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

JUL 12 1988

TO ALL LICENSEES OF OPERATING REACTORS AND HOLDERS OF CONSTRUCTION PERMITS

SUBJECT: NRC POSITION ON RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS AND ITS IMPACT ON PLANT OPERATIONS (GENERIC LETTER 88-11 )

The purpose of this letter is to call your attention to the attached copy of Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," which became effective May 1988. It will be used by the NRC in reviewing submittals regarding pressure-temperature (P-T) limits and for analyses other than pressurized thermal shock (PTS) that require an estimate of the embrittlement of reactor vessel beltline materials.

Licensees and permittees should use the methods described in Revision 2 to Regulatory Guide 1.99 to predict the effect of neutron radiation on reactor vessel materials as required by Paragraph V.A. of 10 CFR Part 50 Appendix G, unless they can justify the use of different methods. The use of the Revision 2 methodology may result in a modification of the pressure-temperature limits contained in Technical Specifications in order to continue to satisfy the requirements of Sec. V of 10 CFR Part 50, Appendix G. Within 180 days of the effective date of Revision 2, licensees should submit the results of their technical analysis and a proposed schedule for whatever actions they propose to take. In the event that such actions are necessary, their schedule is negotiable provided that all actions (hardware, procedures, and/or staff modifications) are completed (fully implemented and operational) within 2 plant outages (approximately 3 years) after the effective date of Revision 2 to Regulatory Guide 1.99.

PWR licensees should note that the Low-Temperature-Overpressure Protection (LTOP) set points and enable temperatures, which are determined from the P-T limits, may also have to be revised as a result of Revision 2. Since Revision 2, in general, results in a lowering of the Appendix G pressure curves and a shift to higher enable temperatures, the resulting narrowing of the operating window may restrict flexibility on heatup and cooldown operations.

Standard Review Plan 5.2.2, "Overpressure Protection," and the associated Branch Position RSB 5-2 is being changed to provide some relief from this impact. Paragraph II.B, which requires protection "at low temperature," is being amended to define the required enable temperature for the LTOP system based on a fracture criterion. Automatic, or passive, protection of the upper end of the P-T limits will not be required but administratively controlled. At the lower end of the P-T limits, for example during startup, automatic protection of the Appendix G P-T limits is still required for anticipated operational occurrences.

As plants age, it is expected that the operating window will continue to narrow and startup operations will become more difficult. Revision 2 accelerates this narrowing of the operating window. Licensees are encouraged to

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JUL 12 1988

review system hardware and operating procedures to determine what changes could be made to reduce the likelihood of LTOP challenges. If changes can be implemented to demonstrate that the frequency of an LTOP event that would exceed Appendix G limits is expected to be much less than one per reactor lifetime, then the staff would consider alternatives to Appendix G LTOP set points with appropriate justification of adequate safety from the standpoint of fracture prevention.

BWR licensees should note that the use of Revision 2 as the basis for P-T limits for BWR pressure tests will require higher pressure test temperatures in many cases. The NRC does not accept the BWR Owners Group position that the margins given by following the procedures of Appendix G, 10 CFR Part 50 can safely be reduced.

With regard to the pressurized thermal shock issue in PWRs, the staff is presently considering an amendment to the PTS Rule, 10 CFR 50.61, that will replace the equations for $RT_{PTS}$ given in paragraph (b)(2) with the calculation procedure given in Section C.1 of Revision 2 to Reg. Guide 1.99, but will not change the screening criterion.

Based on calculations reported in the Regulatory Analysis, a number of reactor vessels will reach the screening criterion sooner, using Revision 2, and in a few cases that date will precede the end of license. To see if their plant falls in this category, licensees may wish to repeat the calculation of $RT_{PTS}$ values submitted to the NRC in response to the PTS Rule (January 23, 1986 submittal) for the critical materials in the vessel beltline, using Section C.1 of Revision 2 to Regulatory Guide 1.99. The purpose of this suggestion is simply to provide early warning that further flux reduction should be considered in some plants.

This request for information is covered by the Office of Management and Budget under Clearance Number 3150-0011, which expires December 31, 1989. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, D.C. 20503.

Sincerely,

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Enclosure:
Revision 2 to R.G. 1.99

Revision 2
May 1988

U.S. NUCLEAR REGULATORY COMMISSION

# REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.99
(Task ME 305-4)

## RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS

### A. INTRODUCTION

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. General Design Criterion 31 also requires that the design reflect the uncertainties in determining the effects of irradiation on material properties. Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," which implement, in part, Criterion 31, necessitate the calculation of changes in fracture toughness of reactor vessel materials caused by neutron radiation throughout the service life. This guide describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels.

The calculative procedures given in Regulatory Position 1.1 of this guide are not the same as those given in the Pressurized Thermal Shock rule (§ 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," of 10 CFR Part 50) for calculating $RT_{PTS}$, the reference temperature that is to be compared to the screening criterion given in the rule. The information on which this Revision 2 is based may also affect the basis for the PTS rule. The staff is presently considering whether to propose a change to § 50.61.

The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 50, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 50 have been cleared under OMB Clearance No. 3150-0011.

### B. DISCUSSION

Some NRC requirements that necessitate calculation of radiation embrittlement are:

1. Paragraph V.A of Appendix G requires the effects of neutron radiation to be predicted from the results of pertinent radiation effects studies. This guide provides such results in the form of calculative procedures that are acceptable to the NRC.

2. Paragraph V.B of Appendix G describes the basis for setting the upper limit for pressure as a function of temperature during heatup and cooldown for a given service period in terms of the predicted value of the adjusted reference temperature at the end of the service period.

3. The definition of reactor vessel beltline given in Paragraph II.F of Appendix G requires identification of regions of the reactor vessel that are predicted to experience sufficient neutron radiation embrittlement to be considered in the selection of the most limiting material. Paragraphs III.A and IV.A.1 specify the additional test requirements for beltline materials that supplement the requirements for reactor vessel materials generally.

4. Paragraph II.B of Appendix H incorporates ASTM E 185 by reference. Paragraph 5.1 of ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels" (Ref. 1), requires that the materials to be placed in surveillance be those that may limit operation of the reactor during its lifetime, i.e., those expected to have the highest adjusted reference temperature or the lowest Charpy upper-shelf energy at end of life. Both measures of radiation embrittlement must be considered. In Paragraph 7.6 of ASTM E 185-82, the requirements for the number of capsules and the withdrawal schedule are based on the calculated amount of radiation embrittlement at end of life.

The two measures of radiation embrittlement used in this guide are obtained from the results of the Charpy V-notch impact test. Appendix G to 10 CFR Part 50 requires that a full curve of absorbed energy versus temperature be obtained through the ductile-to-brittle transition temperature region. The adjustment of the reference temperature, $\Delta RT_{NDT}$, is defined in Appendix G as the temperature shift in the Charpy curve for the irradiated material relative to that for the unirradiated material measured at the 30-foot-pound energy level, and the data that formed the basis for this guide were 30-foot-pound shift values. The second measure of radiation embrittlement is the decrease in the Charpy upper-shelf energy level, which is defined in ASTM E 185-82. This Revision 2 updates the calculative procedures for the adjustment of reference temperature; however, calculative procedures for the decrease in upper-shelf energy are unchanged because the preparatory work had not been completed in time to include them in this revision.

The basis for Equation 2 for $\Delta RT_{NDT}$ (in Regulatory Position 1.1 of this guide) is contained in publications by G. L. Guthrie (Ref. 2) and G. R. Odette et al. (Ref. 3). Both of these papers used surveillance data from commercial power reactors. The bases for their regression correlations were different in that Odette made greater use of physical models of radiation embrittlement. Yet, the two papers contain similar recommendations: (1) separate correlation functions should be used for weld and base metal, (2) the function should be the product of a chemistry factor and a fluence factor, (3) the parameters in the chemistry factor should be the elements copper and nickel, and (4) the fluence factor should provide a trend curve slope of about 0.25 to 0.30 on log-log paper at $10^{19}$ n/cm² (E > 1 MeV), steeper at low fluences and flatter at high fluences. Regulatory Position 1.1 is a blend of the correlation functions presented by these authors. Some test reactor data were used as a guide in establishing a cutoff for the chemistry factor for low-copper materials. The data base for Regulatory Position 1.2 is that given by Spencer H. Bush (Ref. 4).

The measure of fluence used in this guide is the number of neutrons per square centimeter having energies greater than 1 million electron volts (E > 1 MeV). The differences in energy spectra at the surveillance capsule and the vessel inner surface locations do not appear to be great enough to warrant the use of a damage function such as displacements per atom (dpa) (Ref. 5) in the analysis of the surveillance data base (Ref. 6).

However, the neutron energy spectrum does change significantly with location in the vessel wall; hence for calculating the attenuation of radiation embrittlement through the vessel wall, it is necessary to use a damage function to determine $\Delta RT_{NDT}$ versus radial distance into the wall. The most widely accepted damage function at this time is dpa, and the attenuation formula (Equation 3) given in Regulatory Position 1.1 is based on the attenuation of dpa through the vessel wall.

Sensitivity to neutron radiation embrittlement may be affected by elements other than copper and nickel. The original version and Revision 1 of this guide had a phosphorus term in the chemistry factor, but the studies on which this revision was based found other elements such as phosphorus to be of secondary importance, i.e., including them in the analysis did not produce a significantly better fit of the data.

Scatter in the data base used for this guide is relatively significant, as evidenced by the fact that the standard deviations for Guthrie's derived formulas (Ref. 2) are 28°F for welds and 17°F for base metal despite extensive efforts to find a model that reduced the fitting error. Thus the use of surveillance data from a given reactor (in place of the calculative procedures given in this guide) requires considerable engineering judgment to evaluate the credibility of the data and assign suitable margins. When surveillance data from the reactor in question become available, the weight given to them relative to the information in this guide will depend on the credibility of the surveillance data as judged by the following criteria:

1. Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement according to the recommendations of this guide.

2. Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30-foot-pound temperature and the upper-shelf energy unambiguously.

3. When there are two or more sets of surveillance data from one reactor, the scatter of $\Delta RT_{NDT}$ values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82 (Ref. 1).

4. The irradiation temperature of the Charpy specimens in the capsule should match vessel wall temperature at the cladding/base metal interface within ±25°F.

5. The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the data base for that material.

To use the surveillance data from a specific plant instead of Regulatory Position 1, one must develop a relationship of $\Delta RT_{NDT}$ to fluence for that plant. Because such data are limited in number and subject to scatter, Regulatory Position 2 describes a procedure in which the form of Equation 2 is to be used and the fluence factor therein is retained, but the chemistry factor is determined by the plant surveillance data. Of several possible ways to fit such data, the method that minimizes the sums of the squares of the errors was chosen somewhat arbitrarily. Its use is justified in part by the fact that "least squares" is a common method for curve fitting. Also, when there are only two data points, the least squares method gives greater weight to the point with the higher $\Delta RT_{NDT}$; this seems reasonable for fitting surveillance data, because generally the higher data point will be the more recent and therefore will represent more modern procedures.

## C. REGULATORY POSITION

### 1. SURVEILLANCE DATA NOT AVAILABLE

When credible surveillance data from the reactor in question are not available, calculation of neutron radiation embrittlement of the beltline of reactor vessels of light-water reactors should be based on the procedures in Regulatory Positions 1.1 and 1.2 within the limitations in Regulatory Position 1.3.

### 1.1 Adjusted Reference Temperature

The adjusted reference temperature (ART) for each material in the beltline is given by the following expression:

$$\text{ART} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (1)$$

Initial $RT_{NDT}$ is the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code (Ref. 7). If measured values of initial $RT_{NDT}$ for the material in question are not available, generic mean values for that class* of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

$\Delta RT_{NDT}$ is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = (CF)\, f^{(0.28\, -\, 0.10 \log f)} \qquad (2)$$

CF (°F) is the chemistry factor, a function of copper and nickel content. CF is given in Table 1 for welds and in Table 2 for base metal (plates and forgings). Linear interpolation is permitted. In Tables 1 and 2 "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld. If such values are not available, the upper limiting values given in the material specifications to which the vessel was built may be used. If not available, conservative estimates (mean plus one standard deviation) based on generic data may be used if justification is provided. If there is no information available, 0.35% copper and 1.0% nickel should be assumed.

The neutron fluence at any depth in the vessel wall, f ($10^{19}$ n/cm$^2$, E > 1 MeV), is determined as follows:

$$f = f_{surf}\,(e^{\,-0.24x}) \qquad (3)$$

where $f_{surf}$ ($10^{19}$ n/cm$^2$, E > 1 MeV) is the calculated value of the neutron fluence at the inner wetted surface of the vessel at the location of the postulated defect, and x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface. Alternatively, if dpa calculations are made as part of the fluence analysis, the ratio of dpa at the depth in question to dpa at the inner surface may be substituted for the exponential attenuation factor in Equation 3.

The fluence factor, $f^{0.28\, -\, 0.10 \log f}$, is determined by calculation or from Figure 1.

"Margin" is the quantity, °F, that is to be added to obtain conservative, upper-bound values of adjusted reference temperature for the calculations required by Appendix G to 10 CFR Part 50.

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \qquad (4)$$

Here, $\sigma_I$ is the standard deviation for the initial $RT_{NDT}$. If a measured value of initial $RT_{NDT}$ for the material in question is available, $\sigma_I$ is to be estimated from the precision of the test method. If not, and generic mean values for that class of material are used, $\sigma_I$ is the standard deviation obtained from the set of data used to establish the mean.

The standard deviation for $\Delta RT_{NDT}$, $\sigma_\Delta$, is 28 °F for welds and 17 °F for base metal, except that $\sigma_\Delta$ need not exceed 0.50 times the mean value of $\Delta RT_{NDT}$.

### 1.2 Charpy Upper-Shelf Energy

Charpy upper-shelf energy should be assumed to decrease as a function of fluence and copper content as indicated in Figure 2. Linear interpolation is permitted.

### 1.3 Limitations

Application of the foregoing procedures should be subject to the following limitations:

1. The procedures apply to those grades of SA-302, 336, 533, and 508 steels having minimum specified yield strengths of 50,000 psi and under and to their welds and heat-affected zones.

2. The procedures are valid for a nominal irradiation temperature of 550 °F. Irradiation below 525 °F should be considered to produce greater embrittlement, and irradiation above 590 °F may be considered to produce less embrittlement. The correction factor used should be justified by reference to actual data.

3. Application of these procedures to fluence levels or to copper or nickel content beyond the ranges given in Figure 1 and Tables 1 and 2 or to materials having chemical compositions beyond the range found in the data bases used for this guide should be justified by submittal of data.

## 2. SURVEILLANCE DATA AVAILABLE

When two or more credible surveillance data sets (as defined in the Discussion) become available from the reactor in question, they may be used to determine the adjusted reference temperature and the Charpy upper-shelf energy of the beltline materials as described in Regulatory Positions 2.1 and 2.2, respectively.

### 2.1 Adjusted Reference Temperature

The adjusted reference temperature should be obtained as follows. First, if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of $\Delta RT_{NDT}$ should be adjusted by multiplying them by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld. Second, the surveillance data should be fitted using Equation 2 to obtain the relationship of $\Delta RT_{NDT}$ to fluence. To do so, calculate the chemistry factor, CF, for the best fit by multiplying each adjusted $\Delta RT_{NDT}$ by its corresponding fluence factor, summing the products, and dividing by the sum of the squares of the fluence factors. The resulting value of CF when entered in Equation 2 will give the relationship of $\Delta RT_{NDT}$ to

*The class for estimating initial $RT_{NDT}$ is generally determined, for the welds with which this guide is concerned, by the type of welding flux (Linde 80 or other); for base metal, by the ASTM Standard Specification.

TABLE 1

CHEMISTRY FACTOR FOR WELDS, °F

| Copper, Wt-% | Nickel, Wt-% 0 | 0.20 | 0.40 | 0.60 | 0.80 | 1.00 | 1.20 |
|---|---|---|---|---|---|---|---|
| 0 | 20 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.01 | 20 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.02 | 21 | 26 | 27 | 27 | 27 | 27 | 27 |
| 0.03 | 22 | 35 | 41 | 41 | 41 | 41 | 41 |
| 0.04 | 24 | 43 | 54 | 54 | 54 | 54 | 54 |
| 0.05 | 26 | 49 | 67 | 68 | 68 | 68 | 68 |
| 0.06 | 29 | 52 | 77 | 82 | 82 | 82 | 82 |
| 0.07 | 32 | 55 | 85 | 95 | 95 | 95 | 95 |
| 0.08 | 36 | 58 | 90 | 106 | 108 | 108 | 108 |
| 0.09 | 40 | 61 | 94 | 115 | 122 | 122 | 122 |
| 0.10 | 44 | 65 | 97 | 122 | 133 | 135 | 135 |
| 0.11 | 49 | 68 | 101 | 130 | 144 | 148 | 148 |
| 0.12 | 52 | 72 | 103 | 135 | 153 | 161 | 161 |
| 0.13 | 58 | 76 | 106 | 139 | 162 | 172 | 176 |
| 0.14 | 61 | 79 | 109 | 142 | 168 | 182 | 188 |
| 0.15 | 66 | 84 | 112 | 146 | 175 | 191 | 200 |
| 0.16 | 70 | 88 | 115 | 149 | 178 | 199 | 211 |
| 0.17 | 75 | 92 | 119 | 151 | 184 | 207 | 221 |
| 0.18 | 79 | 95 | 122 | 154 | 187 | 214 | 230 |
| 0.19 | 83 | 100 | 126 | 157 | 191 | 220 | 238 |
| 0.20 | 88 | 104 | 129 | 160 | 194 | 223 | 245 |
| 0.21 | 92 | 108 | 133 | 164 | 197 | 229 | 252 |
| 0.22 | 97 | 112 | 137 | 167 | 200 | 232 | 257 |
| 0.23 | 101 | 117 | 140 | 169 | 203 | 236 | 263 |
| 0.24 | 105 | 121 | 144 | 173 | 206 | 239 | 268 |
| 0.25 | 110 | 126 | 148 | 176 | 209 | 243 | 272 |
| 0.26 | 113 | 130 | 151 | 180 | 212 | 246 | 276 |
| 0.27 | 119 | 134 | 155 | 184 | 216 | 249 | 280 |
| 0.28 | 122 | 138 | 160 | 187 | 218 | 251 | 284 |
| 0.29 | 128 | 142 | 164 | 191 | 222 | 254 | 287 |
| 0.30 | 131 | 146 | 167 | 194 | 225 | 257 | 290 |
| 0.31 | 136 | 151 | 172 | 198 | 228 | 260 | 293 |
| 0.32 | 140 | 155 | 175 | 202 | 231 | 263 | 296 |
| 0.33 | 144 | 160 | 180 | 205 | 234 | 266 | 299 |
| 0.34 | 149 | 164 | 184 | 209 | 238 | 269 | 302 |
| 0.35 | 153 | 168 | 187 | 212 | 241 | 272 | 305 |
| 0.36 | 158 | 172 | 191 | 216 | 245 | 275 | 308 |
| 0.37 | 162 | 177 | 196 | 220 | 248 | 278 | 311 |
| 0.38 | 166 | 182 | 200 | 223 | 250 | 281 | 314 |
| 0.39 | 171 | 185 | 203 | 227 | 254 | 285 | 317 |
| 0.40 | 175 | 189 | 207 | 231 | 257 | 288 | 320 |

fluence that fits the plant surveillance data in such a way as to minimize the sum of the squares of the errors.

To calculate the margin in this case, use Equation 4; the values given there for $\sigma_\Delta$ may be cut in half.

If this procedure gives a higher value of adjusted reference temperature than that given by using the procedures of Regulatory Position 1.1, the surveillance data should be used. If this procedure gives a lower value, either may be used.

For plants having surveillance data that are credible in all respects except that the material does not represent the critical material in the vessel, the calculative procedures in this guide should be used to obtain mean values of shift, $\Delta RT_{NDT}$. In calculating the margin, the value of $\sigma_\Delta$ may be reduced from the values given in the last paragraph of Regulatory Position 1.1 by an amount to be decided on a case-by-case basis, depending on where the measured values fall relative to the mean calculated for the surveillance materials.

### 2.2 Charpy Upper-Shelf Energy

The decrease in upper-shelf energy may be obtained by plotting the reduced plant surveillance data on Figure 2 of this guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data. This line should be used in preference to the existing graph.

2. Holders of licenses and permits should use the methods described in this guide to predict the effect of neutron radiation on reactor vessel materials as required by Paragraph V.A of Appendix G to 10 CFR Part 50, unless they can justify the use of different methods. The use of the Revision 2 methodology may result in a modification of the pressure-temperature limits contained in Technical Specifications in order to continue to satisfy the requirements of Section V of Appendix G, 10 CFR Part 50.

3. The recommendations of Regulatory Position 3 are essentially unchanged from those used to evaluate construction permit applications docketed on or after June 1, 1977.

TABLE 2

CHEMISTRY FACTOR FOR BASE METAL, °F

| Copper, Wt-% | Nickel, Wt-% 0 | 0.20 | 0.40 | 0.60 | 0.80 | 1.00 | 1.20 |
|---|---|---|---|---|---|---|---|
| 0 | 20 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.01 | 20 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.02 | 20 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.03 | 20 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.04 | 22 | 26 | 26 | 26 | 26 | 26 | 26 |
| 0.05 | 25 | 31 | 31 | 31 | 31 | 31 | 31 |
| 0.06 | 28 | 37 | 37 | 37 | 37 | 37 | 37 |
| 0.07 | 31 | 43 | 44 | 44 | 44 | 44 | 44 |
| 0.08 | 34 | 48 | 51 | 51 | 51 | 51 | 51 |
| 0.09 | 37 | 53 | 58 | 58 | 58 | 58 | 58 |
| 0.10 | 41 | 58 | 65 | 65 | 67 | 67 | 67 |
| 0.11 | 45 | 62 | 72 | 74 | 77 | 77 | 77 |
| 0.12 | 49 | 67 | 79 | 83 | 86 | 86 | 86 |
| 0.13 | 53 | 71 | 85 | 91 | 96 | 96 | 96 |
| 0.14 | 57 | 75 | 91 | 100 | 105 | 106 | 106 |
| 0.15 | 61 | 80 | 99 | 110 | 115 | 117 | 117 |
| 0.16 | 65 | 84 | 104 | 118 | 123 | 125 | 125 |
| 0.17 | 69 | 88 | 110 | 127 | 132 | 135 | 135 |
| 0.18 | 73 | 92 | 115 | 134 | 141 | 144 | 144 |
| 0.19 | 78 | 97 | 120 | 142 | 150 | 154 | 154 |
| 0.20 | 82 | 102 | 125 | 149 | 159 | 164 | 165 |
| 0.21 | 86 | 107 | 129 | 155 | 167 | 172 | 174 |
| 0.22 | 91 | 112 | 134 | 161 | 176 | 181 | 184 |
| 0.23 | 95 | 117 | 138 | 167 | 184 | 190 | 194 |
| 0.24 | 100 | 121 | 143 | 172 | 191 | 199 | 204 |
| 0.25 | 104 | 126 | 148 | 176 | 199 | 208 | 214 |
| 0.26 | 109 | 130 | 151 | 180 | 205 | 216 | 221 |
| 0.27 | 114 | 134 | 155 | 184 | 211 | 225 | 230 |
| 0.28 | 119 | 138 | 160 | 187 | 216 | 233 | 239 |
| 0.29 | 124 | 142 | 164 | 191 | 221 | 241 | 248 |
| 0.30 | 129 | 146 | 167 | 194 | 225 | 249 | 257 |
| 0.31 | 134 | 151 | 172 | 198 | 228 | 255 | 266 |
| 0.32 | 139 | 155 | 175 | 202 | 231 | 260 | 274 |
| 0.33 | 144 | 160 | 180 | 205 | 234 | 264 | 282 |
| 0.34 | 149 | 164 | 184 | 209 | 238 | 268 | 290 |
| 0.35 | 153 | 168 | 187 | 212 | 241 | 272 | 298 |
| 0.36 | 158 | 173 | 191 | 216 | 245 | 275 | 303 |
| 0.37 | 162 | 177 | 196 | 220 | 248 | 278 | 308 |
| 0.38 | 166 | 182 | 200 | 223 | 250 | 281 | 313 |
| 0.39 | 171 | 185 | 203 | 227 | 254 | 285 | 317 |
| 0.40 | 175 | 189 | 207 | 231 | 257 | 288 | 320 |

### 3. REQUIREMENT FOR NEW PLANTS

For beltline materials in the reactor vessel for a new plant, the content of residual elements such as copper, phosphorus, sulfur, and vanadium should be controlled to low levels.* The copper content should be such that the calculated adjusted reference temperature at the 1/4T position in the vessel wall at end of life is less than 200°F. In selecting the optimum amount of nickel to be used, its deleterious effect on radiation embrittlement should be balanced against its beneficial metallurgical effects and its tendency to lower the initial $RT_{NDT}$.

*For more information, see the Appendix to ASTM Standard Specification A 533 (Ref. 8).

## D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in this guide will be used as follows:

1. The methods described in Regulatory Positions 1 and 2 of this guide will be used by the NRC staff in evaluating all predictions of radiation embrittlement needed to implement Appendices G and H to 10 CFR Part 50.

## REFERENCES


1. American Society for Testing and Materials, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM E 185-82, July 1982.*

2. G. L. Guthrie, "Charpy Trend Curves Based on 177 PWR Data Points," in "LWR Pressure Vessel Surveillance Dosimetry Improvement Program," NUREG/CR-3391, Vol. 2, prepared by Hanford Engineering Development Laboratory, HEDL-TME 83-22, April 1984.**

3. G. R. Odette et al., "Physically Based Regression Correlations of Embrittlement Data from Reactor Pressure Vessel Surveillance Programs," Electric Power Research Institute, NP-3319, January 1984.†

4. S. H. Bush, "Structural Materials for Nuclear Power Plants," in *Journal of Testing and Evaluation*, American Society for Testing and Materials, November 1974.*

5. American Society for Testing and Materials, "Standard Practice for Characterizing Neutron Exposures in Ferritic Steels in Terms of Displacements per Atom (DPA)," ASTM E 693-79, August 1979.*

6. W. N. McElroy, "LWR Pressure Vessel Surveillance Dosimetry Improvement Program: LWR Power Reactor Surveillance Physics—Dosimetry Data Base Compendium," NUREG/CR-3319, prepared by Hanford Engineering Development Laboratory, HEDL-TME 85-3, August 1985.**

7. American Society of Mechanical Engineers, Section III, "Nuclear Power Plant Components," of *ASME Boiler and Pressure Vessel Code*, New York (updated frequently).††

8. American Society for Testing and Materials, "Standard Specification for Pressure Vessel Plates, Alloy Steel, Quenched and Tempered, Manganese-Molybdenum and Manganese-Molybdenum-Nickel," ASTM A 533/A 533M-82, September 1982.*


*Copies may be obtained from the American Society for Testing and Materials, 1916 Race Street, Philadelphia, PA 19103.
**Copies may be obtained from the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Wasington, DC 20013-7082.
†Copies may be obtained from the Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, CA 94304.
††Copies may be obtained from the American Society of Mechanical Engineers, 345 E. 47th Street, New York, NY 10017.

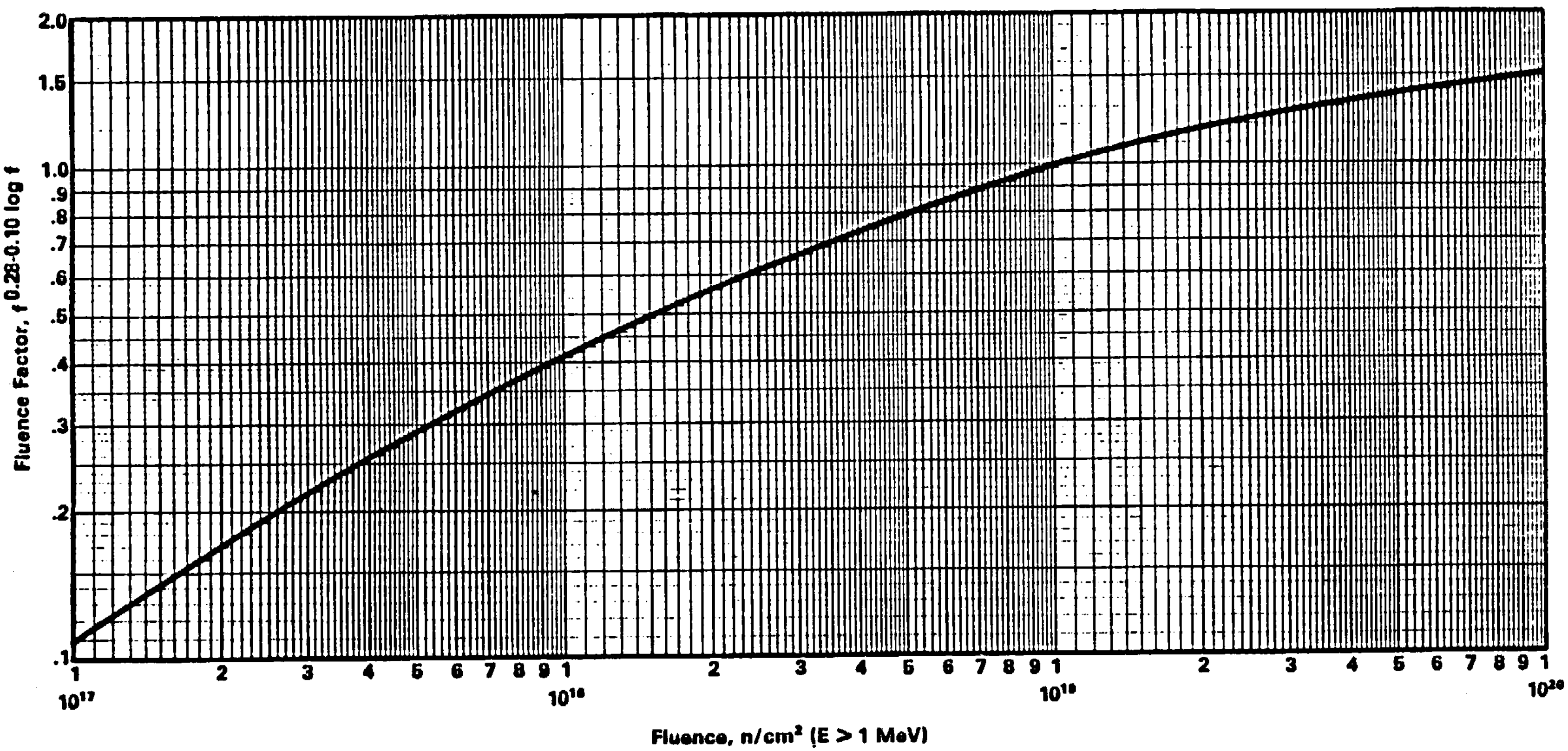


FIGURE 1 Fluence Factor for Use in Equation 2, the Expression for $\Delta RT_{NDT}$

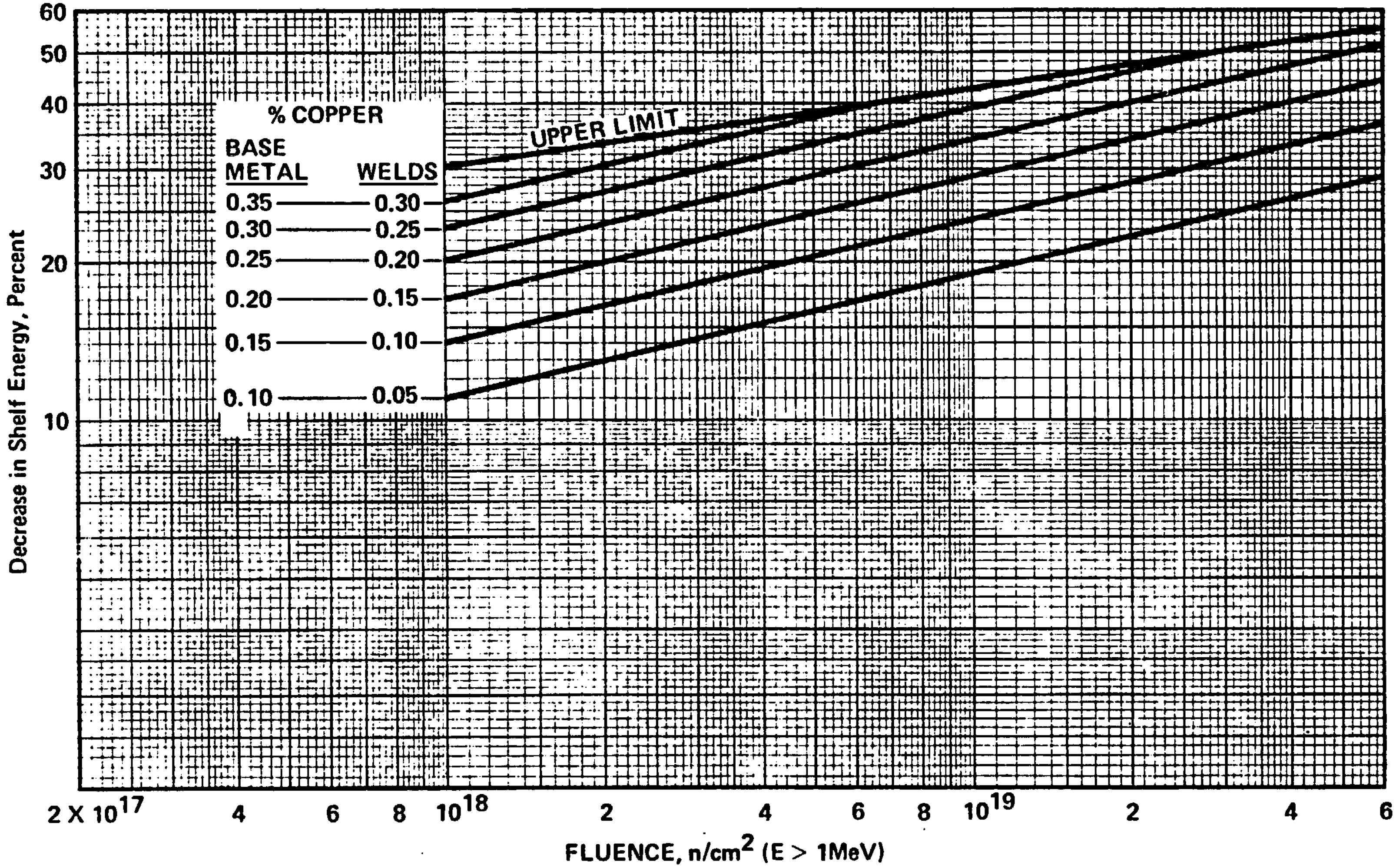


FIGURE 2 Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence

## REGULATORY ANALYSIS

A copy of the regulatory analysis prepared for this Regulatory Guide 1.99, Revision 2, is available for inspection and copying for a fee at the Commission's Public Document Room at 1717 H Street NW., Washington, DC, under Regulatory Guide 1.99, Revision 2.