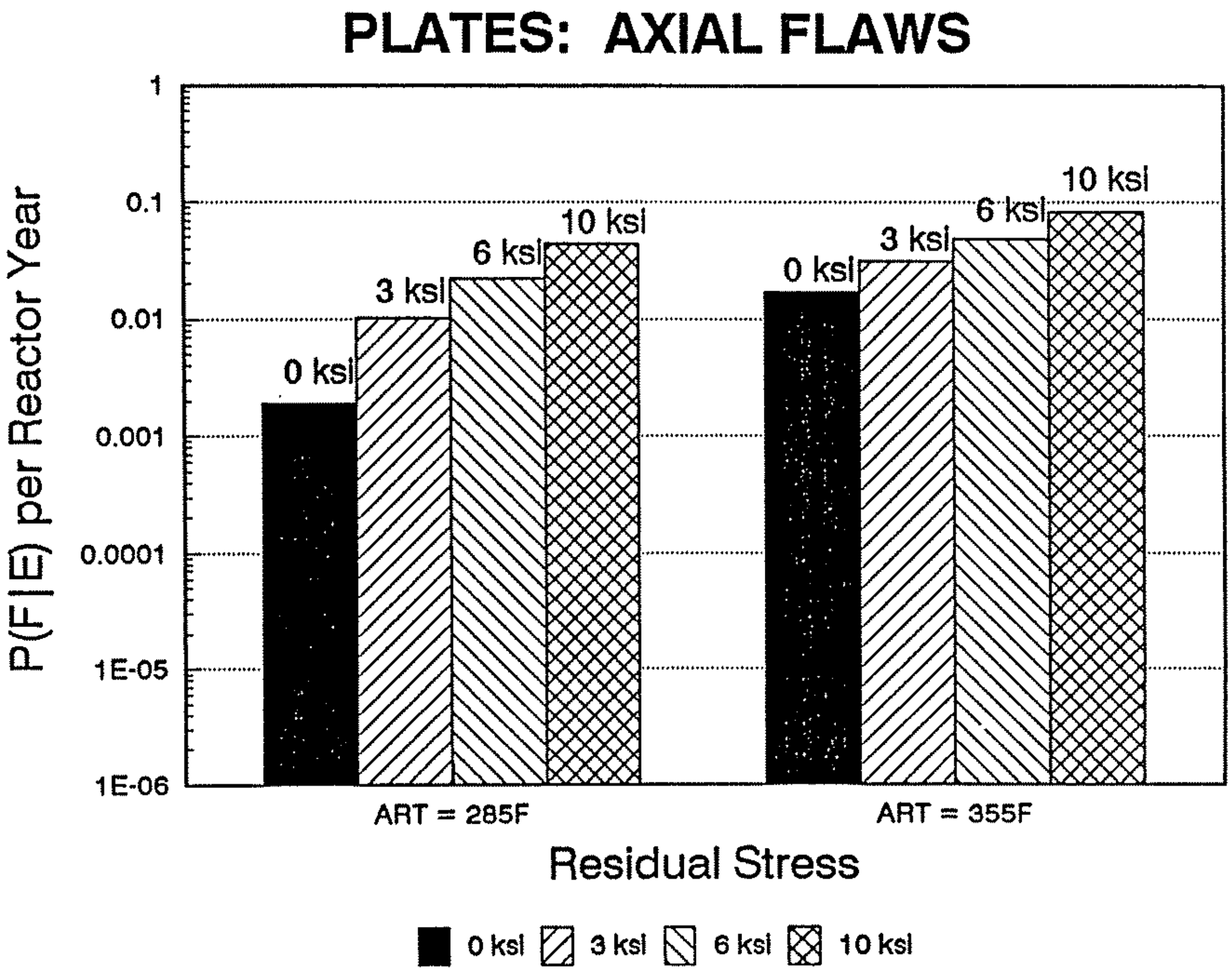


Fig. 13 P(F|E) per reactor year as a function of residual stress, for adjusted reference temperatures (ART) of 285°F and 355°F.

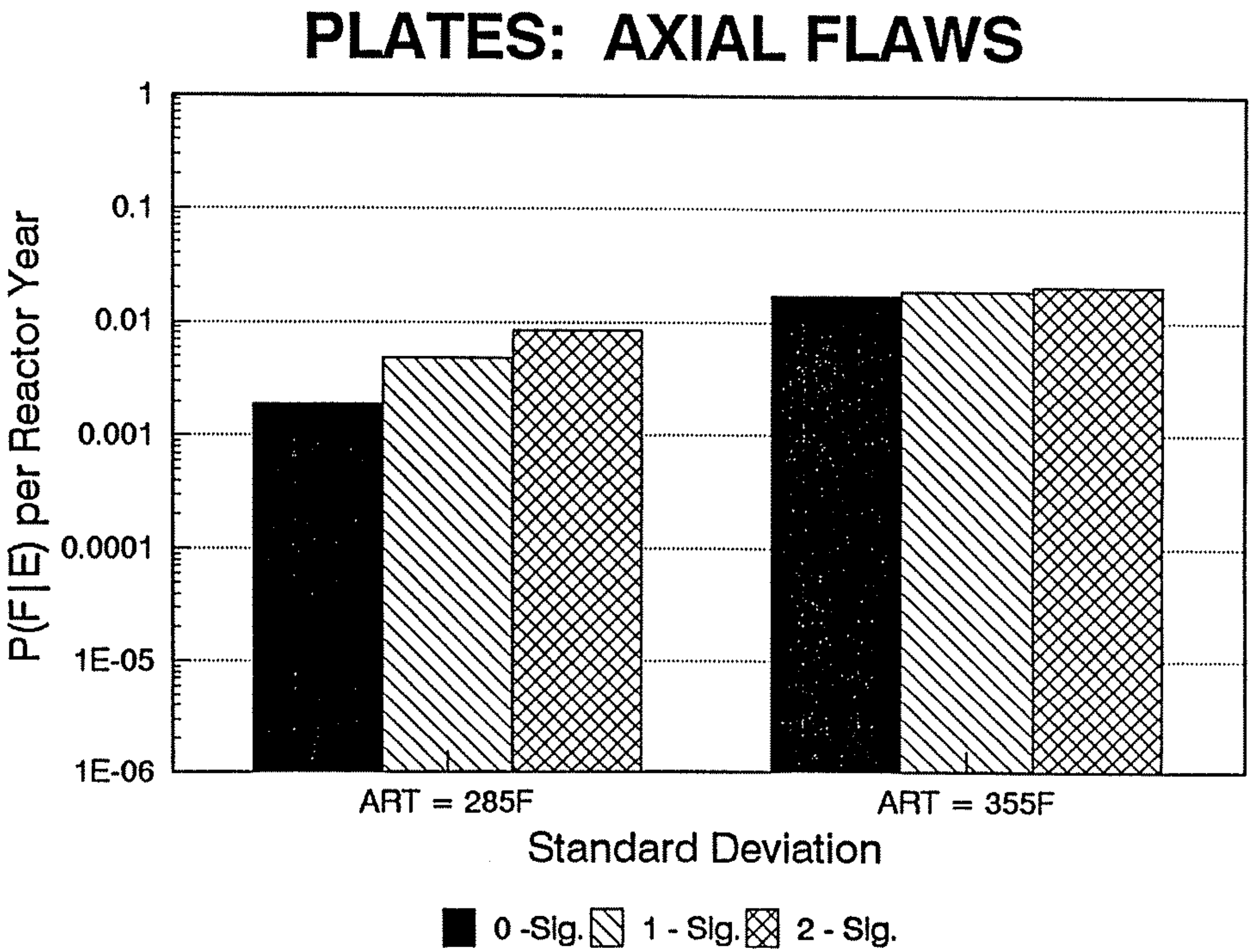


Fig. 14 P(F|E) per reactor year as a function of variability in the initial RT_{NDT} and the reference temperature shift, for ART values of 285°F and 355°F.

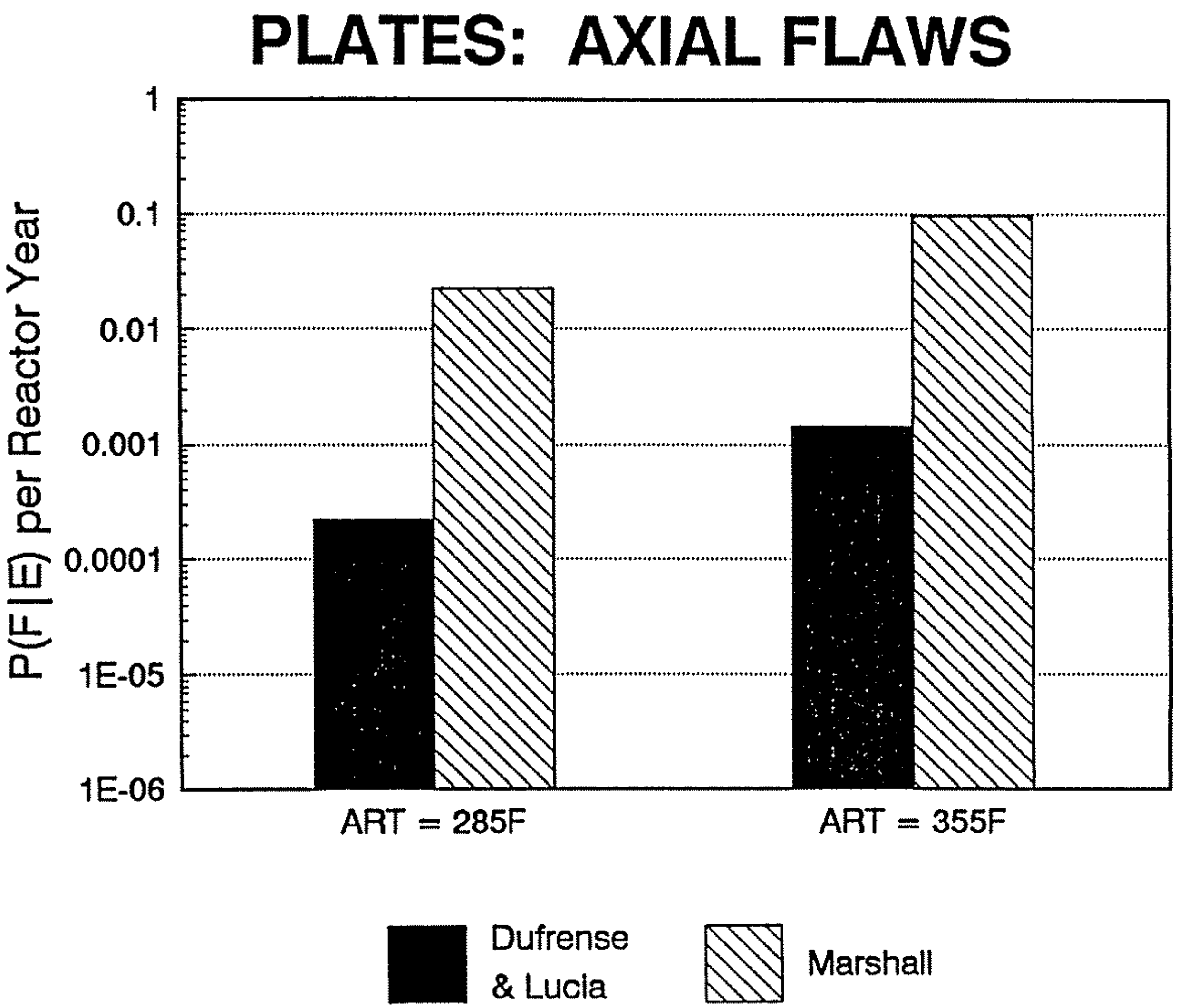


Fig. 15 P(F|E) per reactor year as a function of flaw distribution, for ART values of 285°F and 355°F.

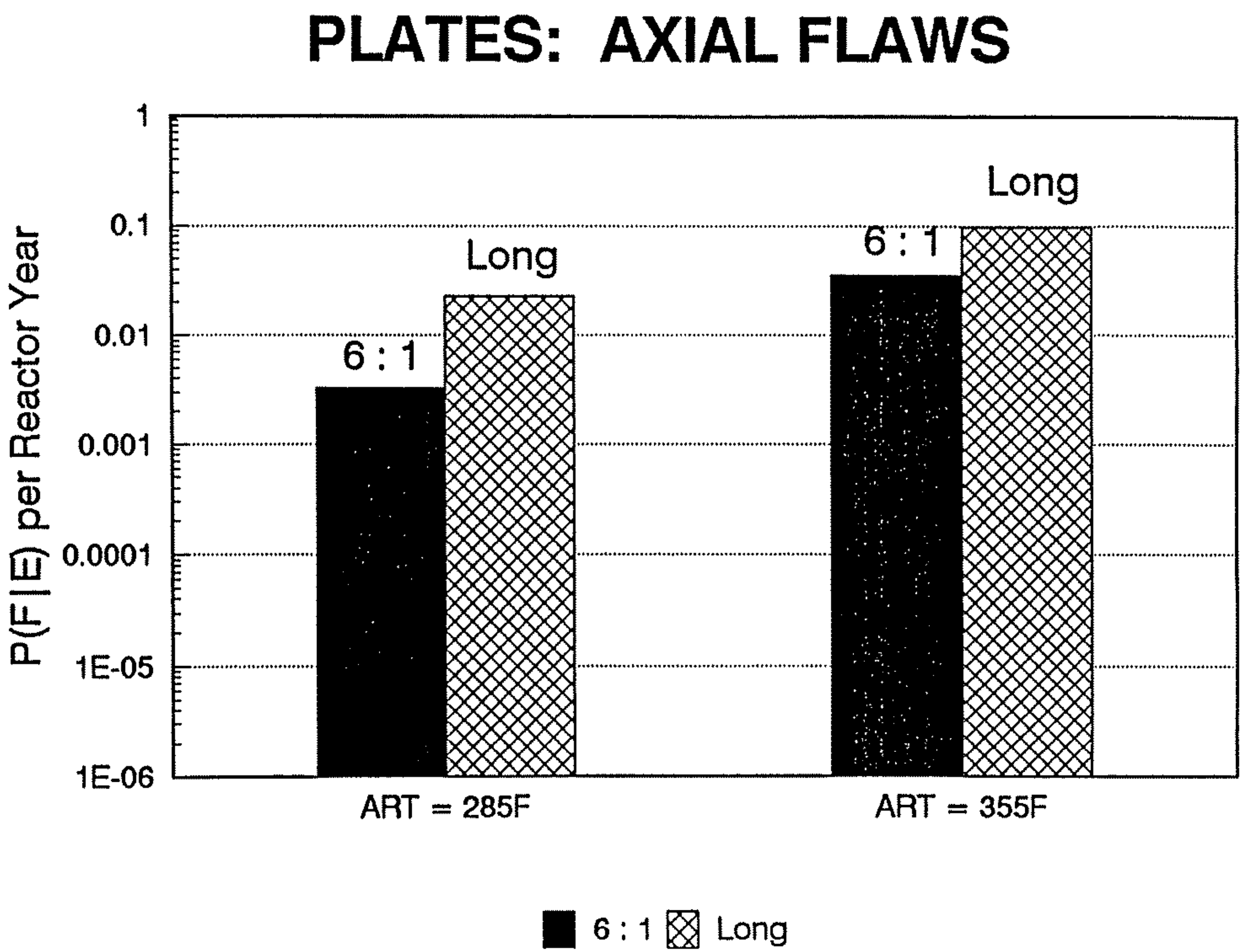


Fig. 16 P(F|E) per reactor year as a function of flaw aspect ratio, for ART values of 285°F and 355°F.

SRM RESPONSE 2

In SECY 82-465 the staff considered the frequency of vessel crack extension without arrest. A plant evaluated to a screening criterion of 270°F is likely to have a true RT_{NDT} of 150-270°F with a mean value of 210°F. For this mean value of RT_{NDT} the staff estimated that the frequency of exceeding this value was $6x10^{-6}$ per reactor year. Using the information in SECY 82-465 the following approximate values were constructed for various initiating events.

Initiating Event	Frequency of Initiating Event Per R/Y	Conditional Probability of Vessel Failure	Vessel Failure Frequency Per R/Y
Small Break LOCA	$2.2x10^{-3}$	$2.7x10^{-5}$	$6x10^{-8}$
Extended High Pressure Injection to 2250 psig (HPI)	$1.4x10^{-4}$	$4.3x10^{-2}$	$6x10^{-6}$
Steam Generator Tube Rupture (SGTR)	$7.4x10^{-3}$	$8.1x10^{-6}$	$6x10^{-8}$
Steam Line Break (SLB)	$3x10^{-4}$	$9.1x10^{-5}$	$3x10^{-8}$
			$6.1x10^{-6}$

In this analysis and the RT_{NDT} range of 150°-270°F the most severe event was HPI since the pressure and temperature (P,T) conditions reached led to the highest conditional vessel failure probability. This sequence was explicitly examined for Yankee Rowe and found to have a very low likelihood of vessel failure from PTS. This is attributable to the following design features at Yankee Rowe:

1. The ECCS shutoff head is 1550 psia.

2. The charging pumps are tripped on an SI signal and therefore will not repressurize the primary system after or during ECCS injection.

3. The charging system maximum flow is small (100 gpm) and therefore, a downcomer cooldown under feed and bleed conditions would be very slow.

4. The probability of the need for feed and bleed cooling (i.e., loss of all heat sink) is low at Yankee because of the variety of ways to feed the steam generators. For example, the charging system can be used to feed the secondary system as well as the primary system.

ATTACHMENT R3

SRM RESPONSE 3

POSSIBLE VARIATIONS FROM MEAN VALUES IN YANKEE SURVEILLANCE IRRADIATED CHARPY DATA

A. RTndt SHIFT.

ASTM E 185 - 82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels", requires that the radiation induced transition temperature shifts be measured by measuring the difference in the 30 ft-lb index temperatures before and after irradiation. The index temperatures are to be obtained from average curves drawn through the Charpy data generated for unirradiated and irradiated data sets. This is illustrated in the attached Figures 1 and 2.

The ASTM Codes relating to Charpy tests do not provide guidance on standard deviations which could exist in determining RTndt shifts nor are standard deviations required in the determination of RTndt shift. Uncertainties in Charpy test results are handled in the ASTM Codes by requiring Charpy test machines to undergo calibration checks with standard material samples and by specifying strict test procedures.

In order to assess a variance in the Yankee reported RTndt shift values, Yankee looked at the two industry recognized methods to fit Charpy curves to data. One technique is fitting the data by eye; the other is a hyperbolic tangent function fit (Tanh). In both cases, the Charpy data is fit to a mean curve.

To determine the variation possible in reported shifts, Yankee looked at its unirradiated Charpy data and determined standard deviations using three methods. The first method is by eye where most of the data is enveloped at high and low temperatures. The second was by using a linear regression technique to fit data in the region of the 30 ft-lb index temperature and thus calculate the standard deviation. The third was to calculate the standard deviation based on the Tanh computer routine and its calculated standard deviation.

Figure 3 shows the unirradiated data for Yankee surveillance plate. Using the "by eye" technique it is noted that the initial value of RTndt at the 30 ft-lb index is +10F. This value could be as low as -20F or as high as +45F at the 30 ft-lb index (see dotted lines of Figure 3). This means that the uncertainty in RTndt shift value could be 30F at the high end (if the initial value was 30F lower) or the uncertainty could be -35F at the low end.

The linear regression technique on baseline data produced a variation of plus or minus 22F from the mean data. This analysis was done by plotting temperature on the "Y" axis and energy on the "X" axis to generate a standard deviation for temperature. The result is a less conservative uncertainty than the above approach.

The last technique used on unirradiated data was the Tanh fit. Using error bands calculated by the "EPRIPLOT" Tanh fit computer routine for data, the variation in RTndt (at the 30 ft-lb index) was calculated to vary plus or minus 10F. The deviation shown in Figure 3 remains conservative.

To perform this type of uncertainty analysis for irradiated data is difficult because the amount of Charpy data available for a given irradiation fluence is small. The ASTM Code requires that the Charpy curve be fit through the existing irradiated data. This is shown in Figure 4. One can assume that if many irradiated specimens were tested, the scatter would be similar to that experienced in unirradiated tests. To also apply and add the additional uncertainty band for irradiated data to the uncertainty for unirradiated data would compound the error band to an unrealistic value. ASTM Codes do not require the use of standard deviations for the determination of 30 ft-lb index values. If one were to attempt to calculate such a value however, Yankee shows an uncertainty band for RTndt shift for its data of +30F, -35F. These are the vertical lines of Figure 5.

B. Fluence variation.

The possible variation in fluence measurements is plus or minus 10%. This is based on calculations performed by Westinghouse and SCK/CEN (BR3) physicists and documented in the following reports.

- Westinghouse: Letter, BYR 91-019, J. D. Haseltine to P. Sears, "Reactor Pressure Vessel Fluence Uncertainty", dated February 20, 1991.

- SCK/CEN: Work Document, "Revisiting the Irradiation Data for the Yankee-Rowe Surveillance Plate Specimens Exposed at the BR3.", From A. Fabry and J. Van de Velde, dated October 18, 1989.

C. Reg. Guide 1.99 Two Sigma Margin (34F).

Figure 5 shows the Yankee and BR3 surveillance data bounded by two dotted lines. These lines represent the margin required to be applied to data by Regulatory Guide 1.99. Figure 5 demonstrates that even with the conservative estimated standard deviations to the Yankee data, Reg. Guide margins bound the data set. In all Yankee PTS calculations, the Reg. Guide 34F margin was added to Yankee estimates of RTndt shift. Thus, the Yankee calculations used bounded (conservative) values of RTndt.

FIGURE 1

UNIRRADIATED CHARPY DATA
EXAMPLE OF CURVE FIT

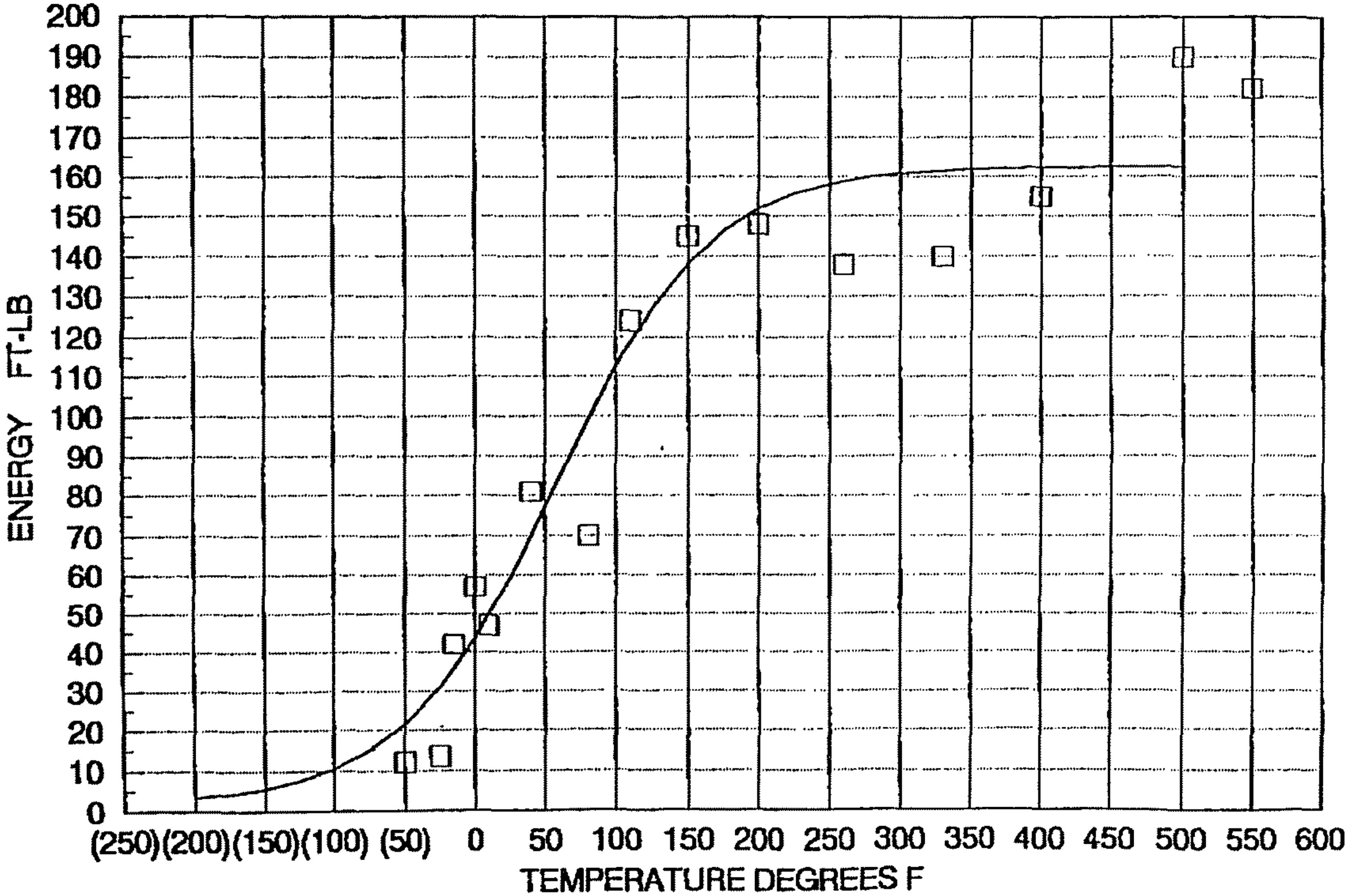


FIGURE 2

CHARPY IMPACT ENERGY TEST RESULTS
PRE- AND POST-IRRADIATION

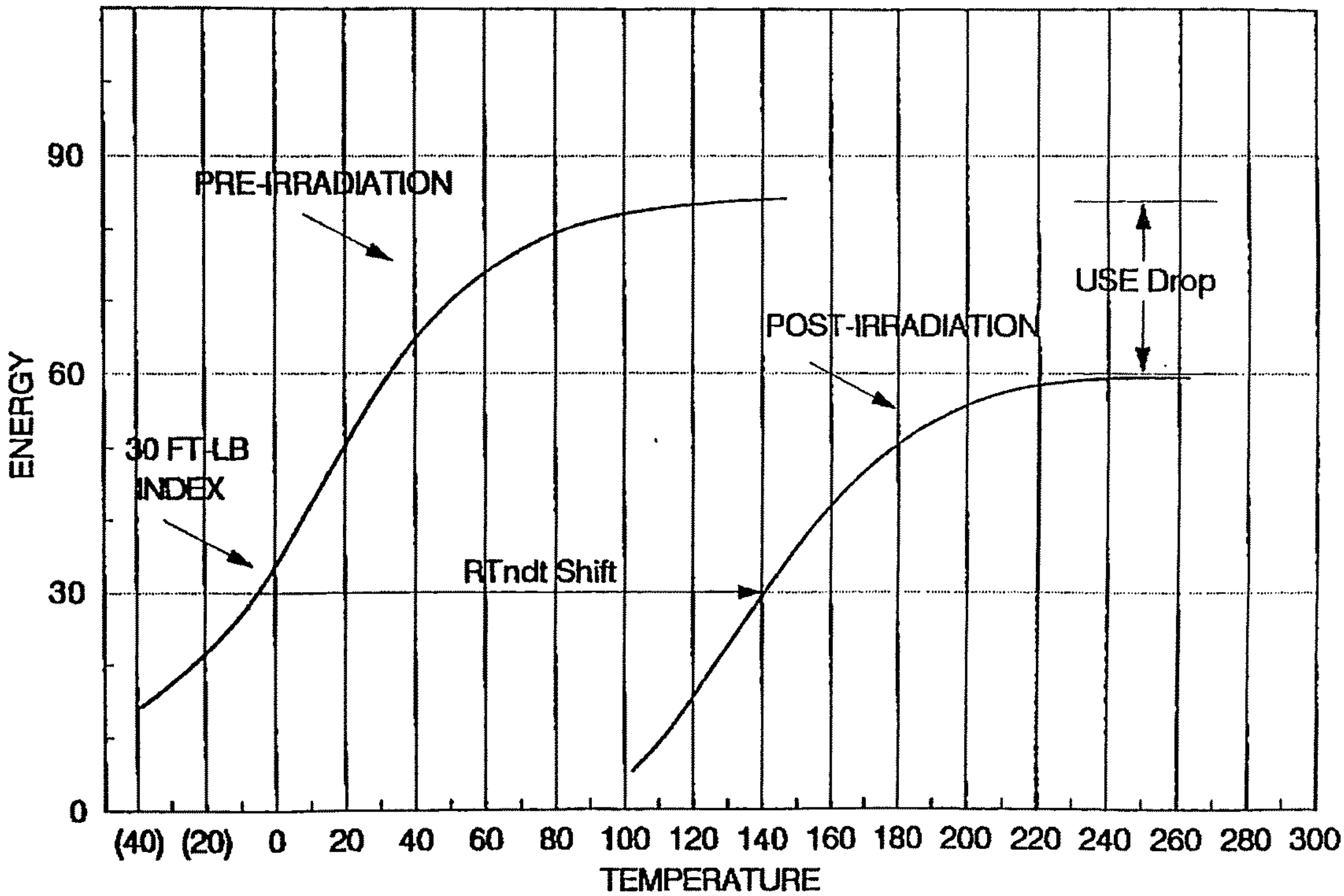


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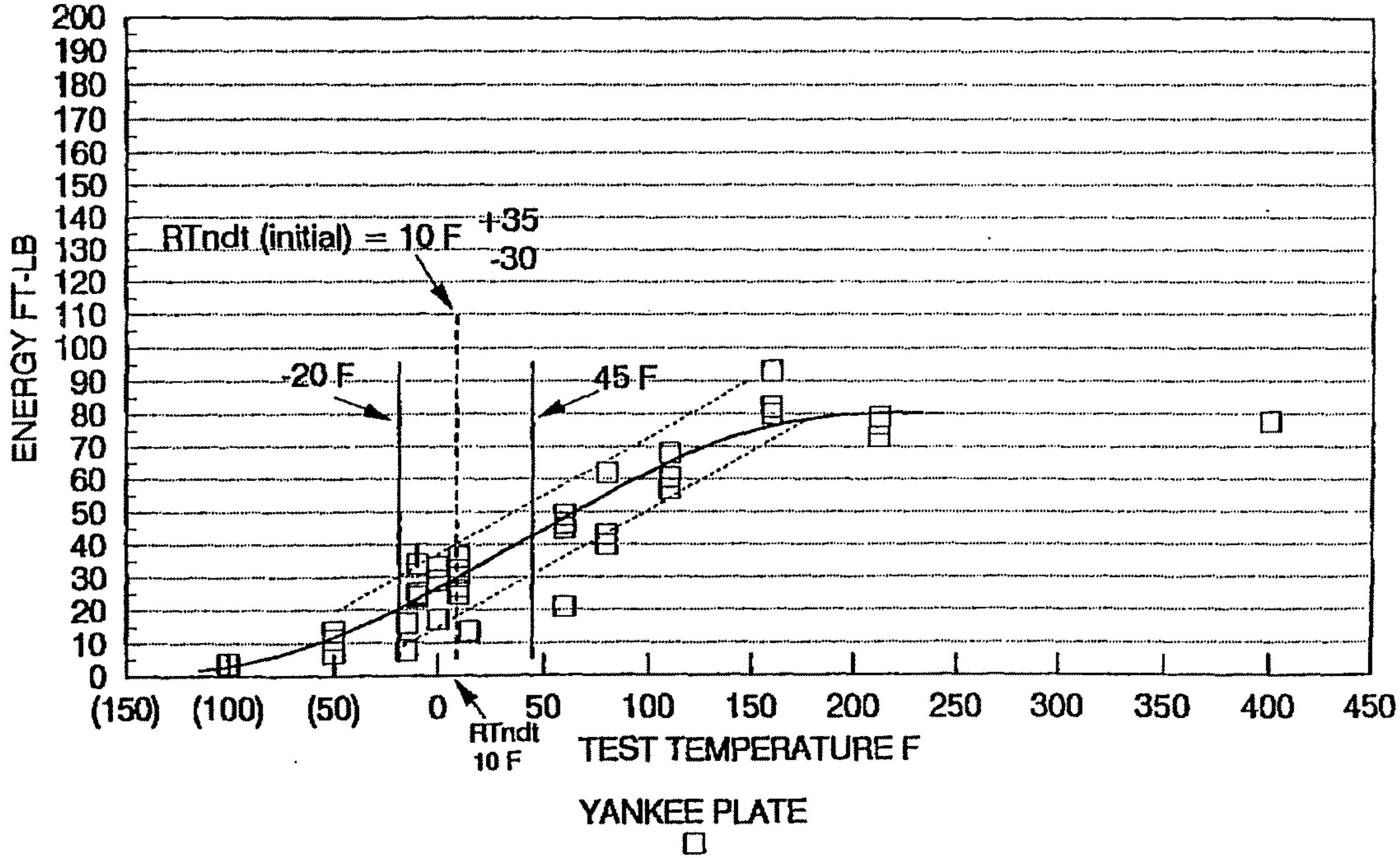
FIGURE 3

UNIRRADIATED CHARPY DATA FOR YANKEE SURVEILLANCE PLATE (UPPER SHELL PLATE) DEPICTS POSSIBLE VARIANCE IN INITIAL RTndt



Longitudinal specimens only

FIGURE 4

IRRADIATED CHARPY DATA
EXAMPLE OF CURVE FIT

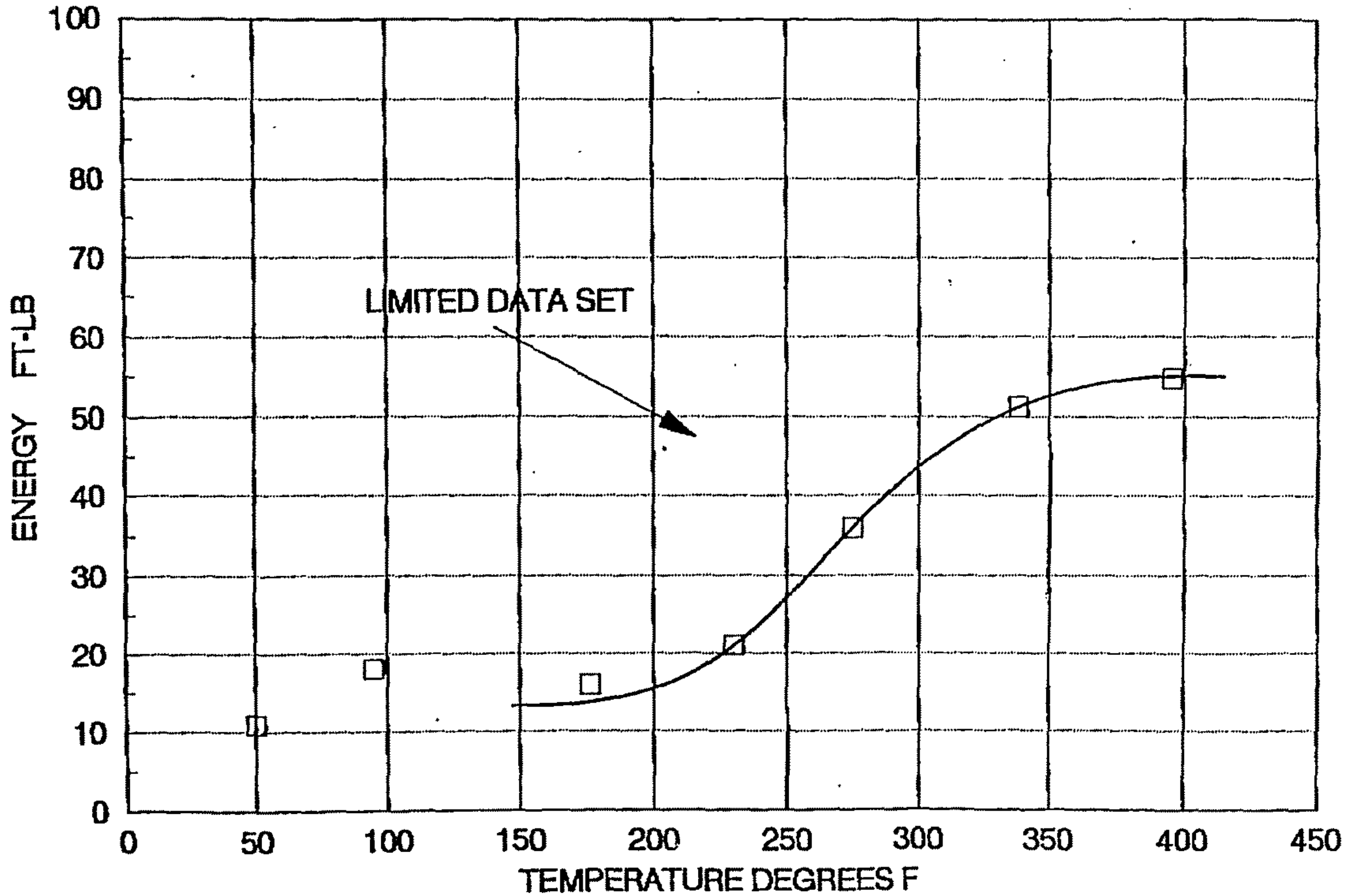
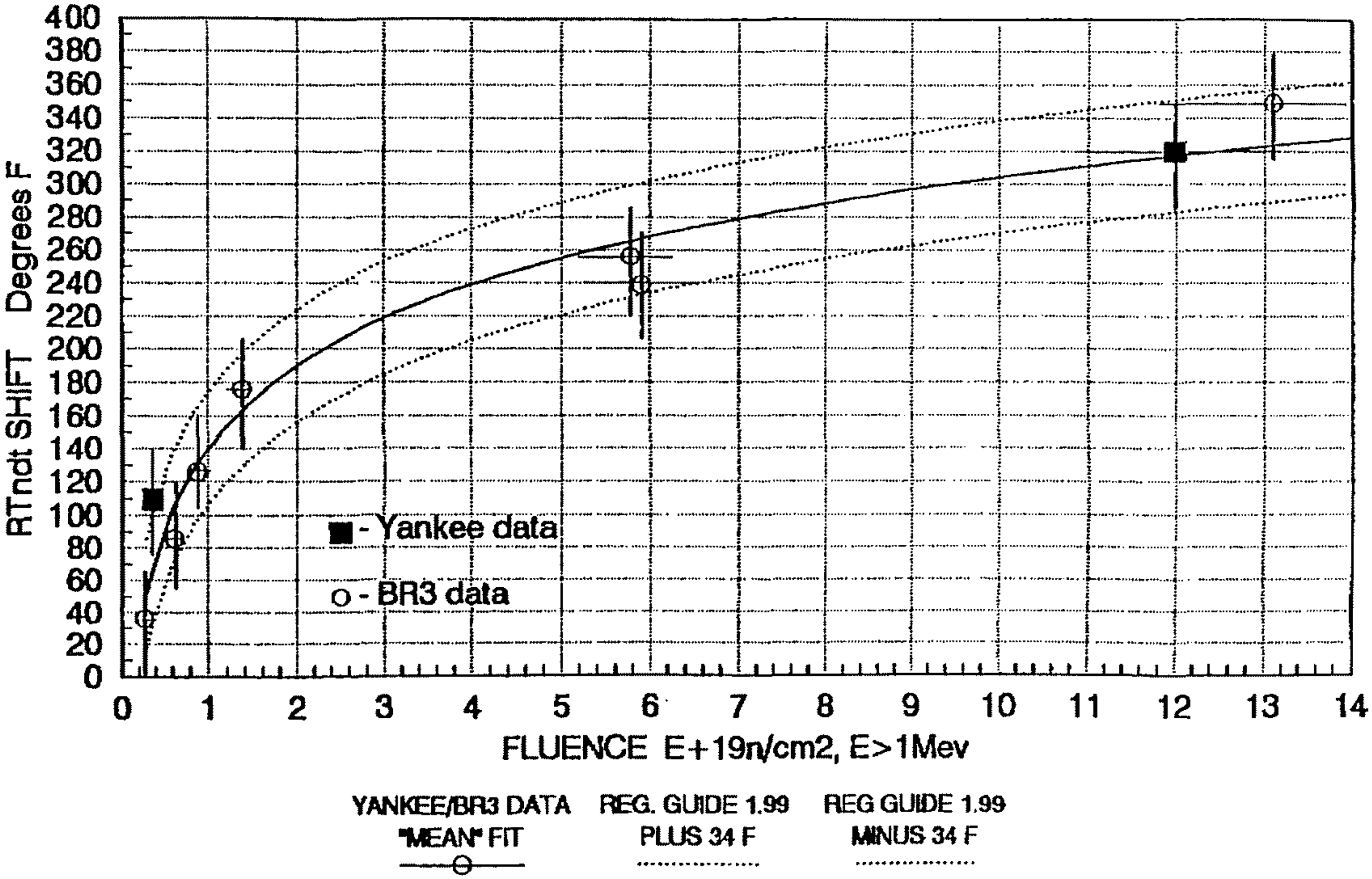


FIGURE 5

YANKEE-BR3 SURVEILLANCE DATA VERSUS REG. GUIDE 1.99 MARGINS FOR PLATE OF 2 SIGMA (PLUS AND MINUS 34 DEGREES F)



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

ATTACHMENT R5

SRM RESPONSE R5 AND R6

OFFICE OF THE
EXECUTIVE DIRECTOR
FOR OPERATIONS

July 23, 1991

TO: D. Rathbun, OCM/IS
J. Scarborough, OCM/KR
J. Gray, OCM/JC
J. Guttman, OCM/FR

FROM: James L. Blaha, AO/OEDO

SUBJECT: YANKEE ROWE INFORMATION

The enclosures respond to Commissioner Curtiss' request at the July 11, 1991 Commission meeting. Please note that Enclosure 4 contains proprietary data. This information will also be provided with the staff proposal on July 24, 1991.

James L. Blaha, AO/OEDO

Enclosure:
As stated

cc: SECY
OGC
J. Taylor, EDO
J. Sniezek, DEDR
B. Elliot, NRR

Since the issuing of this note, new information was obtained indicating that Enclosure 4 can be released to the public. See next page for further details.

Jose A. Calvo

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

July 23, 1991

MEMORANDUM FOR: James L. Blaha
Assistant for Operations
Office of the Executive Director for Operations

FROM: William T. Russell, Associate Director
for Inspection and Technical Assessment
Office of the Nuclear Reactor Regulation

SUBJECT: COMMISSIONER CURTISS' REQUEST FOR INFORMATION DURING A COMMISSION HEARING - BRIEFING ON YANKEE ROWE PRESSURE VESSEL EMBRITTLEMENT ISSUES, JULY 11, 1991

During the NRC staff presentation to the Commission on Yankee Rowe Pressure Vessel Embrittlement Issues on July 11, 1991, Commissioner Curtiss requested additional information related to Appendix G of 10 CFR Part 50. Enclosure 1 responds to Commissioner Curtiss' request. Enclosures 2, 3 and 4 provide background information. I request that you provide this information to Commissioner Curtiss and the other Commissioners. Please note that Enclosure 4 contains Babcock & Wilcox proprietary data.

William T. Russell

William T. Russell, Associate Director
for Inspection and Technical Assessment
Office of Nuclear Reactor Regulation

Enclosures:
As Stated

cc: J. Taylor
T. Murley
F. Miraglia
J. Partlow
B. D. Liaw
S. Varga
J. Calvo
P. Sears

CONTACT: B. Elliot, DET/EMCB
X-20709

Since the issuing of this memorandum, two minor corrections were made on advise from Babcock and Wilcox to Figure 7-11 in Enclosure 4 (data was not changed) that removed the need to identify this figure as proprietary. The proprietary label was also deleted from other pages. Thus Enclosure 4 of Attachment R5 does not contain any proprietary information and can be released to the public.

W T Russell
W.T. Russell 7/23/91

ENCLOSURE 1

INFORMATION RELATED TO APPENDIX G
OF 10 CFR PART 50

1. During the July 11, 1991 meeting, Commissioner Curtiss said "give me an example of a case where the upper-shelf energy requirements of IV.1 have not been met, the authority or the flexibility that the Director of NRR has under IV.1 is exercised to prevent operation at a lower level but the requirements of Appendix V.C. would nevertheless kick in." He clarified his question by saying, "is there any case like that?" Commissioner Curtiss further clarified the question by saying, "explain the instances where the requirements of C.1, .2 and .3 in Appendix G.V would kick in."

Answer

The technical staff in an internal memorandum dated September 24, 1987 (Enclosure 2) reported that Turkey Point Units 3 and 4, Point Beach Unit 1 and 2 and Ginna had reactor vessel beltline welds with Charpy upper shelf energies less than 50 ft-lb. Yankee Rowe was not identified because the welds were thought to have low amounts of copper and the plates were not evaluated. At the time of this activity, the ASME Code had not identified criteria for demonstrating compliance with paragraph IV.A.1. Since, the licensees could not comply with the requirements in paragraph IV.A.1, they had to comply with the requirements in paragraphs V.C.1, .2 and .3. On November 20, 1989, NRC received from the subgroup chairman of ASME, criteria for performing evaluations for upper shelf energy (USE), to demonstrate equivalent margins to the ASME Code Appendix G. This was forwarded May 1, 1990 to Yankee Atomic Energy Company with additional NRC requirements for use in demonstrating equivalent margins to the ASME Code, Appendix G.

2. Commissioner Curtiss also asked, "Have we processed -- have we had instances where the IV.1 upper-shelf energy requirements have not been met and that have been processed exclusively under IV.1?"

Answer

As discussed previously, no plants were processed exclusively under IV.A.1 because we had no criteria to process the plants. Yankee Rowe is the first plant processed exclusively under IV.A.1. The staff reviewed fracture toughness analysis of the beltline welds in the Turkey Point reactor vessel in a letter dated May 31, 1988 (Enclosure 3). The staff in a letter dated July 6, 1989 (Enclosure 4) identified actions taken by the licensees for Turkey Point, Point Beach and Ginna to ensure their reactor vessels meet paragraphs V.C.1, .2 and .3.

Enclosure 2

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

September 24, 1987

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

THRU: Richard W. Starostecki, Associate Director
for Inspection and Technical Assessment

FROM: Lawrence C. Shao, Director
Division of Engineering and Systems Technology

SUBJECT: REACTOR VESSELS MOST SUSCEPTIBLE TO HAVING THEIR CHARPY UPPER SHELF ENERGY REDUCED BELOW 50 FT-LB

In response to your request we are providing a list of reactor vessels that as a result of neutron irradiation are most susceptible to having their Charpy upper shelf energy (USE) reduced below 50 ft-lb. Table I of the enclosure contains the requested list. Susceptibility is greatest for vessels containing high copper welds made with Linde 80 weld flux, characteristic of many early vessels fabricated by Babcock and Wilcox Co. The initial upper shelf energy was low, because of the flux, and the copper content makes them susceptible to neutron embrittlement. High fluence is of course a factor, and this is a characteristic of some Westinghouse reactors, such as those which lead the susceptibility list.

Table I contains the list of 17 PWRs having high copper, Linde 80, welds, which we believe are the reactor vessels most susceptible to having their Charpy USE fall below 50 ft-lb. The table identifies the calculated Charpy USE for the limiting reactor vessel beltline weld on January 1, 1986 and at the end of the plant's license. The calculation was performed using the methodology recommended in Proposed Regulatory Guide 1.99, Rev. 2. The calculation indicates that as a result of neutron irradiation, several PWR reactor vessels had weld metal with Charpy USE at or less than 50 ft-lb on January 1, 1986. The table indicates the licensees that have taken action to resolve this concern. Licensees for Point Beach and Ginna have not instituted a program to resolve this issue. These licensees should be informed that based on the calculation methods in Proposed Regulatory Guide 1.99, Rev. 2, the Charpy USE for their limiting reactor vessel welds are predicted to be less than 50 ft-lb. Based on this calculation, the licensees are subject to the requirements of Appendix G, 10 CFR 50, paragraph V.C. This paragraph requires: (1) inspection for beltline flaws, (2) materials tests and (3) analyses of vessel integrity to see if "...lower values of upper shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code." The licensees should within one year of receipt of the staff letter, provide the staff with a plan for implementing the requirements of Appendix G, 10 CFR 50, paragraph V.C.

In addition, as more reactor vessel material surveillance data becomes available the NRC staff and licensees should evaluate whether the effect of neutron irradiation on the Charpy USE of Linde 80 weld metal and other

Contact: B. Elliot, EMTB/NRR
X-27895

SEP 24 1987

materials such as plate can be predicted with statistical engineering confidence rather than using bounding values.

Original Signed by
L. C. Shao

Lawrence C. Shao, Director
Division of Engineering and Systems
Technology

Enclosure:
As stated

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R. Starostecki
F. Miraglia
J. Richardson
C. Y. Cheng
B. Elliot
K. Wichman
W. Hazelton
N. Randall

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Enclosure

Charpy Upper Shelf Energies
for Reactor Vessel Materials

Regulatory Background

Appendix G, 10 CFR 50, specifies fracture toughness requirements for reactor vessel beltline materials to provide adequate margins of safety during normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the vessel may be subjected over its service lifetime. Section IV.A.1 of Appendix G states, in part, that reactor vessel beltline materials must maintain Charpy upper-shelf energy (USE) throughout the life of the vessel no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, NRR, that lower values of upper shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.

NUREG-0744 Rev. 1 dated October 1982 provided an engineering method, based on elastic-plastic fracture mechanics for analyzing reactor vessel beltline materials with Charpy USE below 50 ft-lb to determine whether they have adequate resistance to fracture. The NUREG did not specify the criteria for acceptability for various plant conditions. These criteria are being reviewed by an ASME code subcommittee, in which the staff has been actively involved.

The effect of neutron irradiation on a material's Charpy USE can be predicted using figure 2 in Regulatory Guide (R.G.) 1.99. This figure indicates that as the amounts of copper in the weld and neutron fluence increase the Charpy USE decreases. This regulatory guide is undergoing a change from revision 1 to 2. However, figure 2 has not changed.

Discussion

Susceptibility for reducing Charpy USE below 50 ft-lb is greatest for material with high copper, low unirradiated Charpy USE and high neutron fluence. Weld metals fabricated by Babcock & Wilcox (B&W) using a submerged arc weld process with Linde 80 flux appears to be the materials most susceptible to having their Charpy USE reduced below 50 ft-lb. Other vessel fabricators such as Combustion Engineering (CE) and Chicago Bridge and Iron (CBI) used weld metal with high upper shelf energies. Hence, welds in CE and CBI fabricated vessels are less susceptible to having their Charpy USE reduced below 50 ft-lb than welds fabricated by B&W. In addition to vessels for their own NSSS design, B&W supplied reactor vessels for BWR's and Westinghouse NSSS designs. BWR reactor vessels receive significantly less neutron fluence than either Westinghouse or B&W reactor vessels. Hence, BWR reactor vessels with Linde 80 flux welds are less susceptible to having their Charpy USE reduced below 50 ft-lb than PWR reactor vessels. We have not made an exhaustive survey of USE characteristics of all plate materials in reactor vessel beltlines. However, plate materials, typically have higher unirradiated Charpy USE than Line 80 weld metal. Hence, they are less susceptible to having their Charpy USE reduced to 50 ft-lb than Linde 80 weld metal.

Table I lists the PWR reactor vessels, which contain Linde 80 weld metal fabricated by B&W. The table identifies the calculated Charpy USE for the limiting reactor vessel beltline weld on January 1, 1986 and at the end of the plant's license. The calculation for Charpy USE at end of license was performed using the methodology recommended in Proposed R.G. 1.99, Rev. 2. This Guide recommends that the calculation be performed using a line drawn parallel to the existing trend curves and bounding all the data when credible surveillance data is available. This method, although conservative, is necessary when plant specific data is sparse and scattered. When credible surveillance data was not available, the calculation was performed by linear interpolation between trend curves contained in figure 2 of the guide based on the copper content.

The calculation for Charpy USE on January 1, 1986 was performed using the methodology recommended; (1) by Commonwealth Edison Co. (CECo) for Zion, Unit 1 and 2; (2) by the B&W Owners Group for B&W plants participating in their owners group; and (3) in Proposed R.G. 1.99, Rev. 2 for the rest of plants. Both methods (1 and 2) provided "best estimates" for when plants would reach 50 ft-lb. CECo combined data from Units 1 and 2 surveillance program to estimate when the limiting beltline weldment would reach 50 ft-lb. The Zion surveillance program contains weld metal samples that are equivalent to the limiting beltline weldment. The Zion surveillance program has produced 5 credible surveillance data points with very little data scatter. Based on the number of credible surveillance data points and the limited amount of scatter, the Zion procedure is an acceptable alternative to R.G. 1.99 Rev. 2 method for calculating when Charpy USE will reach 50 ft-lb. CECo estimates Zion, Units 1 and 2 will reach 50 ft-lb no earlier than 1994. The B&W Owners Group performed a statistical analysis of all the existing surveillance Linde 80 weld metal. The staff considers a statistical analysis of all surveillance data points an acceptable alternative to the R.G. 1.99 Rev. 2 method for calculating when Charpy USE will reach 50 ft-lb. The owners group procedure indicates all B&W plants will reach 50 ft-lb no earlier than 1997. The B&W plants are identified in Table I. CECo and B&W Owners Group procedures for calculating Charpy USE were not used for estimating the Charpy USE at the end of the plants licenses because there is insufficient surveillance data at end-of-license neutron fluences.

Yankee Rowe and Byron-1 have been included on the list as vessels containing Linde 80 weld metal. However, since both of these vessels have low copper Linde 80 welds, they are less susceptible to having their Charpy USE reduced by neutron irradiation than the other PWR reactor vessels.

Florida Power & Light (FPL) Company has recognized that the Turkey Point -3, -4 reactor vessel welds are susceptible to having their Charpy USE reduced below 50 ft-lb. FPL has provided a fracture mechanics analysis to demonstrate that lower values of upper shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. This analysis is under review by the staff.

Since the Charpy USE for reactor vessel welds in Point Beach -1, -2 and Ginna are calculated, in accordance with R. G. 1.99, to be less than 50 ft-lb, the licensees are subject to the requirements of Appendix G, 10 CFR 50, paragraph V.C. which requires: (1) inspection for beltline flaws, (2) materials tests and (3) analyses of vessel integrity to see if "...lower values of upper shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code." (App. G, 10 CFR 50, paragraph IV.A.1).

The exact nature of the materials tests and analyses required by the regulation in general terms has been the subject of research for several years. The question was addressed in USI A-11, and a final report (NUREG 0744) was issued in October 1982. However, the formulation of acceptance criteria that satisfied critics in the industry and the fracture community was not completed. Instead, a question was sent to the chairman of the Subcommittee on Nuclear Inservice Inspection of the ASME Boiler and Pressure Vessel Code (Section XI) for help on that issue. The final report of the Working Group on Flaw Evaluation to whom it was assigned is not out yet, partly because there is continued evolution in the technology of elastic-plastic fracture analysis. The tentative criteria for normal operation and anticipated operational occurrences (Levels A and B loading, in Code language) are that crack instability shall not occur at pressure of 110% of Design with a safety factor of 2.0 on pressure stresses and 1.0 on thermal stress. Simplified analyses have been made from time to time which confirmed our belief that safety margins against low-energy ductile fracture were adequate for materials having Charpy USE values in the 40 - 50 ft-lb range. This is the principal reason that we have been willing to wait for development of the fracture analysis techniques before pursuing this issue further. Another reason is that there are no known, accident scenarios that overpressure the reactors much beyond the setpoint of the safety valves (which are sized to limit pressure to 110% of Design). The ATWS Rule (10 CFR 50.62) requires certain hardware and systems installations to minimize the frequency and severity of ATWS transients. The other accident scenarios that have been identified stress the vessel in the transition temperature range and are analyzed by linear elastic fracture mechanics methods.

<u>Recommendations</u>

Point Beach and Ginna should be informed that based on the calculative methods in Proposed R. G. 1.99, Rev. 2, the Charpy USE for their limiting reactor vessel beltline welds are predicted to be less than 50 ft-lb. Based on this calculation, the licensees are subject to the requirements of Appendix G, 10 CFR 50, paragraph V.C. The licensees should within one year of receipt of the staff letter, provide the staff with a plan for implementing the requirements of Appendix G, 10 CFR 50, paragraph V.C.

As more reactor vessel material surveillance data becomes available, the NRC staff and licensee should evaluate whether the effect of neutron irradiation on the Charpy USE of Linde 80 weld metal and other materials such as plate can be predicted with statistical engineering confidence rather than using bounding values.

Table I

Calculated Charpy Upper Shelf Energies (USE)
for PWR Reactor Vessels with Linde 80 Weld Metal

PWR Plant	End of License	Charpy USE at End of License (ft-lb)	Charpy USE on Jan. 1, 1986 (ft-lb)
Point Beach -2 (W)	2013	34	39
Point Beach -1 (W)	2010	38	43
Turkey Point -3 (W)	2007	40	44*
Turkey Point -4 (W)	2007	40	44*
Ginna (W)	2006	42	47
Arkansas One -1 (B&W)	2008	44	>50**
Rancho Seco (B&W)	2008	44	>50**
Crystal River -3 (B&W)	2008	44	>50**
TMI-1 (B&W)	2008	44	>50**
Oconee-1 (B&W)	2013	44	>50**
Oconee-3 (B&W)	2014	44	>50**
Surry-2 (W)	2008	46	51
Zion-1 (W)	2008	47	>50***
Zion-2 (W)	2008	49	>50***
Oconee-2 (B&W)	2013	49	>50**
Surry-1 (W)	2008	53	57
Davis Besse (B&W)	2011	56	>50**
Yankee Rowe (W)	1997	Low Copper Welds	
Byron-1 (W)	2024	Low Copper Welds	

*Florida Power & Light Company has provided analyses to demonstrate that the weld metal in these reactor vessels have adequate margin.

**B&W Owners Group has provided analysis to demonstrate that the Charpy USE for these reactor vessel welds will reach 50 ft-lb no earlier than 1997.

***Commonwealth Edison Co. has provided analyses to demonstrate that the Charpy USE for these reactor vessel welds will reach 50 ft-lb no earlier than 1994.

Enclosure 3

UNITED STATES
NUCL .AR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 31, 1988

Docket Nos. 50-250
and 50-251

Mr. W. F. Conway
Senior Vice President - Nuclear
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408

Dear Mr. Conway:

SUBJECT: TURKEY POINT UNITS 3 AND 4 REACTOR VESSEL FRACTURE TOUGHNESS (TAC NOS. 68249 AND 55042)

The purpose of this letter is to transmit our review of two reports submitted by Florida Power and Light Company (FP&L). These reports were intended to demonstrate by analysis the existence of adequate safety margins in the Turkey Point Units 3 and 4 reactor vessels when the Charpy upper-shelf energy is below the 50 ft-lb requirement in Appendix G of 10 CFR 50.

By letters dated May 3, 1984 and March 25, 1986, FP&L submitted for NRC's review a fracture toughness analysis of the beltline welds for the Turkey Point reactor vessels. In our Safety Evaluation (enclosed) we have identified the need for additional analysis and data acquisition. Until this information is provided, we cannot complete our review of the reports.

We request that within one year of the date of this letter, FP&L provide a revised analysis incorporating the information requested and a plan for data acquisition. The analysis should include an estimate of the Charpy upper-shelf energy at both the next refueling outage following the submittal date of the report and at expiration of the Turkey Point Units 3 and 4 licenses. The Charpy upper-shelf energy estimates for the beltline weld should be based on the plant's anticipated future fuel management plan and extrapolation of the surveillance data from the Turkey Point Integrated Surveillance Program using the method recommended in Regulatory Guide 1.99, Rev. 2.

We also recommend that FP&L contact the ASME Code Section XI Committee (Working Group on Flaw Evaluation) to determine the status of the Committee's development of recommended safety margins and any impact they would have on Turkey Point if approved for use by the NRC.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than 10 respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Gordon E. Edison, Sr. Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosure: As stated

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Mr. W. F. Conway
Florida Power and Light Company

Turkey Point Plant

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

ENCLOSURE

SAFETY EVALUATION RELATED TO REACTOR VESSEL FRACTURE TOUGHNESS FLORIDA POWER AND LIGHT COMPANY TURKEY POINT UNITS 3 AND 4 DOCKET NOS. 50-250, 251

INTRODUCTION

The licensee indicates that although the Charpy upper-shelf energy (USE) for the Turkey Point Units 3 and 4) (TP) reactor vessel limiting beltline material will be below 50 ft-lb, their fracture mechanics analysis indicates that the material will meet the safety margins of Appendix G of the ASME Code for at least 40 effective full power years (EFPY). However, our review of the analysis indicates that further analysis and data acquisition are necessary. The need for additional analysis and data acquisition is discussed in this Safety Evaluation.

Section IV.A.1 of Appendix G, 10 CFR 50 requires, in part, that the Charpy upper-shelf energy (USE) for all reactor vessel beltline materials be above 50 ft-lb throughout the life of the vessel, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.

Section V.C.3 of Appendix G, 10 CFR 50 requires that the licensee perform analyses to demonstrate the existence of equivalent margins of safety when the Charpy USE is predicted to be less than 50 ft-lb. In letters to the Office of Nuclear Reactor Regulation, USNRC dated May 3, 1984 and March 25, 1986, the licensee provided analyses, which are intended to demonstrate that at 40 EFPY, which corresponds to a neutron fluence of 2.88×10^{19} n/cm^2 (E>1MeV) at the vessel's inside surface, the fracture toughness of each of the reactor vessels meets the safety margins of Appendix G of the ASME Code.

Appendix G of the ASME Code presents a procedure for calculating the allowable pressure for pressure vessels. The procedure is based on the principles of linear elastic fracture mechanics (LEFM). This ASME Code procedure postulates that the Turkey Point pressure vessels have a sharp surface defect normal to the direction of maximum stress that has a depth of one-fourth of the section thickness (1/4 T) and a length six times its depth. For Levels A and B service conditions, the safety margin on the allowable pressure is required to be a factor of 2. Appendix G does not contain fracture toughness limits for Levels C and D service conditions.

In NUREG-0744, Rev. 1 dated July 1982, the staff provided guidance for performing the analyses required by Section V.C.3 of Appendix G, 10 CFR 50. The recommended procedure to be followed is based on the J-Integral Elastic Plastic Fracture Mechanics (EPFM) method. In this method the material fracture resistance is measured using the parameters J, the intensity of the plastic stress-strain field surrounding the crack tip, and T, the tearing modulus. These parameters must be determined by testing of neutron irradiated material, which is equivalent to the material in the reactor vessel beltline. The test limits on these parameters depend upon the amount of J-controlled crack growth. The maximum load-carrying capability of the irradiated reactor vessel occurs when the calculated values J_{app} and T_{app} for the reactor vessel with the postulated flaw equals the J_{mat} and T_{mat} for the irradiated material. When J_{app} exceeds J_{mat}, the postulated flaw is considered to be unstable. The NUREG indicates that the value of the ratio of J/T for surface cracks is dependent upon the material's flow stress, the postulated crack size, a geometry correction factor, and a stress correction factor.

The NUREG indicates that the margins for Levels A and B service conditions should be no less than that now required by the ASME Code, Appendix G. The NUREG does not specify the margins required for Levels C and D service conditions. In a letter from R. E. Johnson to L. T. Chockie dated April 20, 1982, the NRC requested that the ASME Code Committee develop safety margins for all service conditions.

DISCUSSION

The margins of safety against fractures were determined by comparing the predicted value of J_{mat} at instability to values of J_{app} due to normal operating (Levels A and B) stresses acting on the ASME Code postulated flaw. J_{mat} values for the TP limiting beltline welds were extrapolated from a Heavy Section Steel Technology (HSST) welds, which was fabricated using the same heat of weld wire and flux as used in the limiting TP welds. However, the HSST data was irradiated in a test reactor, which has a much higher neutron flux than a commercial reactor.

The J-T curves used to determine the material elastic plastic fracture resistance were developed from 1.6 T compact toughness (CT) specimens. As a result of specimen size limitations the amount of J-controlled crack growth is limited to approximately 5 mm. NUREG-0744 describes a method for extrapolating beyond the J-controlled growth limits when small specimens are used to determine the material's fracture resistance. This method was not followed in the licensee's analyses. Extrapolation of data beyond the J-controlled growth limits is being studied by the Office of Nuclear Regulatory Research and the ASME Code Section XI Committee, "Working Group on Flaw Evaluation."

To determine the material fracture resistance curve (J_{mat},T_{mat}) as a function of neutron fluence, the licensee extrapolated HSST data using a relationship observed between J_{mat} and T_{mat}, an empirically derived relationship between J_{mat} and Charpy USE, and the relationship between Charpy USE and neutron fluence reported in Regulatory Guide 1.99, Rev. 1. The relationships observed and derived in the analysis provide values for J and T beyond the J-controlled growth limits. The licensee has not provided material test data to demonstrate that these relationships apply beyond the J-controlled growth limits. The licensee must provide supplemental fracture toughness data from a commercial reactor surveillance program that demonstrates their analysis, which used HSST data, applies to material irradiated in a commercial reactor. In accordance with Section III.B of Appendix G, 10 CFR 50, the test methods used to provide the supplemental data must be submitted to and approved by the Director, Office of Nuclear Reactor Regulation, prior to testing.

The staff recommends that the relationship between neutron fluence and Charpy USE for the TP reactor vessel beltline materials be predicted using the methodology recommended in Proposed Regulatory Guide (R.G.) 1.99, Rev. 2. This guide recommends that the calculation be performed using a line drawn parallel to the existing trend curves and bounding all the data when credible surveillance data is available. This method, although conservative, is necessary when plant-specific data are sparse and scattered. To date, only three capsules that contain weld metal specimens have been withdrawn and tested. The licensee should use all weld metal surveillance data from these three capsules to determine the relationship between Charpy USE and neutron fluence.

In a letter dated April 22, 1985, the staff approved an Integrated Surveillance Program for Turkey Point (TP) Unit Nos. 3 and 4. The test results from material irradiated in surveillance capsules in these vessels are to be used to determine the vessel's fracture toughness resulting from neutron irradiation. Our evaluation of the fracture toughness data derived from the last capsule withdrawn from TP-3 is contained in a letter from D.G. McDonald to C.O Woody, dated October 30, 1987. The Safety Evaluation contained in that letter indicates that the formula in R.G. 1.99, Rev. 2 conservatively predicts the effect of neutron irradiation on the limiting weld metal in the TP-3 and TP-4 reactor vessels.

When the licensee used the empirically derived relationship between J_{mat} and Charpy USE to determine the Turkey Point material fracture resistance, the licensee assumed that the J_{mat} values from the HSST data corresponded to the Charpy USE values from R.G. 1.99, Rev. 2. This assumption is incorrect and results in a non-conservative value for J at instability. The licensee should have used actual Charpy USE data from the HSST program to determine the relationship between Charpy USE and J_{mat} for the Turkey Point beltline materials.

To determine the value of J at instability the flow stress must be known. In the licensee's analysis the flow stress for the Turkey Point material was derived from the HSST data. Based on the TP surveillance program test results, the value of the flow stress at the end of the plant's license was underestimated. However, lower values of flow stress produce conservative values for J at instability.

The J at instability was determined for a neutron fluence of 1.73×10^{19} n/cm^2 (E>1MeV). This was calculated to be the neutron fluence at the tip of the postulated ¼ T depth flaw when the neutron fluence at the inside surface is 2.88×10^{19} n/cm^2 (E>1MeV) and the TP reactor vessels reach 40 EFPY. The attenuation of neutron fluence from the inside surface to ¼ T depth was performed using a non-conservative method. To determine the effect of neutron irradiation on the TP beltline materials, the neutron fluence through the vessel wall should be attenuated using the formula for displacements per atom in R.G. 1.99, Rev. 2 or SECY 82-465, "Pressurized Thermal Shock."

The licensee's calculation of J_{app} at the tip of the ¼ T postulated flaw included an elastic component, but did not include a plastic component. The stress calculation includes values for the membrane stress from internal pressure, the pressure on the crack surface, the temperature changes during heatup and cooldown and residual weld stress. When these values are summed the author indicates that the value is low enough to permit the use of only the elastic component for calculating J_{app}. However, when the allowable pressure is doubled, in accordance with the safety margins required by Appendix G, the applied stress is near the irradiated material's yield stress. When the applied stress is near the materials yield stress the plastic component of J_{app} can be large and should be considered in the analysis. Hence, to demonstrate that the postulated ¼ T flaw meets the safety margin requirements of Appendix G during Levels A and B service conditions the plastic component of J_{app} must be added to the elastic component.

In addition to Levels A and B service conditions, the reactor vessel's design must consider Levels C and D service conditions. The licensee's analysis does not consider these service conditions. The safety margins for fracture resistance during all service conditions are currently under discussion in the ASME Code Section XI Committee, "Working Group on Flaw Evaluation." When the Committee provides reactor vessel fracture resistance safety margins for all service conditions and when they have been approved by NRC, the licensee should determine whether TP can meet these safety margins.

CONCLUSION

In our Safety Evaluation, we indicate that additional analysis and material test data are needed to confirm that the TP reactor vessels will meet the safety margins of Appendix G of the ASME Code and 10 CFR 50 for 40 EFPY. Until this information is supplied, we can not complete our review of the licensee's submittals.

Dated: May 31, 1983

Principal Contributor:

B. Elliot

Enclosure 4

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

July 6, 1989

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

THRU: Frank J. Miraglia, Jr., Associate Director
for Inspection and Technical Assessment

FROM: James E. Richardson, Acting Director
Division of Engineering and Systems Technology

SUBJECT: CHARPY UPPER SHELF ENERGY FOR REACTOR VESSEL BELTLINE MATERIALS

On June 29, 1989, the staff met with members of the ACRS, R. Fraley, E. Igne, and P. Shewmon, the Chairman of the ACRS Sub-committee on Materials and Metallurgy, to discuss the issue of low Charpy Upper Shelf Energy (USE) of reactor vessel beltline materials. A list of attendees are attached. As reactor vessel beltline materials are irradiated their high temperature fracture resistance is decreased. This reduction in fracture resistance is measured on irradiated surveillance materials by a reduction in their Charpy USE.

The staff discussed with the ACRS the regulatory requirements for Charpy USE, which are contained in Appendix G, 10 CFR 50. Paragraph IV.A.1 of Appendix G requires that "the reactor vessel beltline materials must have Charpy USE... of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code."

Based on the method of predicting the Charpy USE that are contained in Regulatory Guide 1.99, Rev. 2, we identified 5 plants that are presently predicted to have beltline welds with Charpy USE below 50 ft-lbs. The staff identified actions these plants have taken and are planning to ensure reactor vessel integrity and meet regulatory requirements. Licensees have performed: (a) neutron flux reductions, (b) a fracture mechanics analysis to demonstrate margins of safety against fracture, and (c) volumetric examinations of beltline welds. Licensees are planning to up-date their fracture mechanics analyses and provide supplementary irradiated fracture toughness data from Babcock and Wilcox Owners Group Surveillance Capsules.

CONTACT: Barry Elliot, EMCB/DEST
X20931

The information presented by the staff to the ACRS is attached. Dr. Shewmon appeared to be satisfied with the staff's presentation. The staff will meet with the ACRS Main Committee to discuss this issue on July 13, 1989.

Original Signed by

James E. Richardson, Acting Director
Division of Engineering and Systems
Technology

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LIST OF ATTENDESS AT JUNE 29, 1989 MEETING
BETWEEN NRC STAFF AND ACRS

NRC

L. Shao
J. Richardson
C. Y. Cheng
K. Wichman
G. Edison
B. Elliot

ACRS

P. Shewmon
R. Fraley
E. Igne

LOW CHARPY UPPER SHELF ENERGY (USE) FOR

REACTOR VESSEL BELTLINE (RVB) WELDS

- APPENDIX G, 10 CFR 50 REGULATORY REQUIREMENTS

  - RVB MATERIALS MUST HAVE CHARPY USE GREATER THAN 50 FT-LB UNLESS IT IS DEMONSTRATED IN A MANNER APPROVED BY THE DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION, THAT LOWER VALUES OF USE WILL PROVIDE MARGINS OF SAFETY AGAINST FRACTURE EQUIVALENT TO THOSE REQUIRED BY APPENDIX G OF THE ASME CODE

  - WHEN CHARPY USE IS LESS THAN 50 FT-LB, THE LICENSEE MUST:
    - PERFORM 100% VOLUMETRIC EXAMINATION
    - PROVIDE IRRADIATED SUPPLEMENTARY FRACTURE TOUGHNESS DATA
    - PROVIDE AN ANALYSIS TO DEMONSTRATE MARGINS OF SAFETY

  - TEST METHODS FOR SUPPLEMENTARY FRACTURE TOUGHNESS TEST MUST BE SUBMITTED TO AND APPROVED BY THE DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION

IMPLEMENTATION OF REGULATORY REQUIREMENTS

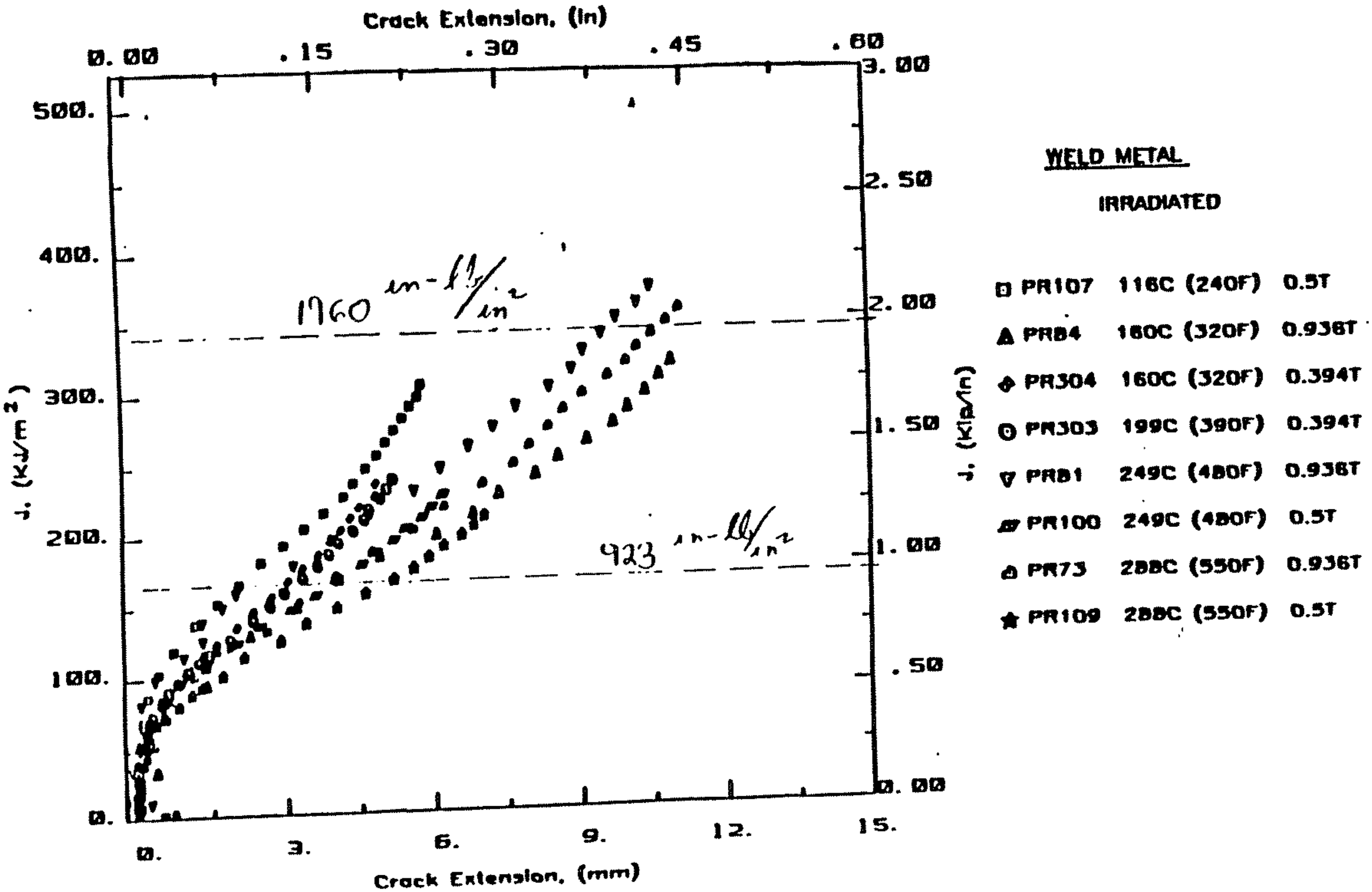
REQUIREMENT	STAFF ACTION
IDENTIFY PLANTS WHICH HAVE LOW CHARPY USE PROBLEM	PLANTS IDENTIFIED IN A SEPT. 24, 1987 LETTER TO MURLEY FROM STAROSTECKI - TURKEY PT. 3 & 4, POINT BEACH 1 & 2, AND GINNA
PROVIDE INDUSTRY WITH A METHOD FOR EVALUATING LOW CHARPY USE WELDS	NUREG-0744, "RESOLUTION OF THE TASK A-11 REACTOR VESSEL MATERIALS TOUGHNESS SAFETY ISSUE," OCT. 1982
SUPPLEMENTARY VOLUMETRIC EXAMINATION REQUIREMENTS FOR BELTLINE WELDS	REGULATORY GUIDE 1.150, "ULTRA-SONIC TESTING OF REACTOR VESSEL WELDS DURING PRESERVICE AND INSERVICE EXAMINATION," FEB. 1983
SUPPLEMENTARY FRACTURE TOUGHNESS DATA	INITIALLY USED DATA FROM HSST PROGRAM - WILL PROVIDE TEST DATA FROM A COMMERCIAL REACTOR THROUGH BWOG - STAFF MEETS PERIODICALLY WITH BWOG TO REVIEW PROGRAM AND TEST RESULTS
TEST METHOD REVIEW	STAFF APPROVED BWOG METHOD
ANALYSIS TO DEMONSTRATE MARGINS OF SAFETY	LICENSEE INDICATES PLANT CAN MEET SAFETY MARGINS FOR 40 EFPY - STAFF AGREES WITH CONCLUSION BUT REQUESTS ADDITIONAL INFORMATION

TURKEY POINT FRACTURE MECHANICS ANALYSIS

- ANALYSIS METHODS PER NUREG-0744, "RESOLUTION OF THE TASK A-11 REACTOR VESSEL MATERIALS TOUGHNESS SAFETY ISSUES", OCT. 1982
- EXTRAPOLATION OF IRRADIATED HSST DATA
- LEVEL A & B SERVICE CONDITIONS
  - SURFACE ELLIPTICAL FLAW 11.625 X 1.9375 INCH (AXIALLY ORIENTED) IN A CIRCUMFERENTIAL WELD
  - PRESSURE OF 5000 PSI
- ACCEPTABLE FOR 40 EFPY, FLUENCE OF 1.73 X 10^{19} N/CM^2 (E>1MEV) AT 1/4T LOCATION
- J_M CRITICAL OF 1960 IN-LB/IN^2
- J_{APPL} OF 923 IN-LB/IN^2

8.45 × 10[18] n/cm² (E > 1 MeV)

Figure 7-11. Comparison of J-R Curves for Irradiated Weld Metal Showing Effects of Specimen Size and Test Temperature



BAW-1920P

STATUS OF PLANTS WITH CHARPY USE CALCULATED TO BE LESS THAN 50 FT. LB.

PLANT	NEUTRON FLUENCE AT ¼T ON 1/1/89 AT CRITICAL WELD ($N/CM^2/E>1MEV$)	FLUX REDUCTION PROGRAM	DATE WELDS VOLUMETRICALLY EXAMINED 100%	FRACTURE MECHANICS ANALYSIS	SUPPL. IRRADIATED FRACT. TOUGH. DATA
TURKEY POINT 3	0.85E19	HAFNIUM NEUTRON ABSORBERS	JULY 1981 RG 1.150	SUBMITTED PER NUREG 0744 - WILL USE BWOG ANALYSIS METHOD	WILL SUBMIT DATA BWOG
TURKEY POINT 4	0.85E19	HAFNIUM NEUTRON ABSORBERS	JULY 1982 RG 1.150	"	"
POINT BEACH 2	1.08E19	HAFNIUM NEUTRON ABSORBERS	FALL 1989 RG 1.150	WILL USE BWOG ANALYSIS METHOD	"
POINT BEACH 1	0.68E19	HAFNIUM NEUTRON ABSORBERS	MAY 1987 RG 1.150	"	"
GINNA	1.12E19	LOW LEAKAGE CORES	APRIL 1989 RG 1.150	"	"

JUL 23 1991

ATTACHMENT R6

SRM RESPONSE 7

MEMORANDUM FOR: The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: PARTIAL RESPONSE TO SRM FROM BRIEFING ON YANKEE ROWE PRESSURE VESSEL (M910711A)

At the July 11, 1991 Commission meeting on the Yankee Rowe vessel, the staff agreed to provide a list of any other plants which have required an exemption to the requirements of Appendix H to 10 CFR 50. Since the information immediately available to us is not complete, we will require some additional time to respond.

The staff did not request licensee's to re-evaluate their surveillance programs that were in effect in 1983 when 10 CFR 50.60 (which implements Appendix H) was promulgated. The staff reviews licensee programs when it receives surveillance data and Technical Specification amendments for changes to pressure-temperature limits.

Section II.B.1 of Appendix H states:

> That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. Later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal after July 26, 1983, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practical for the configuration of the specimens in the capsule. For each capsule withdrawal prior to July 26, 1983 either the 1973, the 1979, or the 1982 edition of ASTM E 185 may be used.

While we believe that most licensees comply with these requirements, the staff is considering issuance of a generic communication to determine the extent of compliance with Appendix H.

Original Signed by:
James M. Taylor
James M. Taylor
Executive Director
for Operations

cc: SECY
OGC

SRM RESPONSE 8

The effects of raising ECCS temperature from 120 degrees F to 170 degrees F is as follows:

Effect on PTS Response:

The effect on PTS response of a higher ECCS injection water temperature will be to slow the cooldown rate and raise the final downcomer temperature. Both of these effects are beneficial from a PTS perspective. It is estimated that the conditional vessel failure probability for the limiting SB-LOCA would be reduced by a factor of about 4 to 5 as a result of increasing the ECCS water temperature to 170 degrees F.

Effect on ECCS Pump Operation:

The licensee has determined that increasing the ECCS temperature will result in an available net positive suction head (NPSH) below that required for the ECCS pumps. As a result pump cavitation would be expected. Therefore, if the proposed higher ECCS temperature is adopted, it will be necessary to replace the LPI pump impellers with the low NPSH impellers. The licensee is also considering installing throttle valves in the LPI discharge to throttle the ECCS flow and thereby alleviate pump cavitation concerns.

Effect on Steam Line Break Response:

Because of the negative moderator temperature coefficient, a higher ECCS temperature is beneficial for steam line break response.

Effect on Large Break LOCA:

A higher ECCS temperature has two effects: quicker boil off by decay heat, and reduction in the ECCS flow.

The ECCS flow is bypassed during the blowdown phase, and therefore, a higher ECCS temperature will have an effect on the Large Break LOCA transient only during refill and reflood phases.

Since the existing Yankee Rowe LBLOCA analysis showed a PCT of 2197 degrees F, there is a possibility that the plant would need to be derated (less than 5%) in order to accommodate the effects of higher ECCS temperature. This assessment is based on comparing the difference of decay heats required to boil off the ECCS water of 120 and 170 degrees F, respectively. The licensee has not performed an analysis to determine the effect of higher ECCS temperature. However, they indicated that the current analysis assumed saturated ECCS water during reflood phase, and they believe no derating would be necessary. It is our understanding the licensee will perform a preliminary analysis using the proposed higher ECCS water temperature by July 25, 1991 to determine the effects.

Effect on Small Break LOCA Response:

The existing SBLOCA analyses indicate a PCT of 1601 degrees F. This result is based upon an assumption of ECCS water temperature of 200 degrees F but ECCS flow rate corresponding to 120 degrees F. Even thought the ECCS flow rate corresponding to the proposed higher temperature would be lowered, we believe that there is enough margin in the PCT and the assumption of 200 degrees F to cover the reduced ECCS flow rate. The licensee is expected to perform an analysis to confirm this conclusion.

Effect on Containment Performance:

The increased ECCS temperature will have minimal effect on the pressure and temperature of the containment during a LOCA. The initial containment pressure and temperature responses are controlled by the stored energy of the primary system which is independent of ECCS water temperature. Also, since the Yankee containment does not have a spray system the ECCS water temperature does not play a significant role in long term containment response.

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

ENCLOSURE 3

JUL 23 1991

NOTE FOR: David C. Trimble, Technical Assistant
Office of the Commissioners

FROM: James L. Blaha, Assistant for Operations
EDO

SUBJECT: RESPONSE TO COMMISSIONER CURTISS' QUESTIONS ON YANKEE ROWE

Attached are the responses to the questions received from Commissioner Curtiss regarding the Yankee Rowe reactor pressure vessel issue. Any questions regarding these responses may be directed to me or Ashok Thadani. Copies of the responses are being provided to the other Commissioners as well.

James L. Blaha, Assistant for Operations
EDO

Enclosure:
As stated

cc: D. Rathbun, OCM/IS
J. Scarborough, OCM/KR
J. Guttman, OCM/FR
SECY
J. Taylor, EDO
J. Sniezek, DEDR

RESPONSES TO COMMISSIONER CURTISS' QUESTIONS (EDO 6770)

On July 17, 1991, Commissioner Curtiss requested information for other plants besides Yankee Rowe concerning PTS events and situations where the Commission has permitted continued operation of a plant based on the probability of the unlikelihood of accidents. The responses to Commissioner Curtiss' questions are as follows:

QUESTION 1(a)

Provide a listing of industry events which staff considers to be potential PTS initiating events. Briefly describe each event, including the cause of the event and resultant pressure, temperature and timing combinations which created the PTS concern.

Staff Response

See Attachment RC 1(a)

QUESTION 1(b)

List the subset of the above events that resulted in pressure/temperature/timing combinations that would have represented a PTS threat (could have caused failure of the vessel) if they had occurred at Yankee Rowe. For each of these events, provide the probability of event occurrence at Yankee Rowe or describe the design feature differences or other reasons why the event would not occur at Yankee.

Staff Response

See Attachment RC 1(b)

QUESTION 2

Dr. Murley made reference to other situations where the Commission has permitted continued operation of a plant on an interim basis where the probabilities of severe accident/large release are in the range of what the staff has currently estimated for Yankee Rowe. Please provide a listing of those plants and brief description of the situation and the probabilities involved.

Staff Response

See Attachment RC 2

RESPONSE RC 1(a)

QUESTION 1(a) FROM COMMISSIONER CURTISS' STAFF RESPONSE

Attached are descriptions of the overcooling events which were examined in the process of development of the PTS rule 10 CFR 50.61. These descriptions are taken from SECY 82-465. Since 1982 there have been no significant overcooling events (events in which the primary system experienced an uncontrolled temperature decrease to below 350 degrees Fahrenheit.)

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 no decay heat, the plant cooled down 213°F in one-half hour to 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 was reestablished prior to shutdown.

The transient data for this event are provided in Figure 1.

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.

Transient data for this event are provided in Figure 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 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 3.

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 4.

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 5.

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 6.

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.

Simple 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 of the HPI with the minimum vent valve flow (1 vent valve) in the downcomer, using the TRAC code and Creare data for thermal mixing. The mean mixed value for T_f is approximately 250°F based on an approximate time span of 20 minutes prior to the operator regaining control of the transient.

8. Prairie Island 2, SGTR (10/02/79)

This event was similar to the Ginna SGTR; however, the minimum temperature was 350°F for a cooldown period of approximately 20 minutes. No plots of temperature and pressure data were available.

9. Summary of Operating Experience

In addition to the eight events described above, 24 other events which could have led to PTS concern have been identified. 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 were examined. Other than the eight events discussed above, however, all of the remaining events maintained temperature above approximately 400°F and therefore are not discussed further.

H.B. ROBINSON SLB 04/28/70

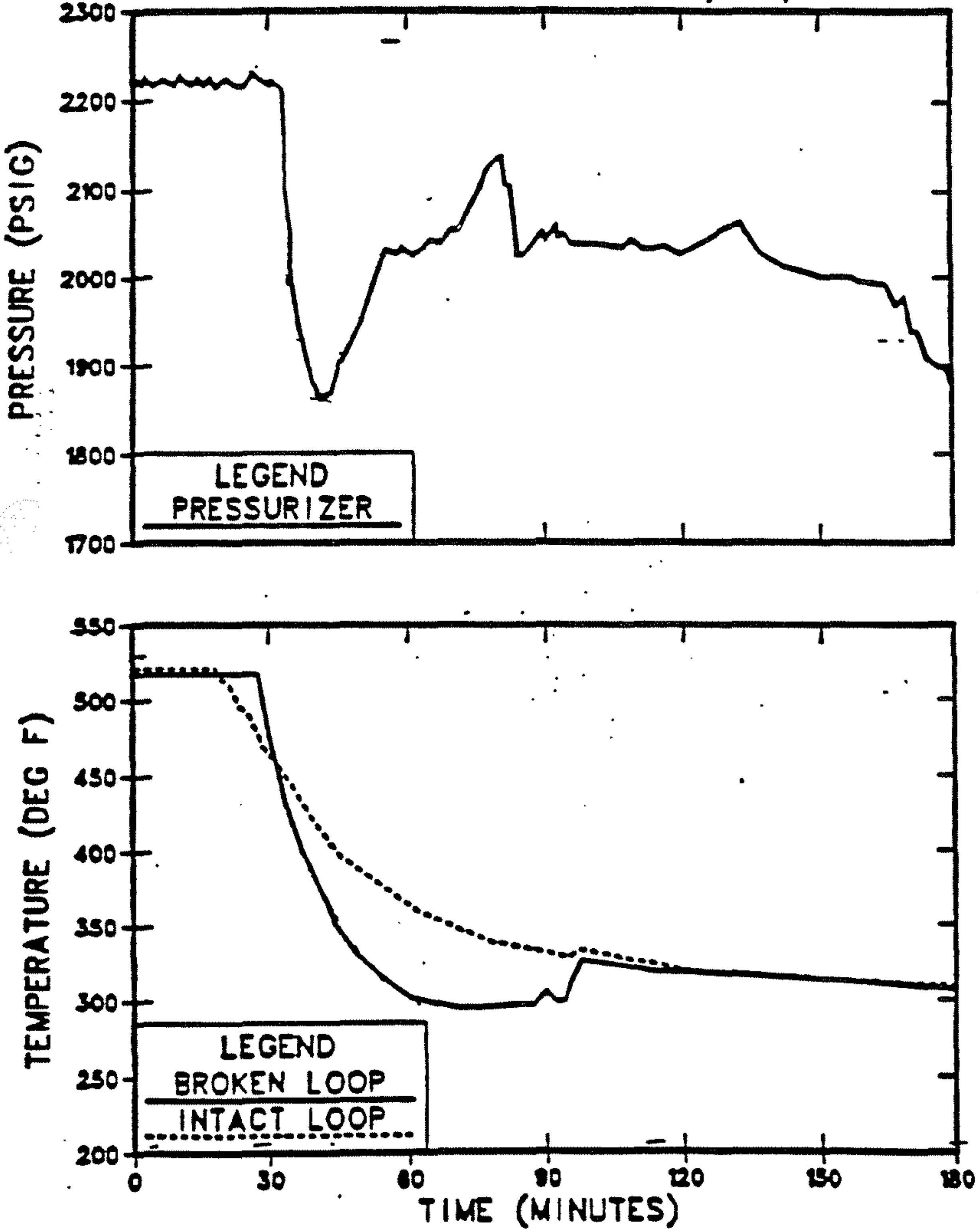


FIGURE 1

~~note 17 and 18 deleted~~

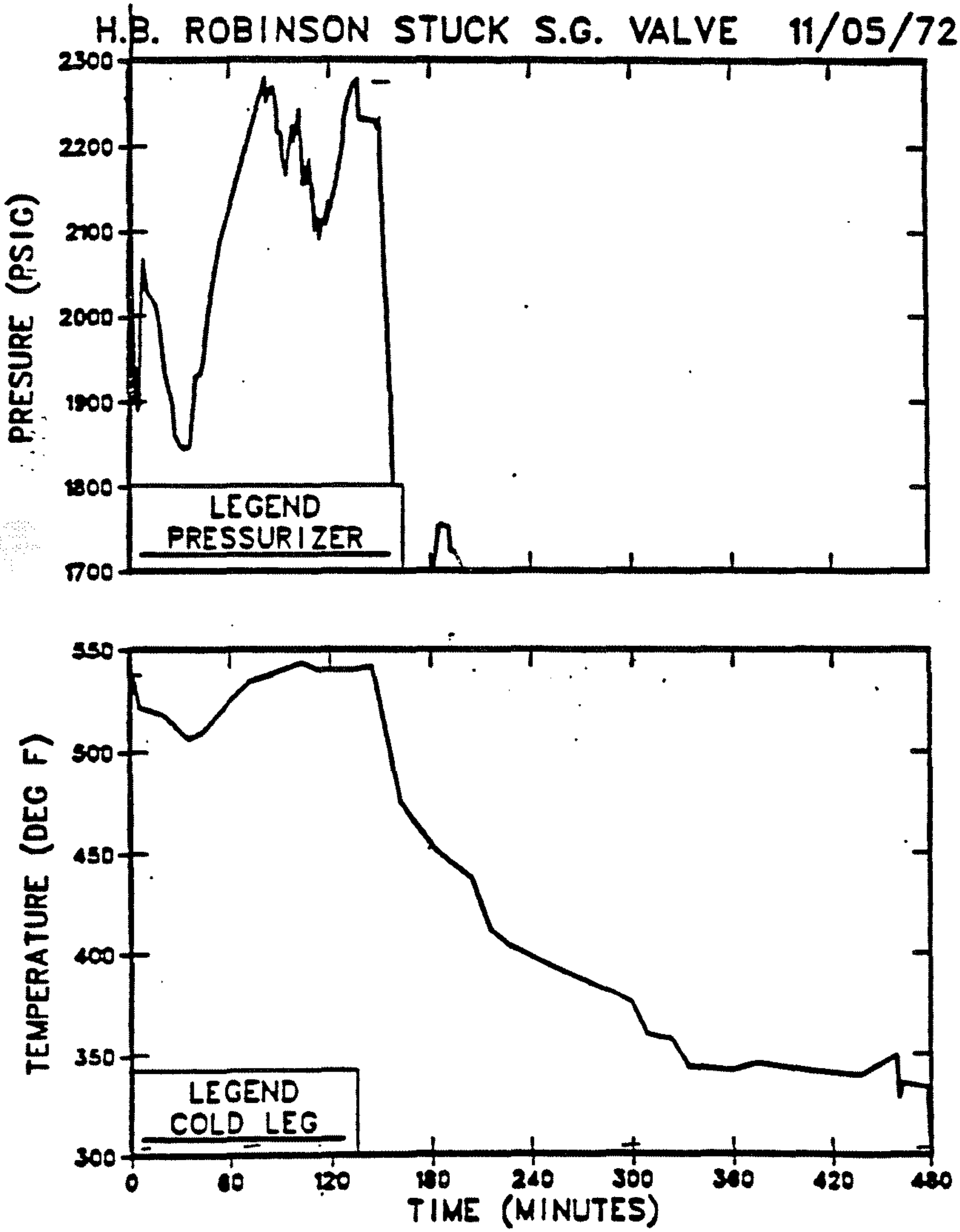


FIGURE 2

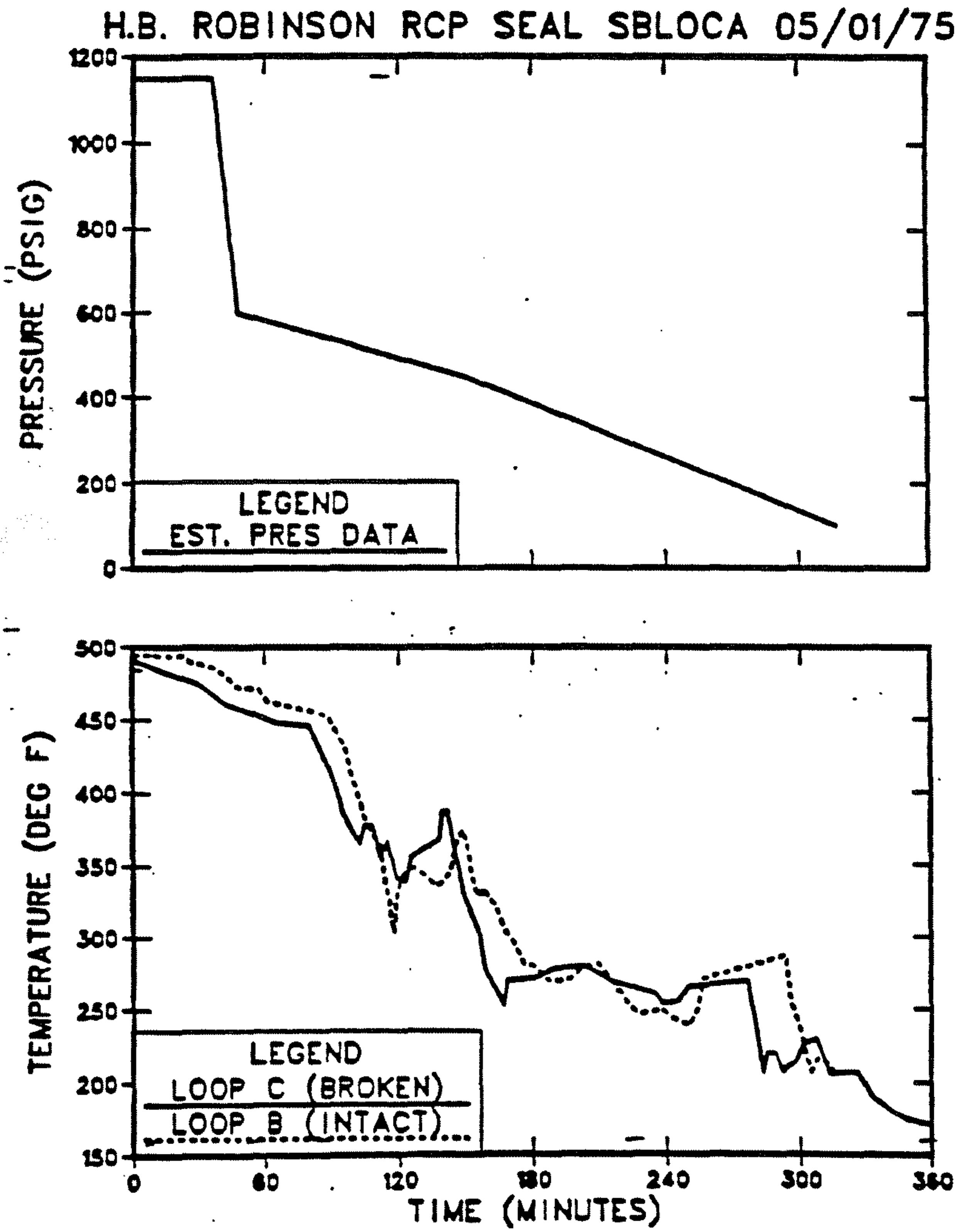

H.B. ROBINSON RCP SEAL SBLOCA 05/01/75
PRESSURE (PSIG)
1200
1000
800
600
400
200
0
LEGEND
EST. PRES DATA
TEMPERATURE (DEG F)
500
450
400
350
300
250
200
150
LEGEND
LOOP C (BROKEN)
LOOP B (INTACT)
0
60
120
180
240
300
360
TIME (MINUTES)

FIGURE 3

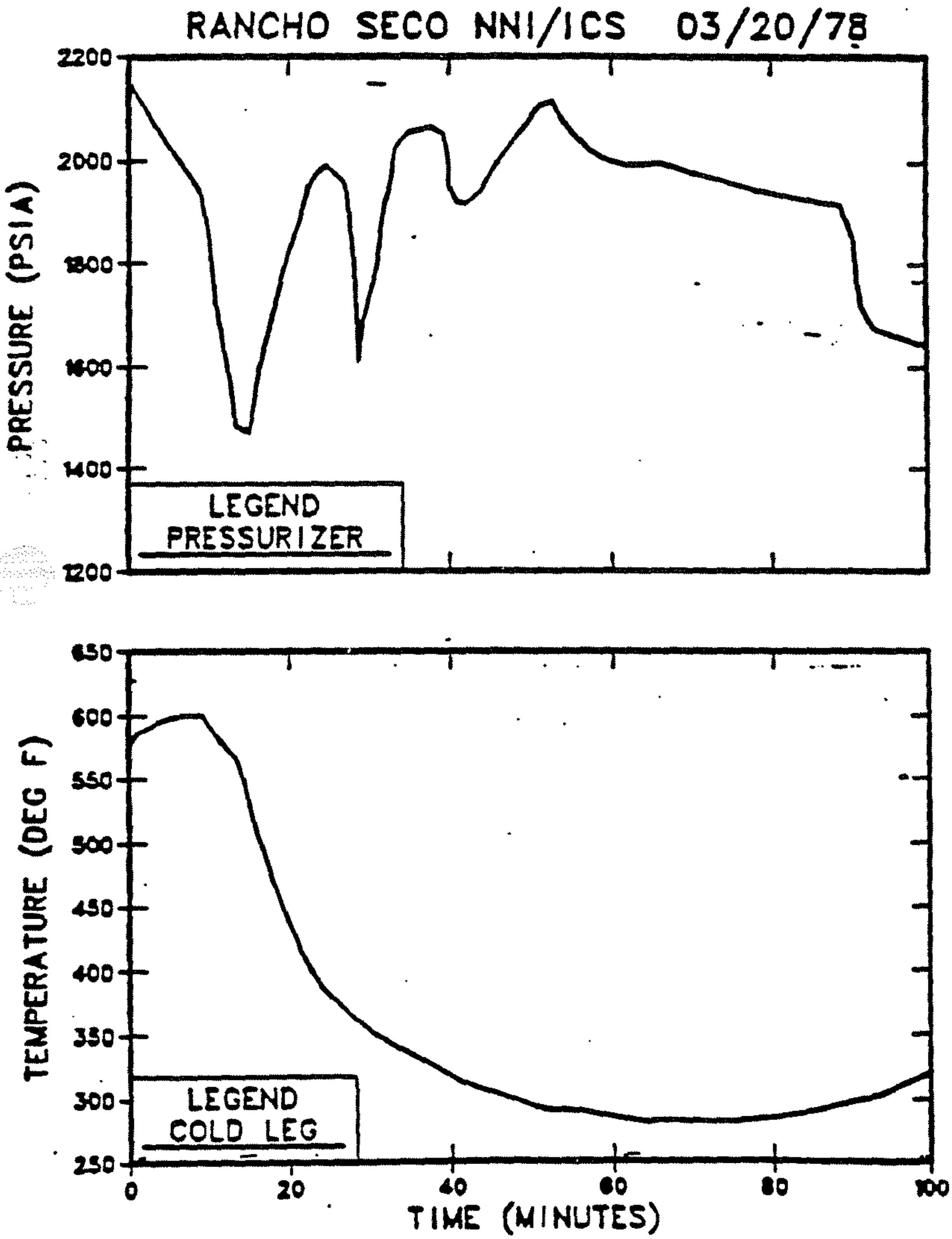

RANCHO SECO NNI/ICS 03/20/78
PRESSURE (PSIA)
2200
2000
1800
1600
1400
1200
LEGEND
PRESSURIZER
TEMPERATURE (DEG F)
650
600
550
500
450
400
350
300
250
LEGEND
COLD LEG
0
20
40
60
80
100
TIME (MINUTES)

FIGURE 4

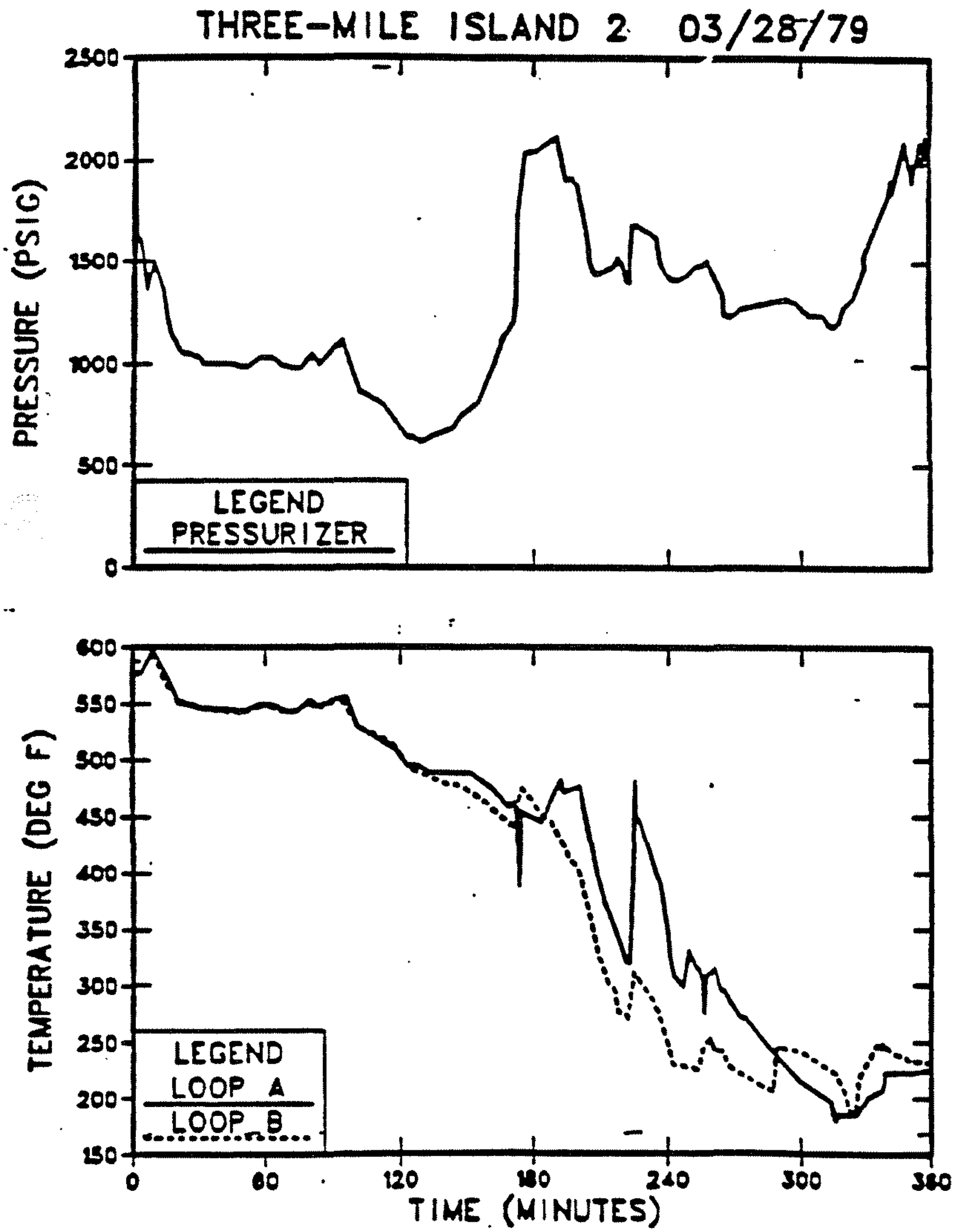
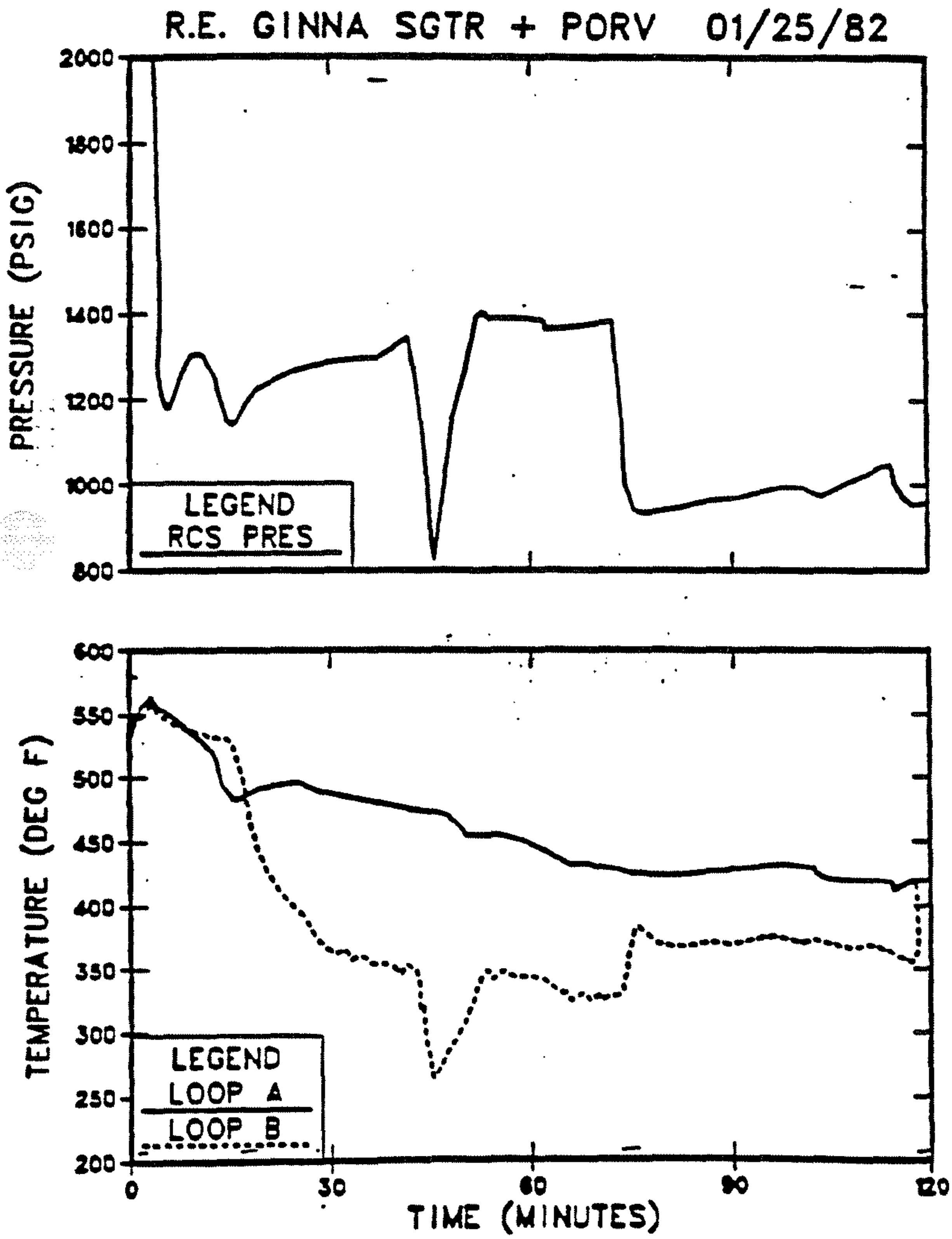

THREE-MILE ISLAND 2 03/28/79
PRESSURE (PSIG)
2500
2000
1500
1000
500
0
LEGEND
PRESSURIZER
TEMPERATURE (DEG F)
600
550
500
450
400
350
300
250
200
150
LEGEND
LOOP A
LOOP B
0
60
120
180
240
300
360
TIME (MINUTES)

FIGURE 5



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

FIGURE 6

RESPONSE RC 1(b)

QUESTION 1(b) FROM COMMISSIONER CURTISS' STAFF RESPONSE

Six of the events described above are estimated to have reached a final temperature below 325°F. An examination of each of these events relative to Yankee Rowe is given below. Note that conditional vessel failure probabilities are for the upper axial welds and assume a conservative copper content of 0.35%.

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

This event resulted from a 6-in schedule 80 pipe break and resulted in a plant cooldown to 320°F in one-half hour with primary system pressure generally being maintained above 2000 psig. The magnitude and rate of the cooldown was large because no fuel was loaded in the plant.

For Yankee Rowe a spectrum of steam line breaks was analyzed. The results of a large steam line break on one steam generator, which is larger than the specific 6-inch line break which occurred at H. B. Robinson, resulted in a cooldown to 307°F while primary system pressure was maintained at 1550 psig by the safety injection system. The conditional vessel failure probability for this case was calculated to be approximately 7×10^{-4}.

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

This event is not applicable as canned rotor pumps are used at Yankee Rowe.

Rancho Seco NNI/ICS (03/20/78)

This event resulted from a loss of power to the non-nuclear instrumentation (NNI) and its effect on the integrated control system. Heat removal by the once through steam generators resulted in an RCS cooldown from 582°F to 285F° in slightly more than one hour while primary system pressure was maintained around 2000 psig.

This event is not directly applicable to Yankee Rowe because the control systems are not integrated, and a power supply failure would not be propagated through the control system. However, the impact of control system failures on the primary system response was examined for Yankee Rowe. A power supply failure for the control system would result in a loss of feedwater transient, which is not a PTS event. The limiting transient would be excessive feedwater which would be automatically terminated by tripping the main feedwater pumps on receipt of a reactor trip signal. Continued feedwater addition for 20 minutes, which is well after the reactor trip signal would be reached, results in a conditional failure probability of less than 1×10^{-3}.

Three Mile Island 2 (03/28/79)

The PTS aspects of this event occur approximately three hours into the event after core damage had occurred, the PORV had been closed and when continued high pressure injection was used to recover core cooling and pressurize the primary coolant system. The many significant actions taken in response to this event, especially in the area of instrumentation, emergency procedures and operator training, makes this specific scenario very unlikely even for B&W plants.

Yankee Rowe specific features of significantly larger water inventory in the steam generators makes it very unlikely that the PORV would be challenged. In over 31 years of operation of Yankee Rowe the PORV has not been challenged. In addition, the PORV is small (0.8" diameter) leading to a slower event and the safety injection shutoff head is 1550 psi where as at TMI the primary system was pressurized to 2300 psig. Even if one postulates that the conditions reached at TMI are feasible for Yankee Rowe, conservative estimate of vessel failure likelihood would be approximately $3x10^{-2}$.

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

This event was a SGTR event which resulted in a primary system cooldown to around 325°F. The PORV stuck open when the operator used it to equalize the primary and faulted SG pressures per emergency procedures. The operator manually closed the block valve to stop the coolant loss.

This type of event, SGTR, is applicable to Yankee Rowe. Assuming a downcomer temperature of approximately 325°F the likelihood of vessel failure would be well below 1×10^{-4}.

Crystal River 3 NNI/ICS (02/26/80)

This transient was the result of a failure in NNI which resulted in an opening of the PORV. The high pressure injection system was actuated and resulted in the system being repressurized to about 2400 psig. Ultimately, the PORV block valve was closed and the operator proceeded to cooldown the plant to cold shutdown conditions.

As discussed above, the Yankee Rowe control systems are substantially different than the NNI on B&W designed plants. However, the potential PTS aspects of this transient are bounded by the Three Mile Island 2 event above.

Summary

As noted from the discussion of the historical events, such an assessment must of necessity be a plant specific review. Adjusting these events to recognize the Yankee Rowe specific design features, it is highly unlikely that these events would result in vessel failure.

In addition, it should be noted that Yankee Rowe has not experienced an over-cooling transient in its 31 year operating history. In fact, since 1982 there has been no operating reactor events which resulted in cooldown transients which resulted in PTS concern. The staff's safety assessment accounts fully for the plant specific details of the Yankee Rowe plant.

RESPONSE RC2

QUESTION 2 FROM COMMISSIONER CURTISS' STAFF RESPONSE

The general concept of maintaining safety and controlling overall risk by limiting the time exposure to a higher interim level of risk is widely accepted and applied. This concept is the underlying principle of Technical Specification Allowable Outage Times as well as the expressed basis for allowing plant operation while generic issues are being resolved (i.e., the definition of an Unresolved Safety Issue presumes that some changes will be required during the life of the plant). The delay allowed for implementing generic requirements (or commitments) also reflects this principle. The Yankee Rowe decision to permit operation for this cycle assured that the reactor vessel irradiation and the associated RT_{NDT} would not significantly change before a more complete understanding of the issue was developed.

The probability range of 1E-5 to 1E-4 conservatively estimated by the staff for Yankee Rowe is the probability of vessel failure, due to a thermal shock event. Although not every PTS failure event would lead to core damage, the staff has not attempted to estimate the conditional core damage probability because of the difficulty in performing such an estimate. It will be assumed here that the core damage frequency is the same as the vessel failure frequency.

In recent years the Commission has made decisions in several instances to permit interim operation of plants with core damage frequencies in the range of 1E-5 to 1E-4. In a July 11, 1989 memorandum to Victor Stello from Samuel Chilk the staff was directed to initiate plant-specific backfit analyses for each BWR plant with a Mark I containment to evaluate the efficacy of requiring the installation of hardened vents. The directive further indicated that where the requirement was justified by the backfit analyses, installation should be required within three years.

This directive was the response to staff recommendations presented in SECY-89-017. In SECY-89-017 the regulatory analysis was based upon a core melt frequency in the range of 1E-4 to 2E-5 (see page 8 of Attachment 4 to SECY-89-017).

Two other examples are the implementation of the Station Blackout Rule (SBO) (10 CFR 50.63), and the Anticipated Transients Without Scram (ATWS) Rule (10 CFR 50.62). The core damage frequency for SBO sequences was estimated by the staff in NUREG-1109 "Regulatory/Backfit analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout" to be in the range of 1E-5 to 1E-4. The same range was estimated for ATWS. In the case of SBO the rule allows two years for installation of equipment needed to meet the rule requirements. Likewise, the rule allowed two refueling outages for installation of equipment to meet the ATWS Rule.

Another example is the Indian Point case. In an extensive proceeding, the Commission held that the risks posed by the reactors were not "unacceptably high" (CLI-85-6(1985), 21 NRC 1043 at 1057), even though the core damage frequency was estimated to be about 3E-4.

We have not attempted to list all instances where the Commission has permitted operation with core damage frequencies in the range estimated for Yankee Rowe. However, it can be seen from the above examples that many reactor years of operation have been found acceptable where the estimated level of risk is similar to the Yankee Rowe case.

RESPONSE TO TECHNICAL ISSUES RAISED IN DR. PRYOR N. RANDALL'S JULY 15, 1991 LETTER TO THE CHAIRMAN

Dr. Randall's letter summarizes his reasons for believing that the Yankee Rowe plant should not be allowed to operate. He concludes that the NRC is "simply gambling" that a PTS event will not occur, or if it occurs that a critical flaw does not exist in the beltline region of the reactor vessel. The basis for his conclusion is given in several major comments in the letter which are summarized and responded to below.

Dr. Randall's first comment raises concerns about the staff's evaluation of the condition of the vessel. Specifically, he objects to the staff's phrase "may be above the screening criteria (due to uncertainties)." Dr. Randall goes on to say, "It sounds as though the staff is afraid to disagree with YAEC, whose hypothesis that coarse grain size negates nickel effects and irradiation temperature effects was thoroughly rebutted by Professor G. R. Odette, a consultant to RES, at the September 5 ACRS meeting."

The staff agrees with Professor Odette that the case for the coarse grain effect has not been proven. It is for this reason that the staff insisted that the reference temperature be adjusted to account for the lower irradiation temperature and higher nickel content in the plate material. However, the licensee continues to believe, based on advice from noted experts, that the coarse grain does negate the effects of lower temperature and nickel. We believe the coarse grain effect remains unproven and should not be used until experimental data can be obtained to resolve this issue. Thus, at this time, the staff can only say the screening criteria for the plate material may have been exceeded.

The concerns on the uncertainty of the vessel material properties and the possibility that the Yankee Rowe vessel exceeds the screening criteria led the staff to require the licensee to evaluate its plant specific PTS risk. The staff in its evaluation of the Yankee Rowe vessel specifically utilized conservative properties for the welds and plates and neglected the beneficial effects of course grain size proposed by the licensee.

Dr. Randall further states there have been six events in other reactors whose transient temperatures, if applied to the Yankee Rowe vessel and assuming a preexisting crack one half inch deep by about two inches long, would result in failure of the Yankee Rowe vessel. First, the staff judges the probability of a preexisting flaw of this size and at a location critical to PTS to be small. Further, the application of these events directly to the Yankee Rowe vessel is not appropriate. As noted in the staff's safety assessment of August 1990, there are significant differences between the Yankee Rowe plant and other PWRs which make the Yankee Rowe plant less likely to experience a PTS event. Additionally, these events were considered when performing the safety assessment for the Yankee Rowe vessel.

Each of the six historical events versus the Yankee Rowe is discussed below. Note that the conditional vessel failure probabilities given in all cases constructively assume a copper content of 0.35%.

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

This event resulted from a 6-in schedule 80 pipe break and resulted in a plant cooldown to 320°F in one-half hour with primary system pressure generally being maintained above 2000 psig. The magnitude and rate of the cooldown was large because no fuel was loaded in the plant.

For Yankee Rowe a spectrum of steam line breaks were analyzed. The results of a large steam line break on one steam generator, which is larger than the specific 6-inch line break which occurred at H. B. Robinson, resulted in a cooldown to 307°F while primary system pressure was maintained at 1550 psig by the safety injection system. The conditional vessel failure probability for this case was calculated to be approximately 7×10^{-4}.

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

This event is not applicable as canned rotor pumps are used at Yankee Rowe.

Rancho Seco NNI/ICS (03/20/78)

This event resulted from a loss of power to the non-nuclear instrumentation (NNI) and its effect of the integrated control system. Heat removal by the once through steam generators resulted in an RCS cooldown from 582°F to 285F° in slightly more than one hour while primary system pressure was maintained around 2000 psig.

This event is not directly applicable to Yankee Rowe because the control systems are not integrated, and a power supply failure would not be propagated through the control system. However, the impact of control system failures on the primary system response was examined for Yankee Rowe. A power supply failure for the control system would result in a loss of feedwater transient, which is not a PTS event. The limiting transient would be excessive feedwater which would be automatically terminated by tripping the main feedwater pumps on receipt of a reactor trip signal. Continued feedwater addition for 20 minutes, which is well after the reactor trip signal would be reached, results in a conditional failure probability of less than 1×10^{-3}.

Three Mile Island 2 (03/28/79)

The PTS aspects of this event occur approximately three hours into the event after core damage had occurred, the PORV had been closed and when continued high pressure injection was used to recover core cooling and pressurize the primary coolant system. The many significant actions taken in response to this event, especially in the area of instrumentation, emergency procedures and operator training, makes this specific scenario very unlikely even for B&W plants.

Yankee Rowe specific features of significantly larger water inventory in the steam generators makes it very unlikely that the PORV would be challenged. In over 31 years of operation of Yankee Rowe the PORV has not been challenged. In addition, the PORV is small (0.8" diameter) leading to a slower event and the safety injection shutoff head is 1550 psi where as at TMI the primary system was pressurized to 2300 psig. Even if one postulates that the conditions reached at TMI are feasible for Yankee Rowe, conservative estimate of vessel failure likelihood would be approximately $3x10^{-2}$.

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

This event was a SGTR event which resulted in a primary system cooldown to around 325°F. The PORV stuck open when the operator used it to equalize the primary and faulted SG pressures per emergency procedures. The operator manually closed the block valve to stop the coolant loss.

This type of event, SGTR, is applicable to Yankee Rowe. Assuming a downcomer temperature of approximately 325°F the likelihood of vessel failure would be well below 1×10^{-4}.

Crystal River 3 NNI/ICS (02/26/80)

This transient was the result of a failure in NNI which resulted in an opening of the PORV. The high pressure injection system was actuated and resulted in the system being repressurized to about 2400 psig. Ultimately, the PORV block valve was closed and the operator proceeded to cooldown the plant to cold shutdown conditions.

As discussed above, the Yankee Rowe control systems are substantially different than the NNI on B&W designed plants. However, the potential PTS aspects of this transient are bounded by the Three Mile Island 2 event.

As noted from the discussion of the historical events, such an assessment must of necessity be a plant specific review. Adjusting these events to recognize the Yankee Rowe specific design features, it is highly unlikely that these events would result in vessel failure.

In addition, it should be noted that Yankee Rowe has not experienced an overcooling transient in its 31 year operating history. In fact, since 1982 there has been no operating reactor events which resulted in cooldown transients which resulted in PTS concern. The staff's safety assessment accounts fully for the plant specific details of the Yankee Rowe plant.

Dr. Randall also notes that the probability of a significant PTS transient at Yankee Rowe becomes the dominant factor in the PRA since the probability of fracture is high. The staff does not agree with this characterization. The staff has performed a conservative assessment of both the probability of a significant PTS and the conditional probability of vessel failure given the PTS event. This assessment concluded that the limited PTS event was a small break LOCA which results in stagnated flow in the primary coolant system.

Given this event, whose probability was estimated to be about 1 in a 1000, the primary system response result in a cooldown to approximately 150°F in 100 minutes while system pressure was maintained at approximately 700 psig. This transient response was used to evaluate the conditional failure probability for the reactor vessel. Using the OCA-P code, probabilistic fracture mechanics analyses were conducted, which address uncertainties in material properties as well as flaw size, orientation and distribution. The probabilistic fracture mechanics analyses resulted in an estimated conditional failure probability for the limiting Yankee Rowe PTS event of around 3 in 100. Because of uncertainties in the flaw characteristics and the material properties, the staff believed it prudent to assume a conditional probability of vessel failure in the range of 1 in 10 to 1 in 100. Since this estimate was based on conservative assumptions, the staff believes this range of conditional failure probability is judged to be conservative.

We agree that continued operation over this cycle will not significantly impact the conditional failure probability given a PTS challenge. However, since operation did not pose an undue risk, continued operation was deemed appropriate while the licensee developed additional information of the effect of irradiation embrittlement with course grain size and developed techniques for vessel inspection.

Dr. Randall states that the staff has failed to follow some of its own regulatory guides; specifically Regulatory Guide 1.154 which gives a criterion of a thru-wall crack penetration mean frequency of less than 5 in a million reactor years. The guidelines of Regulatory Guide 1.154 were not used since mean values were not available and conservative assumptions were made in reaching the decision to permit operation for a limited time as discussed above.

In summary, the staff was fully cognizant of the issues raised by Dr. Randall when it performed its safety assessment in August 1990. Even if the Yankee Rowe vessel exceeds the screening criteria, it does not follow that operation may not be authorized by the staff in accordance with applicable regulations. The licensee performed, and the staff considered in its safety assessment, a plant specific PTS analysis as required by 10 CFR 50.61(b)(4).

ENCLOSURE 5

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July 23, 1991
BYR 91-093

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Attention: Mr. Patrick Sears
Senior Project Manager
Division Of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Reference: (a) License No. DPR-3 (Docket No. 50-29)

Subject: Responses to Requests for Additional Information

Dear Mr. Sears:

In a public meeting held at the Rowe, Massachusetts Elementary School on July 22, 1991 the NRC requested the Yankee Atomic Electric Company respond to questions regarding the reactor pressure vessel inspection program schedule and potential system changes or operator actions to further reduce the potential for pressurized thermal shock events. The questions and Yankee responses are provided below.

Question 1 What can be done to expedite the inspection and sampling schedule?

Answer We have contacted the two vendors performing inspection and sampling of the reactor vessel to determine if additional actions can be taken to shorten the schedule without compromising the safety or quality of the work. The inspection was originally requested to be accomplished in 1993. It was accelerated by Yankee by over a year in order to accomplish it during 1992. The sampling program was required prior to re-start from the 1992 outage. Both tasks require considerable development, testing in the laboratory and mock-up testing in special facilities to be sure they can be accomplished and done safely. The vendors have been unable to identify anything which will materially shorten this aggressive schedule.

Yankee and our vendors will continue to make our best efforts to shorten the schedule; but, given the developmental nature of the inspection and sampling programs, our judgement indicates that our present schedule will be difficult to meet. It will not benefit the public, the NRC or Yankee to rush this program through without adequate testing and find that the data obtained is insufficient to answer the questions posed.

Question 2 What operator actions or system changes can be accomplished to reduce the potential for pressurized thermal shock events?

Answer We are actively pursuing two options for further reducing the potential for pressurized thermal shock events. These are: 1) allowing two of the four the main coolant pumps to continue operating following certain small break loss of coolant events to enhance mixing in the reactor vessel and thereby keep the coolant temperature at the vessel wall higher and 2) increasing the injection water temperature from 120°F to approximately 170°F.

The first option offers the potential for faster implementation and is the preferred option if additional safety analysis demonstrates that it can be done safely. Our initial analyses indicate a substantial benefit from this option. However, we are now performing a complete safety assessment of the proposed action to be sure it is safe for events other than pressurized thermal shock. By August 26, 1991 we will have completed our safety reanalysis and provide the results to the NRC. Should the results be favorable and safety is assured, we will implement as soon as practical the necessary procedure changes and operator training to accomplish this option.

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Although increasing injection water temperature also shows positive benefit, it is a longer term option due to the need to re-perform loss of coolant safety analyses and make necessary design changes, including the procurement of long lead time equipment, should the safety analyses show favorable results.

Thus, our efforts are aimed at Option 1 since it is quicker to implement and may obviate the need to raise the injection water temperature.

If you should have any questions concerning these responses please notify me.

Sincerely,

Jay K. Thayer

Jay K. Thayer
Vice President and
Manager of Operations

JKT/ram