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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 Introduction

By letters dated March 22, 1978, September 26, 1983 and May 2, 1983 the Niagara Mohawk Power Corporation (the licensee) requested changes to the Nine Mile Point Nuclear Station, Unit No. 1 (NMP-1) Technical Specifications (TS). The licensee provided supplemental and clarifying information supporting the March 22, 1978 request in letters dated April 20 and October 26, 1983. The additional information provided did not change the action noticed in the Federal Register, revision to the pressure-temperature limits, but only provided supplemental information to support the technical review. The letter dated September 26, 1983 requested that Section 6.9.3 be changed to require that the test results from the reactor vessel material surveillance program be submitted to the staff within 12 months of their removal from the NMP-1 vessel. The letter of May 2, 1983 requested that Section 4.2.2 be changed to delete the standby capsule from the reactor vessel surveillance program. The letter of October 26, 1983 provided revised reactor vessel pressure-temperature limits for Section 3.2.2 of the Technical Specification for 10 effective full power years of operation.

2.0 Evaluation

Appendix H, 10 CFR 50, is the regulation that establishes the requirements for the Reactor Vessel Material Surveillance Program. The requested change to NMP-1 Technical Specification Section 6.9.3 dated September 26, 1983 meets the reporting requirements of this regulation. Hence it is acceptable.

In a letter from C. V. Mangan to D. B. Vassallo dated March 23, 1983, the licensee indicated that the change to NMP-1 Technical Specification Section 4.2.2 was necessary because one surveillance capsule was removed from the vessel, placed in spent fuel pool and lost during a cleanup of the spent fuel pool. As a result of losing this capsule, there were only two surveillance capsules remaining in the surveillance program. One capsule was removed from the vessel and its surveillance materials are being tested. The other capsule remains in the reactor vessel, is being irradiated, and is scheduled for removal from the vessel at a later date.

Appendix H, 10 CFR 50, requires that the licensee's reactor vessel surveillance program have three capsules; two for testing and one as standby. Since the licensee's proposed surveillance program has no standby capsule, it would not comply with the requirements of Appendix H, 10 CFR 50. The purpose of a standby capsule is to provide backup material test data if the test results from the material in the other capsules should indicate that the NMP-1 beltline materials are behaving abnormally. In order to meet the standby capsule requirements of Appendix H, 10 CFR 50, the licensee must place an additional surveillance capsule within the NMP-1 reactor vessel. The material to be placed within the standby capsule must be similar to the limiting reactor vessel beltline materials. If the licensee cannot reasonably meet the above stated requirement, Appendix H, 10 CFR 50 has provisions for an integrated surveillance program. Therefore, either reinstallation of an additional surveillance capsule or implementation of an integrated surveillance program are considered acceptable methods for resolution of this item. Therefore we are not acting on your May 2, 1983 amendment request until you commit to either one of these methods.

The change to NMP-1 Technical Specification Section 3.2.2 was required to meet the pressure-temperature limits safety margins of Appendix G, 10 CFR 50. Pressure-temperature limits are calculated in accordance with the requirements of Appendix G, 10 CFR 50, are dependent upon the initial reference temperature for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in reference temperature resulting from neutron irradiation damage to the limiting beltline materials.

The latest editions of the ASME Code require that the initial reference temperature be calculated from the results of Charpy V-Notch and drop weight tests. The NMP-1 reactor vessel was procured to an early edition of the ASME Code that did not specify both of these tests. Hence, the initial reference temperature for each beltline and closure flange region material could not be determined in accordance with the test requirements of the latest ASME Code editions. The licensee has determined the initial reference temperature for these materials using the criteria documented in NRC Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements."

This branch technical position is the method recommended by the staff for determining a material's initial reference temperature, when the licensee has fracture toughness tested material to early ASME Code requirements.

The licensee indicates that the limiting beltline material is plate heat no. P2076 and the limiting closure flange region material is the forging used for fabrication of the vessel flange. The initial reference temperature was estimated for these materials as 10°F and 40°F, respectively using the methods in Branch Technical Position MTEB 5-2.

The increase in reference temperature resulting from neutron irradiation damage was estimated by the licensee using the methodology documented in Regulatory Guide 1.99, Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." As a result of the staff's review of the pressurized thermal shock issue, the staff has revised the

recommended method for predicting the increase in reference temperature resulting from neutron irradiation damage. The revised method is documented in draft Regulatory Guide 1.99, Rev. 2, Working Paper A, December 22, 1983, which is attached as an appendix. The method documented in draft Regulatory Guide 1.99, Rev. 2 for predicting the increase in reference temperature depends upon the amount of neutron fluence, and the amounts of copper and nickel in the material.

The licensee stated that the fluence for ten effective full power years was calculated to be 7.73×10^{17} n/cm² (E > 1Me) at the vessel inside surface and the amounts of copper and nickel in the limiting beltline material are .27 percent and .53 percent, respectively. We have evaluated the pressure temperature limits proposed by the licensee based upon the estimated initial reference temperature for the limiting closure flange and beltline materials, the calculated neutron fluence provided by the licensee, the amounts of copper and nickel reported for the limiting beltline material using our current method for predicting neutron irradiation damage. Our evaluation indicates that the proposed curves meet the safety margins of Appendix G, 10 CFR 50 for a period of time corresponding to 10 effective full power years.

3.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Barry J. Elliot

Attachment:
Working Paper A

Dated: April 18, 1984

APPENDIX

Working Draft A
December 27, 1983
Office of Nuclear Regulatory Research

DRAFT REGULATORY GUIDE 1.99, REVISION 2

RADIATION DAMAGE TO 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, "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. 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 damage to the low-alloy steels currently used for light-water-cooled reactor vessels. The Advisory Committee on Reactor Safeguards will be consulted concerning this guide.

B. DISCUSSION

The principal examples of NRC requirements that necessitate calculation of radiation damage are:

1. Paragraph V.A. of Appendix G requires that "...the effects of neutron radiation...are to be predicted from the results of pertinent radiation effect studies...." This guide provides such results in the form of calculational 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 pressure 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 damage to be considered in the selection of the most limiting material...." Paragraphs III.A. and IV.A.1. specify the additional test requirements for belt line materials that supplement the requirements for reactor vessel materials generally.

4. Paragraph IV.B. of Appendix G requires that vessels be designed to permit a thermal annealing treatment if the predicted value of adjusted reference temperature exceeds 200°F during their service life.

5. Paragraph II.B. of Appendix H incorporates ASTM E185 by reference. Paragraph 5.1 of ASTM E185-82 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 damage must be considered. In Paragraph 7.6 of ASTM E185-82 the requirements for number of capsules and withdrawal schedule are based on the calculated amount of radiation damage at end of life.

The two measures of radiation damage used in this guide are obtained from the results of the Charpy 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, ΔRT_{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. The second measure of radiation damage is the decrease in the Charpy upper-shelf energy level, which is defined in ASTM E185-82. Revision 2 of this guide updates the calculational procedures for the adjustment of reference temperature; however, those for the decrease in upper-shelf energy are left intact, because the preparatory work had not been completed in time to include it in Revision 2.

The basis for the equation for ΔRT_{NDT} given in Position C.1.a.2. is contained in publications by G. L. Guthrie[1] and G. R. Odette.[2] Their data base was surveillance data from commercial power reactors, but the data were analyzed differently. Both authors recommended the following: (1) separate correlation functions for weld and base metal, (2) the function 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 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. Position C.1.a. is a blend of the correlation functions presented by the two 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 Position C.1.b. is that given by Spencer H. Bush.[3]

The measure of fluence used herein 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 inside surface do not appear to be great enough to warrant the use of a damage function such as displacements per atom (dpa). In fact, the neutron energy spectra are sufficiently uniform, plant to plant, that dpa and n/cm² (E>1 MeV) can be used interchangeably, with a constant correlation factor. For example, a recent study by Simons[4] has produced the following result from analysis of 42 capsules from pressurized water reactors:

$$\text{dpa} = 1.6 \times 10^{-21} \text{ n/cm}^2 \text{ (E>1 MeV)} \pm 15\%$$

[1] G. L. Guthrie, "Development of Trend Curves Using Surveillance Data - III," from LWR Pressure Vessel Surveillance Dosimetry Improvement Program, Quarterly Progress Report _____, Hanford Engineering Development Laboratory, NUREG-_____.

[2] G. R. Odette and P. M. Lombrozo, "Physically based regression correlations of of embrittlement data from reactor pressure vessel surveillance programs, Research Project 1240-1, Final Report, August 1983, Prepared for Electric Power Research Institute.

[3] Spencer H. Bush, "Structural Materials for Nuclear Power Plants," 1974 ASTM Gillett Memorial Lecture, published in ASTM Journal of Testing and Evaluation, November 1974, and its addendum, "Radiation Damage in Pressure Vessel Steels for Commercial Light-Water Reactors."

[4] R. L. Simons, "Re-Evaluation of Dosimetry for 42 PWR Surveillance Capsules," LWR Pressure Vessel Surveillance Dosimetry Improvement Program Quarterly Progress Report NUREG-04.

Results from Westinghouse reactors are at the top of the ±15% range and results from Babcock & Wilcox reactors are at the bottom of the range. Boiling water reactors were not included in the study.

For calculation of attenuation of radiation damage through the vessel wall, the neutron energy spectrum changes significantly; hence, a damage function should be used to determine ΔRT_{NDT} at the crack tip of postulated defects. The most widely accepted damage function at this time is dpa[5] and the attenuation formula given in Position C.1.a.(2) is based on the attenuation of dpa through the vessel wall.

As used herein, references to "% Cu" and "% Ni" mean the weight percent of copper and nickel as measured in the surveillance program per ASTM E185. However, if such results are not available, the results of a product analysis may be used.

Sensitivity to neutron radiation damage may be affected by elements other than copper and nickel. Revisions 0 and 1 of this guide had a phosphorus term in the chemistry factor, but studies that provided the basis for this revision 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 severe, as evidenced by the fact that one standard deviation is 28°F for welds and 17°F for base metal despite extensive efforts to fit the data. Thus, the use of surveillance data from a given reactor (in place of the calculation 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 it relative to the information in this guide should 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 damage according to the provisions of this guide.

2. Scatter in the Charpy data should be small enough to avoid large uncertainty in curve fitting.

---

[5]ASTM E 693-79, "Standard Practice for Characterizing Neutron Exposures in Ferritic Steels in Terms of Displacements Per Atom (dpa)."

3. The irradiation temperature of the Charpy specimens in the capsule should match vessel wall temperature at the 1/4T position within ±25°F.

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

C. REGULATORY POSITION

1. When credible surveillance data from the reactor in question are not available, prediction of neutron radiation damage to the beltline of reactor vessels of light water reactors should be based on the following procedures, within the limitations in Paragraph 1.c.:

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

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

(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. In cases where measured values of Initial RT_{NDT} for the material in question are not available, generic values for that class of material may be used if there is sufficient information to establish a mean and standard deviation for the class.*

(2) "ΔRT_{NDT}" is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} \text{ surface} = [CF]f^{(0.28-0.10 \log f)}$$

The chemistry factor, "CF," °F, a function of copper and nickel content, is given in Table I for welds and Table II for base metal (plates and forgings). Interpolation is permitted.

In Tables I and II, "Percent Copper" and "Percent Nickel" are the best-estimate values for the material. If measured values for a base metal or for the weld wire heat number associated with a vessel weld are unavailable,

*Additional guidance is given in the Standard Review Plan, NUREG-0800, Section 5.3.2.

but generic best-estimate values for the class of material are known, one standard deviation should be added to the best-estimate values of copper and nickel used to enter Tables I and II. If an estimate of the standard deviation cannot be made, the best-estimate values should be increased by 0.05% Cu and 0.10% Ni for use in the tables. If copper or nickel content is unknown, upper bound values--0.35% Cu and 1.00% Ni--should be used.

The fluence, "f," is the best-estimate value of the neutron fluence at the inner surface of the vessel at the location of the postulated defect, n/cm² (E>1 MeV) divided by 10^{19} n/cm² (E>1 MeV). If desired, the fluence factor may be read from Figure 1. To calculate ΔRT_{NDT} at the crack tip of the postulated defect (e.g., at 1/4T or 3/4T), the following attenuation formula should be used:

$$\Delta RT_{NDT} = \Delta RT_{NDT}\ \text{surface} \times e^{-0.065x}$$

where "x" is the radial distance in inches from the crack tip to the vessel inside surface.

(3) "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, 10 CFR Part 50.

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

The standard deviation for Initial RT_{NDT}, "σ_I," is obtained as described in paragraph C.1.a.(1) if a generic value of ΔRT_{NDT} is used. If a measured value of Initial RT_{NDT} for the material in question is used, σ_I may be considered to be zero. The standard deviation for ΔRT_{NDT}, "σ_Δ," when obtained by paragraph C.1.a.(2) is 28°F for welds and 17°F for base metal, except the value used need not exceed 0.50 times the mean value of ΔRT_{NDT}.

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

c. 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 damage, and irradiation above 590°F may be considered to produce less damage. The correction factor used should be justified.

(3) Application of these procedures to fluence levels or to copper or nickel content beyond the ranges given in Figure 1 and Tables I and II or to materials having other chemical content beyond that represented by the data bases used for this guide, should be justified by submittal of data.

2. When credible surveillance data as defined in the Discussion Section B 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 Paragraphs a. and b., respectively.

a. The adjusted reference temperature may be obtained by first fitting the surveillance data using the fluence function given in Position C.1.a.(2) to obtain a best-estimate value of ΔRT_{NDT} for the fluence in question to use in the calculation of ART described in paragraph C.1.a. To calculate the Margin for this case, use the procedure given in paragraph C.1.a.(3) if there is only one surveillance data point. If there are two or more, the values given for σ_Δ may be cut in half.

b. The decrease in upper-shelf energy may be obtained by fitting the data, using the fluence function illustrated in Figure 2 to obtain a best-estimate value of percent decrease in upper-shelf energy for the fluence in question. A Margin of 5 percentage points should be added to calculate a conservative, upper-bound value for comparison with the requirements of Appendix G, 10 CFR Part 50.

3. For new plants, the reactor vessel beltline materials should have the content of residual elements such as copper, phosphorus, sulfur, and vanadium controlled to low levels. The levels 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.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for utilizing this regulatory guide.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the positions described in this guide will be used by the NRC staff as follows:

1. The method described in regulatory positions C.1 and C.2 of this guide will be used in evaluating all predictions of radiation damage called for in Appendices G and H to 10 CFR Part 50 submitted on or after (60 days after publication); however, if an applicant wishes to use the recommendations of regulatory position C.1 and C.2 in developing submittals before (60 days after publication), the pertinent portions of the submittal will be evaluated on the basis of this guide.

2. Following publication of this guide in final form, the owners of all operating reactors should review the basis for the pressure-temperature limits in their Technical Specifications for consistency with Position C.1.a. Those for which the allowable operating period is shortened, based on the review, should submit the appropriate revision to their Technical Specifications within one year of the date of publication of Revision 2 in final form.

3. The recommendations of regulatory position C.3 will be used in evaluating construction permit application docketed on or after June 1, 1977; however, if an applicant whose application for construction permit was docketed before June 1, 1977 wishes to use the recommendations of regulatory position C.3 of this regulatory guide in developing submittals for the application, the pertinent portions of the application will be evaluated on the basis of this guide.

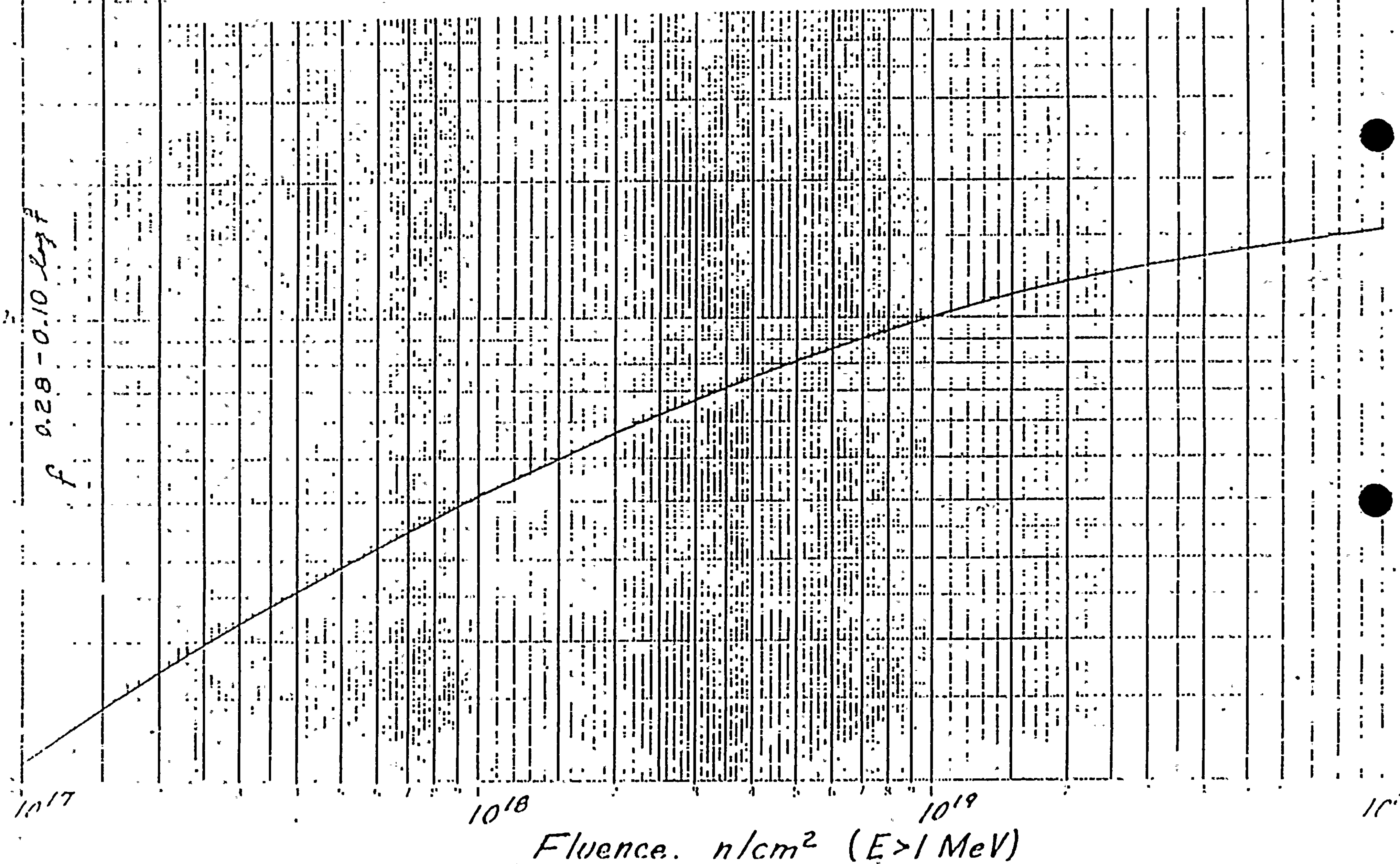
TABLE 1
CHEMISTRY FACTOR FOR WELDS, °F

Percent Copper	Percent Nickel 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	173	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

TABLE II
CHEMISTRY FACTOR FOR BASE METAL, °F

Percent Copper	Percent Nickel 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

Fig. 1 FLUENCE FACTOR FOR USE IN THE EXPRESSION FOR ΔRT_{NDT}, GIVEN IN POSITION C.1.a.(2)



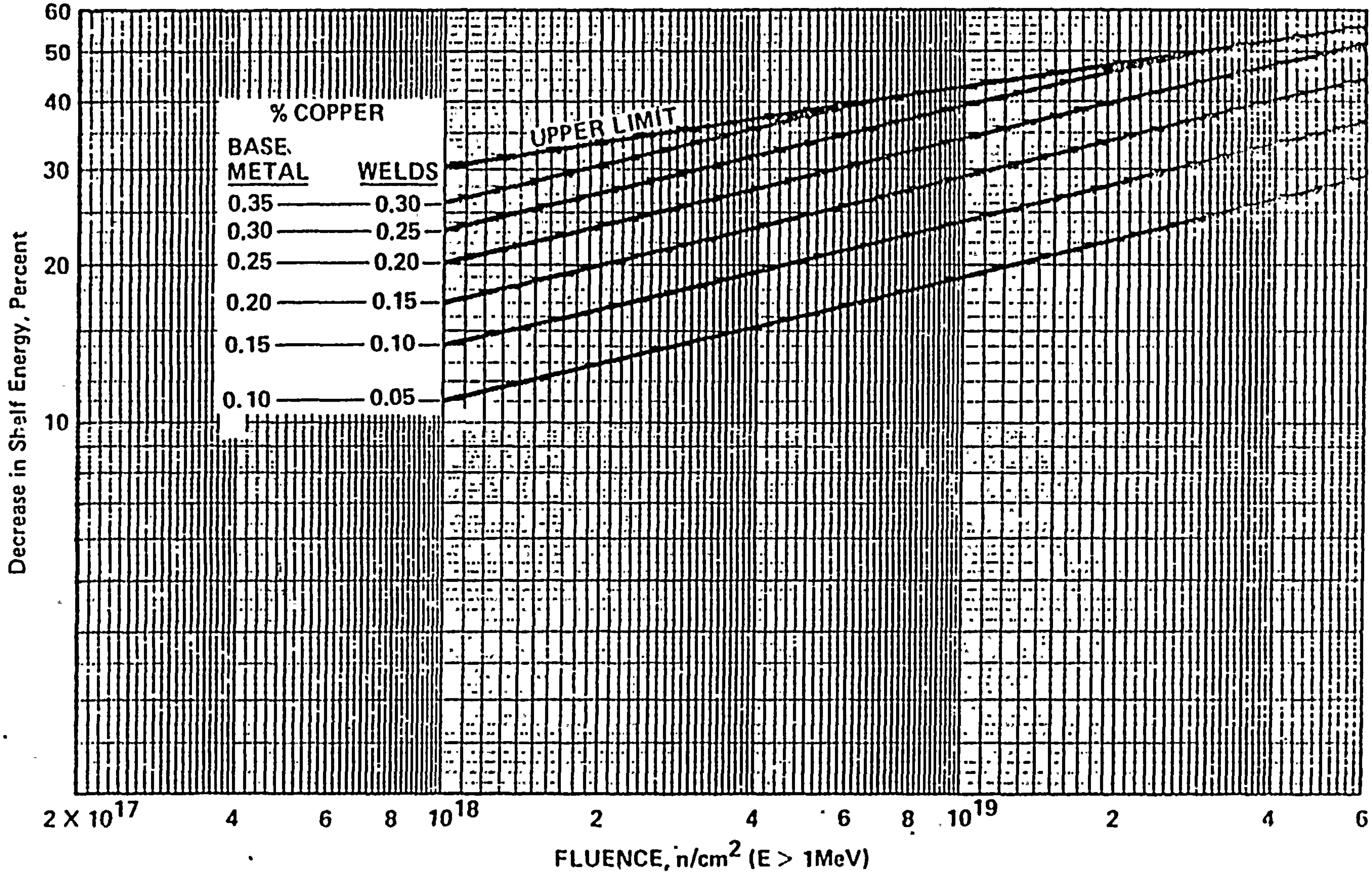


Figure 2 Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence.