BASIS FOR REVISION 2 OF U.S. NRC REGULATORY GUIDE 1.99

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Abstract

Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials," is being updated to reflect recent studies of the physical basis for neutron radiation damage and efforts to correlate damage to chemical composition and fluence. Revision 2 of the Guide contains several significant changes. Welds and base metal are treated separately. Nickel content is added as a variable and phosphorus removed. The exponent in the fluence factor is reduced, especially at high fluences. And, guidance is given for calculating attenuation of damage through the vessel wall. This paper describes the basis for these changes in the Guide.

Key Words

Neutron irradiation, nuclear reactor materials, low alloy steels, fluence, copper, nickel, fracture toughness.

INTRODUCTION

Revision 2 of Regulatory Guide 1.99 "Radiation Damage to Reactor Vessel Materials" (the Guide), is an outgrowth of many activities: (1) experience with the application of Revision 1 in licensing work since 1977, (2) technical contacts at meetings of ASTM Committee E-10 and Metal Properties Council Subcommittee 6 Task Groups on radiation damage plus an American Nuclear Society seminar in 1983, (3) accumulation of surveillance data from commercial power reactors, (4) resolution of the pressurized thermal shock issue, which required best-estimate calculative procedures and careful attention to uncertainties, (5) extensive help in data analysis by G. L. Guthrie[1] and (6) interaction with G. R. Odette[2].

The objective of the Guide is to provide calculative procedures for the adjusted reference temperature (ART) that are acceptable to the NRC. The purpose of this paper is to describe the basis for those procedures. As given in the Guide,

ART = Initial RT_{NDT} + Δ RT_{NDT} + Margin

where

RT_{NDT} = reference temperature, nil-ductility transition, deg. F.

Initial RT_{NDT} = the reference temperature for the unirradiated material.

ΔRT_{NDT} = the adjustment of reference temperature, commonly called the Charpy shift, i.e., the temperature shift (measured at the 30 ft lb level) in the average Charpy curve for the irradiated material relative to that for the unirradiated material.

Margin = the quantity, deg. F, that is to be added to obtain conservative, upperbound values of ART.

[1]From Hanford Engineering and Development Laboratories, a contractor to NRC.

[2]From University of California, Santa Barbara, CA, a contractor to the Electric Power Research Institute.

This paper focuses on the calculation of ΔRT_{NDT} from chemistry and fluence information, which will be a typical procedure for all plants until they have credible surveillance data. As used in the Guide, "ΔRT_{NDT} surface" is the product of a chemistry factor and a fluence factor. The latter is based on the fluence at the inside surface of the vessel beltline at the location of interest. However, in the fracture mechanics calculations that are the basis for pressure-temperature limits for reactor heatup and cooldown, the fracture toughness value of interest is that at the tip of the postulated flaw. (Typically, toughness data are given in terms of temperature relative to RT_{NDT}.) Thus one is also required to calculate the attenuation of ΔRT_{NDT} through the vessel wall. This paper will also discuss the formula given in the Guide for this purpose.

THE GUTHRIE DATA BASE

Commercial power reactor surveillance data were used exclusively (no test reactor data) in the analyses made by Guthrie. There were 51 weld and 126 base metal (plate and forging) data points, taken from surveillance reports by P. N. Randall and rechecked by Guthrie. Shift values were those read from the hand-drawn Charpy curves at the 30 ft lb level by the authors of the surveillance reports. In those few cases where the authors reported only 50 ft lb shift values, Randall and Guthrie made their own estimate of the 30 ft lb shift from the plotted data. Fluence values for the surveillance capsules were those reported by Simons(2) when available, otherwise the surveillance report information was used directly. Copper and nickel content were as reported except in the case of welds made by Babcock and Wilcox for which newer information was available(3).

The distribution of the data with respect to copper and nickel content is illustrated in Figure 1 for welds and Figure 2 for base metal, taken from one of Guthrie's computer printouts. To produce the data shown, each measured value of ΔRT_{NDT} was first normalized to a fluence of 10^{19} n/cm^2 using Guthrie's fluence function (described below) then placed in the "box" corresponding to its copper and nickel content. If there were two or more entries in a box, they were averaged. Thus, in the lower left corner of Figure 1 the entry "77(2)" indicates that there were two pieces of data for welds having

0.10 ± 0.025 percent Cu and 0.15 ± 0.025 per cent nickel, and the average of their normalized shift values was 77°F (43°C). To get a feel for the task of developing correlations, it is instructive to attempt to draw isoshift "contour" lines on Figs. 1 and 2 for several values of shift. Two characteristics of the data base will become obvious from this exercise: the degree of scatter is significant, and there are clumps of data and sizeable blank areas where there are no data.

In Revision 2 of the Guide, it was necessary to give values of the chemistry factor for copper content ranging from 0 to 0.40 percent and for nickel ranging from 0 to 1.2 percent to provide guidance over the full range of expected compositions. Admittedly, these values somewhat exceed the ranges of the data base; hence, the Guide violates a restriction placed on the correlation functions by Guthrie. The most likely occurrence is for welds having copper content less than 0.12 percent and nickel in the 0.6 percent range. Clearly, application of the Guide at the fringes of the data base should be made with caution and supported by additional data. Unfortunately, these will likely be test reactor data, and their applicability to a correlation useful for operating reactors is still in doubt.

There is also a dearth of surveillance data for materials having low copper and higher than normal phosphorus contents. The upper limit on phosphorus in the data base is 0.020 percent for welds and 0.017 for base metal. Application of the Guide to cases where the phosphorus content is significantly higher should not be made without supporting data, which again brings up the question of the applicability of test reactor data.

To observe the distribution of fluence values in the data base, refer to Figure 11, which will be described in the discussion of residuals (measured minus calculated values) as a function of fluence. The range of fluence values was from 7.3×10^{17} to 7.8×10^{19} n/cm^2 (E>1MeV), and the distribution within that range was reasonably uniform.

THE ODETTE DATA BASE

The data base used by Odette (4) was the EPRI data base (5) which contained 65 weld data and 151 base metal data. It overlapped the Guthrie data base almost completely and the ranges of copper, nickel and fluence were about the same. There are more data from boiling water reactors in the Odette data base. The principle difference is in the derivation of shift values. In the EPRI data base, the plots of Charpy energy as a function of temperature were refitted using a hyperbolic tangent function and the 30 ft lb shift values were then recalculated. A spot check showed the results are about the same as those in the Guthrie data base.

DERIVATION OF THE CHEMISTRY FACTOR FROM THE GUTHRIE AND ODETTE CORRELATION FUNCTIONS

Guthrie and Odette reached similar conclusions in several areas: (1) separate correlations are needed for welds and base metal, (2) the expression should be the product of a chemistry factor and a fluence factor, (3) the elements in the chemistry factor should be copper and nickel, and (4) the fluence factor should provide a trend curve slope when plotted on log-log paper of about 0.25 to 0.30 at 10^{19} n/cm², and it should be steeper at lower fluences and flatter at higher fluences.

For welds, their correlation functions for ΔRT_{NDT} are as follows:

Guthrie: $\Delta RT_{NDT} = [624\ Cu - 331\sqrt{CuNi} + 251\ Ni][f^{0.282-0.0409\ \ln f}]$

1 Standard Deviation = 28°F (16°C)

Odette: $\Delta RT_{NDT} = 360\ Cu\ [1+1.38(\mathrm{erf}\{\frac{0.3\ Ni-Cu}{Cu}\}+1)][1-\exp(\frac{-f}{0.11})]^{1.36}\ [f^{0.18}]$

1 Standard Deviation = 27°F (15°C)

For base metal their correlation functions for ΔRT_{NDT} are as follows:

Guthrie: $\Delta RT_{NDT} = [-38+556\ Cu + 480\ Cu\ \tanh 0.353\ \frac{Ni}{Cu}][f^{0.266-0.0449\ \ln f}]$

1 Standard Deviation = 17°F (10°C)

Odette: $\Delta RT_{NDT} = 389\ Cu[1+0.33(erf\{\frac{0.77\ Ni-Cu}{Cu}\}+1)][f^{0.28}]$

1 Standard Deviation = 23°F (13°C)

The units of ΔRT_{NDT} are degrees Fahrenheit. (The equations by Odette have been converted from degrees Centigrade.) Copper and nickel are given in weight percent, and fluence, "f", is in units of 10^{19} n/cm^2 (E>1 MeV). Values for the error function, "erf", are given in Table A-3 of Reference 4.

To compare the correlation functions given by Guthrie and Odette, the first step is to compare the chemistry factors, the expressions obtained by setting the fluence equal to 1 x 10^{19} n/cm^2. Figures 3 and 4 compare the Guthrie and Odette chemistry factors for welds for 2 nickel contents. There is remarkable agreement at the higher copper levels. However, the function chosen by Odette passes through zero at zero copper whereas that by Guthrie intercepts the zero-copper ordinate at increasingly higher values as nickel increases. This difference is understandable in view of the lack of weld data below 0.10 percent copper. Because it has been generally accepted that the effect of nickel is a synergistic copper-nickel effect, if follows that ΔRT_{NDT} should be low when the copper content is low, regardless of the nickel content. Therefore, the Odette curves were used throughout for welds with the exception of the cutoff at the lower end, discussed below.

For base metal, Figures 5 and 6 compare the Guthrie and Odette chemistry factors with regard to the effects of copper. In this case, the curves cross. Those by Guthrie are higher at high copper levels, but those by Odette are higher at low copper levels. For the Guide, the higher curve was used, with two exceptions. First, when the Guthrie curves for base metal exceeded those for welds, which they did at high copper levels (see Figure 6), the latter were used, the justification being that there were no base metal data for copper above 0.25 percent and there is no basis to believe that base metal should be more sensitive to radiation than welds. Second, at very low copper levels, the curves for both welds and base metal were levelled off at CF = 20°F. Again, the weld data base is lacking in the range 0-0.10 percent copper, but some guidance is needed in this range because newer plants will have low copper.

Therefore, test reactor data were used as described below to assist in setting the CF function for very low copper values.

Test reactor data (6,7) for low copper, high nickel (0.7 percent, nominal) materials were normalized to a fluence of 10^{19} n/cm^2 by dividing the measured shift by the quantity: $(f)^{0.5}$, the fluence function favored by the authors of the reports. Most test fluences were in the range 2 to 8 x 10^{19} n/cm^2, hence the normalized values of shift were felt to be as low as one could justify. The results are plotted in Figure 7 for welds and Figure 8 for base metal, superimposed on the chemistry factor data as given in the Guide. For base metal, there is adequate surveillance data to support the Odette and Guthrie work, but for welds the data are sparse. Nevertheless, in the light of the test reactor data plotted in Figures 7 and 8, it seemed prudent to establish a minimum at 20°F (11°C) for the chemistry factor. In addition, for base metal the curves in Figure 8 were faired in to the minimum at 0.05 percent copper, which meant raising the curves for 0 and 0.2 percent nickel about 15°F (8°C).

A comparison of the chemistry factors for welds and base metal is given in Figure 9. The fact that the differences disappear at copper levels above about 0.25 percent is an artifact of the procedure used to draw the curves, described above.

DERIVATION OF THE FLUENCE FACTOR

Guthrie found only small differences in the constants of the fluence factors for welds and base metal. (See the correlation functions given earlier.) In Figure 10, the two factors are plotted over the range 2 x 10^{17} to 10^{20}. They differ by less than 4 percent. Consequently, in the interests of simplicity, the fluence factor used for both was: $f \exp (0.28 - 0.0434 \ln f)$ or $f \exp (0.28 - 0.10 \log f)$ as it is given in the Guide. For clarity this curve is not shown in Figure 10. It would fall between the Guthrie curves for weld and base metal. The fluence factor for welds derived by Odette, also shown in Figure 10, gives good agreement with that obtained by Guthrie except at fluences below 1.5 x 10^{18}, where the Odette fluence factor drops off sharply. For base metal, Odette used a uniform slope of 0.28, which (happily) agrees

with that found by Guthrie at 10^{19} n/cm^2. Therefore, it was an easy decision to use Guthrie's fluence factor with the constants given above.

JUSTIFICATION FOR THE CALCULATIVE PROCEDURES GIVEN IN THE GUIDE

To show that the calculative procedures given in the Guide are faithful to the data base, they were used to calculate a shift value based on the copper, nickel and fluence values for each line of data in the Guthrie data base. The residual (observed minus calculated value) is plotted versus fluence, copper and nickel content in Figures 11, 12, and 13, respectively. Scatter about the zero residual axis is fairly well balanced between overprediction and underprediction. One exception is seen in Figure 12 where for base metal the perturbation seen at low copper values is a reflection of the adjustment of chemistry factors made to reflect test reactor data and provide a conservative minimum.

Another purpose in showing these plots of residuals is to demonstrate that the blending of Guthrie's and Odette's results to get the calculative procedures for the Guide has not invalidated the use of twice the standard deviation from Guthrie's regression analysis to provide suitable margin. The "two-sigma" limits, ±56°F (±31°C) for welds and ±34°F (±19°C) for base metal, plotted on the Figures, do indeed show that only one weld and two base metal data points will be underpredicted if the margin on ΔRT_{NDT} is made twice the standard deviation.

In considering the requirement for the amount of margin to be added, there was a question about the choice of margin for very low values of calculated shift. A more general question was: should the margin be some function of the shift? To answer this question, the residuals were plotted against the calculated value of shift as shown in Figure 14. There is no clear evidence of a relationship of the residuals to the calculated value. Consequently, it was decided to add twice the standard deviation across the board except at low values where it was arbitrarily decided to add 100 percent of the calculated value, as shown in Figure 14.

As given in the Guide, the margin to be added in calculating conservative values of RT_{NDT} for use in Appendix G (10 CFR Part 50) evaluations includes margin on initial RT_{NDT} as well as margin on ΔRT_{NDT}. Following the precedent set in the analyses for the pressurized thermal shock problem, the two are combined in the expression:

$$\text{Margin} = 2\sqrt{\sigma_o^2 + \sigma_\Delta^2}$$

where σ_o is the standard deviation on initial RT_{NDT} when a generic mean value is used, and σ_Δ is the standard deviation on ΔRT_{NDT}.

ATTENUATION OF RADIATION DAMAGE WITHIN THE VESSEL WALL

The changes in neutron energy spectra with depth of penetration in the wall are significant; and to take this into consideration, it was decided to use a "dpa equivalent" attenuation formula. This is one change in the method used previously. Another change is brought about by the change in the fluence factor from a simple power law with an exponent of 0.50 to one with an exponent of (0.28-0.10 log f). In the face of considerable uncertainty about the various elements in this calculation, we elected to use the following simplified procedure.

The starting point was the attenuation formula used for a number of years:

$$f = (f_{surface})\, e^{-0.33x}$$

where f is the fluence in units of n/cm² (E> MeV) and "x" is depth in the wall, in inches, measured from the inside surface. This formula came from a staff review of surveillance reports made several years ago. To convert to a "dpa equivalent" formula we used some calculations reported at the 4th ASTM - Euratom Symposium (8), which showed that dpa attenuation through an 8.0 in. vessel wall is less than the attenuation of fluence, n/cm² (E> 1 MeV) by a factor of 2.06, the average of six calculations made for different reactor

vessels. To achieve this reduction in attenuation, the equation for fluence attenuation becomes:

$$f = (f_{surface})\ e^{-0.24x}$$

For simplicity, the relationship of ΔRT_{NDT} to fluence is taken to be a simple power function with an exponent of 0.28 with the result:

$$\Delta RT_{NDT} = [\Delta RT_{NDT\ surface}]\ e^{-0.067x}$$

This is a best-estimate expression. The uncertainty is assumed to be accounted for in the margin term described earlier.

CONCLUSION

The work of two independent investigators, working from separate data bases, yet coming to very similar conclusions, has provided a sound basis for the calculative procedures for adjustment of reference temperature given in Revision 2 of Regulatory Guide 1.99. The Guide will receive peer review when published for public comment and will be reviewed again within the NRC in response to those comments.

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(8) G. L. Guthrie, W. N. McElroy and S. L. Anderson, "A Preliminary Study of the Use of Fuel Management Techniques for Slowing Pressure Vessel Embrittlement," Paper presented at 4th ASTM - Euratom Symposium, March, 1982.

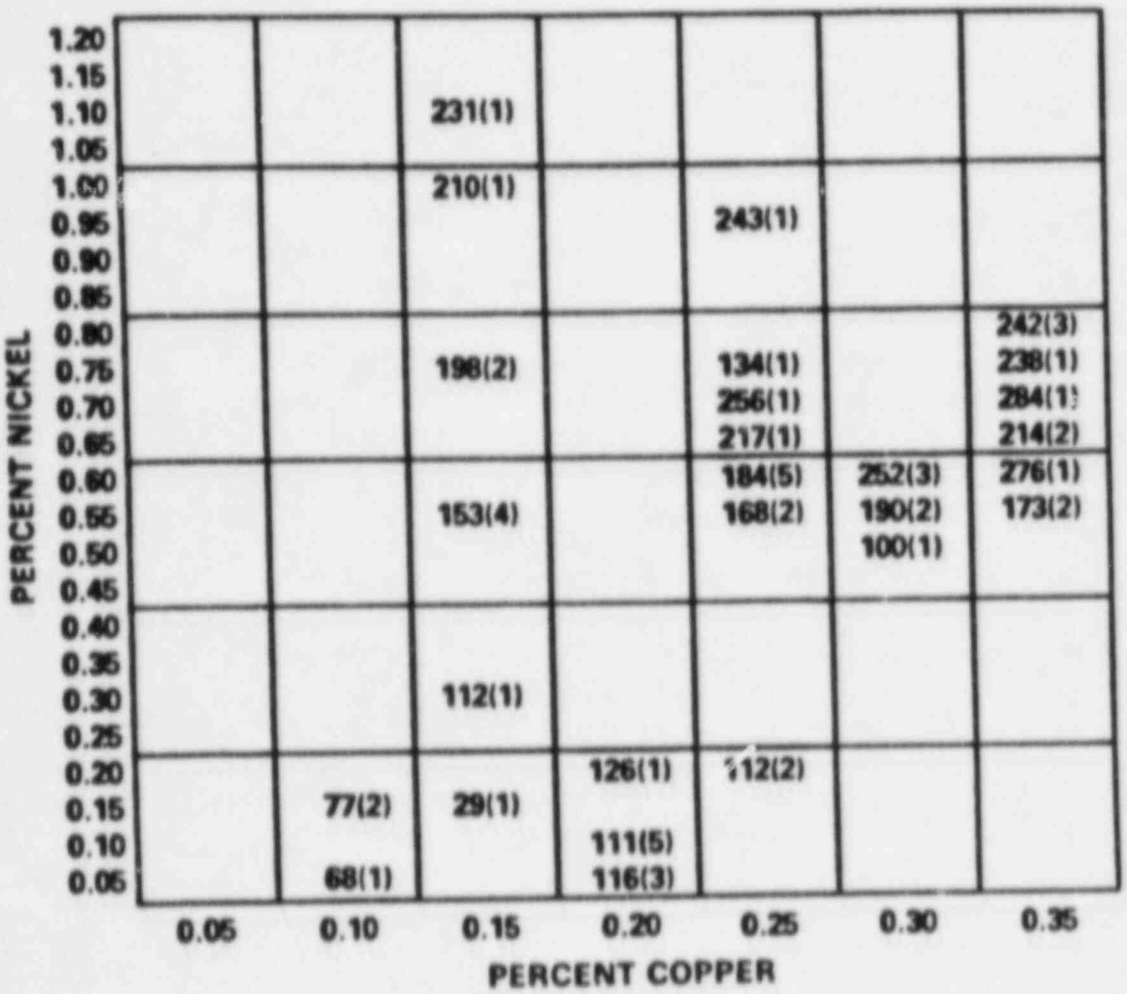

FIG. 1 DISTRIBUTION OF GUTHRIE'S WELD DATA BASE IN TERMS OF COPPER AND NICKEL CONTENT

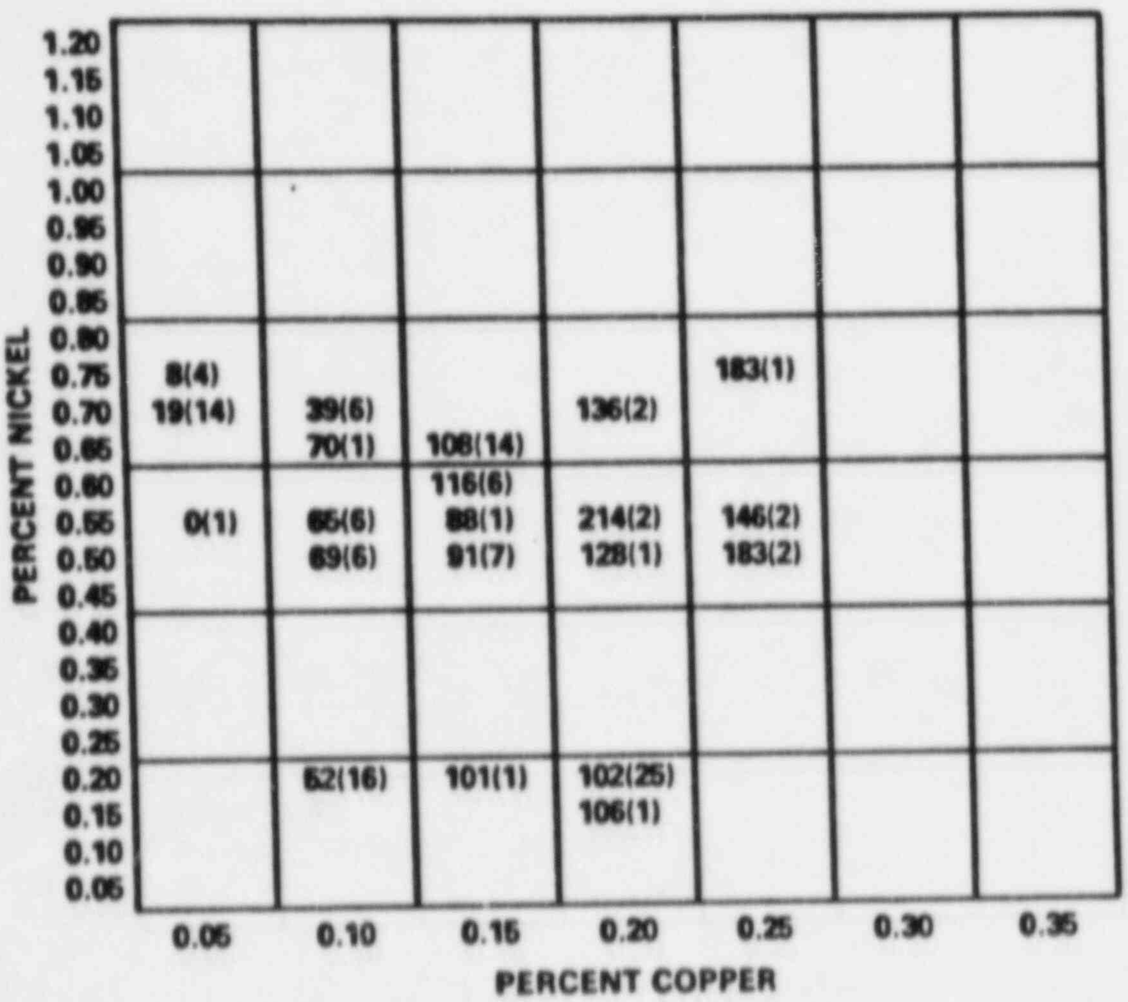

FIG. 2 DISTRIBUTION OF GUTHRIE'S BASE METAL DATA BASE IN TERMS OF COPPER AND NICKEL CONTENT

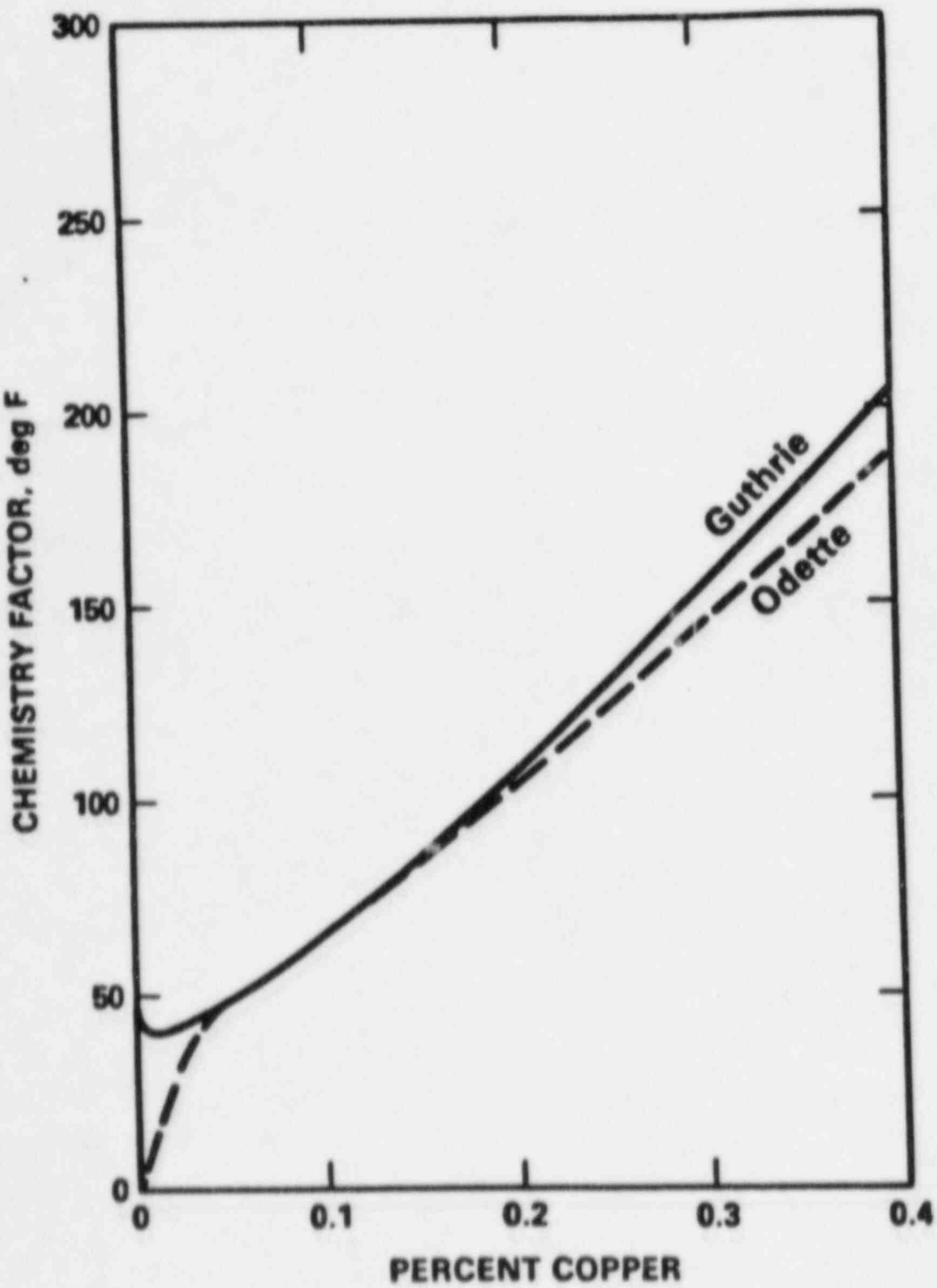

FIG. 3 COMPARISON OF GUTHRIE AND ODETTE STUDIES ON THE EFFECT OF COPPER ON THE CHEMISTRY FACTOR -- WELDS WITH 0.2 PERCENT NICKEL

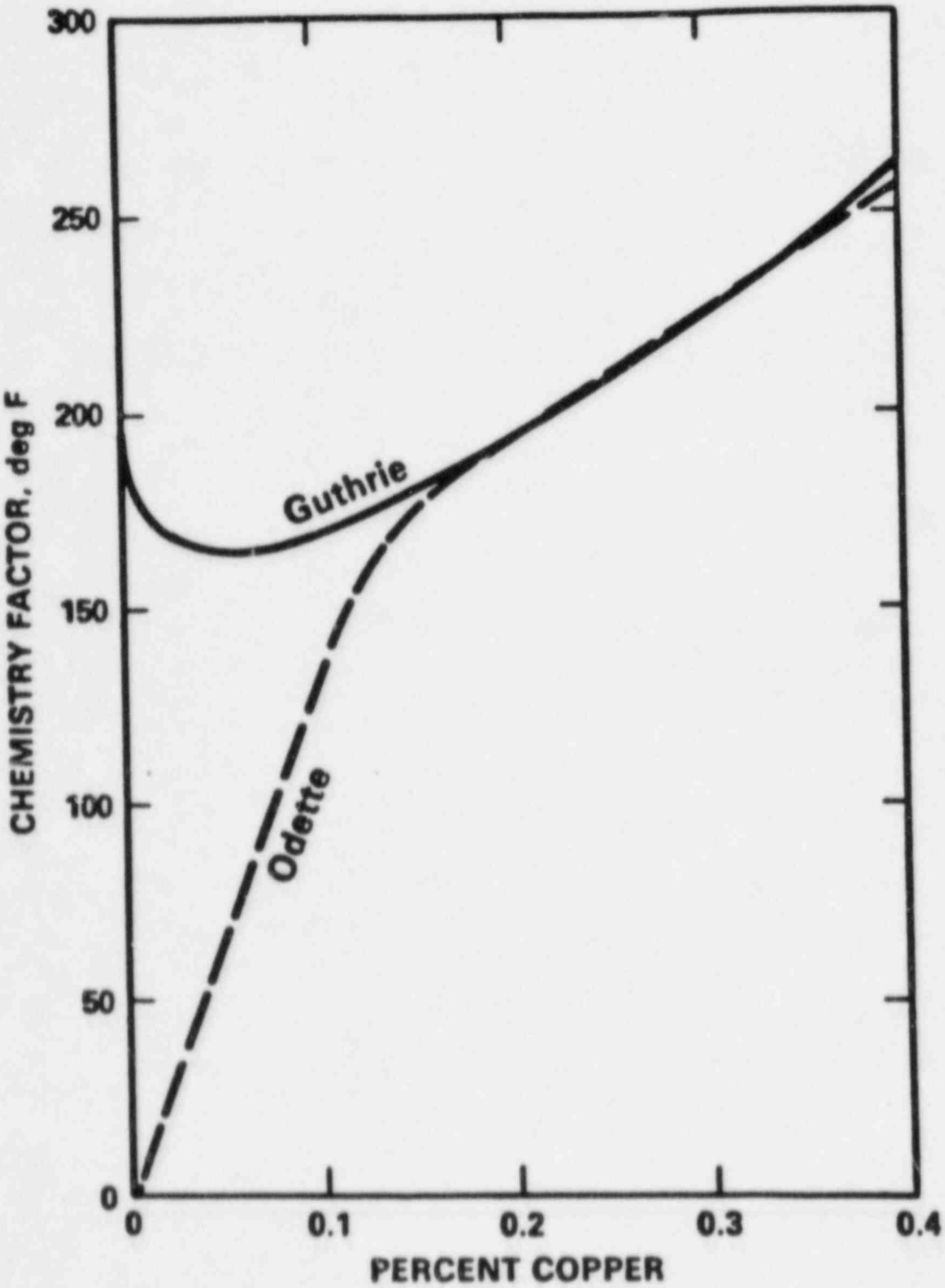

FIG. 4 COMPARISON OF GUTHRIE AND ODETTE STUDIES ON THE EFFECT OF COPPER ON THE CHEMISTRY FACTOR -- WELDS WITH 0.8 PERCENT NICKEL

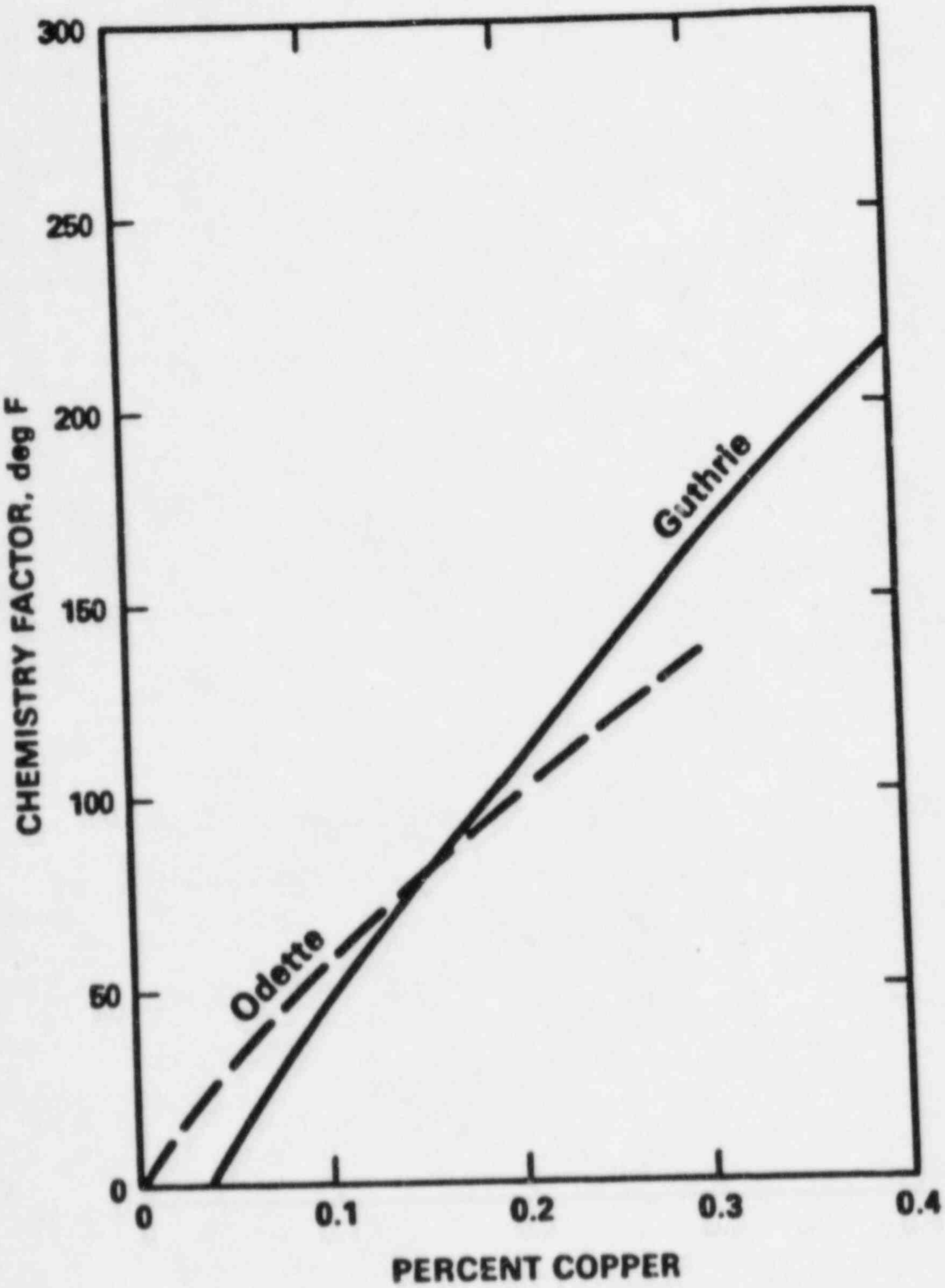

FIG. 5 COMPARISON OF GUTHRIE AND ODETTE STUDIES ON THE EFFECT OF COPPER ON THE CHEMISTRY FACTOR -- BASE METAL WITH 0.2 PERCENT NICKEL

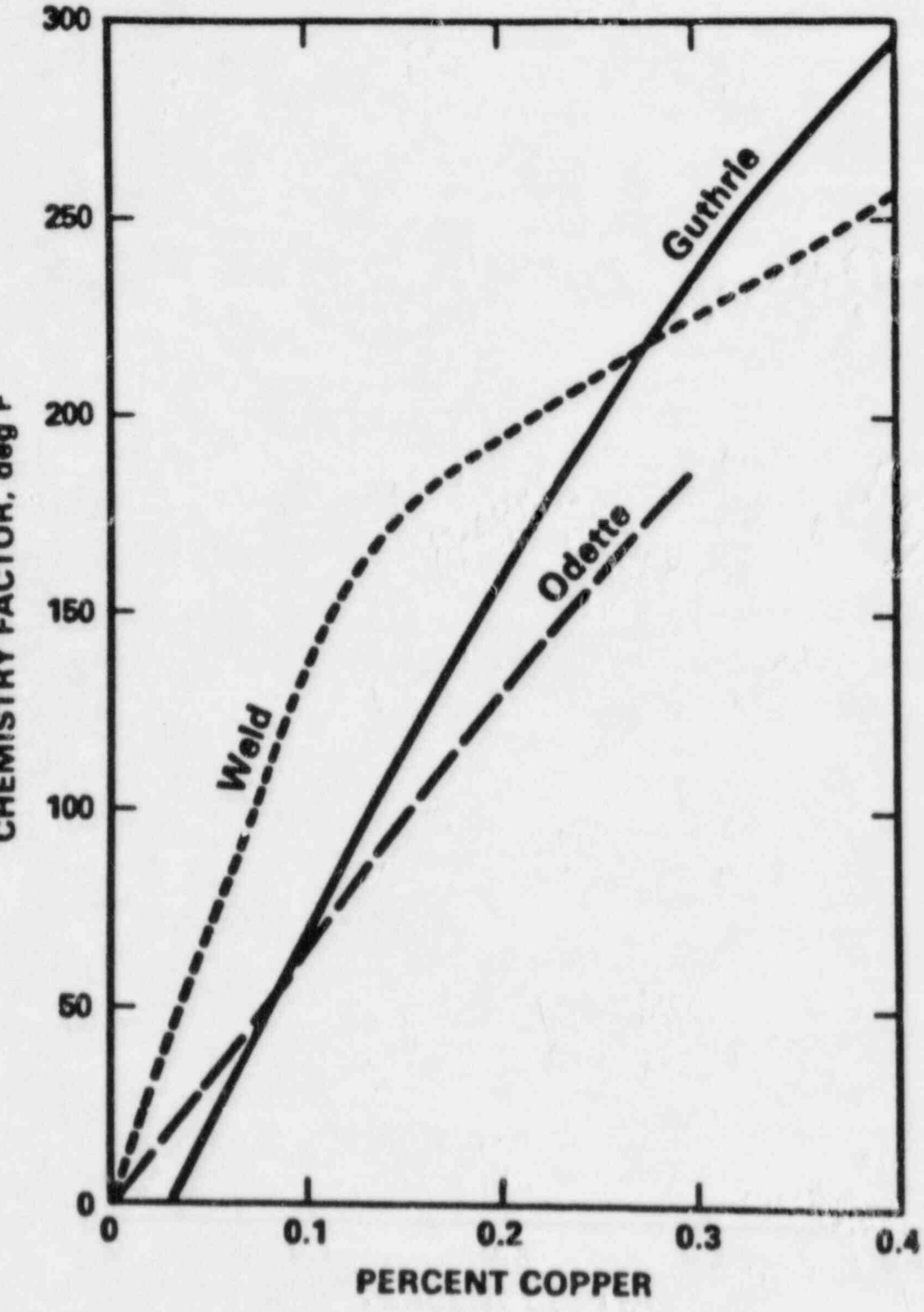

FIG. 6 COMPARISON OF GUTHRIE AND ODETTE STUDIES ON THE EFFECT OF COPPER ON THE CHEMISTRY FACTOR -- BASE METAL WITH 0.8 PERCENT NICKEL

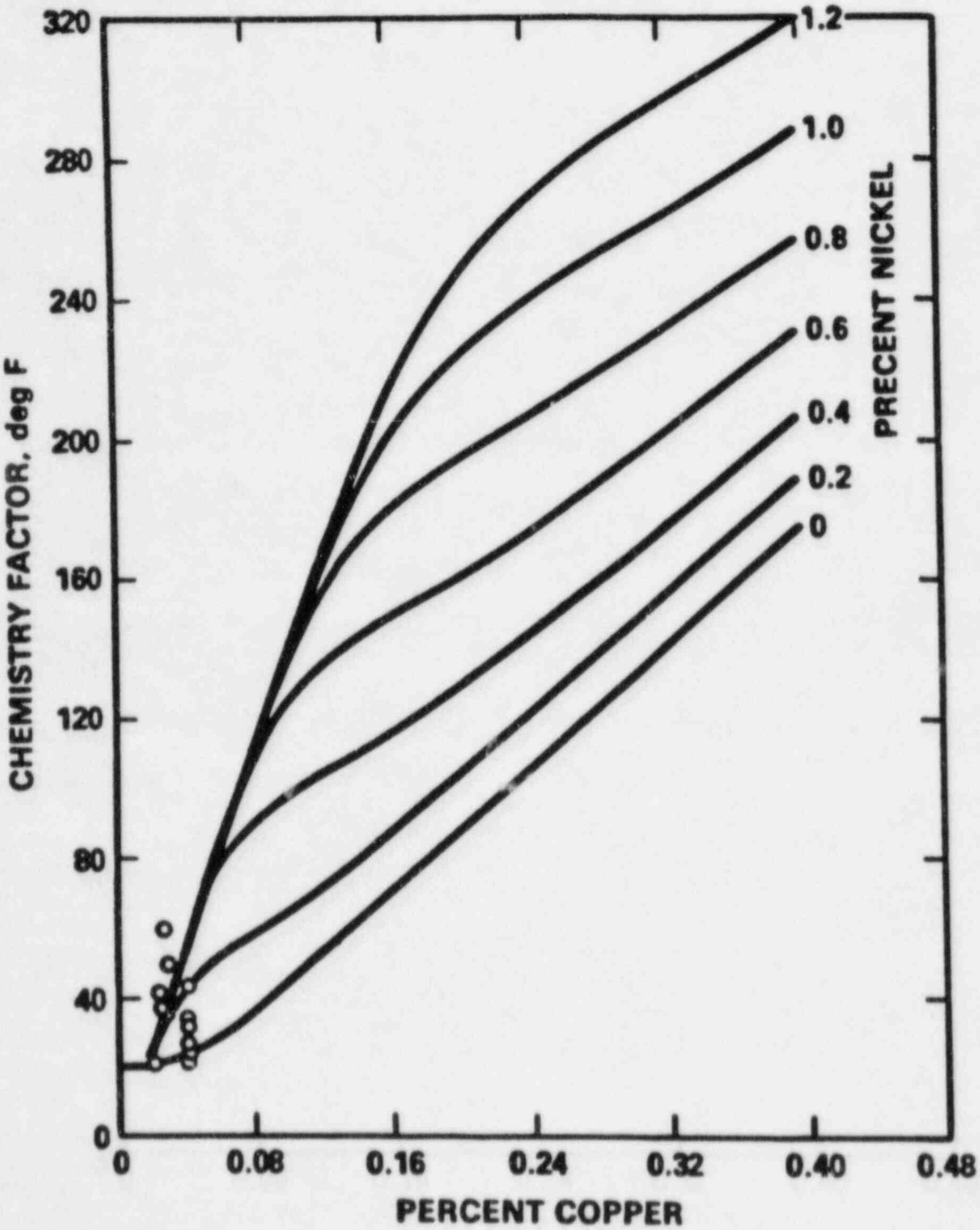

FIG. 7 TEST REACTOR DATA FOR LOW COPPER WELDS, NORMALIZED TO 10^{19}n/cm^2, COMPARED TO THE CURVES FOR CHEMISTRY FACTOR VERSUS COPPER FROM REVISION 2

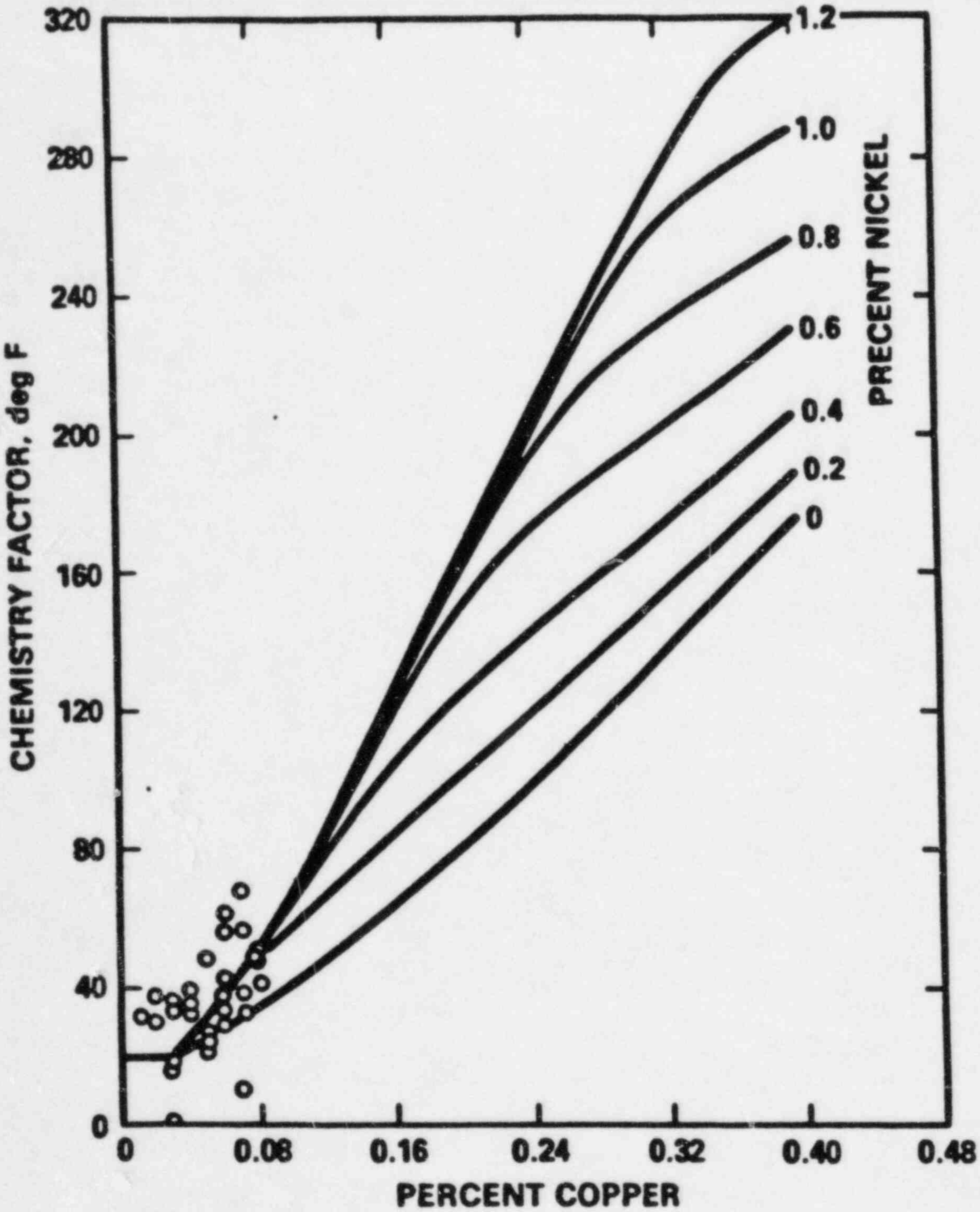

FIG. 8 TEST REACTOR DATA FOR LOW COPPER BASE METAL, NORMALIZED TO $10^{19} n/cm^2$, COMPARED TO THE CURVES FOR CHEMISTRY FACTOR VERSUS COPPER FROM REVISION 2

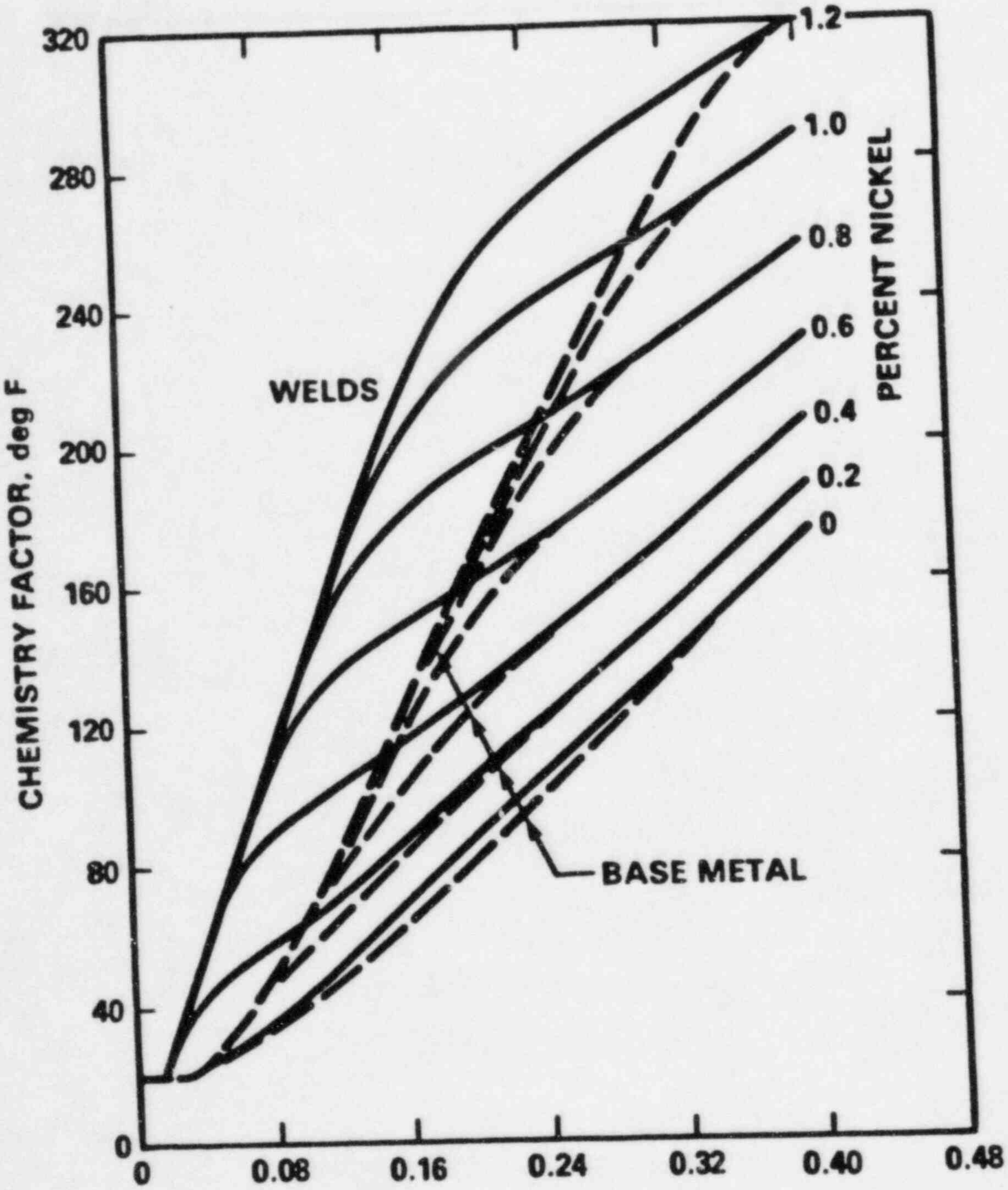

FIG. 9 COMPARISON OF CHEMISTRY FACTORS FOR WELDS AND BASE METAL, GIVEN IN REVISION 2

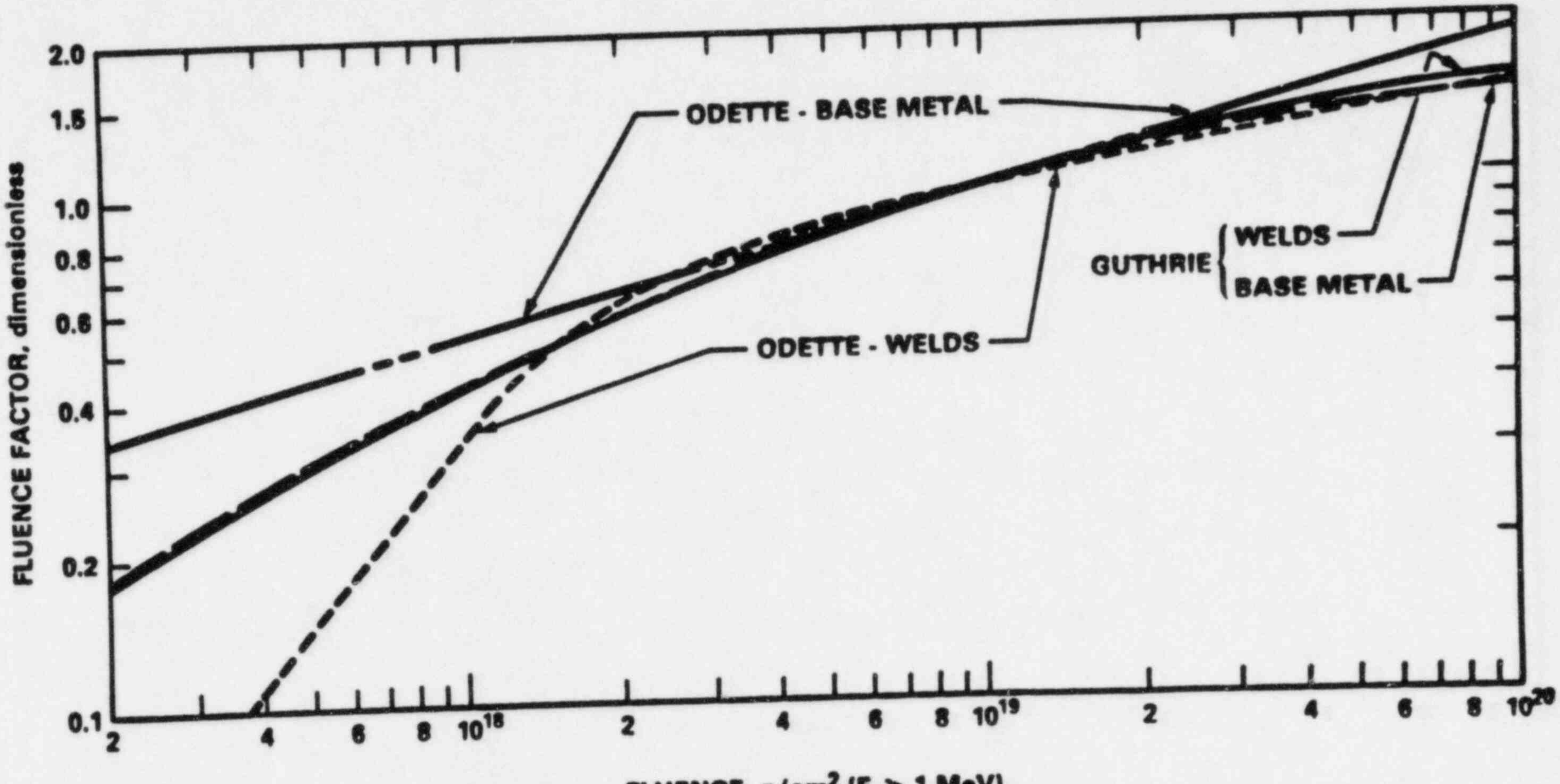

FIG. 10 COMPARISON OF FLUENCE FACTORS

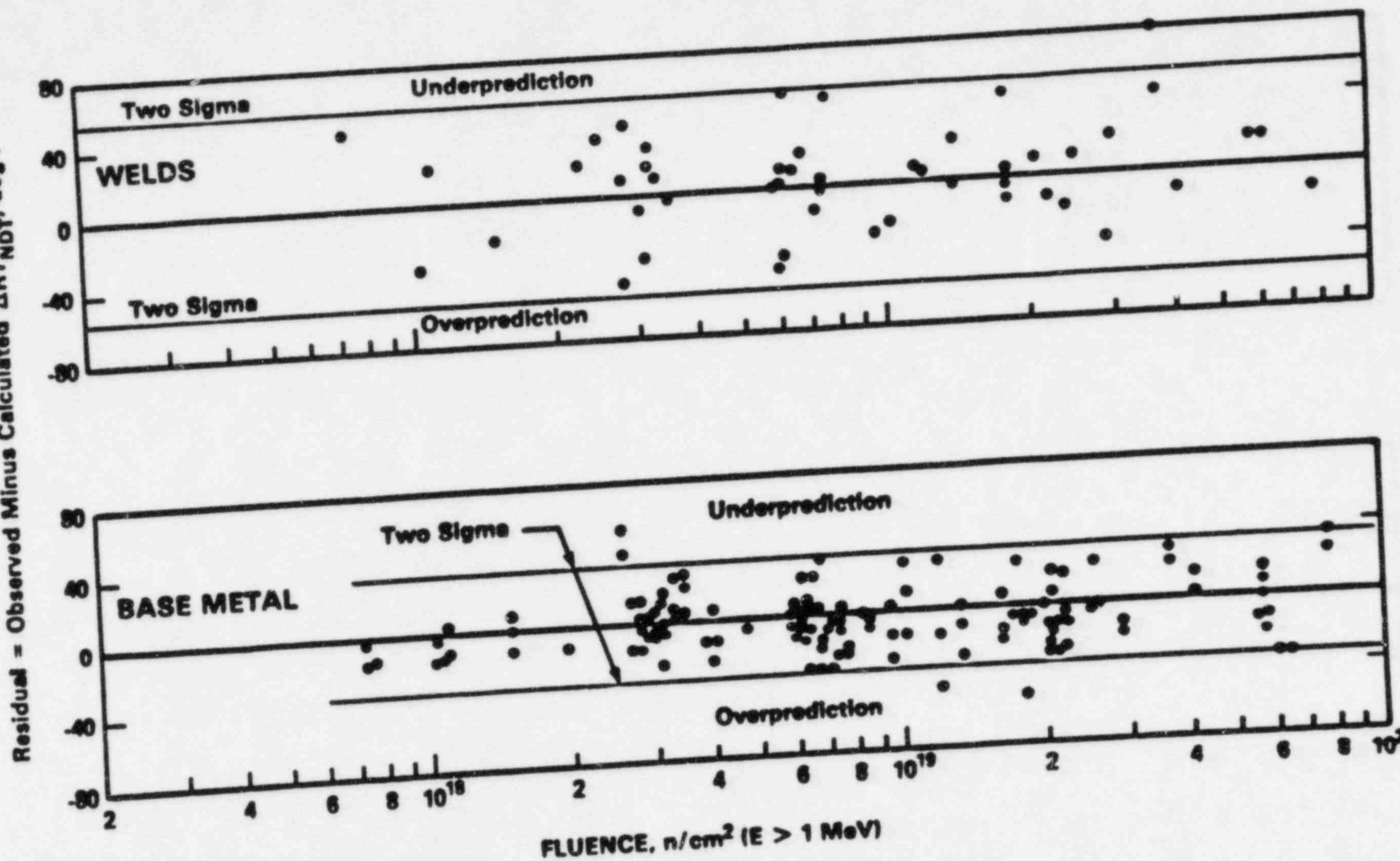

FIG. 11 PLOTS OF RESIDUALS VERSUS FLUENCE FOR 51 WELD AND 126 BASE METAL DATA POINTS

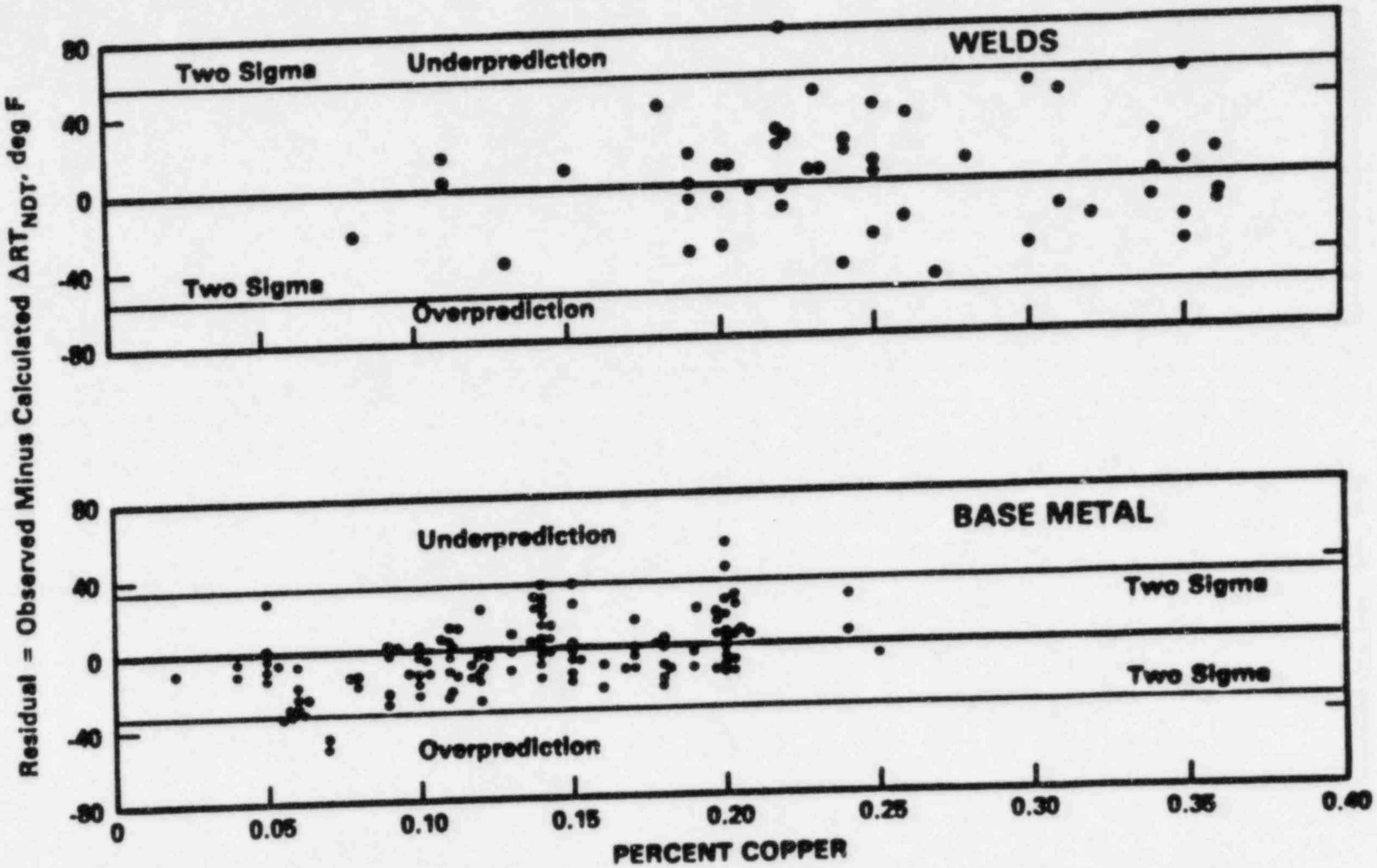

FIG. 12 PLOTS OF RESIDUALS VERSUS COPPER CONTENT FOR 51 WELD AND 126 BASE METAL DATA POINTS

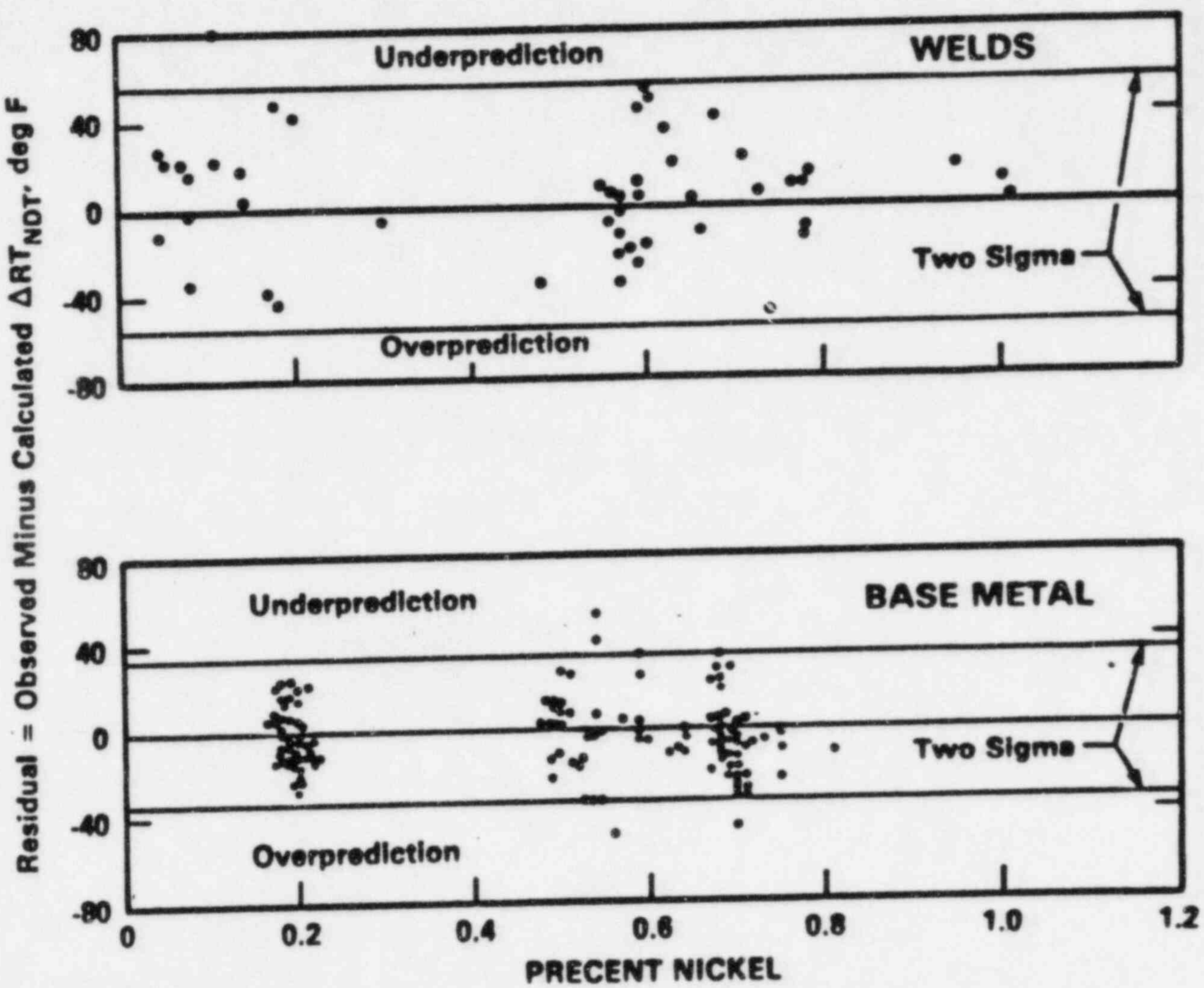

FIG. 13 PLOTS OF RESIDUALS VERSUS NICKEL CONTENT FOR 51 WELD AND 126 BASE METAL DATA POINTS

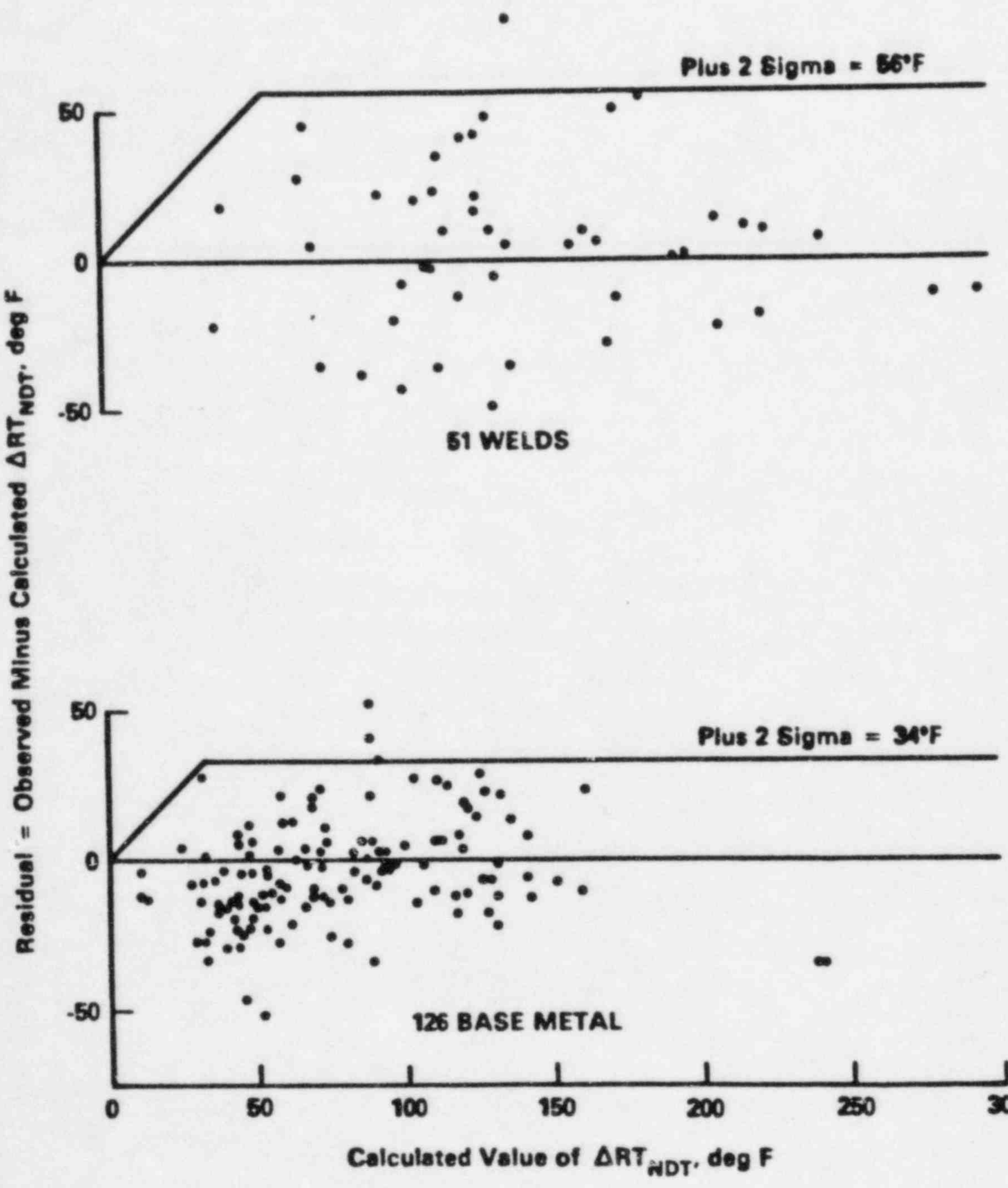

FIG. 14 PLOTS OF RESIDUALS VERSUS CALCULATED VALUE OF ΔRT_{NDT} FOR BOTH WELDS AND BASE METAL

Enclosure 2

BACKGROUND INFORMATION FOR CRGR REVIEW
OF REGULATORY GUIDE 1.99, REVISION 2

The following information is provided in the format required by the CRGR charter.

1. The proposed generic action is issuance of Revision 2 of Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials," for public comment. It is Enclosure 1 to this memorandum. The proposed implementation schedule is Section D of the Guide.

2. A staff paper giving the technical basis for the procedures for calculating the extent of radiation damage given in the Guide is Enclosure 3. Copies of the references in Enclosure 3 and in the Guide are available.

3. A brief description of each of the steps anticipated that licensees must carry out in order to comply with the recommendations of the Guide is as follows.

3.1 Are there separate short-term and long-term requirements?

When Revision 2 becomes effective, it will be used in our review of all submittals of P-T limit updates by owners of operating reactors and applicants for OLs. Revision 2 will also be used in analyses of transients that threaten the integrity of the reactor vessel beltline and in the evaluation of flaws found in the reactor vessel beltline. Licensees may continue with schedulesfor review of the P-T limits presently given in their Technical Specifications, for a maximum of three years. Within that period, those for whom the allowable operating period has been reduced or has expired when judged by the criteria of Revision 2, should revise their operating procedures and submit the appropriate revision to their Technical Specifications.

3.2 Is it the definitive, comprehensive position on the subject or is it the first of a series of requirements to be issued in the future?

Revision 2 is an update of the Revision 1 procedures for calculating the adjustment of the reference temperature caused by radiation damage. Another revision will be required to update the procedures for calculating the decrease in upper-shelf energy when the technical basis for that because available in a year or two. That change will not affect plant P-T limits. The technology for prediction of radiation damage is in such a state that further revision of the calculative procedures probably will be necessary in a few years time.

3.3 How does this requirement affect other requirements? Does this requirement mean that other items or systems or prior analyses need to be reassessed?

Paragraph 5.3.2 of the Standard Review Plan should be changed to refer to Revision 2 of Regulatory Guide 1.99. As an example of the editorial changes, the references to "copper and phosphorus" should be changed to "copper and nickel." The changes of most significance involve the use of mean values and the separate calculation of margin described in Revision 2. The proposed amendments to the Standard Review Plan are given in Enclosure 4.

Issuance of Revision 2 and the associated changes in the Standard Review Plan will eventually affect the Tech Spec P-T limits for most plants and near-term OLs.

Issuance of Revision 2 for public comment will <u>not</u> affect the procedure for calculating RT_{NDT} relative to the screening criteria given in the proposed final rule on pressurized thermal shock, SECY 85-60, February 20, 1985. That procedure is associated with screening criteria (270°F for base metal and axial welds, 300°F for circumferential welds), which were justified by a probabilistic analysis that considered all identifiable uncertainties including those in the calculation of RT_{NDT}. Only if a licensee expects a screening criterion to be exceeded 3 years hence will Revision 2 (or the then-current revision) be applied in the reanalysis of susceptibility to

pressurized thermal shock for that particular reactor vessel. Note, however, the action recommended in paragraph 4.b.1 of Enclosure 5.

3.4 Is it only computation? Or, does it require or may it entail engineering design of a new system or modification of any existing system?

It entails the computation of new P-T limits, processing a Tech Spec change, and making the resulting changes in operating procedures during heatup and cooldown. Such changes occur every few years anyway. It will also be used in the analysis of transients that affect the reactor vessel beltline and the evaluation of flaws found in inservice inspection of the beltline.

3.5 There are no plant hardware changes involved in issuance of Revision 2.

4. Revision 2 will apply to all owners of operating reactors and to all applicants for an operating license.

5. In the regulatory analysis (Enclosure 5), Table 1 gives a breakdown of the benefits and impacts to operating reactors and applicants for OLs, and to PWRs and BWRs.

5.1. Risk reduction assessments and cost assessments given in Enclosure 5 are based on a value/impact analysis made by Pacific Northwest Laboratories (Enclosure 6) and a review of the costs by the NRC's Cost Analysis Group (Enclosures 7 and 8).

5.2. The basis for requiring implementation as described in Section D of the Guide is given in the Regulatory Analysis (Enclosure 5), Sections 5 and 6. Section D states in effect that Revision 2 will be used in evaluating all predictions of radiation damage submitted after Revision 2 becomes effective. However, as given in § 3.1, the implementation schedule will require that every utility review its pressure-temperature limits by 3 years after Revision 2 becomes effective. The reasons for doing so are discussed in the Regulatory Analysis, Enclosure 5.

5.3 The schedule for staff actions to get Revision 2 published in final form will involve: ACRS review, final editing, a 2-month public comment period, resolution of comments and redrafting of the final version, resubmittal to CRGR, and publication in final form.

5.4 Implementation of Revision 2 will crowd the normal calendar for review of P-T limits somewhat, especially toward the end of the 3-year period described in § 3.1, but the NRC staff schedule will not be impacted severely.

6. Each proposed requirement implements existing regulations, namely General Design Criterion 31 and Appendices G and H, 10 CFR Part 50.

7. Section D of Revision 2 describes the implementation. To ensure a proper and timely response, copies of the Guide when issued in final form will be sent to each licensee and applicant for an operating license as part of the distribution of Revision 2. The concurrence of OELD and DL was obtained when the completed package was sent for Office concurrence to transmit to CRGR.

8. The OMB clearance package when required under the Paperwork Reduction Act, and the Regulatory Flexibility Statement are not required. Regulatory Guide 1.99 does not impose reporting requirements. It describes acceptable procedures for fulfilling certain requirements of Appendices G and H, 10 CFR Part 50 for which OMB clearance has been obtained and for which the Commission has certified that it will not have a significant impact on a substantial number of small entities (Federal Register Vol. 48, Number 104, May 27, 1983, p. 24008).

Enclosure 4

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

See changes marked with a horizontal tic, - on pages 5.3.2-2
-3
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-17

5.3.2 PRESSURE-TEMPERATURE LIMITS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

1. Pressure-Temperature Limits

The regulations requiring the imposition of pressure-temperature limits on the reactor coolant pressure boundary are the following:

Paragraph 50.55a of 10 CFR Part 50, "Codes and Standards," requires that structures, systems, and components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. In addition, General Design Criterion 1 of Appendix A of 10 CFR Part 50, "Quality Standards and Records," requires that the codes and standards used to assure quality products in keeping with the safety function be identified and evaluated to determine their adequacy.

General Design Criterion 14 of Appendix A of 10 CFR Part 50, "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. Likewise, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance and testing, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. Further, in order to assess the structural integrity of the reactor vessel, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires, in part, an appropriate materials surveillance program for the reactor vessel beltline region.

The pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed in this section of the Standard Review Plan (SRP) to assure adequate safety margins of structural integrity for the ferritic components of the reactor coolant pressure boundary.

II. ACCEPTANCE CRITERIA

The requirements of paragraph 50.55a and General Design Criteria 1, 14, 31 and 32 of Appendix A of 10 CFR Part 50 are met by the assurance that material of the reactor coolant pressure boundary possess adequate fracture toughness properties to resist rapidly propagating failure and act in a nonbrittle manner when stressed under operating, maintenance, testing, and anticipated operational conditions. The requirement, in part, of General Design Criterion 32 is met by conducting a surveillance program to monitor the change in fracture toughness properties of the ferritic materials in the reactor vessel.

The fracture toughness requirements for ferritic materials in the pressure-retaining components of the RCPB are specified for testing and operational conditions, including anticipated operational occurrences, in Section IV of Appendix G of 10 CFR Part 50. This appendix requires the acceptance and performance criteria of Appendix G of Section III of the ASME Boiler and Pressure Vessel Code. Pressure-temperature calculation procedures are described in Appendix G of the ASME code; while the detailed technical basis for the ASME code requirement is provided by the Welding Research Council (WRC) Bulletin 175, "PVRC Recommendation on Toughness Requirements for Ferritic Materials." Changes in the fracture-toughness properties of materials in the beltline region, resulting from neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance to the requirements of Appendix H of 10 CFR Part 50. The effect of neutron fluence on the shift in the nil-ductility temperature of pressure vessel steel is predicted by Regulatory Guide 1.99, Revision 2, "~~Effect of Residual Elements on Predicted~~" Radiation Damage to Reactor Vessel Materials."

1. Applicable Regulations, Codes, and Basis Documents

Appendices G and H of 10 CFR Part 50 describe the conditions that require pressure-temperature limits and provide the general basis for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in the ASME Boiler and Pressure Vessel Code (hereinafter "the Code"), Section III, Appendix G, "Protection Against Nonductile Failure," during heatup, cooldown, and test conditions. Appendix G to 10 CFR Part 50 also requires additional safety margins whenever the reactor core is critical (except for low-level physics tests).

2. Technical Bases

Since many of the fracture toughness requirements for the ferritic materials in the pressure-retaining components were not required at the time some of the reactor facilities were designed and constructed, the Materials Engineering Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," describe procedures for making estimates and assumptions on the fracture toughness properties of materials in the older plants. Calculations are required, and an evaluation is made by the reviewer to show compliance with the regulations and to show an adequate margin of

quality and safety for the facility. When it has been determined that certain requirements of Appendices G or H have not been strictly complied with by these older plants, and when it has been determined that an equivalent level of quality and safety, as required by the regulations exist, then exemption to the specific requirements of these appendices will be granted by the Commission.

a. The principles of linear elastic fracture mechanics (LEFM) are used to determine safe operational conditions. The basic parameter of LEFM is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. An analytical method is used to determine the effects of real or postulated flaws. The minimum K_I that can cause failure is defined as the critical stress intensity factor, K_{Ic}, and is the material parameter used in this method. The K_{Ic} of the material is either directly measured as a function of temperature, or is conservatively estimated, using information from other fracture toughness tests.

b. The Code specifies the maximum K_{Ic}, as a function of temperature, that can be assumed for the specific material, based on results of tests on the material used. This value is called K_{IR}, reference stress intensity factor. The Code also provides rules for calculating K_I, including definitions of postulated flaws, and specifies the safety factors to be applied. The acceptance criterion is that the K_{IR} of the material must always be higher than the K_I calculated.

c. Direct measurement of the K_{Ic} as a function of temperature is expensive and time consuming and requires more sample material than is usually available. Correlations between the K_{Ic} determined directly and results of simpler fracture toughness tests are not exact, but may be used if appropriate allowances are made for variations in material behavior and data scatter. The Code gives values of K_{IR} as a function of temperature relative to a conservative determination of the reference temperature of the material. This reference temperature, RT_{NDT}, is determined for the ferritic materials of components for which operating and testing limit curves must be calculated. The effects of radiation on the fracture toughness of the material in the beltline region of the reactor vessel is accounted for by adjusting the RT_{NDT} of the affected material upward. The amount of upward shift depends on the composition of the steel (especially its copper and ~~phosphorous~~ nickel content), and the neutron fluence. Conservative predictions of the effect of radiation on RT_{NDT} ~~based on data in Regulatory Guide 1.99 are factored into the original limit curves~~ are achieved by adding margin to the best estimates as described in Regulatory Guide 1.99, Revision 2. The continued conservatism of these predictions throughout plant life is verified by a mandatory material surveillance program described in Appendix H of 10 CFR Part 50.

d. The Code specifies the stress components that must be used for the K_I calculations, and the factors that must be applied to each to

provide adequate safety margins. The Code, by reference to WRC-175, specifies the expression to use for calculating the K_I, using the applied stresses and the postulated flaw geometry. Although calculations are usually made by a computer, curves are provided in the Code to facilitate the use of conservative hand calculations if desired.

3. Pressure-Temperature Requirements

The requirements for the pressure-temperature limits are as follows:

a. Pressure-Temperature Limits for Preservice Hydrostatic Tests

During preservice hydrostatic tests (if fuel is not in the vessel), the K_{IR} must be greater than the K_I caused by pressure. The expression used is:

$$K_I = K_I(\text{pressure}) < K_{IR}$$

b. Pressure-Temperature Limits for Inservice Leak and Hydrostatic Tests

During performance of inservice leak and hydrostatic tests, the K_{IR} must be greater than 1.5 times the K_I caused by pressure. The expression used is:

$$K_I = 1.5\ K_I(\text{pressure}) < K_{IR}$$

c. Pressure-Temperature Limits for Heatup and Cooldown Operations

At all times during heatup and cooldown operations, the K_{IR} must be greater than the sum of 2 times the K_I caused by pressure and the K_I caused by thermal gradients. The expression used is:

$$K_I = 2K_I(\text{pressure}) + K_I(\text{thermal}) < K_{IR}$$

d. Pressure-Temperature Limits for Core Operation

At all times that the reactor core is critical (except for low power physics tests) the temperature must be higher than that required for inservice hydrostatic testing, and in addition, the pressure-temperature relationship shall provide at least a 40°F margin over that required for heatup and cooldown operations.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. Preliminary Safety Analysis Report (PSAR)

Information in the PSAR is reviewed for a commitment that the fracture toughness of the ferritic materials in the reactor coolant pressure boundary will comply with the requirements of Appendix G of 10 CFR Part 50,

as detailed in Section III of the ASME Boiler and Pressure Vessel Code and that the materials in the beltline region of the reactor vessel will comply with the requirements of Appendices G and H of 10 CFR Part 50 and Regulatory Guide 1.99, Revision 2 ~~"Effects of Residual Elements on~~ "Predicted Radiation Damage to Reactor Vessel Materials."

2. <u>Final Safety Analysis Report (FSAR)</u>

The limits in the plant Technical Specifications will be shown using real temperature. These curves and their bases are reviewed to determine acceptability in the following areas:

a. The limiting RT_{NDT} has been properly determined, and radiation effects are included in a conservative manner.

b. Limits are shown for all required conditions.

c. The limits proposed are consistent with the acceptance criteria described in II. above.

d. The procedures for updating the limit curves, in conjunction with scheduled tests on material surveillance specimens, are well defined and included in the Technical Specifications.

3. <u>Acceptability Determination Methods</u>

The reviewer evaluates each limit curve for acceptability by performing check calculations using the simplified methods referenced in the Code and WRC Bulletin 175 that have been verified by the Materials Engineering Branch to yield conservative values. These methods are described in detail by examples below, and the curves necessary to perform the calculations are included herein as Figures 1, 2 and 3.

a. Preservice Hydrostatic Tests

The preservice hydrotest at 1.25 design pressure corresponds to the standard Code component hydrotest usually performed in the shop, but in this case it is the hydrotest for field welds, so it may involve the entire reactor coolant system.

The Code recommends that component hydrostatic tests be run at a temperature no lower than RT_{NDT} + 60°F, but also recommends that system tests should have more stringent requirements. The MTEB position is that the minimum temperature for the preservice test, if fuel is not in the vessel, be determined using the methods of Code Section III, Appendix G, using less stringent factors.

First, the RT_{NDT} of the vessel material must be determined. This is defined by the Code for new plants, and is essentially a conservative value of the NDTT as determined by drop weight test. Guidelines for estimating the RT_{NDT} if the prescribed tests have not been run are given by Branch Technical Position - MTEB 5-2, "Fracture Toughness Requirements." Technical justification for all estimates of RT_{NDT} must be provided by the applicant.

The toughness of the material is a function of the difference between the RT_{NDT} of the material and the temperature of interest. The Code provides a curve (Figure G-2210.1) for the allowable calculated stress intensity factor (K_{IR}) as a function of the temperature relative to RT_{NDT}. Refer to Figure 2 herein.

The Code also provides a recommended basis for calculating K_I, including recommendations for assumed flaw size and shape, and appropriate front and back surface correction factors. Because the assumed flaw size is proportional to the wall thickness, t (flaw depth = 0.25 t and length = 1.5 t), the K_I expressions are simplified to multiples that are a function only of wall thickness and stress level. These factors, M_m for membrane stresses and M_B for bending stresses, are provided in graphical form in Figure G-2214.1 of the Code. Refer to Figure 1 herein.

The criterion recommended by MTEB can be expressed as

$$K_I < K_{IR} \text{ for the shell region.}$$

To get K_I, the stress level and wall thickness must be known. The pressure for the hydrostatic test is 1.25 times the design pressure, so that the higher of two simple methods described below to approximate the membrane stress should be accurate enough for this purpose:

$$\text{stress} = 1.25 \text{ times the Code allowable } (S_m)$$

$$\text{stress} = \frac{Pr}{t}$$

where P is the test pressure and r is the vessel radius. As an example, assume a vessel with a design pressure of 2500 psig, made of steel with an S_m of 26,700 psi, and a minimum yield strength of 50,000 psi. The stress for the preservice hydrotest is then

$$26{,}700 \times 1.25 = 33{,}400 \text{ psi, or}$$

$$\frac{(1.25)\ (2500)\ (95)}{9} = 33{,}400 \text{ psi, for a vessel with a radius of 95 inches and a wall thickness of 9 inches.}$$

The next step is to determine the factor to apply to this stress to obtain K_I. Figure G-2214.1 (reproduced here as Fig. 1) provides several curves, depending on the ratio of the stress level to the yield strength of the material. In this case, the stress level is 33,400; the yield strength is conservatively assumed to be 50,000 so the curve for a ratio of .7 should be used. (A ratio equal to or higher than the actual ratio must be used for conservatism.) For a 9-in. thick vessel ($\sqrt{t} = 3$), the value of M_m from Figure G-2214.1 is 2.94. The K_I for this case is then:

$K_I = (M_m)$ (Membrane Stress)

$K_I = (2.94)(33{,}400) = 98{,}300$ psi $\sqrt{in.}$

From Figure G-2210.1 (reproduced here as Fig. 2), a temperature of at least RT_{NDT} + 120°F is necessary for a K_I of this level.

If, for example, an original RT_{NDT} of 40°F is assumed, the required temperature is then 40 + 120, or 160°F.

b. Inservice Leak and Hydrotest.

The temperatures for the inservice leak and hydrotest, performed at operating pressure and about 1.1 operating pressure, respectively, are calculated in essentially the same way. The differences are that a factor of 1.5 must be applied to the calculated K_I to provide extra margin, and the stress levels are lower, so the value of M_m is taken from a lower ratio curve.

Using the same vessel as an example, with a normal operating pressure (P_o) of 2250 psi, the membrane stress for the leak test can be approximated as:

$$\frac{\text{operating pressure}}{\text{design pressure}} \times \text{allowable stress}$$

$$\text{or } \frac{2250}{2500} \times 26{,}700 = 24{,}000 \text{ psi}$$

This is about half of the minimum yield strength, so the M_m is taken from the 0.5 ratio curve, and is 2.87. The calculated K_I that must be assumed is then:

$K_I = (1.5)(M_m)$ (Membrane Stress)

or $K_I = (1.5)(2.87)(24{,}000) = 103{,}500$ psi $\sqrt{in.}$

From the K_{IR} curve, a temperature of about RT_{NDT} + 125°F is required. As this is an inservice test, the RT_{NDT} would probably have been increased from its original value of + 40°F by some shift caused by radiation. Assume this shift is 100°F, thus the temperature for the leak test must be at least:

40 + 100 + 125 = 265°F

The inservice hydrotest temperature (at 1.1 P_o) is determined in exactly the same way, and requires a minimum temperature of about RT_{NDT} + 133°F, or 273°F.

c. Heatup, Cooldown, and Normal Operation

For normal operation, which includes upset conditions and startup and shutdown procedures, operating limit curves must be provided

that show the maximum permissible pressure at any temperature from cold shutdown conditions to full pressurization conditions.

Reactor vendors have developed computer codes to perform the necessary calculations, because thermal stresses must be included, and hand calculations of even moderate sophistication are very time consuming. WRC Bulletin 175 includes a set of curves derived from computer programs that can be used to approximate the K_I caused by thermal stresses, as a function of wall thickness and rate of temperature change. Pressure-temperature curves developed using these approximations agree fairly well with those determined using much more rigorous procedures, and can be used with confidence to evaluate the proposed operating limits given in Technical Specifications. These curves require the calculation of only 3 to 5 points. Either allowable pressure at a given temperature, or allowable temperature at a given pressure can be calculated. It is usually more convenient to calculate allowable minimum temperature, so this method will be used in the example.

Using the same reactor vessel as in the previous example, and a rate of temperature change of 50°F per hour, calculations of required temperatures for several pressures are illustrated. The curves for thermal effects given in WRC Bulletin 175 are very conservative, thus no additional margin need by applied to the K_I from thermal stress, but a factor of 2.0 is used on primary stresses. The basic expression is then:

$$K_{IR} \geq 2\ K_I(\text{membrane}) + K_I\ (\text{thermal})$$

K_I(membrane) is calculated exactly as in the previous examples. K_I(thermal) for a 9-in. thick wall, at 50°/hr is about 12,000 psi $\sqrt{\text{in.}}$ from Figure 4-5, WRC Bulletin 175 (reproduced here as Fig. 3).

Thus, for a pressure of 2250 psig, a membrane stress of 24,000 psi, and M_m of 2.87, the basic expression is given by

$$K_{IR} > (2)(24{,}000)(2.87) + 12{,}000 = 150{,}000 \text{ psi } \sqrt{\text{in.}}$$

From the K_{IR} curve, a temperature of RT_{NDT} + 158°F is required. With an RT_{NDT} of 140°F (including irradiation effects), the temperature required for operating pressure at a heatup or cooldown rate of 50°/hr is then

$$140 + 158 = 298°\text{F}$$

For a pressure of 1/2 of operating (1125 psig), the membrane stress is 1/2 of that at operating pressure, or 12,000 psi.

The M_m can be taken from the 0.5 $\frac{\sigma}{\sigma y}$ ratio curve in Figure G-2214.1 (reproduced as Figure 1 herein), so is again 2.87.

$$K_{IR} \geq (2)(12{,}000)(2.87) + 12{,}000 = 81{,}000 \text{ psi } \sqrt{\text{in.}}$$

From the K_{IR} curve, the minimum temperature is RT_{NDT} + 100°F, or 140 + 100 = 240°F.

The same calculation for a pressure of 1/5 operating pressure (450 psig and 4800 psi stress) is similar, but in this case the stress is less than .1 of the yield strength, so the M_m (from the .1 ratio curve) is only 2.82.

$$K_{IR} \geq (2)(4800)(2.82) + 12,000 = 39,000 \text{ psi } \sqrt{\text{in.}}$$

The K_{IR} curve shows that the minimum temperature is RT_{NDT} + 0°F, or 140°F.

Three points on a 50°/hr operating limit curve for this vessel at this time in its service lifetime have thus been calculated:

Pressure (psig)	Min. Temperature (Fahrenheit)
450	140
1150	240
2250	298

A smooth curve drawn through these points will very closely approximate the results using more rigorous methods.

d. Core Operation

Appendix G, 10 CFR Part 50, specifies pressure-temperature limits for core operation to provide additional margin during actual power production.

The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum pressure-temperature curve for heatup and cooldown calculated as described in the preceding section. The minimum temperature for the inservice hydrostatic test for the vessel used in the preceding example was 273°F. A vertical line at 273°F on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve as determined in the preceding section, constitutes the limit for core operation for this example.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this SRP section and that the completeness and technical adequacy of his evaluation will support the following statement in the staff's safety evaluation report:

The pressure-temperature limits imposed on the reactor coolant system for all operating and testing conditions to assure adequate safety margins against nonductile or rapidly propagating failure are

in conformance with the fracture toughness criteria of Appendix G of 10 CFR Part 50 and Section III, including Appendix G, "Protection Against Nonductile Failure," of the ASME Boiling and Pressure Vessel Code. The change in fracture toughness requirements of the pressure vessel during operation will be determined by Appendix H of 10 CFR Part 50. The use of operating limits, based upon the criteria defined in Standard Review Plan Section 5.3.2, provides reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the requirements of paragraph 50.55a of 10 CFR Part 50 and General Design Criteria 1, 14, 31 and 32 of Appendix A of 10 CFR Part 50.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plan to using this SRP section.

Except in those cases in which the applicant proposed an acceptable alternative method for complying with specific portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guide.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criteria 1, 14, 31, and 32.

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

3. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

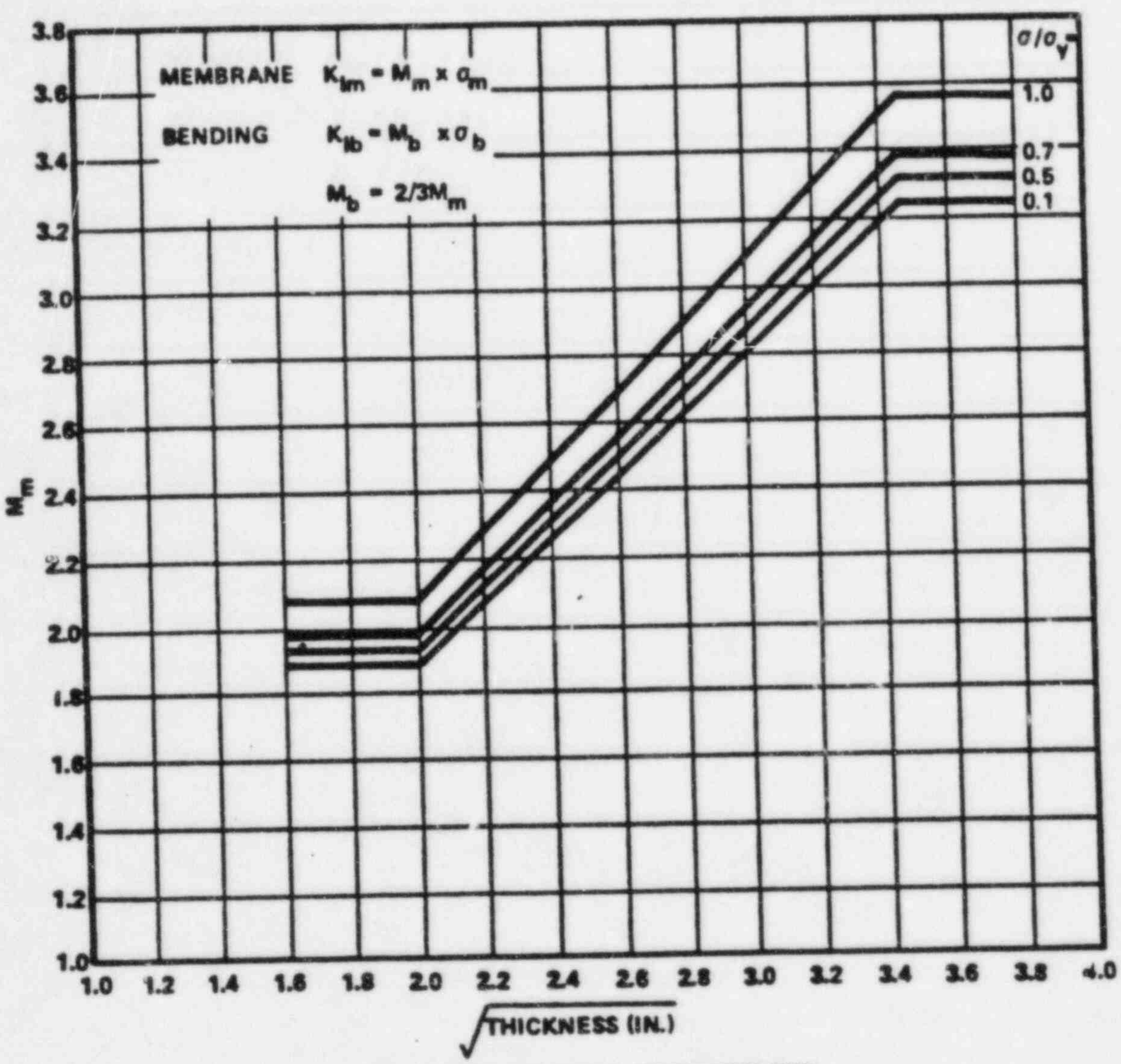
4. ASME Boiler and Pressure Vessel Code, Section III, including Appendix G, "Protection Against Nonductile Failure," American Society of Mechanical Engineers.

5. WRC Bulletin 175, "PVRC Recommendation on Fracture Toughness," Welding Research Council.

6. Branch Technical Position - MTEB 5-2, "Fracture Toughness Requirements for Older Plants," attached to this SRP section.

7. Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

8. 10 CFR Part 50, paragraph 50.55a, "Codes and Standards."



M_m AND M_b VS. WALL THICKNESS FOR

SEMI-ELLIPTICAL SURFACE FLAW ¼T DEEP AND 1½T LONG

FIGURE 1

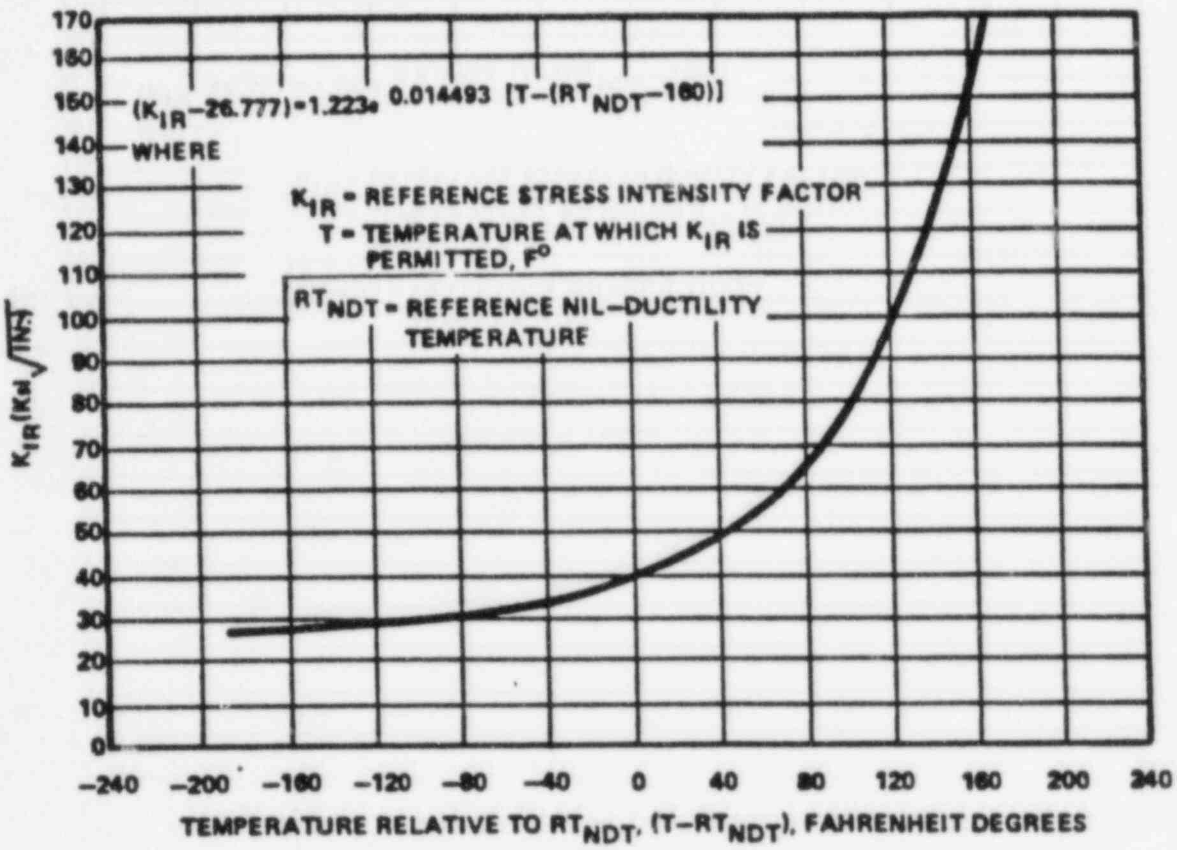
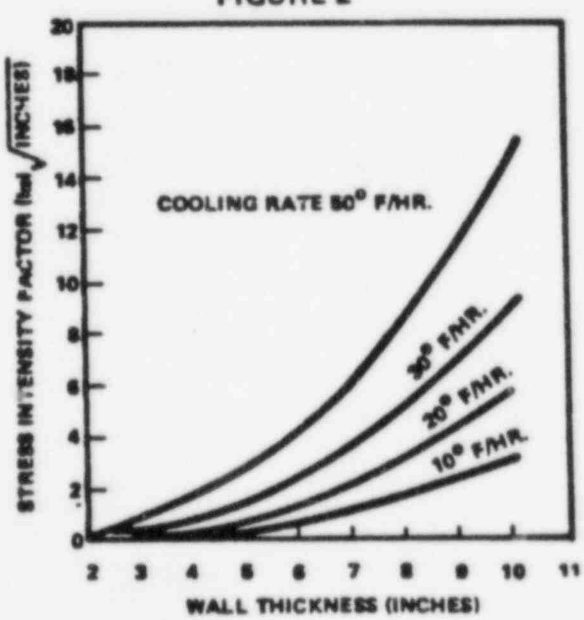


FIGURE 2



4-5 – STRESS INTENSITY FACTOR CAUSED BY THERMAL STRESS FOR CYLINDERS WITH RADIUS/THICKNESS = 10

FIGURE 3

BRANCH TECHNICAL POSITION - MTEB 5-2
FRACTURE TOUGHNESS REQUIREMENTS

A. Background

Current requirements regarding fracture toughness, pressure-temperature limits, and material surveillance are covered by the ASME Code and Appendices A, G, and H to 10 CFR Part 50. The purpose of this branch technical position is to summarize these requirements and provide clarification, as necessary.

Since many of these requirements were not in force when some plants were designed and built, this position also provides guidance for applying these requirements to older plants. Also included is a description of acceptable procedures for making the conservative estimates and assumptions for older plants that may be used to show compliance with the new requirements. It should be noted that the applicants must present adequate technical justifications for any estimates of material properties required by the regulations before exemption to the regulations may be granted.

B. Branch Technical Position

1. Preservice Fracture Toughness Test Requirements.

The fracture toughness of all ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary shall be evaluated in accordance with the requirements of Appendix G, 10 CFR Part 50, as augmented by Section III of the ASME Code. The fracture toughness test requirements for plants with construction permits prior to August 15, 1973 may not comply with the new Codes and Regulations in all respects. The fracture toughness of the materials for these plants must be assessed by using the available test data to estimate the fracture toughness in the same terms as the new requirements. This must be done because the operating limitations imposed on old plants must provide the same safety margins as are required for new plants.

1.1 Determination of RT_{NDT} for Vessel Materials

Temperature limitations are determined in relation to a characteristic temperature of the material, RT_{NDT}, that is established from results of fracture toughness tests. Both drop weight NDTT tests and Charpy V-notch tests must be run to determine the RT_{NDT}. The NDTT temperature, as determined by drop weight tests (ASTM E-208) is the RT_{NDT} if, at 60°F above the NDTT, at least 50 ft-lbs of energy and 35 mils lateral expansion are obtained in Charpy V tests on specimens oriented in the weak direction (traverse to the direction of maximum working).

In most cases, the fracture toughness testing performed on vessel material for older plants did not include all tests required to determine the RT_{NDT} in this manner. Acceptable estimation methods for the most common cases, based on correlations of data from a large number of heats of vessel material, are provided for guidance in determining RT_{NDT} when measured values are not available.

(1) If dropweight tests were not performed, but full Charpy V-notch curves were obtained, the NDTT for SA-533 Grade B, Class 1 plate and weld material may be assumed to be the temperature at which 30 ft-lbs was obtained in Charpy V-notch tests, or 0°F, whichever was higher.

(2) If dropweight tests were not performed on SA-508, Class II forgings, the NDTT may be estimated as the lowest of the following temperatures:

(a) 60°F.

(b) The temperatures of the Charpy V-notch upper shelf.

(c) The temperature at which 100 ft-lbs was obtained on Charpy V-notch tests if the upper-shelf energy values were above 100 ft-lbs.

(3) If transversely-oriented Charpy V-notch specimens were not tested, the temperature at which 50 ft-lbs and 35 mils LE would have been obtained on traverse specimens may be estimated by one of the following criteria:

(a) Test results from longitudinally-oriented specimens reduced to 65% of their value to provide conservative estimates of values expected from transversely oriented specimens.

(b) Temperatures at which 50 ft-lbs and 35 mils LE were obtained on longitudinally-oriented specimens increased 20°F to provide a conservative estimate of the temperature that would have been required to obtain the same values on transversely-oriented specimens.

(4) If limited Charpy V-notch tests were performed at a single temperature to confirm that at least 30 ft-lbs was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 45 ft-lbs was obtained if the specimens were longitudinally oriented. If the minimum value obtained was less than 45 ft-lbs, the RT_{NDT} may be estimated as 20°F above the test temperature.

1.2 Estimation of Charpy V Upper-Shelf Energies

For the beltline region of reactor vessels, the upper shelf toughness must be adequate to accommodate degradation by neutron radiation. The original minimum shelf energy must be 75 ft-lbs for vessels with an estimated end of life neutron fluence (> 1 MeV) of 1×10^{19} and over. A value of 70 ft-lbs is considered adequate for material for vessels that will be subjected to lower fluences.

If upper-shelf Charpy energy values were not obtained, conservative estimates should be made using results of tests on specimens from the first surveillance capsule removed.

If tests were only made on longitudinal specimens, the values should be reduced to 65% of the longitudinal values to estimate the transverse properties.

1.3 Reporting Requirements

Fracture toughness information required by the Code and by Appendix G, 10 CFR Part 50, must be reported in the FSAR to provide a basis for evaluating the adequacy of the operating limitations given in the Technical Specifications. In the case of older plants, the data may be estimated, using the procedures listed above, or other methods that can be shown to be conservative.

2. Operating Limitations for Fracture Toughness

2.1 Required Pressure-Temperature Operating Limitations

As required by Appendix G, 10 CFR Part 50, the following operating limitations shall be determined and included in the Technical Specifications. The basis for determination shall be reported, and is the responsibility of the applicant, but in no case shall the limitations provide less safety margin than those determined in accordance with Appendix G, 10 CFR Part 50, and Appendix G to Section III of the Code.

(1) Minimum temperatures for performing any hydrostatic test involving pressurization of the reactor vessel after installation in the system.

(2) Minimum temperatures for all leak and hydrostatic tests performed after the plant is in service.

(3) Maximum pressure-minimum temperature curves for operation, including startup, upset, and cooldown conditions.

(4) Maximum pressure-minimum temperature curves for core operation.

2.2 Recommended Bases for Operating Limitations

2.2.1 Leak and Hydrostatic Tests

(1) It is recommended that no tests at pressures higher than design pressure be conducted with fuel in the vessel.

(2) Tests at pressures less than design pressure should be conducted at temperatures calculated according to Appendix G of Section III of the Code for the beltline region (including conservative estimates of radiation damage, see Section 3.0 below) if the maximum calculated primary stress in no other region of the vessel exceeds 1.25 S_m during the test, and the RT_{NDT} of the beltline is assumed to be at least 30°F above that of the higher stressed regions. If primary stresses are calculated to be over 1.25 S_m in any region during the test, the RT_{NDT} of the vessel must be assumed to be at least 50°F higher than that of

any region where the calculated primary stresses are over 1.25 S_m.

(3) Alternatively, a fracture mechanics analysis, with technical justification for all assumptions and bases, may be made to determine the minimum test temperature. In no event shall the minimum temperature be lower than that resulting from calculations for the beltline region in accordance with Appendix G of the Code.

2.2.2 Heatup and Cooldown Limit Curves

Heatup and cooldown pressure-temperature limit curves may be determined using single $\frac{pr}{t}$ stress calculations, using the method given in Appendix G of the Code. The effect of thermal gradients may be conservatively approximated by the procedures in Appendix G of the Code or from Figure 4-5 in WRC Bulletin 175.

Calculations need only be performed for the beltline region, if the RT_{NDT} of the beltline is demonstrated to be adequately higher than the RT_{NDT} for all higher stressed regions.

Alternatively, more rigorous analytical procedures may be used, provided that the intent of the Code is met, and adequate technical justification for all assumptions and bases is provided.

2.2.3 Core Operation Limits

To provide added margins during actual core operation, Appendix G, 10 CFR Part 50 requires a minimum temperature during core operation, and a 40°F margin in temperature over the pressure-temperature limits as determined for heatup and cooldown in 2.2.2 above. The minimum temperature, regardless of pressure, is the temperature calculated for the inservice hydrostatic test according to 2.2.1 above.

2.2.4 Upset Conditions

The pressure-temperature limits described in 2.2.2 and 2.2.3 above are applicable to upset conditions. Normal operating procedures must permit variations from intended operation, including all upset conditions, without exceeding the limit curves.

2.2.5 Emergency and Faulted Conditions

It is recognized that the severity of a transient resulting from an emergency or faulted condition is not directly related to operating conditions, and resulting temperature-stress relationships in the reactor coolant boundary components are primarily system dependent, and therefore not under direct control of the operator.

For these reasons, operating limits for emergency and faulted conditions are not a requirement of the Technical Specifications.

The SAR should present descriptions of the continued integrity of all vital components of the RCPB during postulated faulted conditions. It is recommended that such descriptions be made in as realistic a manner as possible, avoiding grossly overconservative assumptions and procedures.

2.3 Reporting Requirements

The Technical Specifications must include the operating and test limits discussed above, and the basis for their determination. The Technical Specifications must also include information on the intended operating procedures, and justify that adequate margins between the expected conditions and the limit conditions will be provided to protect against unexpected or upset conditions.

3. Inservice Surveillance of Fracture Toughness

The reactor vessel may be exposed to significant neutron radiation during the service life. This will affect both the tensile and toughness properties. A material surveillance program in conformance with Appendix H, 10 CFR Part 50, must be carried out.

3.1 Surveillance Program Requirements

The minimum requirements for the surveillance program are covered by Appendix H, 10 CFR Part 50. It is strongly recommended that consideration by given to the desirability of additional surveillance methods, such as the inclusion of CT, DWT, DT, or other specimens to provide the capability of redundant test methods and analytical procedures, particularly if the estimated neutron fluence is over 2×10^{19}, or the toughness of the vessel material is marginal.

The selection of material to be included in the surveillance program should be in accordance with ASTM E-185-~~73~~ 82 unless the intent of the program is better realized by using more rigorous criteria. For example, the approach of estimating the actual RT_{NDT} and upper shelf toughness of each plate, forging, or weld in the beltline as a function of service life, and choosing as the surveillance materials those that are expected to be most limiting, may be preferable in some cases. This would include consideration of the initial RT_{NDT}, the upper shelf toughness, the expected radiation sensitivity of the material (based on copper and ~~phosphorous~~ nickel content, for example) and the neutron fluence expected at its location in the vessel.

3.2 SAR Requirements

The adequacy of the surveillance program cannot be evaluated unless all pertinent information is included in the SAR. Information requested for beltline materials includes the following:

(1) Tensile properties.

(2) DWT and Charpy V test results used to determine RT_{NDT}.

(3) Charpy V test results to determine the upper shelf toughness.

(4) Composition, specifically the copper and phosphorous content.

(5) Estimated maximum fluence for each beltline material.

(6) List of materials included in the surveillance program, with basis used for their selection.

3.3 Surveillance Test Procedures

Surveillance capsules must be removed and tested at intervals in accordance with Appendix H, 10 CFR Part 50. The proposed removal and test schedule shall be included in the Technical Specifications.

3.4 Reporting Requirements

All information used to evaluate results of the tests on surveillance materials, evaluation methods, and results of the evaluation should be submitted with the evaluation report. This should include:

(1) Original properties and compositions of the materials.

(2) Fluence calculations, including original predictions, for both surveillance specimens and vessel wall.

(3) Test results on surveillance specimens.

(4) Basis for evaluation of changes in RT_{NDT} and upper shelf toughness.

(5) Updated prediction of vessel properties.

3.5 Technical Specification Changes

Changes in the operating and test limits recommended as a result of evaluating the properties of the surveillance material, together with the basis for these changes, shall be submitted to the Division of Licensing for approval.