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ENCLOSURE

Consists of review and responses to request for additional information concerning compliance with 10CFR50, Appendix G for the Duane Arnold Energy Center...

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FOR ACTION/INFORMATION

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Iowa Electric Light and Power Company

General Office
Cedar Rapids, Iowa

Lee Liu
VICE PRESIDENT – ENGINEERING

October 12, 1977
IE-77-1875

Mr. George Lear, Chief
Operating Reactors Branch 3
Division of Operating Reactors
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Lear:

Your letter of February 9, 1977, requested us to review and provide responses to your request for additional information concerning compliance with 10CFR50, Appendix G for the Duane Arnold Energy Center.

Included herewith are responses to your request for information. Revised Technical Specifications are not included. Analyses are continuing which will further refine the information to be considered in the Technical Specifications. These responses are being provided prior to the completion of these further analyses to aid in resolution of the questions.

We expect to receive the results of further analyses and formulate amended Technical Specifications prior to January 1, 1978.

Very truly yours,

Lee Liu
Vice President, Engineering

LL/KAM/ms
Encls.
cc: K. Meyer
D. Arnold
R. Clark (NRC)
R. Lowenstein
L. Root
File A-117
A-107
A-286

QUESTION 1

The change in transition temperature as a function of fluence shown in Figure 3.6.1 deviates from Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials". Provide justification for the deviations.

ANSWER

Introduction

The purpose of this study was to provide an evaluation of 10CFR50 Appendix G compliance of the reactor pressure vessel steel irradiation effect prediction methods, as applied to the Duane Arnold plant, for the purposes of justifying the GE fluence shift curves in lieu of Regulatory Guide 1.99 upper limit curve, and if necessary, provide a revised curve based on test data of samples with known copper contents.

The work performed consists of two parts. First, a review of fabrication procedures was performed to assure that abnormally high copper contents were not present in the reactor pressure vessel materials, thus assuring the conservative adequacy of the revised upper limit General Electric operating curve for prediction of Charpy Impact Transition temperature shifts as a function of neutron fluence.

Secondly, an evaluation of the existing General Electric transition temperature shift prediction curve based on the acquisition of a substantial amount of new data in the BWR operating fluence range was performed. This evaluation has resulted in the creation of revised upper limit prediction curve and, after final computer regression analysis of the total data bank now available, a new family of transition temperature shift prediction curves as a function of copper and phosphorus content similar to those presented by the NRC in Regulatory Guide 1.99 will be developed.

Vessel Fabrication Practice Review:

A review of reactor pressure vessel manufacture and fabrication practices has been made to characterize the general ranges of copper contents expected in BWR vessels and to identify those procedures which may have resulted in abnormally high copper contents in the vessel beltline region. In general, the copper contents resulting from standard vessel manufacturing processes are all within a well defined acceptable range.

Information characterizing the standard vessel-manufacture practices is presented in Table 1-1. This information, obtained from material for which General Electric has accompanying irradiation data, shows the typical range of copper contents resulting from the various practices generally used in vessel manufacture. Also included in Table 1-1 are the actual practices used for vessel manufacture for the Duane Arnold vessel.

As can be seen, the copper content in the reactor pressure vessel plate is consistently in the 0.15 to 0.20 weight percent range. Discussions with Lukens steel representatives (Domestic RPV steel supplier) revealed that because of their electric furnace process, the copper content is determined by the amount of copper in their scrap steel input and that their plate falls almost exclusively in the 0.15 to 0.20 percent range unless special low-copper scrap selection controls are employed.

The characterization of weld practices also indicates a predominance in the 0.15 to 0.20 percent copper range. A combination of the measured copper contents for four plants weld metal and the estimated copper levels for Duane Arnold and one other plant reveals that none of the six plants in question should significantly exceed the 0.20 weight percent copper level in their pressure vessel welds.

The significance of the results from this fabrication practice review lies in the fact that the copper level of the upper limit data for the revised General Electric transition temperature shift curve in the fluence range of operating BWR's is at the 0.20 percent level. Thus, if a plant in question is at or below this copper level, the General Electric upper operating curve will conservatively predict the shift in transition temperature for the reactor pressure vessel materials as a function of neutron fluence. The actual transition temperature shifts will be lower than this upper limit curve depending on the actual copper and phosphorous contents of the vessel materials in question, and when a final General Electric analysis of recent data is concluded, reactors with known copper contents will be able to eliminate the extra conservatism involved with the upper limit curve by predicting their transition shifts as a function of actual copper levels in their pressure vessel materials.

Transition Temperature Shift Evaluation:

The irradiation response of ferritic Reactor Pressure Vessel Steels is described using Charpy Impact Data as a function of test temperature. Idealized curves for the non-irradiated and irradiated conditions are shown in Figure 1-1. With irradiation the transition temperature (vertical portion of the curve) shifts to higher temperatures and the upper shelf (horizontal portion of the curve at high temperature) decreases in terms of ft-lbs absorbed.

The existing GESSAR curve for irradiation response of Reactor Pressure Vessel plate, forging, and weld metal is shown in Figure 1-2. All three product forms are governed by the same plot. The shift in temperature at which 30 ft-lbs is absorbed is the only attribute of the irradiation response curves described. Longitudinally oriented specimens were used to generate the plot. Surveillance data from four BWR's were used. Copper and phosphorous contents, which have been shown to govern irradiation response were not specifically evaluated but an overall upper bound curve was used. Almost all the data were collected over the fluence regime from

about 5×10^{18} nvt (> 1 mev) to about 5×10^{19} nvt (> 1 mev).

Regulatory Guide 1.99 contains curves for irradiation response of Reactor Pressure Vessel steels. Both the transition temperature and the upper shelf attributes are predicted in these plots. Cu and P contents are specifically taken into account. The transverse sample orientation is used since this results in more conservative data. The more conservative of the 35 mil lateral expansion or the 50 ft-lb transition temperature is used instead of the 30 ft-lb transition temperature to assure proper measure of the vertical portion of the charpy energy vs. temperature curve and to avoid measurements on the lower shelf portion of the charpy vs. temperature curve. Weld metal behavior and wrought metal behavior are treated separately. The transition temperature shift curve is shown in Figure 1-3. The decrease in upper shelf curve is shown in Figure 1-4. Almost all the data used to generate the curves are in the fluence regime from

5×10^{18} to 2×10^{20} nvt (> 1 mev). The data are extrapolated into the lower fluence regime down to an increase in 50 ft-lb transition temperature of 50°F.

Neither the existing GESSAR curves nor the Regulatory Guide curves are based on extensive data in the BWR fluence regime

{about 1×10^{17} to 5×10^{18} nvt (> 1 mev) after 40 years.} In order to remedy this problem, material from operating plants not previously included in the test data bank were irradiated to the BWR fluence regime - three values for each heat of material between

2×10^{17} and 4×10^{18} nvt (> 1 mev) and tested. Four heats of plate, three heats of weld metal, and three heats of forging were tested. Cu and P values ranged from 0.01 to 0.21 and 0.007 to 0.02, respectively. In addition, test specimens from three heats of weld metal and two heats of plate material from already existing data were subjected to chemical analysis to determine the Cu and P contents. Finally, available data from three additional operating plants were added into the data base.

With all this information, data from ten operating plants are available. Cu contents range from .01 to 0.3 and P contents range from 0.007 to 0.02. Most of the data is concentrated in the BWR fluence regime.

Figure 1-5 is an upper bound curve for all product forms based on the more conservative of the 35 mil lateral expansion or the 50 ft-lb transition temperature for transverse oriented samples. In general, there was little difference between the curve shown in Figure 1-4 and a curve based on 30 ft-lb transition temperature. On occasion, however, use of the 30 ft-lb transition temperature gave fictitiously large shifts because the lower shelf was being measured rather than the vertical portion of the curve.

All of the test data is currently being subjected to computer regression analysis to separate effects of product form, fluence, Cu, and P. This, it is estimated will be completed by September 1, 1977. In the interim, the curves shown in Figures 1-5 and 1-6 should be used. Cu and P contents for the Duane Arnold vessel lie within the envelope of Cu and P contents used to generate Figure 1-5.

This revised upper limit operating curve conservatively represents the behavior of all expected copper levels in BWR pressure vessels. When a finalized set of curves based on copper and phosphorous content becomes available, those plants with known copper contents will be able to utilize these curves to eliminate the temporary excess conservatism of the upper limit curve. These finalized curves will be available before the Duane Arnold plant reaches a fluence level at which the transition temperature shifts plays a significant role.

Figure 1-6 is a similar upper band curve for decrease in upper shelf energy vs. nvt (> 1 Mev). Data used to generate this plot must also be subjected to additional analysis.

TABLE 1-1 - CHARACTERIZATION OF REACTOR PRESSURE VESSEL MANUFACTURING PRACTICES

VESSEL	PLATE Cu CONTENT	WELD PRACTICE	WELD Cu
Location 1	-	Submerged metal arc	0.08
Location 2	0.10	Submerged metal arc	0.27
Location 3	0.19	Electroslag Submerged metal arc	- 0.18
Location 4	0.17	Submerged metal arc	0.17
Location 5	0.10	Shielded metal arc	0.01
Location 6	0.16	Submerged metal arc	0.21
Location 7	0.21	Submerged metal arc	0.19
PVRC Test Plate	0.20	Electroslag	0.19
Location 8	0.10 to 0.17	Electroslag/Submerged metal arc	0.16 to 0.21
Location 9	0.10 to 0.17	Electroslag/Submerged metal arc	0.16 to 0.21
Duane Arnold	< 0.20*	Submerged metal arc/ Shielded metal arc	< 0.20**
Location 10	< 0.20*	Submerged metal arc	< 0.20**

* Estimate Based on Lukens Steel Process Capability
** Estimate Based on Weld Process Capability

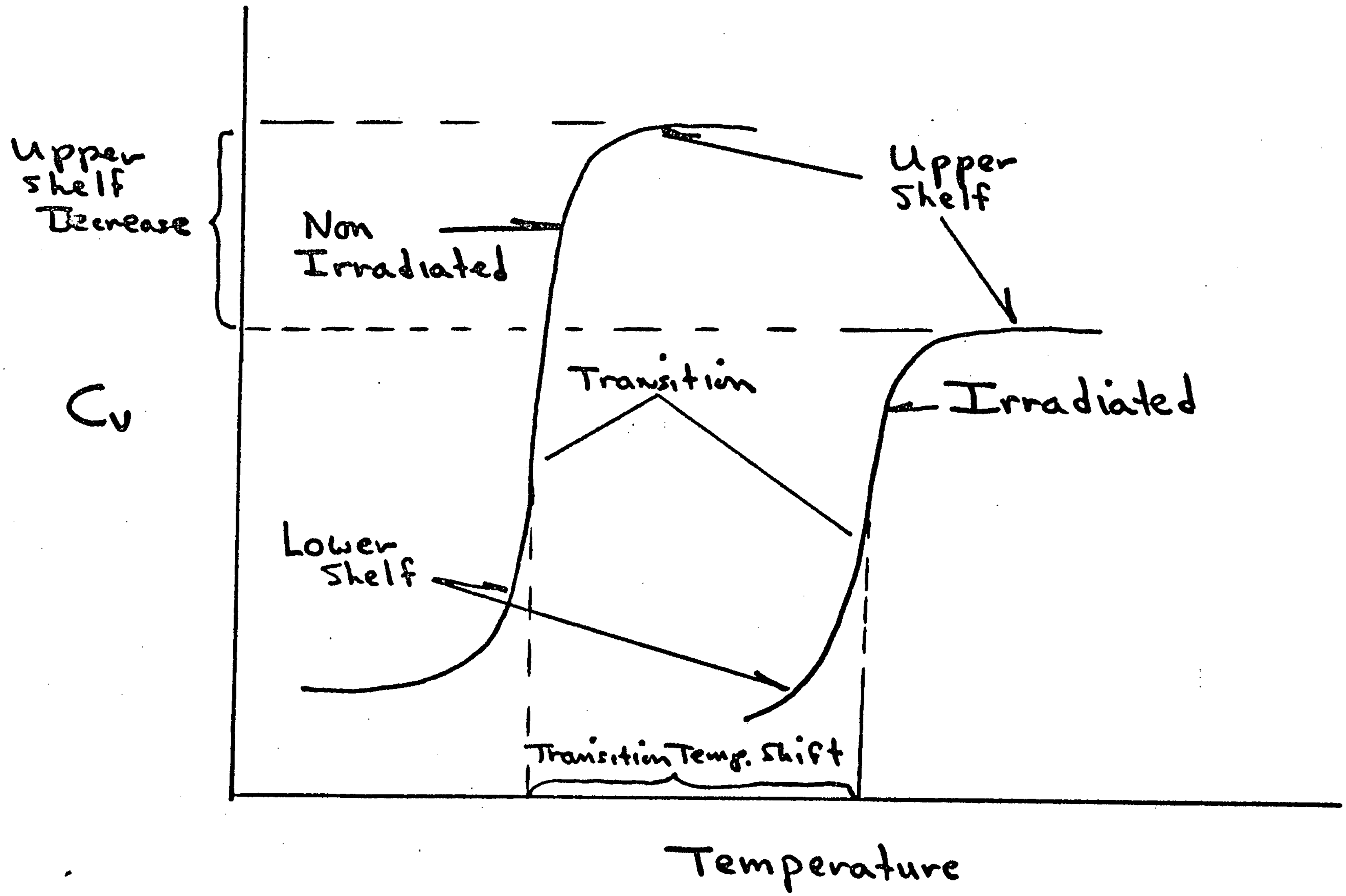


FIGURE 1-1 SCHEMATIC OF CHARPY CURVES

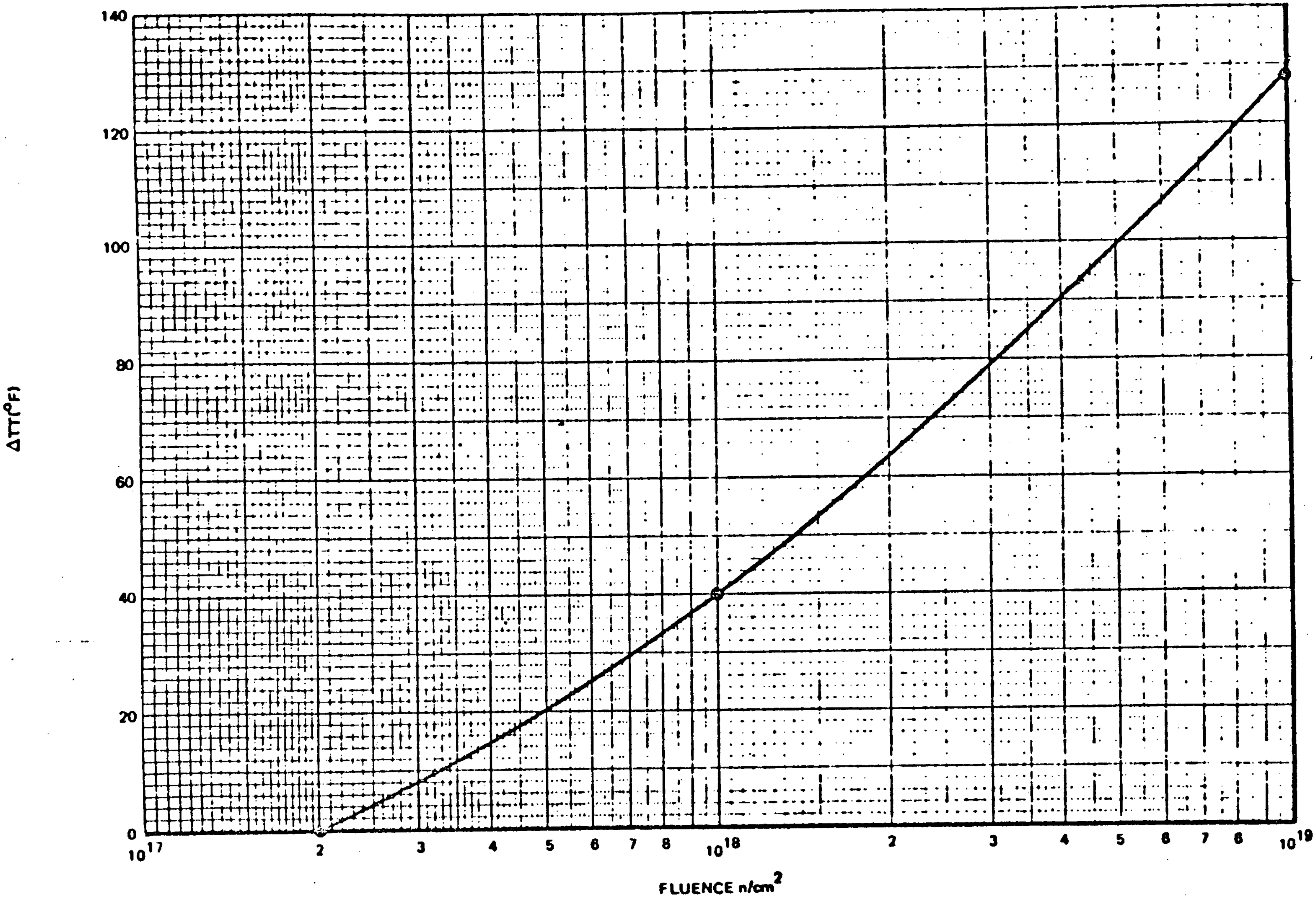


FIGURE 1-2 BWR RADIATION EFFECTS DESIGN CURVE

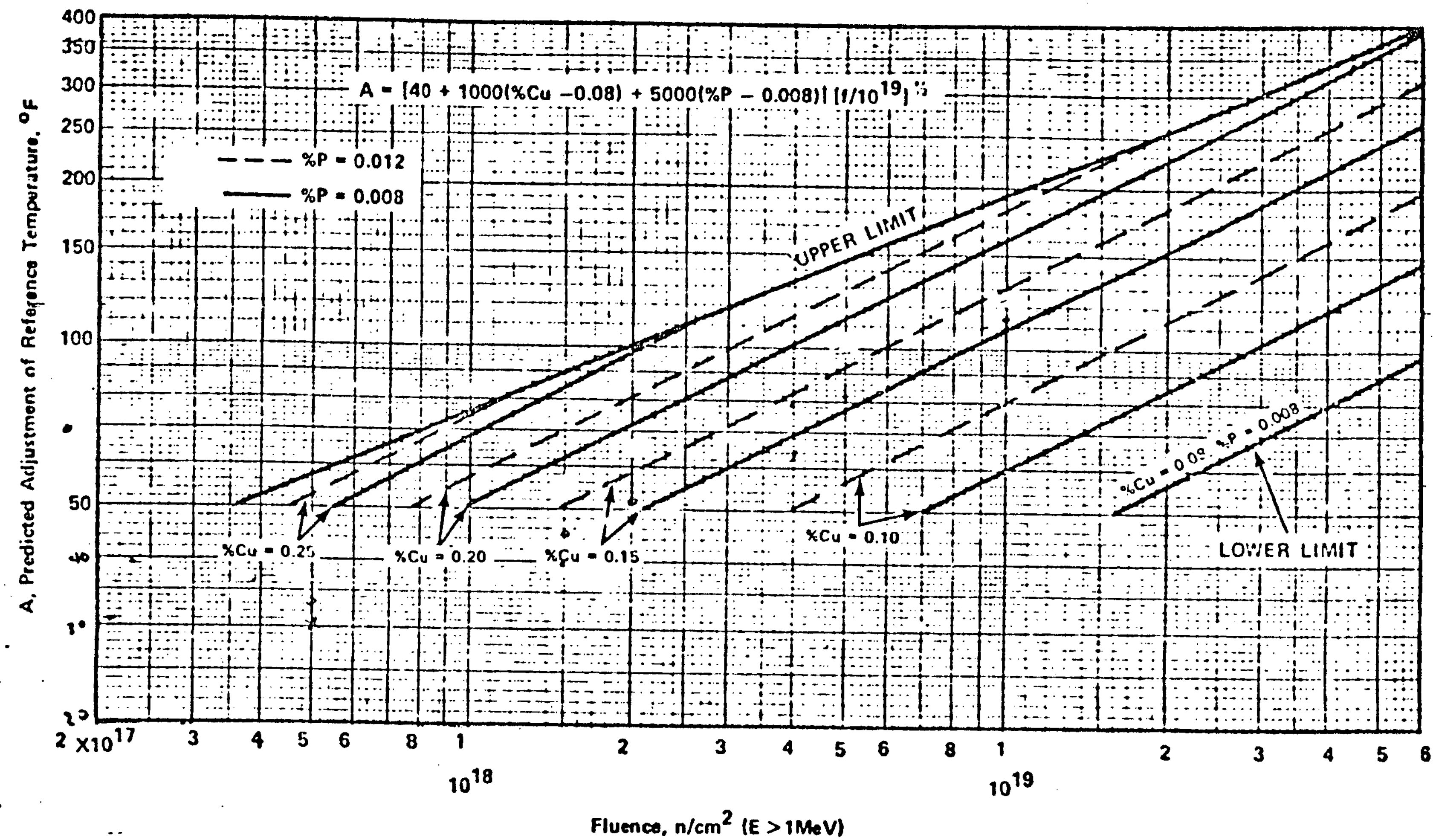


FIGURE 1-3 PREDICTED ADJUSTMENT OF REFERENCE TEMPERATURE AS A FUNCTION OF COPPER AND PHOSPHOROUS CONTENT AND FLUENCE

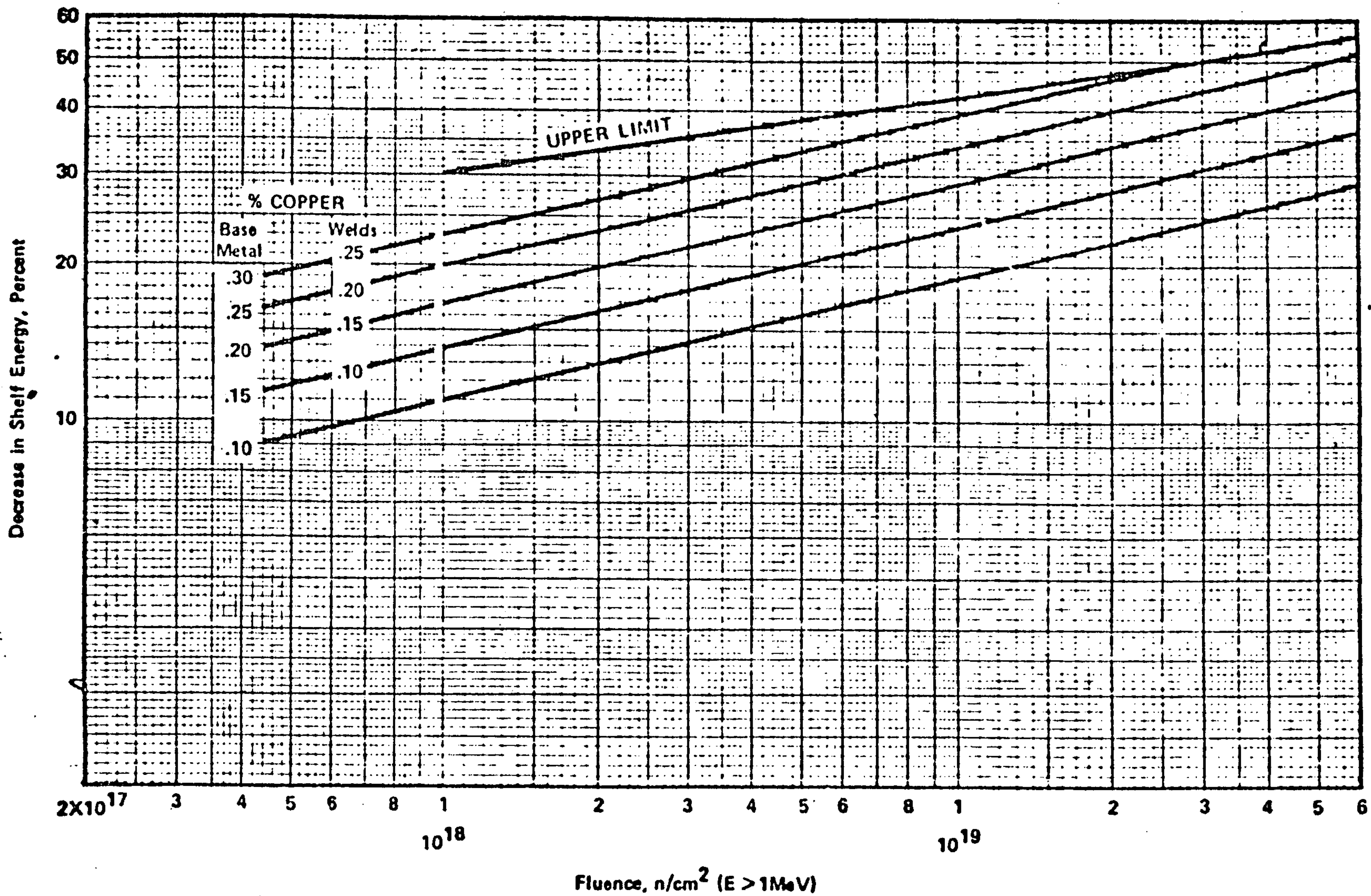


FIGURE 1-4 PREDICTED DECREASE IN SHELF ENERGY AS A FUNCTION OF COPPER CONTENT AND FLUENCE

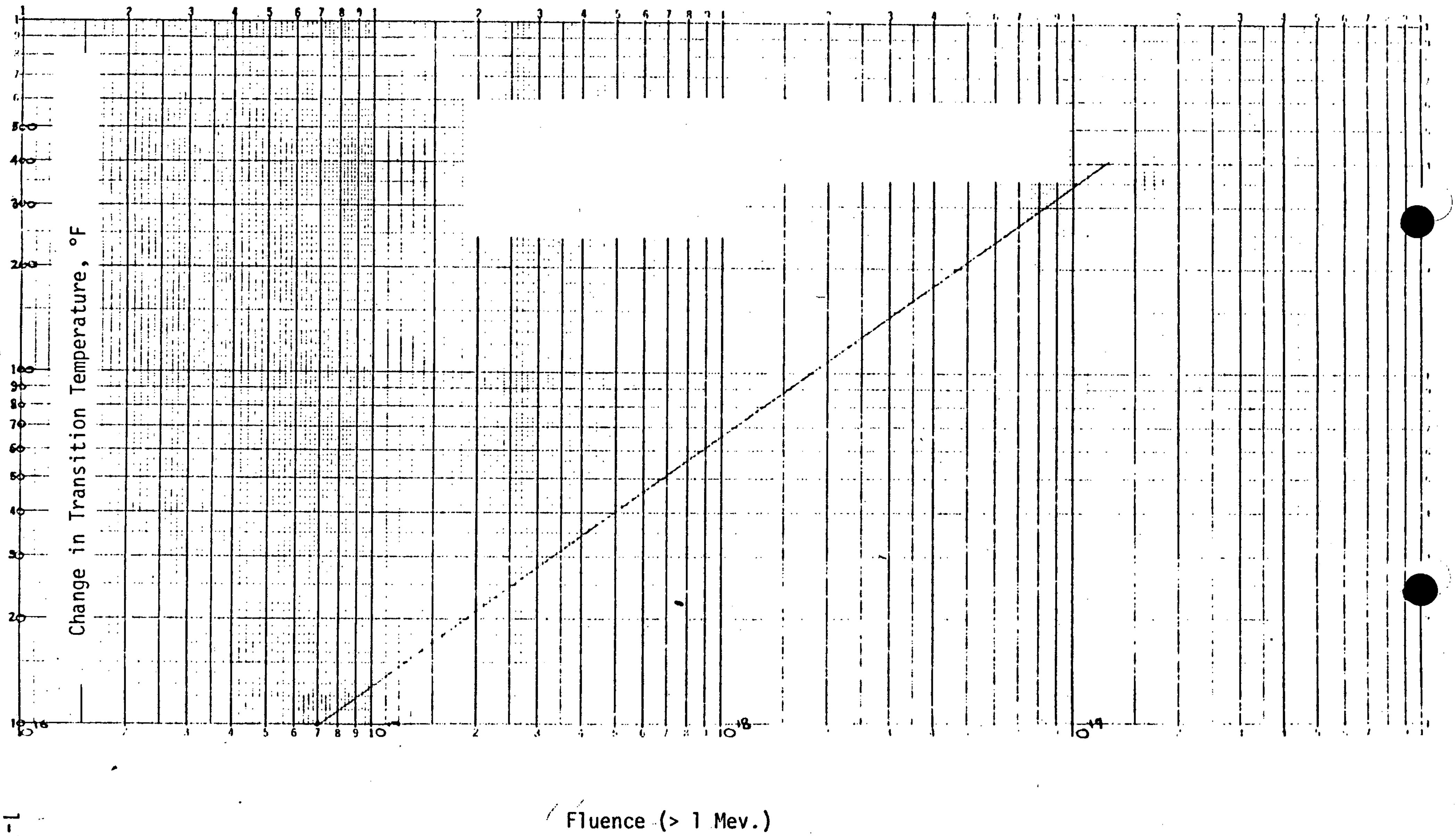
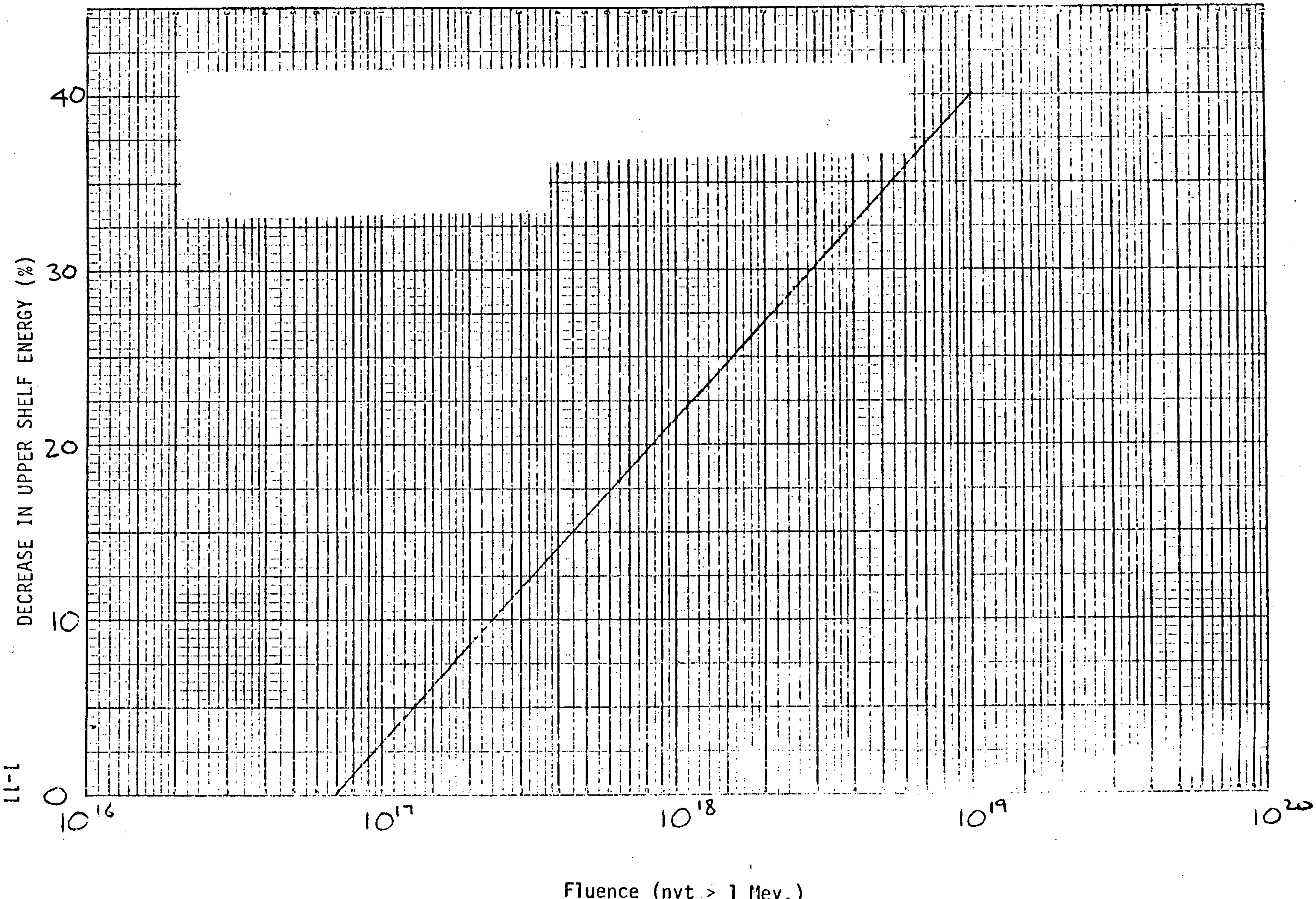


FIGURE 1-5 UPPER BOUND CURVE (INTERIM) - TRANSITION TEMPERATURE SHIFT VS FLUENCE

FIGURE 1-6 UPPER BOUND CURVE (INTERIM) - TRANSITION TEMPERATURE SHIFT VS FLUENCE



QUESTION 2

Please submit a proposed Technical Specification change to incorporate a figure showing predicted fluence as a function of time.

ANSWER

The predicted maximum fluence for the Duane Arnold reactor pressure vessel was re-calculated to be $2.8 \times 10^{18} \frac{n}{cm^2}$ (>1 Mev) at 1/4 vessel wall thickness. This calculation was based on "as-built" pressure vessel dimensions and reactor availability of 90% at 90% power level for 40 years or equivalent to 1×10^9 full-power seconds. In addition, a factor of 1.3 has been used to compensate for the angular variation in flux that occurs because of core bundle pattern. The predicted fluence as a function of time is a straight line relation from 0 to 2.8×10^{18}nvt.

QUESTION 3

As stated in Branch Technical Position MTEB 5-2 (attached to NRC Standard Review Plan 5.3.2), Positions 2.2.1 and 2.2.2, the staff requires calculations of pressure-temperature limits for regions other than the beltline unless the RT_{NDT} of the beltline is at least 50^{o} F above the RT_{NDT} for all higher stressed regions. Please submit calculations of pressure-temperature limits for higher stressed regions or provide documentation that shows the RT_{NDT} of the beltline is at least 50^{o} F above the RT_{NDT} for all higher stressed regions.

ANSWER

The thermal and stress analysis information which is uniquely needed to establish operating limit curves for this reactor were not included in the original stress analysis. The original stress analysis was completed prior to the issuance of 10CFR50, Appendix G, Fracture Toughness Requirements, and it did not include comprehensive analyses of stresses at the lower temperatures which are needed to establish operating limit curves. The following approach has been taken for this reactor to show compliance with the intent of the new requirements of 10CFR50, Appendix G:

1. Re-analyze the reactor vessel shell and head regions for this reactor at locations remote from discontinuities in accordance with ASME Code Section III and 10CFR50, Appendix G, and define operating limit curves based on an assumed 1/4t flaw depth.

2. Make a similar analysis for the BWR/6 251 Standard reactor for regions remote from discontinuities using the same calculational models and methods used for this reactor as described in item 1.

3. Compare the results of item 2 with the results of a more comprehensive analysis of the BWR/6 Standard reactor which was specifically made to show adequacy with current 10CFR50, Appendix G, limits. This analysis included discontinuity regions such as the nozzles and flanges. The purpose of this comparison is to identify the adjustments which are needed as a result of the discontinuity analysis.

4. Adjust the operating limit curves derived from the re-analysis of item 1 for this reactor based on BWR/6 discontinuity analysis results. BWR/6 discontinuity results, adjusted for the RT_{NDT} differences between this reactor and BWR/6, are used for the adjustment.

The results of this approach are summarized on Figures 1 through 5. Figures 1, 2 and 3 show comparisons for the BWR/6 251 Standard reactor vessel and Figures 4 and 5 show comparisons for this reactor.

Figure 1 shows that the limits for regions remote from discontinuities (with an assumed 1/4t flaw) do provide reasonable temperature-pressure limits for pressure tests provided the RT_{NDT} of the nozzles is at least 30° below the RT_{NDT} of the regions remote from discontinuities. With this 30° difference, the feedwater nozzle results closely correspond to those derived for the region remote from discontinuities. The flange results shown on Figure 1 are based on the ability to detect a surface crack which is equal to or less than 0.24 inches deep at the outer junction of the head with the flange (at point 8 as shown on Figure 6). A 1/4t crack can be accommodated in the flange discontinuity regions except at surface location points 8, 18, and 20 shown in Figure 6. A crack of 0.24 inch depth can easily be detected by outside surface examination techniques at these locations. Summarizing, the flange results are not limiting compared to the limits for the region remote from discontinuities with a 1/4t flaw because it is possible to detect a flange discontinuity flaw smaller than 0.24 inch depth at the locations of concern by surface examination methods. Also, volumetric examination methods can be used for supplemental evaluation. The curves derived for regions remote from discontinuities are satisfactory for pressure tests provided the nozzle RT_{NDT} is at least 30°F below that for the remote regions.

Figure 2 shows that the feedwater nozzle is somewhat more limiting at lower operating pressures under conditions of non-nuclear heatup or cooldown following nuclear shutdown. The feedwater nozzle becomes more limiting because the flow of 40°F feedwater into the nozzle causes higher thermal stresses than those that occur at regions remote from discontinuities. Therefore, the feedwater nozzle limits supersede the limits at regions remote from discontinuities for initial reactor operation as shown on Figure 2. As irradiation shifts the reactor beltline RT_{NDT}, the region remote from discontinuities again becomes more limiting.

Figure 3, which defines the minimum temperature for core operation, is constructed by adding a 40°F margin to Figure 2 limits. The lower temperature limit is established by the 1100 psig inservice hydrostatic test pressure point from Figure 1. Both of these requirements are defined in 10CFR50, Appendix G.

Figure 4 compares the feedwater nozzle and closure head flange with the points remote from discontinuities for the Duane Arnold reactor vessel. The feedwater nozzle curve is similar to that for BWR/6 shown on Figure 1; however, it has been adjusted to the Duane Arnold feedwater nozzle RT_{NDT} of 40°F from a BWR/6 RT_{NDT} of -20°F. Also, the flange curve is similar to that for BWR/6 shown on Figure 1; however, it has been adjusted to the Duane Arnold flange RT_{NDT} of 40°F from a BWR/6 RT_{NDT} of 10°F. The comparison shows that the feedwater nozzle results are limiting instead of the region remote from discontinuities for pressure tests. This is caused by the relatively high RT_{NDT} of the Duane Arnold nozzle material of 40°F (tests were not made on the nozzle material to establish a lower RT_{NDT} on this reactor).

Figure 5 is a comparison similar to that made for Figure 4. The BWR/6 results of Figure 2 were used as the basis. The feedwater nozzle results are limiting for the Duane Arnold reactor instead of the region remote from discontinuities.

In conclusion, the initial operating limits newly identified on Curve 5 are more restrictive than the earlier limits based on regions remote from discontinuities. This is mainly due to the relatively high (40°F) RT_{NDT} of the feedwater nozzle. After adjustment for irradiation, the beltline region (which is remote from discontinuities) will become limiting.

It is General Electric Company's opinion that the additional margins for core operation required by 10CFR50, Appendix G, should not be required for boiling water reactors. For instance, 10CFR50, Appendix G, Paragraph IV.A.2.c calls for a pressure margin equal to inservice hydrostatic test pressure but in no case less than 40°F margin above limits established on the basis of ASME Code Appendix G calculations to take into account such factors as the potential for overstress and thermal shock during anticipated operational occurrences in the control of reactivity. Postulated boiling water reactor accidents have been analyzed and reported in Chapter 15 of the BWR/6 Standard Safety Analysis Report (GESSAR) as well as final safety analysis reports for this reactor. The results of these analyses do not support the need for the additional margin called for in Paragraph IV.A.2.c. In particular, the continuous rod withdrawal and control rod drop accidents analyzed and reported in GESSAR, Paragraphs 15.1.11, 15.1.12, and 15.1.38 report that fuel power excursions from these events are below a 280 cal/gm enthalpy addition to the affected fuel. The power excursions are so localized and the mechanical conversion efficiencies are so low that there is a negligible reactor pressure rise effect associated with these postulated excursions. Application of 10CFR50, Appendix G, and MTEB 5-2 criteria to Duane Arnold would require that these reactors be heated up by non-nuclear means to temperatures not considered in the design of the plant auxiliary equipment. For these reasons, it is recommended that Figure 5 be used for both core operation and non-nuclear heatup or cooldown.[3]

[3]An additional discussion of the safety factors implicit in the fracture toughness of the feedwater nozzle analysis is given in Chapter 4 of General Electric Company Report NEDE-21480, Boiling Water Reactor Feedwater Nozzle/Sparger Interim Program Report, Class III, February 1977. This report has been submitted to the NRC.

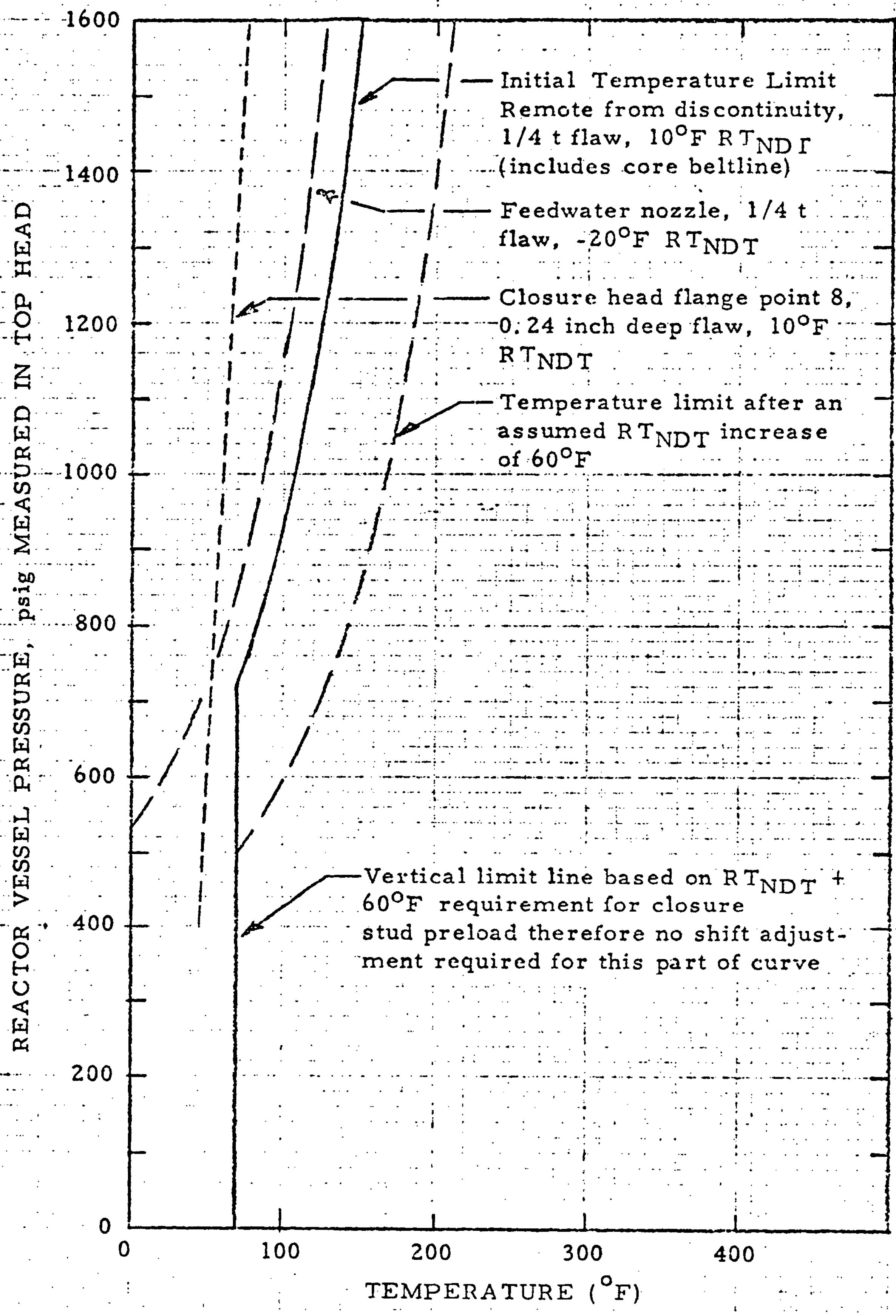


Figure 1. BWR/6 251 STANDARD PLANT- Minimum Temperature for Pressure Tests Such as Required by Section XI.

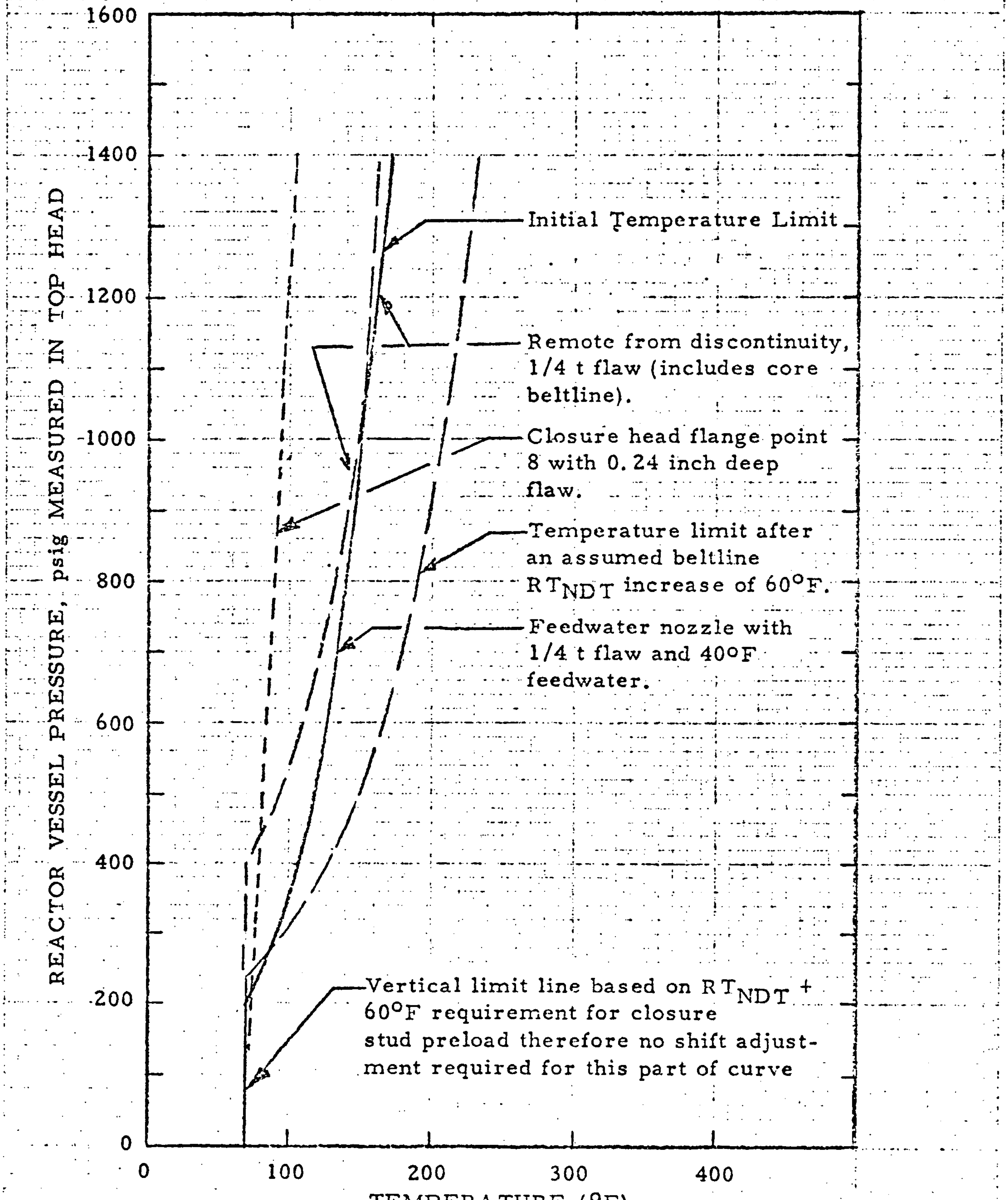


Figure 2. BWR/6 251 STANDARD PLANT - Minimum Temperature for Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown.

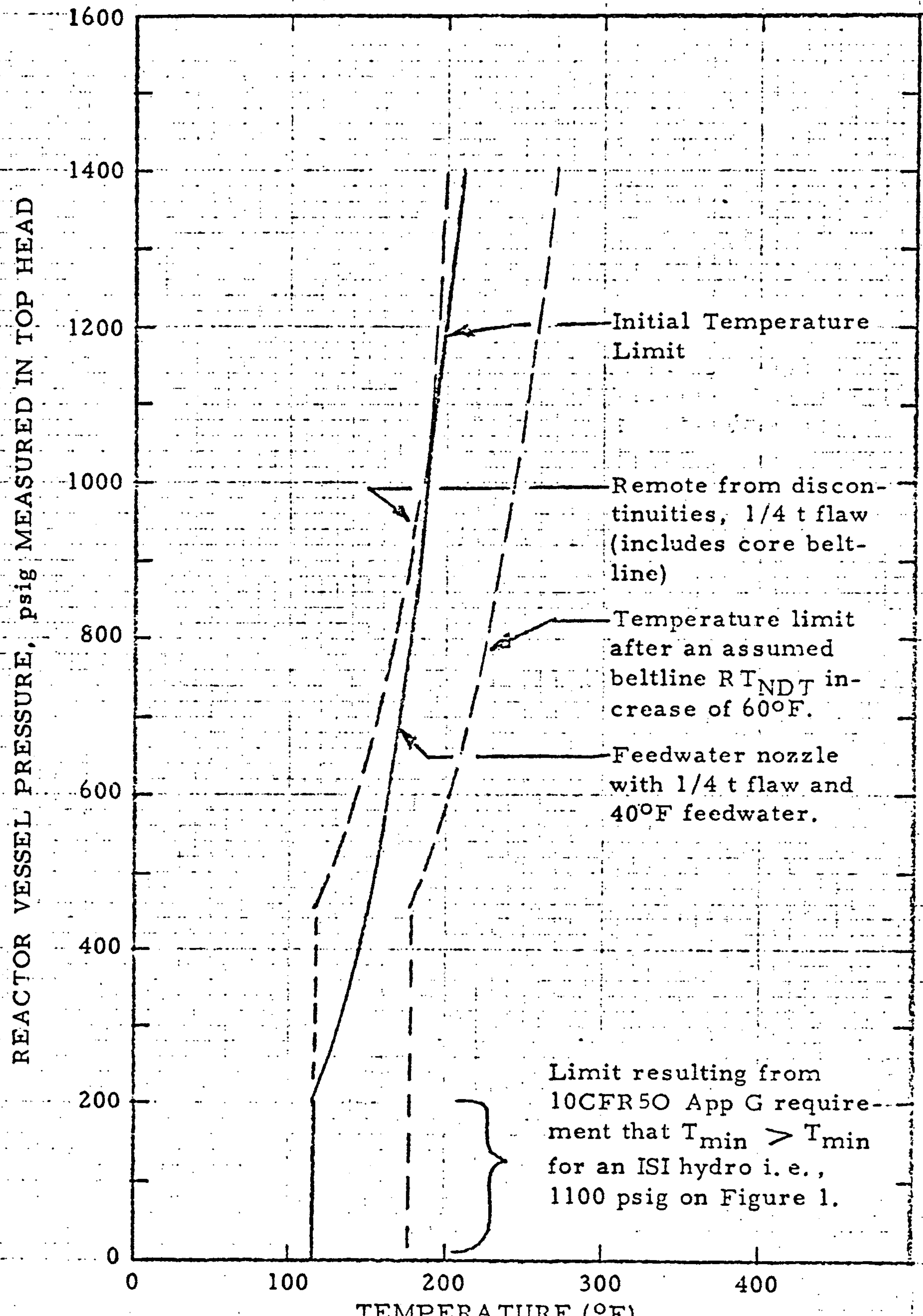


Figure 3. BWR/6 251 STANDARD PLANT - Minimum Temperature for Core Operation (Criticality). (Includes Additional 40°F Margine Required by 10CF50 Appendix G)

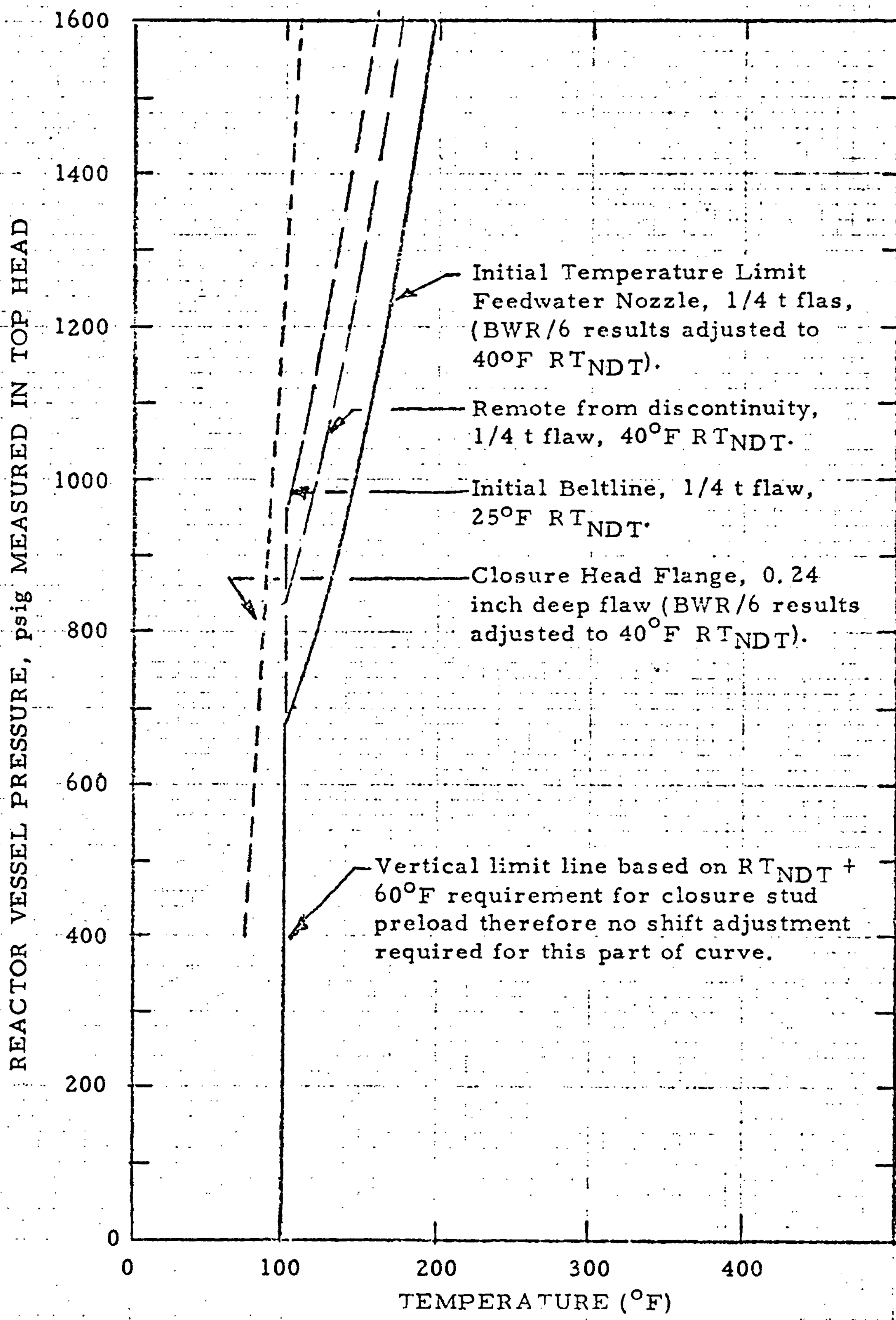


Figure 4. DUANE ARNOLD - Minimum Temperature for Pressure Tests Such as Required by Section XI.

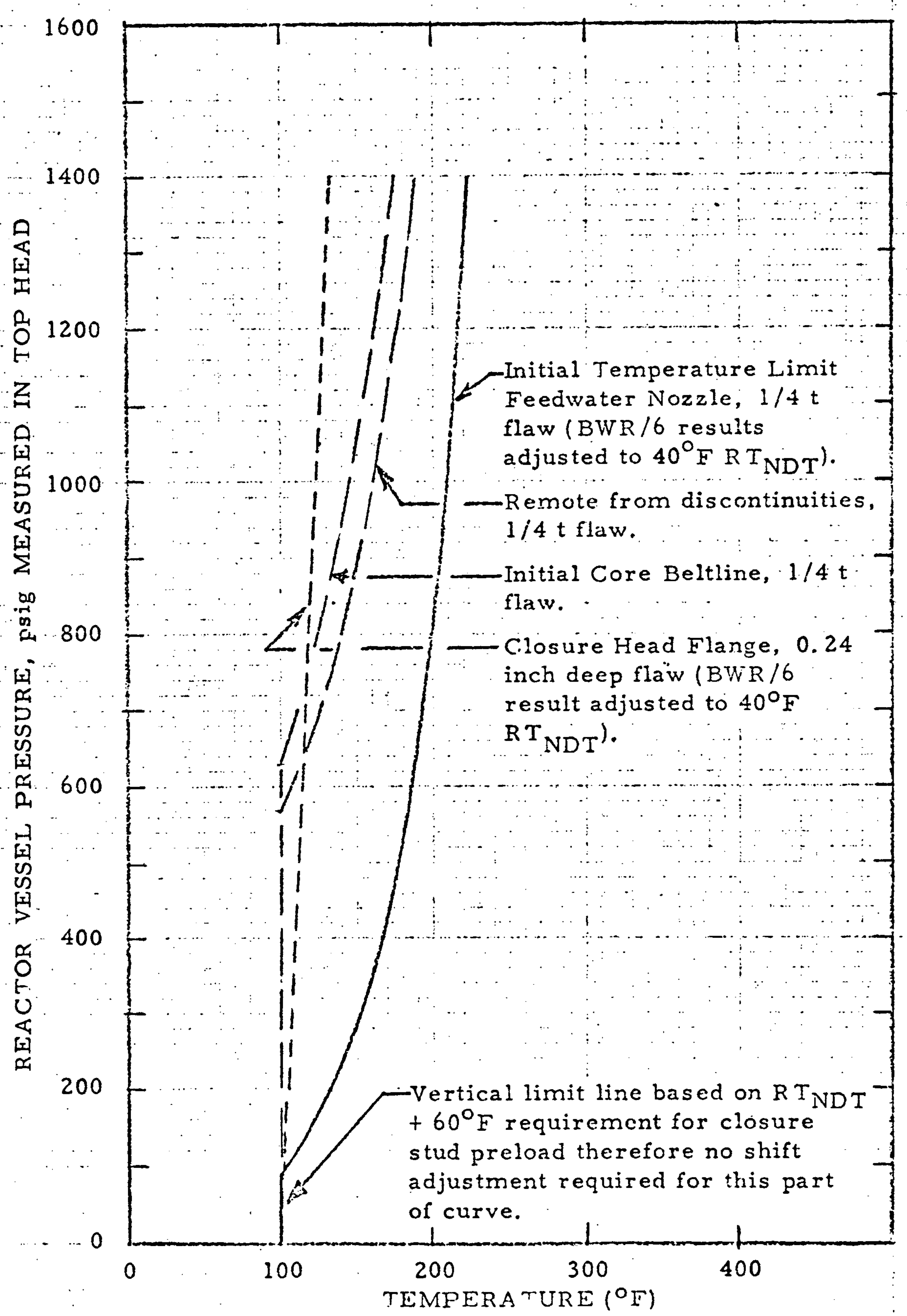


Figure 5. DUANE ARNOLD - Minimum Temperature for Mechanical Heatup or Cooldown Following Nuclear Shutdown.

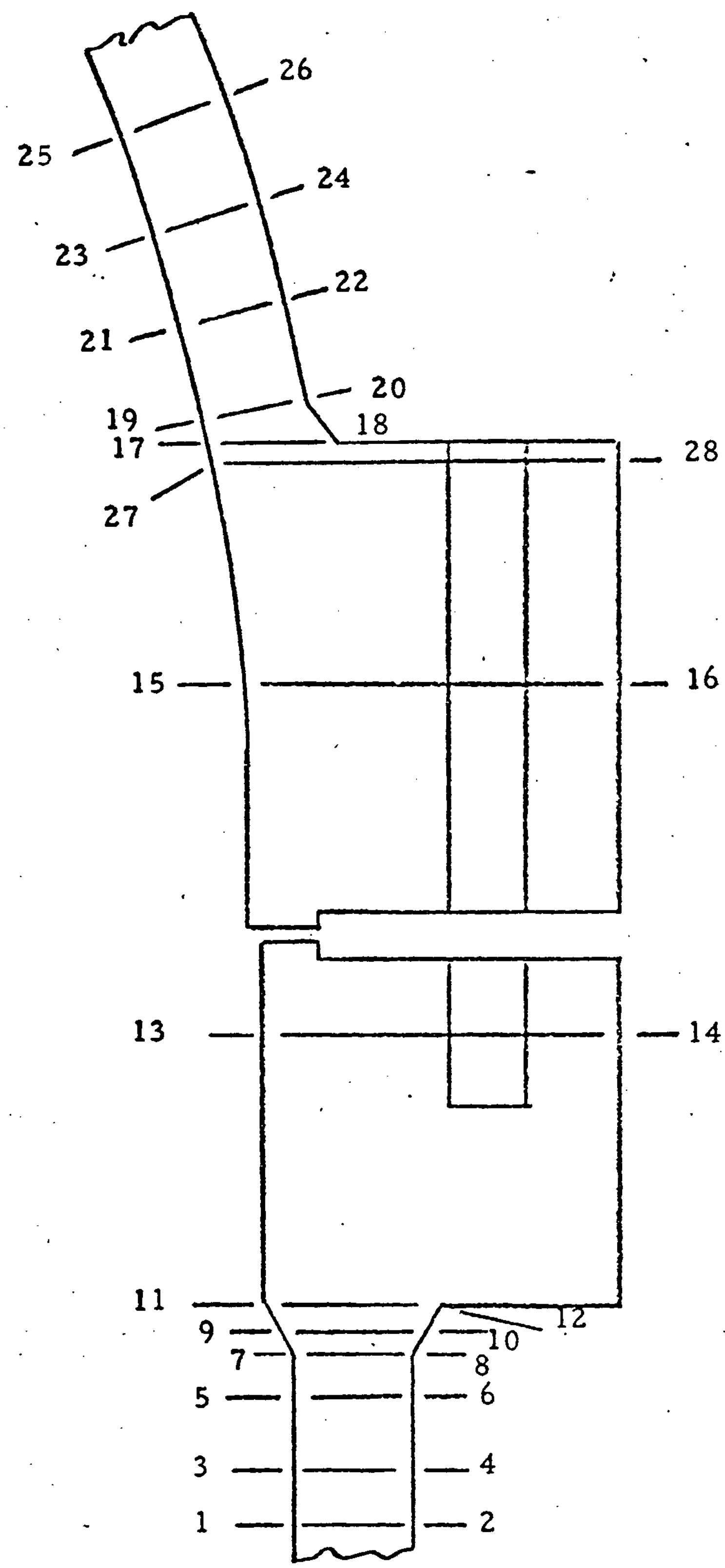


Figure 6. ANALYSED STRESS LOCATION POINTS - MAIN CLOSURE FLANGE.

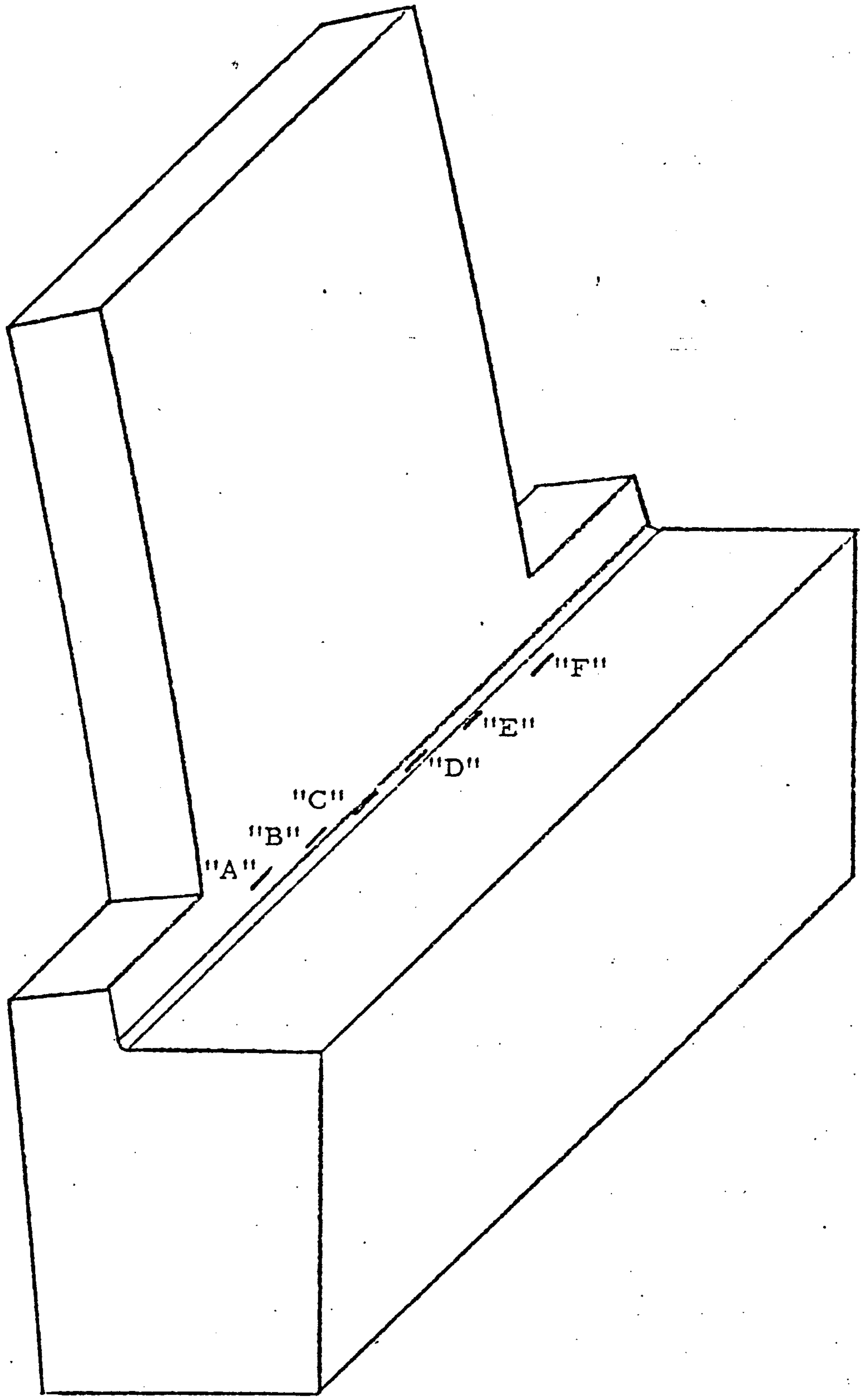


Figure 7. HEAD FLANGE TEST GROOVE LOCATIONS.

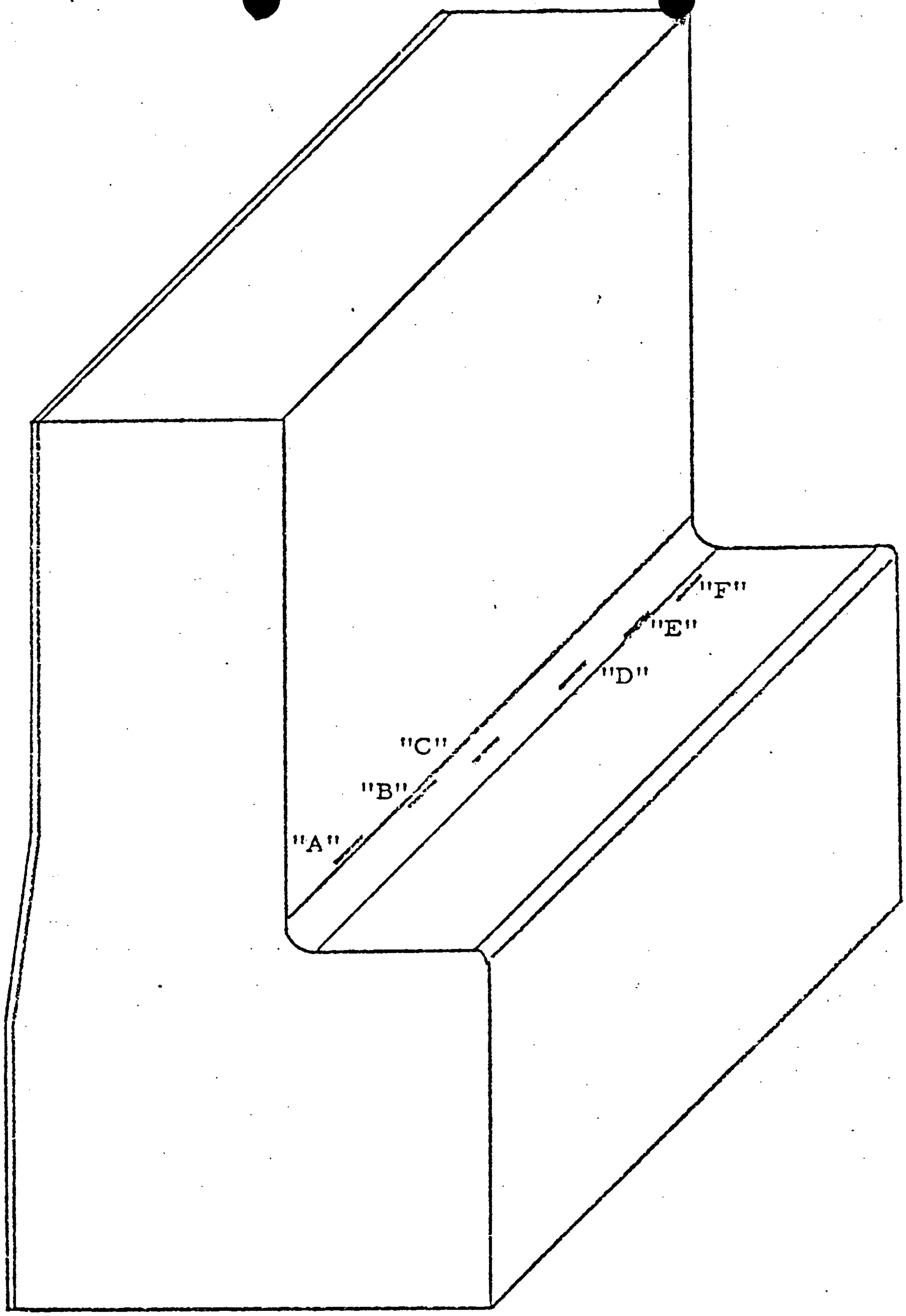


Figure 8. SHELL FLANGE TEST GROOVE LOCATIONS.