VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23209

December 15, 1978

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. A. Schwencer, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Serial No. 702
LQA/ESG:esh

Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Dear Mr. Denton:

AMENDMENT TO OPERATING LICENSES
SURRY POWER STATION UNIT NOS. 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGE NO. 74

Pursuant to 10 CFR 50.90, the Virginia Electric and Power Company hereby requests an amendment, in the form of changes to the Technical Specifications, to Operating Licenses DPR-32 and DPR-37 for the Surry Power Station, Unit Nos. 1 and 2. The proposed changes are enclosed and have been designated as Change No. 74.

By our letter Serial No. 081, dated February 15, 1978, we informed the Staff that our NSSS Supplier, Westinghouse Electric Corporation, had confirmed that the material in the reactor vessel surveillance capsules was not spawned from the material contained in the vessels themselves. Consequently, the use of material fracture toughness data from these capsules is not considered valid in evaluating radiation damage to the vessels, and the only appropriate means of determining this damage is by measurement of flux or fluence. It is also evident that present Technical Specification Figure 3.1-1, specifying heatup and cooldown limitations, incorporated a shift of 280°F in 1975, based on invalid surveillance specimen data, and is consequently extremely conservative.

As discussed in Attachment 1, it has been demonstrated that the applicability of present TS Figure 3.1-1 can be extended to 32 EFPY. The revised figure is included in Attachment 2. Also included are revisions to Specification 3.1B which provide new procedures for updating Figure 3.1-1, if necessary, based on future fluence calculations on surveillance samples. Table 4.2-1 (Item 7.1) has been revised to specify dosimetry as the examination method for reactor vessel irradiation surveillance.

The enclosed changes have been reviewed and approved by the Station Nuclear Safety and Operating Committee and the System Nuclear Safety and Operating Committee. It has been determined that this request does not involve an unreviewed safety question.

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We have evaluated this request in accordance with the criteria specified in 10 CFR 170.22. Since it has been demonstrated that the present limitations are based on extremely conservative data, and the specification provides for updating the limitations should future data not be bounded by the present curves, the staff should be able to determine that this request does not involve a significant hazards consideration. Accordingly, this request has been determined to be Class III for Unit 1. The duplicate revision for Unit 2 has been designated Class I. A check in the amount of $4,400.00 is attached in payment of the amendment fees.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President-Power Supply
and Production Operations

Attachments:
1. Safety Evaluation and Description of Change
2. Change No. 74
3. Check in amount of $4,400.00

cc: Mr. James P. O'Reilly, Director
Office of Inspection and Enforcement
Region II

COMMONWEALTH OF VIRGINIA)
) S. S.
CITY OF RICHMOND)

Before me, a Notary Public, in and for the City and Commonwealth aforesaid, today personally appeared C. M. Stallings, who being duly sworn, made oath and said (1) that he is Vice President-Power Supply and Production Operations, of the Virginia Electric and Power Company, (2) that he is duly authorized to execute and file the foregoing Amendment in behalf of that Company, and (3) that the statements in the Amendment are true to the best of his knowledge and belief.

Given under my hand and notarial seal this 15th day of December, 1978.

My Commission expires January 22, 1981.

Robert M. Neil
Notary Public

(SEAL)

ATTACHMENT 1

SAFETY EVALUATION

Proposed Changes to Technical Specification 3.1B

The proposed change to Technical Specification 3.1B is based on the desire to avoid further revisions to the document's text and figures. Analysis of the irradiation samples removed from the Unit One vessel during the first refueling revealed charpy values inconsistent with predictions. As per our letter to Mr. Robert W. Reid, NRC Chief Operating Reactors Branch 4, of February 15, 1978, Serial No. 081, it was confirmed by our NSSS supplier, Westinghouse, that the material contained in the surveillance capsules was not spawned from the material contained in our reactor vessel.

The basic consequence of this new information is that the data regarding material fracture toughness obtained from surveillance capsules are not applicable to the reactor vessel from which it was removed. Therefore, the use of charpy data obtained from surveillance capsule samples is not considered valid in evaluating radiation damage to the reactor vessel in regard to meeting the requirements of Appendix G of 10CFR50 for continued unit operation. It is now evident that the present heatup and cooldown curve, T.S. Figure 3.1-1, incorporated a shift of 280°F in 1975 based on invalid surveillance specimen data and consequently is overly conservative. Since the station has used, and is familiar with, the current heatup and cooldown curve without operational difficulties, the validity of T.S. Figure 3.1-1 will be demonstrated to 32 EFPY rather than liberalizing the heatup and cooldown curve based on a revised T.S. Figure 3.1-1.

Presently, due to our inability to conform to 10CFR50 Appendix G, as stated above, the only applicable means of determining radiation damage to the reactor vessel, based on the surveillance capsule data, is by flux or fluence. The fluence data obtained from the initial specimens from each unit coincides with the Westinghouse prediction of fluence versus EFPY. The second and most recent sample was from Unit 1 and was within the accuracy of the analysis. The error in the

accuracy was associated with the uncertainty of the lead factor assigned by Westinghouse. The flux will be determined by dosimetry as the capsules continue to be extracted in accordance with the Technical Specifications. However, charpy V-notch testing will not be performed, on the surveillance capsule samples.

T.S. Figure 3.1-2 will be modified to omit Curve 1 since it is based on nonrepresentative material. The original ΔRT_{NDT} versus fluence T.S. Curves will be used. The individual curves will show base metal and weld metal copper content.

Due to the previous 280°F shift, the present T.S. Figures 3.1-1 is calculated to be valid beyond 32 EFPY. T.S. Figure 3.1-3 will be added to show the relationship between operating time and fluence at the 1/4 thickness of the vessel as described in the FSAR, Section 4. As the note on T.S. 3.1-3 infers, the slope of this curve was established from a 280°F shift for 32 EFPY and is more conservative than any existing data.

To prove the conservatism, we will look at three cases: (1) the slope at the 1/4 T for 32 EFPY, using the predicted data from Westinghouse, is $0.9375 \times 10^{18} \dfrac{n}{cm^2\text{-}EFPY}$; (2) the slope at the 1/4 T for 32 EFPY based on the recent surveillance data, and most conservative, is $1.201 \times 10^{18} \dfrac{n}{cm^2\text{-}EFPY}$; and (3) the slope at 1/4 T for a 280°F shift (current shift in Surry's heatup and cooldown curve) for 32 EFPY is $1.203 \times 10^{18} \dfrac{n}{cm^2\text{-}EFPY}$. Thus, Figure 3.1-3 is more conservative than all the applicable data.

When, according to Technical Specification, another surveillance sample or samples are withdrawn and the fluence calculated, the data point should be compared to the 1/4 T line of T.S. Figure 3.1-3. If this data point is above the line, a new line shall be constructed through the origin such that it is above all the applicable data points. Once Figure 3.1-3 is revised, T.S.

Figure 3.1-1 must be updated, either by a temperature shift, or by revising the applicable period (EFPY) to match the new transition temperature from T.S. Figure 3.1-2.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGE 74

B. HEATUP AND COOLDOWN

Specification

1. Unit 1 and Unit 2 reactor coolant temperature and pressure and the system heatup and cooldown (with the exception of the pressurizer) shall be limited in accordance with TS Figure 3.1-1.

   Heatup:

   Figure 3.1-1 may be used for heatup rates of up to 50°F/hr. below an indicated temperature of 440°F and 100°F/hr. above 440°F.

   Cooldown:

   Allowable combinations of pressure and temperature for specific cooldown rates are below and to the right of the limit line as shown in TS Figure 3.1-1. This rate shall not exceed 50°F/hr. for temperatures at or below an indicated temperature of 440°F. For temperatures above an indicated temperature 440°F, the rate shall not exceed 100°F/hr.

   Core Operation:

   During operation where the reactor core is in a critical condition, except for low level physics tests, vessel metal and fluid temperature shall be maintained above the reactor core criticality limits specified in Figure 3.1-1.

2. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

3. The pressurizer heatup and cooldown rates shall not exceed $100^{o}F/hr.$ and $200^{o}F/hr.$, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than $320^{o}F$.

4. TS Figure 3.1-1 shall be updated periodically in accordance with the following procedures, before the calculated maximum exposure of the vessel exceeds the exposure for which TS Figure 3.1-1 applies. The curve based on 0.25% Cu weld in TS Figure 3.1-2 shall be used to predict the increase in transition temperature based on integrated power.

   a. If measurements on the most recently examined irradiation specimen show that its data point is above the 1/4T (thickness) line of T.S. Figure 3.1-3 then a new line shall be constructed through the origin such that it is above all the applicable data points. Once T.S. Figure 3.1-3 is revised, T.S. Figure 3.1-1 must be updated, either by a temperature shift, as by T.S. 3.1.B.4c below, or by revising the applicable period (EFPY) to match the new transition temperature from TS Figure 3.1-2.

   b. At or before the end of the integrated power period for which TS Figure 3.1-1 applies, the limit lines on the figure shall be updated for a new integrated power period as follows. The total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated neutron exposure. The predicted increase in transition temperature at the end of the new period shall then be obtained from TS Figure 3.1-2.

c. The limit lines in TS Figure 3.1-1 shall be moved parallel to the temperature axis (horizontally) in the direction of increasing temperature a distance equivalent to the transition temperature increase obtained from TS Figure 3.1-2 less the increment used for the end of the present period.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Section 4.1 of the FSAR. During unit startup and shutdown, the rates of temperature and pressure are limited. The maximum plant heatup and cooldown rate of $100^{o}F/hr.$ is consistent with the design number of cycles and satisfies stress limits for cyclic operation.[2]

The allowable pressure vs. temperature is based on a temperature scale relative to the RT_{NDT}. The RT_{NDT} is basically the drop weight NDTT of the material, as determined by ASTM E208. However, to assure that this value is conservative, and to guard against the possibility that material with low upper shelf toughness, or with a low rate of increase of toughness with temperature, is not properly evaluated, Charpy tests may be performed. If 35 mils lateral

expansion or 50 ft-lbs is not obtained at NDTT + 60, the RT_{NDT} is shifted upward until these criteria are met.

This procedure of selecting RT_{NDT} assures that the K_{IR} curve used to calculate allowable pressures will be conservatively applicable to the material.

The procedure for determining the limiting RT_{NDT} for the Reactor System is as follows:

1. Determine the highest RT_{NDT} of the material in the core region of the reactor vessel, using original values and adding to this the predicted shift in RT_{NDT} due to radiation during the service period for which this RT_{NDT} applies. This takes into account the copper content of the material.
2. Examine the data for all other ferritic materials in the reactor system to assure that the RT_{NDT} so selected is the highest in the system. If drop weight data are not available for all materials, the RT_{NDT} of these shall be estimated in a conservative manner using trend data for the materials concerned.
3. For succeeding service periods, the same procedure as given in (1) above will be used unless test data from the surveillance program indicates that this will not be appropriate. In this event, the results of these tests will be used to predict the limiting RT_{NDT}.

Test results on material from the Surry Unit 1 reactor vessel is presented in FSAR Table 4.A-1. Using the above procedure, the highest original RT_{NDT} of the core region plates is $+20^{o}F$. No drop weight NDTT value is

available for the core region weld material but on the basis of actual drop weight data on many similar weld materials, plus the actual Charpy values on this material, the drop weight NDTT is estimated to be 0^oF.

The RT_{NDT} for the first two years of operation included a conservative estimate of the shift in RT_{NDT} caused by radiation of 100^oF. This added to the original RT_{NDT} of 0^oF assumed for the welds, gave a reference RT_{NDT} of 100^oF to be used for the first two years of operation, or until the radiation shift was estimated to be over 100^oF.

In examining the data for the rest of the material in the vessel; as well as the properties for the other ferritic components of the reactor system, it is certain that all other materials initially had RT_{NDT} values significantly lower than 100^oF.

Since the neutron spectra at the samples and vessel inside radius are identical, the measured $(RT)_{NDT}$ shift for a sample can be supplied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel is obtainable from the measured sample data by appropriate application of the calculated azimuthal neutron flux variation.

During cooldown and steady state, the thermal stress varies from tensile at the inner wall to compressive at the outer wall. The internal pressure superimposes a tensile stress on this thermal stress pattern, increasing the stress at the inside wall and relieving the stress at the outside wall. Therefore, the limiting stress always appears at the inside wall and the limit line has a

direct dependence on cooldown rate. For heatup, the thermal stress is reversed and the location of the limiting stress is a function of heatup rate. The 1/4T location is considered conservative since the enhanced metallurgical properties of the surface are not used for the determination of NDTT. The 1/4T location is used for cooldown and steady state and 3/4T location is used for heatup but the 1/4T location is the most restrictive so it will be the controlling curve. In addition, the limiting NDTT for the reactor vessel after operation is based on the NDTT shift due to irradiation. Since the fast neutron dose is highest at the inner surface, usage of the 1/4T NDTT criterion is conservative (FSAR Section 4). The 50°F/hr. heatup and cooldown line on TS Figure 3.1-1 bounds all limit lines for heatup and cooldown rates up to 50°F/hr. for indicated temperatures at or below 440°F, and 100°F/hr. above 440°F. TS Figure 3.1-1 is based on the Standard Review Plan as modified by measured irradiation sample temperature shifts and appropriate vessel attenuation factors and azimuthal neutron flux variations.

TS Figure 3.1-1 defines stress limitations only. For normal operation other inherent plant characteristics, e.g., pump parameter and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure ranges.

The heatup and cooldown rate of 100°F/hr. for the steam generator is consistent with the remainder of the Reactor Coolant System, as discussed in the first paragraph of the basis. The stresses are within acceptable limits for the anticipated usage.

Temperature requirements for the steam generator correspond with the measured NDT for the shell. The spray should not be used if the temperature difference between the pressurizer and spray fluid is greater than 320°F. This limit is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

References:

(1) FSAR, Section 4.1.5

(2) ASME Boiler & Pressure Vessel Code, Section III, N-415

(3) ASME Boiler & Pressure Vessel Code, Section III, proposed non-mandatory Appendix G2000

(4) 10 CFR 50, Appendix A, G, & H

(5) Regulatory Guide 1.99, Revision 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials"

(6) USNRC Standard Review Plan, Section 5.3.2, 11/29/75, "Pressure - Temperature Limits"

(7) Welding Research Council (WRC) Bulletin 175, "PVRC Recommendation on Toughness Requirements for Ferritic Materials"

(8) WCAP - 7924-A, "Basis for Heatup and Cooldown Limit Curves"

(9) Surry Reactor Vessel Radiation Surveillance Program WCAP 7723-Surry 1 (July, 1972), WCAP 8085-Surry 2 (June, 1973)

(10) Battelle Columbus Laboratories Research Reports for Surry Pressure Vessel Irradiation Capsule Program.

(a) Surry 1 examination and analysis of capsule T (June, 1975)
(b) Surry 2 examination and analysis of capsule X (Sept., 1975)

(11) ASTM: E185-73, E208, & E23

(12) Surry T.S. Change 27 (Proposed Change 35)

(13) Vepco letter to Mr. Robert W. Reid, NRC Chief Operating Reactors Branch 4, of February 15, 1978, Serial No. 081

TABLE 4.2-1

SECTION F. VALVE PRESSURE BOUNDARY (Continued)

Item No.	Category	Required Examination Areas	Required Examination Methods	Extent of Examination Planned During First 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
6.7	(Continued)					The support settings of constant and variable spring-type hangers, snubbers and shock absorbers would be inspected to verify proper distribution of design loads among the associated support components.

SECTION G. MISCELLANEOUS INSPECTIONS

Item No.	Category	Required Examination Areas	Required Examination Methods	Extent of Examination Planned During First 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
7.1		Materials Irradiation	Dosimetry	Capsule 1 shall be removed and examined at the first region replacement. Capsule 2 shall be examined after 5 years.	Capsules shall be removed and examined after 10 years	Capsule 4 shall be removed and examined after 20 years. Capsules 5-8 are extra capsules for complementary or duplicate testing.
7.2		Low Head SIS Piping Located in Valve Pit	Visual	(See Remarks)	Not Applicable	This pipe shall be visually inspected at each refueling shutdown.
7.3		Low Pressure Turbine Rotor	Visual and magnetic particle or dye penetrant	100% of blades	Not Applicable	

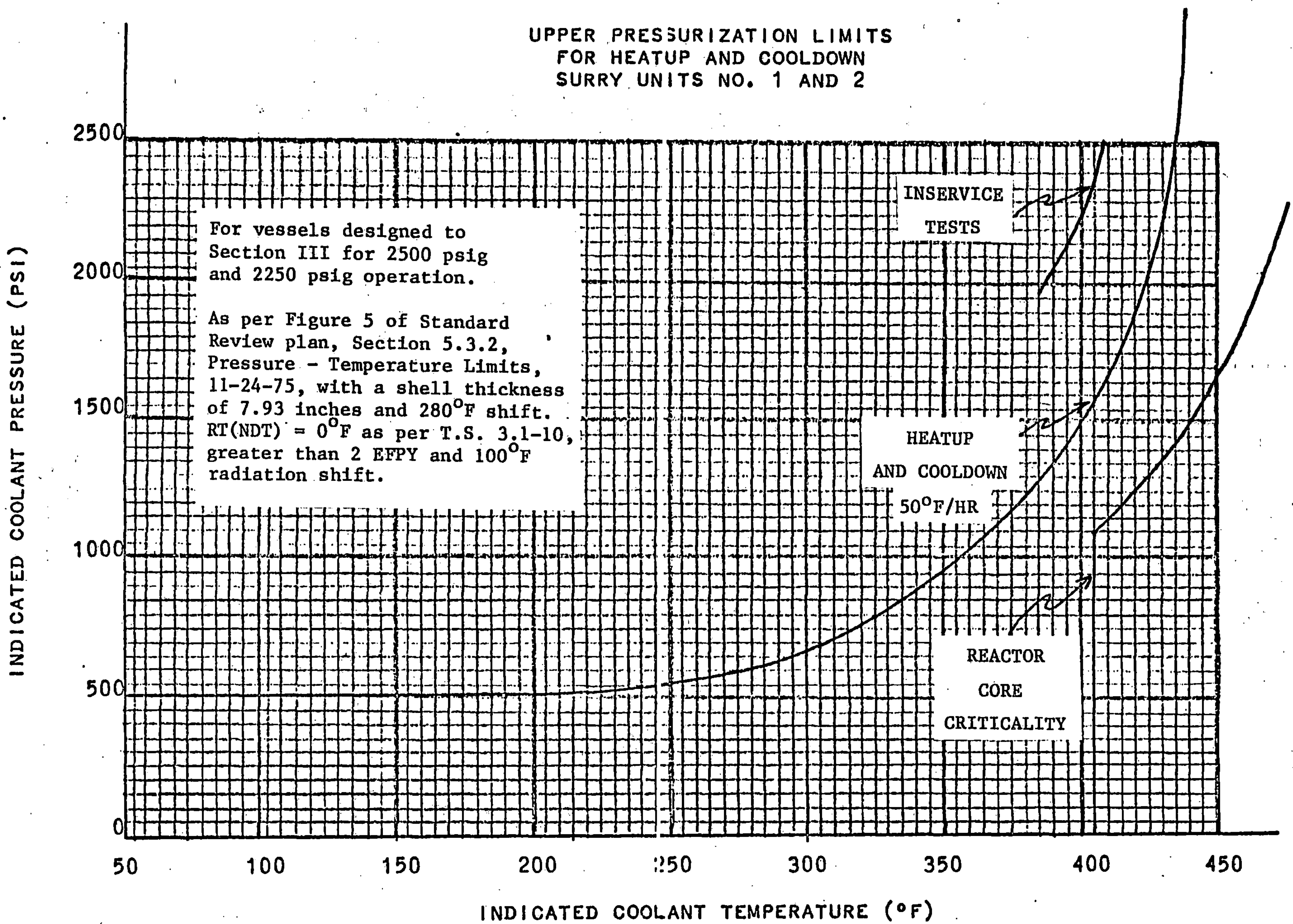
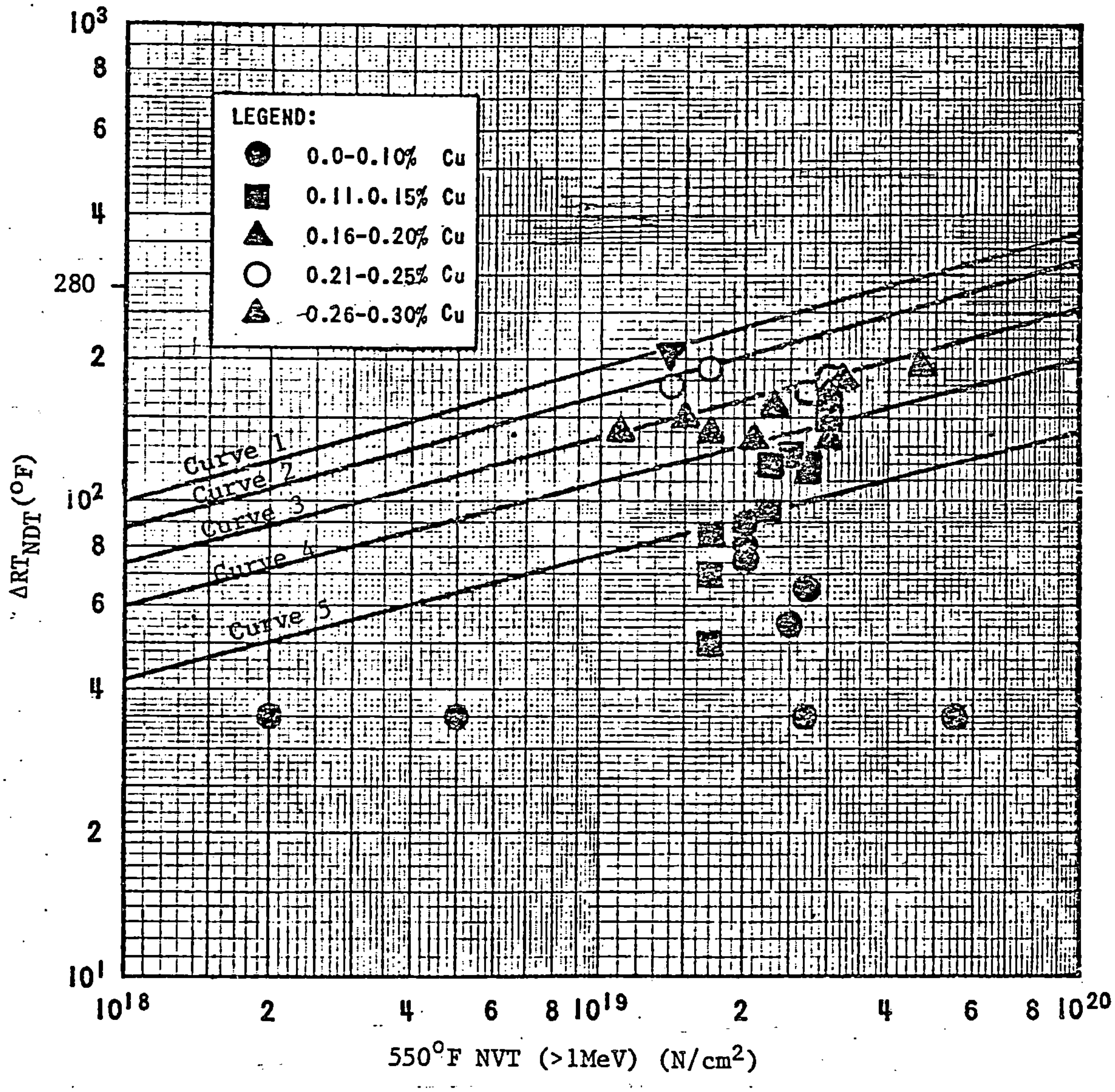


FIGURE VALID UP TO 32 EFPY.

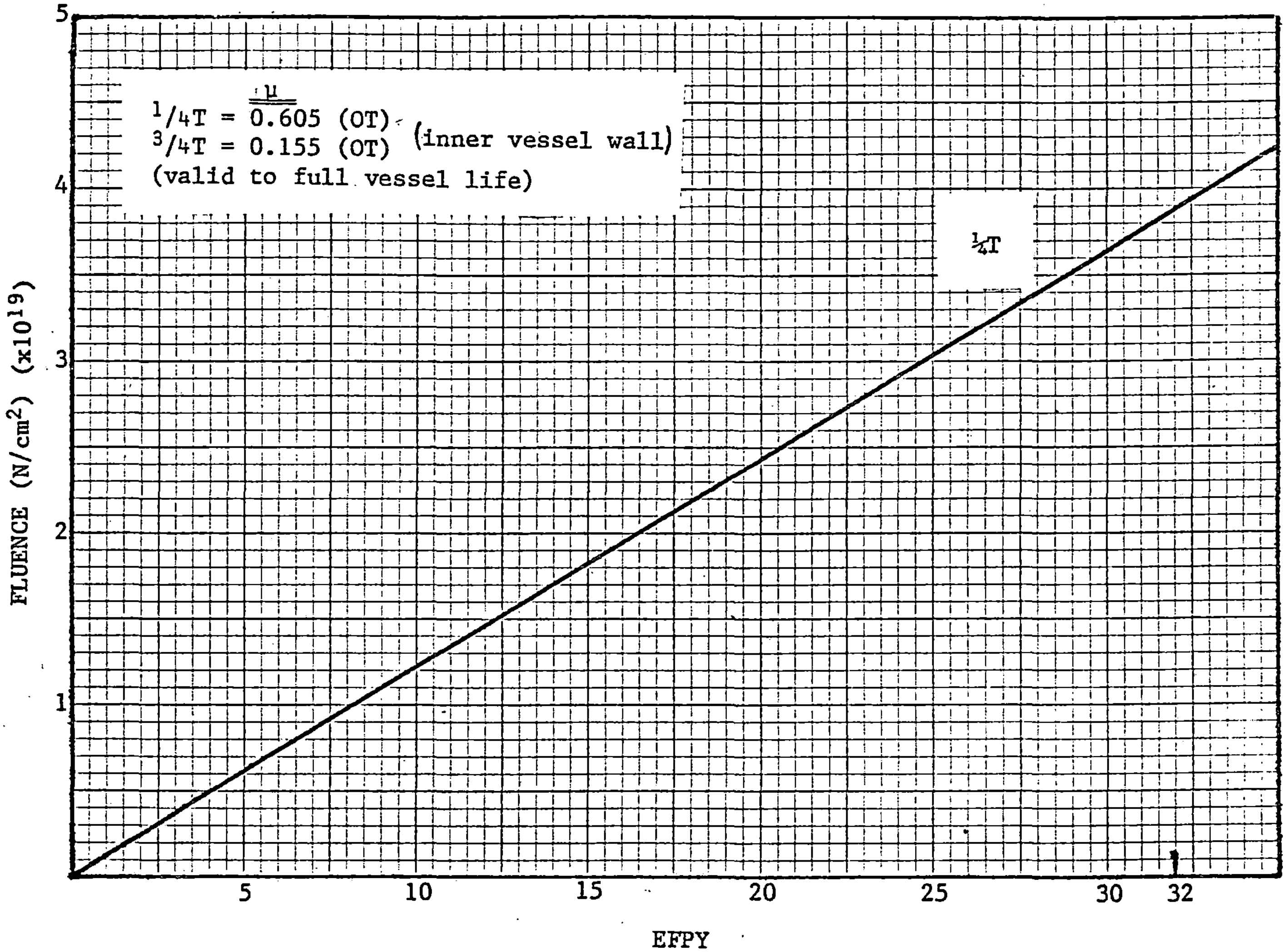
TS FIG 3.1-1



Curve 1 - 0.30% Cu base, 0.25% Cu weld
Curve 2 - 0.25% Cu base, 0.20% Cu weld
Curve 3 - 0.20% Cu base, 0.15% Cu weld
Curve 4 - 0.15% Cu base, 0.10% Cu weld
Curve 5 - 0.10% Cu base, 0.05% Cu weld

Figure 3.1-2. Radiation Induced Increase In Transition Temperature

SURRY 1&2



NOTE: Slope of line (Φ) above was determined from T.S. Figure 3.1-2 with a 0.25% Cu weld for 32 EFPY's and a 280°F shift as determined for T.S. Figure 3.1-1. It was calculated as follows:

$$\Phi = \frac{3.85\text{x}10^{19}\text{N/cm}^2}{32\text{ EFPY}} = 1.203\text{x}10^{18}\text{N/(cm}^2\text{-EFPY)}$$