PDR

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

September 19, 1984

Docket No. 50-247

Mr. John D. O'Toole
Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

SUBJECT: REACTOR VESSEL FLAW AT THE INDIAN POINT NUCLEAR GENERATING PLANT, UNIT NO. 2 (IP-2)

By letter dated Sepember 7, 1984 you submitted the fracture mechanics evaluation regarding the above subject. Our evaluation is based upon the review of the Westinghouse Report WCAP-10651, "Fracture Mechanics Evaluation of Inservice Inspection Indication, Indian Point Unit 2 Reactor Vessel".

In order to determine the safety margin between the ASME Code allowable flaw and the potential flaw in the IP-2 beltline, we request that you respond to the questions and concerns which are contained in Attachment 1. In addition, attachment 2, Draft Regulatory Guide 1.99, Rev. 2, dated July 23, 1984, is the staff's most "up-to-date" method of estimating the amount of irradiation damage to base metal and weld metal. Although the Draft Regulatory Guide has not been formally approved, its effect upon the safey margins for the potential flaw in the IP-2 reactor vessel should be evaluated.

You earliest response is requested.

The reporting and/or recordkeeping requirements of this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Steven A. Varga

Steven A. Varga, Branch Chief
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Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

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Attachment 1

Consolidated Edison Company of New York
Indian Point Unit No. 2 (IP-2)
Docket No. 50-247

To demonstrate the safety margins against brittle fracture for the potential flaw indication in the IP-2 reactor vessel beltline, the licensee has provided to the staff a fracture mechanics analysis which is contained in Westinghouse Report WCAP 10651 (Proprietary Class 2), "Fracture Mechanics Evaluation of Inservice Inspection Indication Indian Point Unit 2 Reactor Vessel." The Westinghouse report was submitted for staff review in a letter from J. D. O'Toole to S. A. Varga dated September 7, 1984. The following questions and comments relate to the analysis documented in the report.

1. The events analyzed in determining the ASME Code allowable flaw indication should include the Turkey Point Unit 4 LTOP event which occurred on November 28 and 29, 1981. Based upon the frequency of this type of event in all operating PWRs, the licensee should determine whether the event is considered upset or emergency and faulted. In analyzing this event for the IP-2 vessel, the pressures and temperatures to be considered should be those which would occur if the event were terminated by lifting of the IP-2 Pressurizer Safety Valve. If the Turkey Point set of events had occurred at IP-2, without operator action to terminate the transient, how much time would it take for the pressure to reach the Pressurizer Safety Valve set point?

2. If the flaw indication were located in the adjacent HAZ or base metal (Plate B 2003-1), what would be the ASME Code allowable flaw indication during normal, upset, test, emergency and faulted conditions?

3. Compare the end-of-life RT_{NDT} and ASME Code allowable flaw indication using the amount of increase in RT_{NDT} predicted by the "Guthrie" formula in Commission Report SECY 82-465 and the model in Draft Regulatory Guide 1.99 Rev. 2 (Attachment 2).

4. Indicate the references and heat numbers, and lot numbers for the weld wire and flux for each weld chemistry in Table 3-1.

5. Indicate the heat number and lot number for the weld wire and flux for the weld in Table 3-2.

6. Figure 3-2 indicates that the current fast neutron exposure at the inside surface - 345° Azimuthal Angle is 1.5×10^{18} n/cm^2. Consolidated Edison has reported to the staff in a telecon that after completing the sixth fuel cycle using a low leakage core, the current fast neutron exposure at the inside surface - 345° Azimuthal Angle is 1.77×10^{18} n/cm^2. Explain the difference in these estimates and use the more accurate number in the analysis.

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: "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 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 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 beltline materials that supplement the requirements for reactor vessel materials generally.

4. 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 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, 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 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 Revision 2.

The basis for equation (2) for ΔRT_{NDT} surface, given in Position C.1.a.2. of this Guide, is contained in publications by G. L. Guthrie[1] and G. R. Odette.[2] Both authors used as their data base surveillance data from commercial power reactors, but their analysis techniques were different. 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 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/cm^3 (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 inner surface locations do not appear to be great enough to warrant the use of a damage function such as displacements per atom (dpa)[4] in the analysis of the surveillance data base.[5]

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[1]G. L. Guthrie, "Charpy Trend Curves Based on 177 PWR Data Points," from LWR Pressure Vessel Surveillance Dosimetry Improvement Program, Quarterly Progress Report April 1983 - June 1983, Hanford Engineering Development Laboratory, NUREG/CR-3391, Vol. 2, HEDL-TME 83-22.

[2]G. R. Odette and P. M. Lombrozo, "Physically Based Regression Correlations of Embrittlement Data From Reactor Pressure Vessel Surveillance Programs," EPRI NP-3319, Final Report, January 1984, 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]ASTM E 693-79, "Standard Practice for Characterizing Neutron Exposures in Ferritic Steels in Terms of Displacements Per Atom (dpa)."

[5]W.N. McElroy, Editor. "LWR Power Reactor Surveillance Physics - Dosimetry Data Base Compendium," NUREG/CR 3319, HEDL TME 84-2 March 1984.

However, the neutron energy spectrum does change significantly with location in the vessel wall; hence for calculation of attenuation of radiation damage through the vessel wall, a damage function should be used to determine ΔRT_{NDT} versus radial distance into the wall. The most widely accepted damage function at this time is dpa and the attenuation formula (3) given in Position C.1.a.(2), is based on the attenuation of dpa through the vessel wall.

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 the studies upon 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 are 28°F for welds and 17°F for base metal, despite extensive statistical analysis. 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 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 plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and the upper shelf energy unambiguously.

3. When there are two or more surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best fit line drawn as described in Position C.2.a. 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.

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.

In using plant surveillance data to develop a plant-specific relationship of ΔRT_{NDT} to fluence, it was deemed advisable (because of scatter) to determine the slope, i.e., the fluence factor, from other than the plant data. Instead, Equation 2, paragraph C.1.a.(2), is to be fitted to 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 ΔRT_{NDT}, which seems reasonable for fitting surveillance data, because generally that datum will be the more recent one 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 damage to the beltline of reactor vessels of light water reactors should be based on the following procedures, within the limitations in Paragraph C.1.c.:

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

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

(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[6] of material may be used if there are sufficient test results to establish a mean and standard deviation for the class. Additional guidance for the estimation of initial RT_{NDT} is given in the Standard Review Plan, NUREG-0800, Section 5.3.2.

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

$$\Delta RT_{NDT} \text{ surface} = [CF] f^{(0.28-0.10 \log f)} \quad (2)$$

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). Linear interpolation is permitted.

In Tables I and II, "Percent Copper" and "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 fluence, "f," is the calculated value of the neutron fluence at the inner surface of the vessel at the location of the postulated defect, n/cm> (E>1 MeV) divided by 10^{19}.

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

To calculate ΔRT_{NDT} at any depth, (e.g., at 1/4T or 3/4T), the following attenuation formula should be used:

$$\Delta RT_{NDT} = [\Delta RT_{NDT} \text{ surface}] \, e^{-0.067x} \quad (3)$$

[6]For the welds with which this guide is concerned, for estimating Initial RT_{NDT}, class is generally determined by the welding flux; for base metal, by the ASTM Standard Specification.

where "x" (in inches) is the depth into the vessel wall measured from the vessel inner 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.

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \tag{4}$$

If a measured value of Initial RT_{NDT} for the material in question is used, σ_I may be taken as zero. If a generic value of Initial RT_{NDT} is used, σ_I should be obtained from the same set of data (see paragraph C.1.a.(1)). The standard deviations for ΔRT_{NDT}, "σ_Δ", are 28°F for welds and 17°F for base metal, except σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT} surface.

b. 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.

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 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 I and II 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 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 the following Paragraphs a. and b., respectively.

a. The adjusted reference temperature may be obtained by first fitting the surveillance data using Equation 2, paragraph C.1.a.(2), to obtain the relationship of ΔRT_{NDT} surface to fluence. To do so, calculate the chemistry factor, "CF," for the best fit as follows. Multiply each measured ΔRT_{NDT} by its corresponding fluence factor, sum the products and divide 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 ΔRT_{NDT} surface to fluence that fits the plant surveillance data in such a way as to minimize the sums of the squares of the errors.

To calculate the Margin in this case, use the procedure given in paragraph C.1.a.(3), except the values given there for σ_{Δ} may be cut in half.

If this procedure gives a higher value of adjusted reference temperature than that given by using the procedures of paragraph C.1.a, the former should be used if the surveillance data meet the criteria for credibility.

b. The decrease in upper-shelf energy may be obtained as follows. Plot the reduced plant surveillance data on Figure 2 of this Guide. Fit 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.

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.

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 and all applicants for an operating license should promptly review the basis for the pressure-temperature limits in their Technical Specifications for consistency with Positions C.1 or C.2 as appropriate. Those for whom the allowable operating period has been reduced or has already expired, when judged by the criteria of Revision 2, should promptly revise their operating procedures, as appropriate, to conform with the criteria of Revision 2 of this guide and submit the appropriate revision to their Technical Specifications within six months of the date of publication of Revision 2 of this guide in final form.

Those for whom the allowable operating period has been extended, when judged by the criteria of Revision 2, should submit the appropriate revision to their TSs no later than 90 days prior to the expiration of their current operating period.

3. The recommendations of regulatory position C.3 are unchanged from those used to evaluate construction permit applications docketed on or after June 1, 1977.

TABlE I
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

TABLE II
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