

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 *[Signature]*

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

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X-27895

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SEP 24 1987

Thomas E. Murley

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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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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 Linde 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.

Recommendations

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

<u>PWR Plant</u>	<u>End of License</u>	<u>Charpy USE at End of License (ft-lb)</u>	<u>Charpy USE on Jan. 1, 1986 (ft-lb)</u>
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
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 31, 1983

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.

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May 1, 1988

Mr. W. F. Conway

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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-1/II
Office of Nuclear Reactor Regulation

Enclosure: As stated

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Florida Power and Light Company

Turkey Point Plant

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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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/cm² (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

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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/cm² ($E > 1$ MeV). This was calculated to be the neutron fluence at the tip of the postulated $\frac{1}{4}$ T depth flaw when the neutron fluence at the inside surface is 2.88×10^{19} n/cm² ($E > 1$ MeV) and the TP reactor vessels reach 40 EFPY. The attenuation of neutron fluence from the inside surface to $\frac{1}{4}$ 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 $\frac{1}{4} T$ 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 $\frac{1}{4} T$ 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