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SUBJECT: Forwards response to GL 92-01,Rev 1,dtd 920306, "Reactor Vessel Structural Integrity."

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July 2, 1992

10 CFR 50.54(f)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Reactor Vessel Structural Integrity

Reference: 1) Generic Letter 92-01, Revision 1, dated March 6, 1992

On March 6, 1992, the Nuclear Regulatory Commission (NRC) issued Revision 1 of Generic Letter 92-01 titled, "Reactor Vessel Structural Integrity," (Reference 1). This Generic Letter requested all licensees to provide certain information regarding fracture toughness and material surveillance for the reactor coolant pressure boundary. Reference 1 requires that this information be submitted to the NRC within 120 days of the date of this generic letter.

Wisconsin Public Service Corporation (WPSC) has previously provided much of this information in submittals pertaining to heatup and cooldown limit curves, the Kewaunee Nuclear Power Plant (KNPP) radiation surveillance program, and in response to generic letters. The purpose of this letter is to transmit the information requested under "Required Information" in the Generic Letter.

WPSC's response to each of the NRC's questions is provided in the attachment to this letter. This information verifies that the KNPP is in compliance with its licensing basis regarding reactor vessel fracture toughness and material surveillance for the reactor coolant pressure boundary.

In accordance with the requirements of 10CFR50.54(f), this submittal has been signed and notarized. Should you have any questions regarding this response, please contact a member of my staff.

Ltr. Encl. A001

ADD: D. McDonald
K. Wickman
B. Elliot

9207080012 920702
PDR ADOCK 05000305
P PDR

Sincerely,

C. R. Steinhardt
Senior Vice President - Nuclear Power

CAT/jac

Attach.

cc: US NRC Region III
Mr. Patrick Castleman, USNRC

Subscribed and Sworn to
Before Me This 2nd Day
of July 1992

Jeanne M. Stein
Notary Public, State of Wisconsin

My Commission Expires:
June 18, 1995

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ATTACHMENT

To

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated July 2, 1992

Response to Generic Letter 92-01, Revision 1

Request 1

Certain addressees are requested to provide the following information regarding Appendix H to 10 CFR Part 50:

> Addressees who do not have a surveillance program meeting ASTM E185-73, 79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).

WPSC Response 1

The Kewaunee Nuclear Power Plant surveillance program complies with Appendix H to 10CFR Part 50 as explained herein. ASTM E185 was originally issued in 1961 and was revised in 1966, 1970, 1973, 1979, and 1982. Appendix H to 10 CFR Part 50, which was first published in 1973, outlines the requirements for compliance with the pertinent version of ASTM E185.

The current version of Appendix H to 10 CFR Part 50 reads:

> That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirements of the edition of the ASTM E185 that is current on the issue date to which the reactor vessel was purchased. Later versions of the ASTM standard may be used, but including only those editions through 1982. For each capsule withdrawal after July 26, 1983, the test procedures and reporting requirements must meet the requirements of ASTM E185-82 to the extent practical for the configuration of the specimens in the capsule. For each capsule withdrawal prior to July 26, 1983 either the 1973, the 1979, or the 1982 edition of ASTM E185 may be used.

The KNPP vessel was designed to the 1968 Edition through Winter 1968 Addenda of the ASME Code. ASTM E185-70 was the standard in place at the time the surveillance program was established. Testing of surveillance capsules after July 26, 1983 has been performed in accordance with ASTM E185-82.

The surveillance program was approved during original plant licensing (a). Subsequently, analysis results from three surveillance capsules have been submitted to the NRC as required by 10 CFR 50, Appendix H III.A (b and c). The resultant fracture toughness data were previously approved by the NRC for the Pressurized Thermal Shock rule and in support of the heatup and cooldown limit curves (d through j). Therefore, the surveillance program for KNPP meets the requirements of Appendix H to 10 CFR Part 50 and an exemption request is not necessary.

Request 2.a

Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

> Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraph C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C. of Appendix G to 10 CFR Part 50.

WPSC Response 2.a

An acceptable method for predicting the decrease in upper shelf energy (USE) as a function of copper content and fluence is given in Figure 2 of Regulatory Guide 1.99, Revision 2. In a recent NRC safety evaluation report dated January 25, 1990 regarding KNPP's response to Generic Letter 88-11, the predicted Charpy USE for the limiting circumferential beltline weld metal at the end of license was verified to be above 50 ft./lb (k). WPSC's preliminary evaluation of end of license Charpy USE also confirms this finding as shown in Table 1:

TABLE 1

KNPP VESSEL TOUGHNESS DATA

Component	Unirradiated NMWD Upper Shelf Energy (Ft-Lb)(1)	Cu%	Fluence ¼T n/cm^2 (E>1.0 MEV)(2)	Corrected USE for L-T Direction (Ft-Lb)	Decrease in Shelf Energy Table 2 Reg. Guide 1.99, Rev. 2(4)	EOL (32 EFPY) USE(4) (Ft-Lb)
Intermediate Shell	141.5	0.06(1)	2.17×10^{19}	92(3)	20%	74
Lower Shell	149	0.06(1)	2.17×10^{19}	97(3)	20%	78
Belt Line Weld	126	0.283(5)	2.17×10^{19}	126	42%	73

(1) Data taken from WCAP-12020 dated November, 1988.

(2) Fluence ¼T = $\text{fluence}_{\text{surface}} e^{-.24X}$
X = .25 Thickness where T = 6.5 inches
$\text{Fluence}_{\text{surface}}$ and thickness taken from WCAP-13229 dated March, 1992

(3) Correction to USE for Charpy specimens oriented in longitudinal (L-T) direction by reducing by a factor of .65 per NRC Std. Review Plan Section 5.3.2.

(4) EOL USE determined from Table 2 Reg. Guide 1.99, Rev. 2

(5) Cu% taken from WCAP-13257 dated March, 1992.

Request 2.b

Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the considerations given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

1. The results from all Charpy and drop weight tests for all unirradiated beltline materials, and the unirradiated reference temperature for each beltline material, and the method for determining the unirradiated reference temperature from the Charpy and drop weight test;

WPSC Response 2.b.1

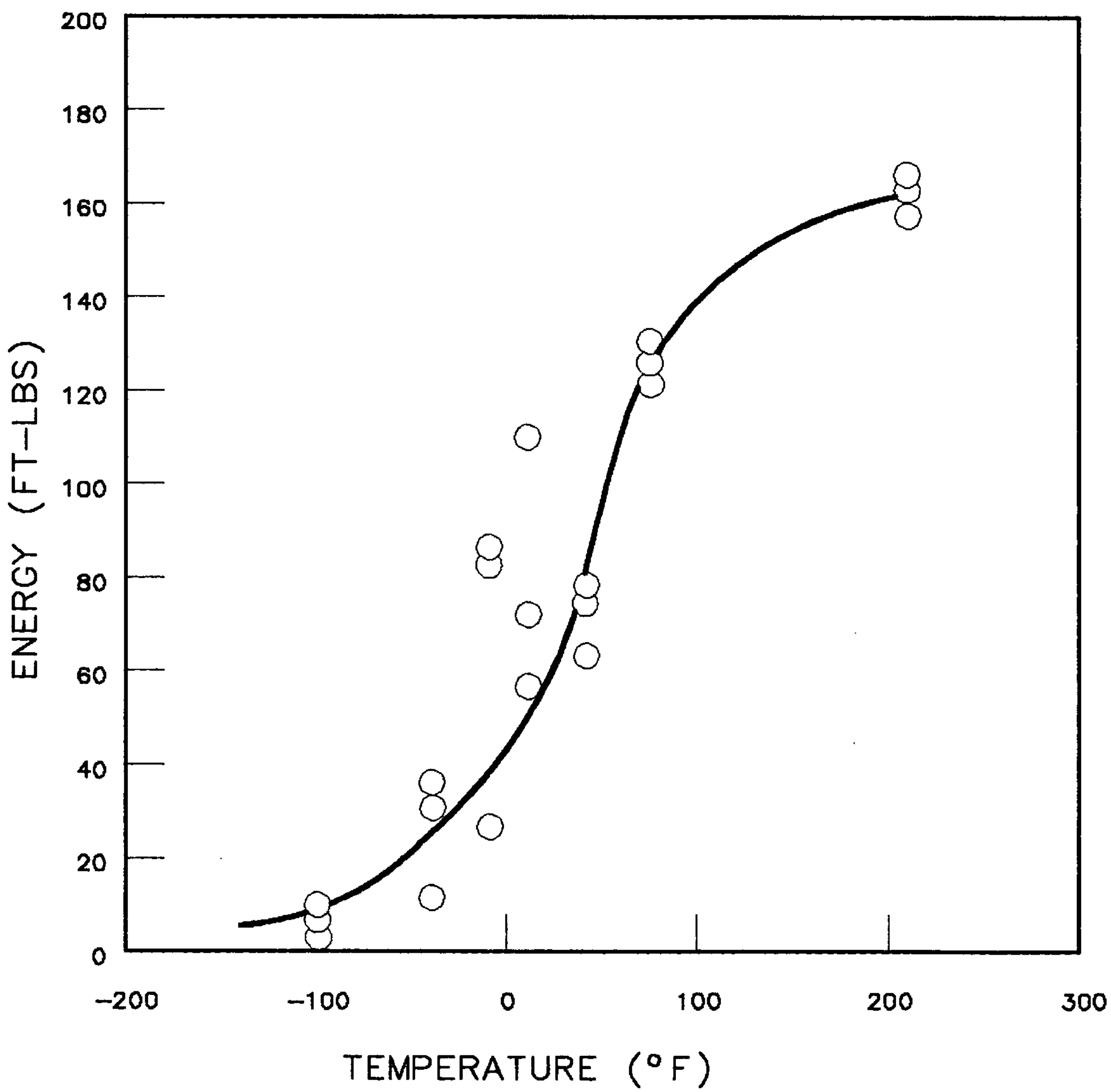
Both Charpy and drop weight testing of KNPP reactor vessel unirradiated beltline materials was performed by Westinghouse Electric Corporation. The material was supplied by Combustion Engineering, Inc. from two steel ring forgings (heats 122X208VA1 and 123X167VA1) which were used in the intermediate and lower shell forgings of the vessel, respectively. A weldment which joined these two heats of material was also supplied by Combustion Engineering, Inc.

Test material from each shell forging was heat-treated with the shells. All test specimens were machined from the 1/4 thickness location of the forgings after performing a simulated post-weld stress-relieving treatment on the test material. The test specimens represent material taken at least one forging thickness (6-1/2 inches) from the quenched ends of the forging. Specimens were machined from weld and heat-affected zone metal from a stress-relieved weldment which joined sections of the two shell ring forgings. All heat-affected zone specimens were obtained from the weld heat-affected zone of forging 122X208VA1.

Charpy V-notch impact tests were performed on the vessel shell ring forgings 122X208VA1 and 123X167VA1 at various temperatures from -150 to 210°F to obtain a full Charpy V-notch transition curve. The results are reported in Tables 2 through 5. Charpy V-notch impact tests were performed on weld metal and HAZ metal at various temperatures from -200 to 210°F. The results are reported in Tables 6 and 7, respectively.

TABLE 2 (1)
PREIRRADIATION CHARPY V-NOTCH IMPACT PROPERTIES
FOR THE
KEWAUNEE NUCLEAR POWER PLANT REACTOR PRESSURE
VESSEL INTERMEDIATE SHELL FORGING 122X208VA1

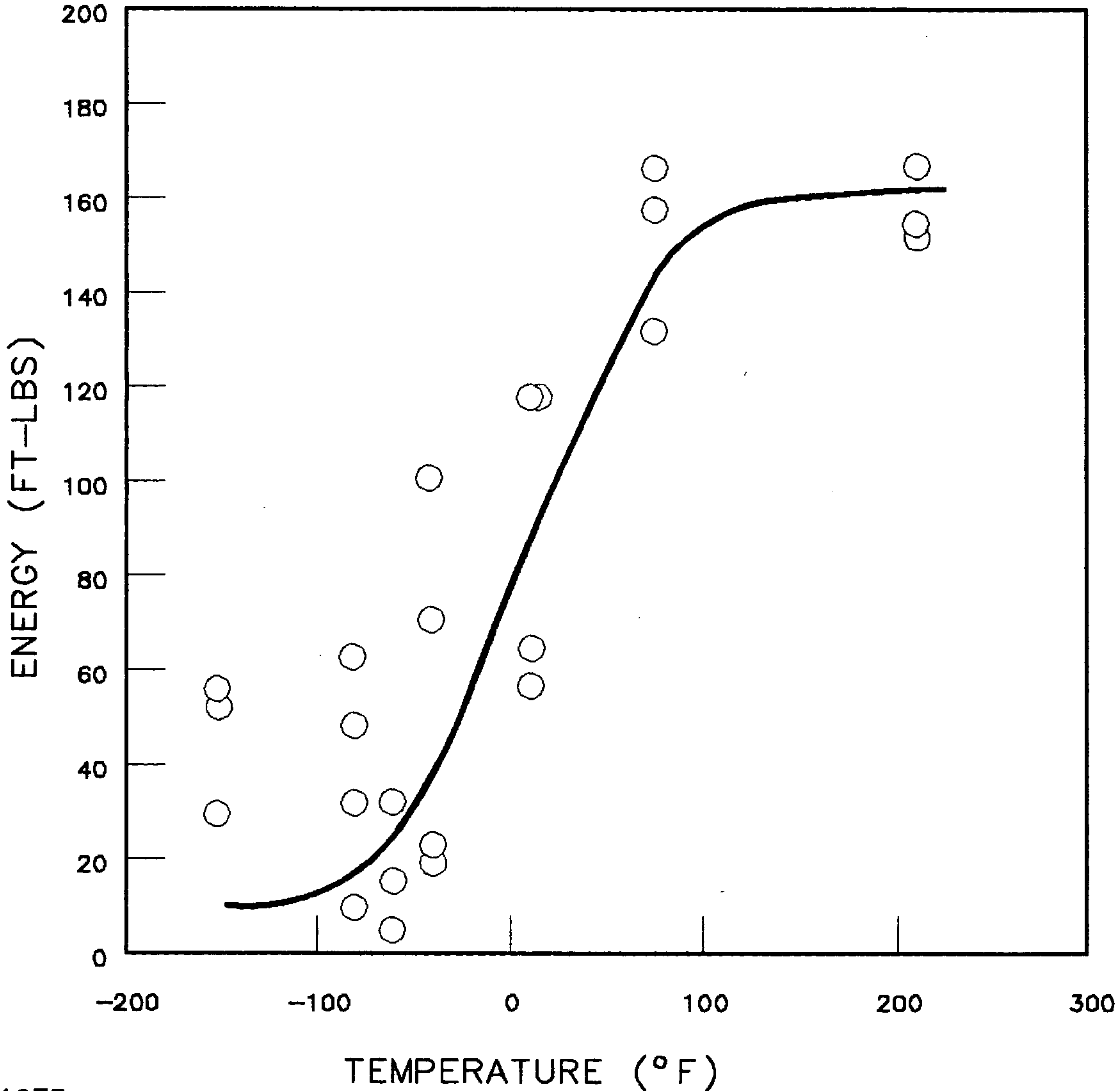
TEST TEMP. (°F)	IMPACT ENERGY (ft-lb)	SHEAR (%)	LATERAL EXPANSION (mils)
-100	4.5	5	7
-100	2.0	3	2
-100	9.5	5	7
-40	12.0	13	12
-40	31.5	17	26
-40	37.0	21	30
-10	27.5	23	27
-10	83.0	43	71
-10	85.0	47	72
10	57.0	30	48
10	72.0	36	64
10	110.0	37	50
40	63.0	37	54
40	73.0	36	62
40	78.0	41	65
75	119.5	73	89
75	123.5	73	88
75	129.5	75	90
210	156.0	100	96
210	162.0	100	92
210	165.0	100	96



(1) DATA TAKEN FROM WCAP-8107, DATED APRIL, 1973

(1)
TABLE 3
PREIRRADIATION CHARPY V-NOTCH IMPACT DATA
FOR THE
KEWAUNEE NUCLEAR POWER PLANT REACTOR PRESSURE
LOWER SHELL FORGING 123X167VA1

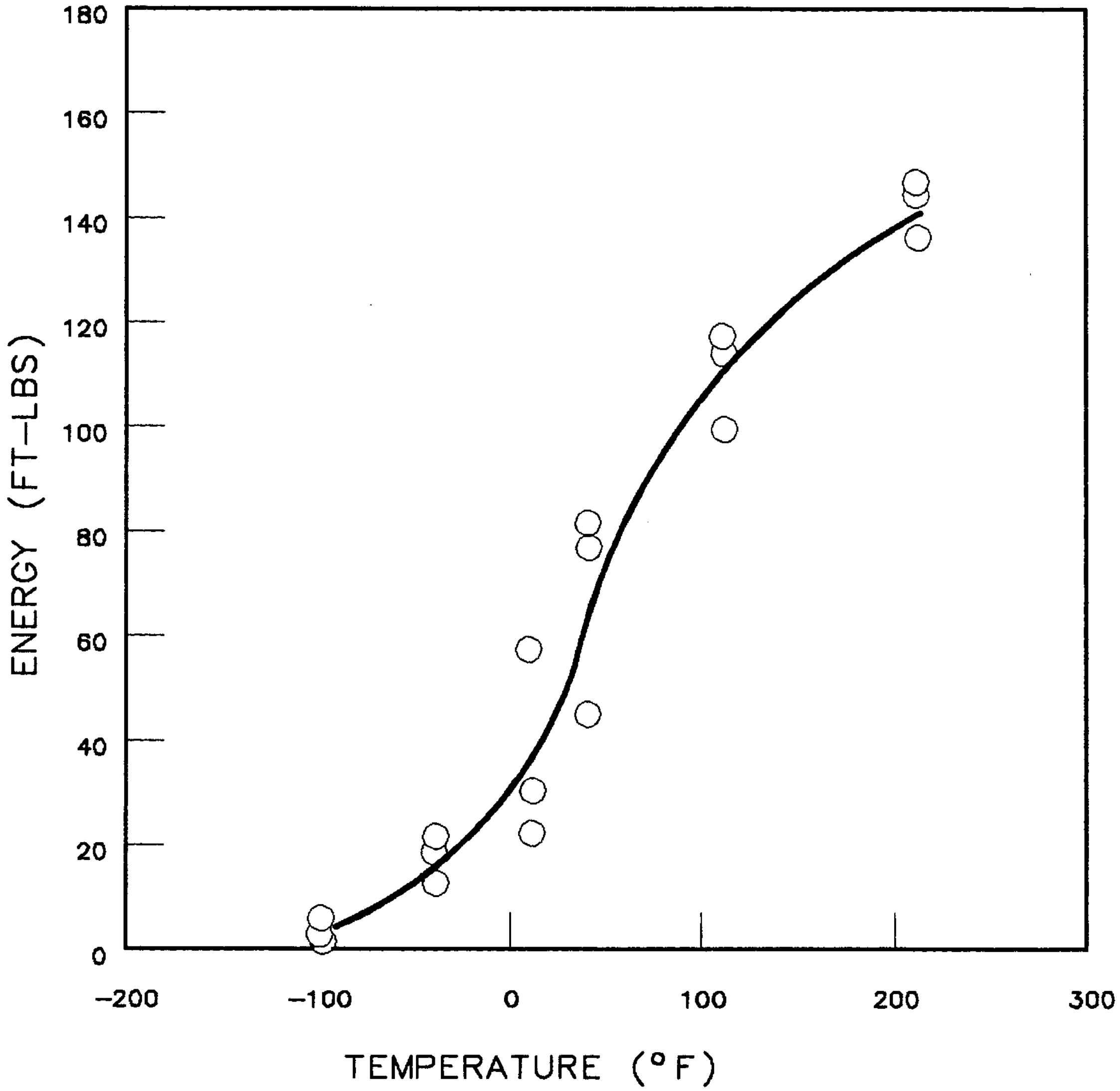
TEST TEMP. (°F)	IMPACT ENERGY (ft-lb)	SHEAR (%)	LATERAL EXPANSION (mils)
-150	29.0	23	26
-150	52.0	41	52
-150	55.5	27	46
-80	31.0	27	28
-80	10.0	9	8
-80	47.5	27	42
-80	62.5	37	52
-60	14.5	13	14
-60	4.5	5	4
-60	31.0	25	28
-40	22.0	21	19
-40	18.5	13	17
-40	70.5	43	60
-40	100.0	63	82
10	64.0	43	54
10	56.0	53	85
10	117.0	59	86
10	117.0	57	86
75	156.0	100	86
75	164.5	100	99
75	130.5	73	88
210	153.0	100	92
210	165.0	100	96
210	150.0	100	95



(1) DATA TAKEN FROM WCAP-8107, DATED APRIL, 1973

TABLE 4 (1)
SUPPLEMENTARY CHARPY V-NOTCH IMPACT DATA
FOR THE
KEWAUNEE NUCLEAR POWER PLANT REACTOR PRESSURE
VESSEL SHELL FORGING 122X208VA1
(SPECIMENS ORIENTED IN THE AXIAL DIRECTION)

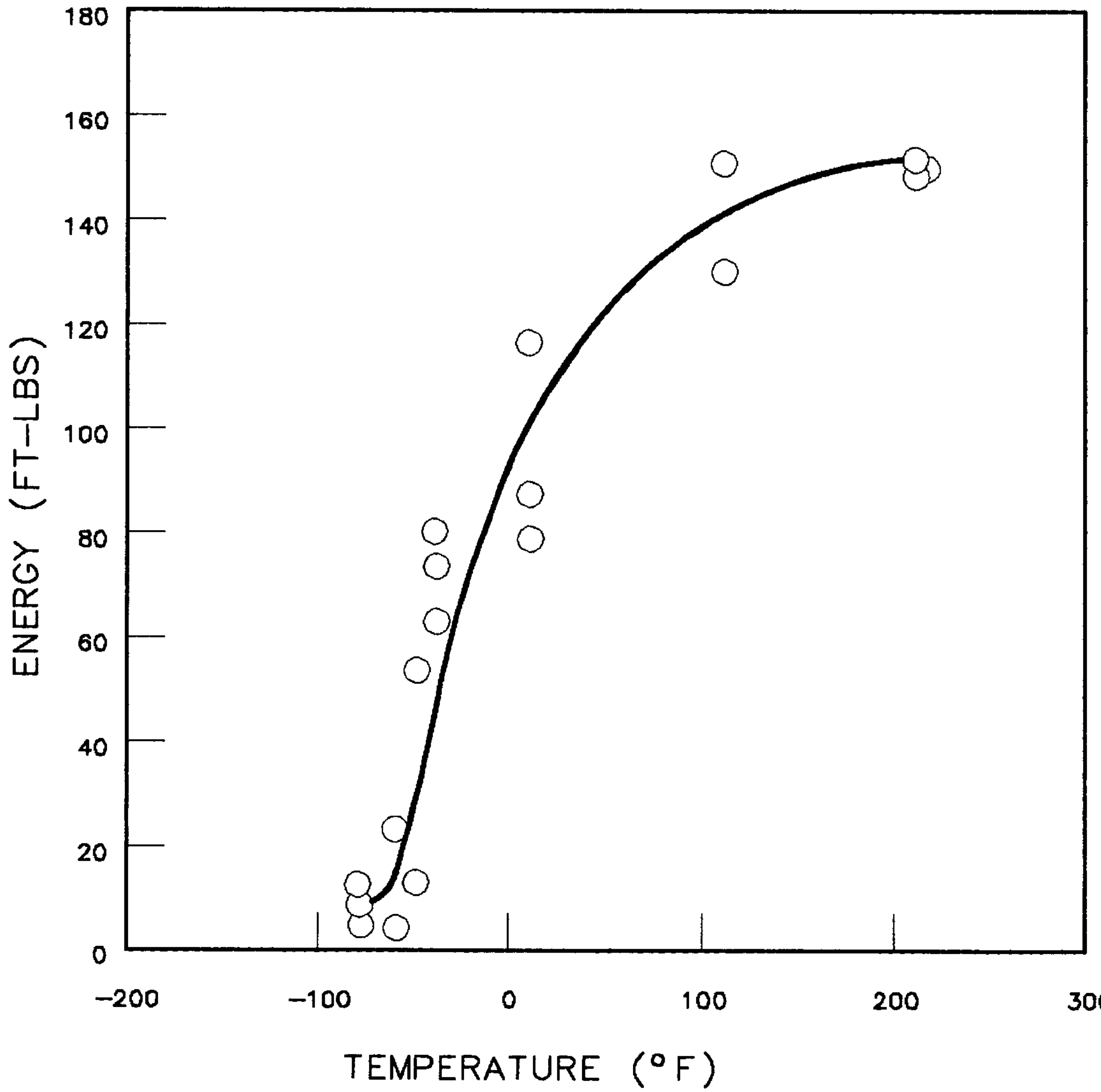
TEST TEMP. (°F)	IMPACT ENERGY (ft-lb)	SHEAR (%)	LATERAL EXPANSION (mils)
-100	2.0	5	5
-100	2.0	5	4
-100	5.5	5	7
-40	12.5	13	11
-40	19.5	18	18
-40	20.5	19	18
10	21.0	13	21
10	57.0	30	51
10	30.0	30	59
40	77.0	39	65
40	45.0	29	40
40	81.5	53	69
110	98.5	56	79
110	114.0	76	82
110	116.0	76	85
210	144.0	100	97
210	145.0	100	94
210	135.5	100	95



(1) DATA TAKEN FROM WCAP-8107, DATED APRIL, 1973

TABLE 5 (1)
SUPPLEMENTARY CHARPY V-NOTCH IMPACT DATA
FOR THE
KEWAUNEE NUCLEAR POWER PLANT REACTOR PRESSURE
VESSEL SHELL FORGING 123X167VA1
(SPECIMENS ORIENTED IN THE AXIAL DIRECTION)

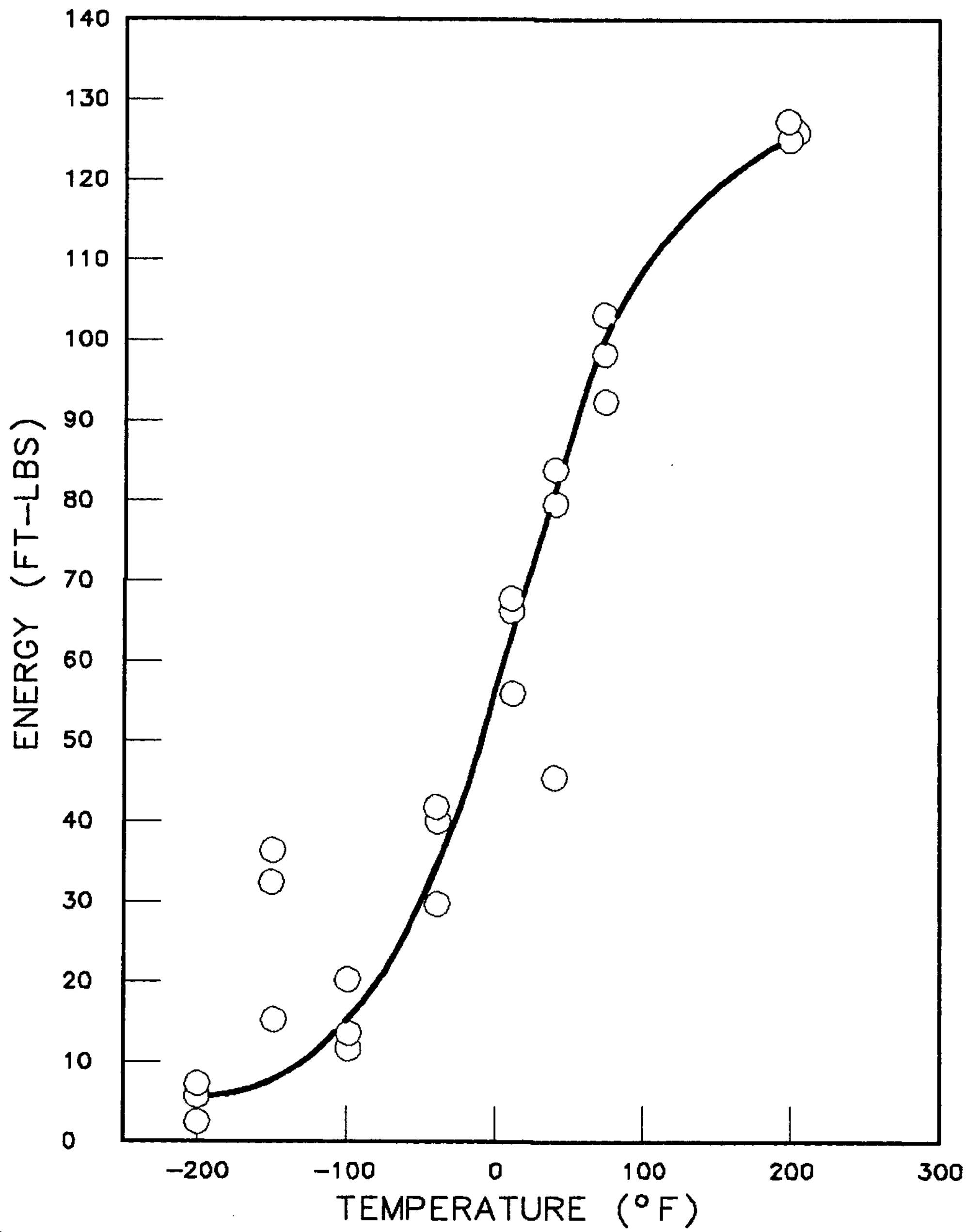
TEST TEMP. (°F)	IMPACT ENERGY (ft-lb)	SHEAR (%)	LATERAL EXPANSION (mils)
-80	12.0	25	36
-80	10.0	5	11
-80	6.5	5	6
-60	23.0	21	22
-60	4.5	3	4
-50	53.0	37	50
-50	13.0	12	12
-40	63.0	43	55
-40	73.5	42	66
-40	80.0	43	69
10	79.0	42	66
10	87.0	42	72
10	116.5	67	93
110	130.0	88	95
110	150.0	100	96
210	148.0	100	100
210	149.0	100	97
210	149.5	100	96



(1) DATA TAKEN FROM WCAP-8107, DATED APRIL, 1973

TABLE 6 (1)
PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE KEWAUNEE NUCLEAR POWER PLANT REACTOR PRESSURE VESSEL WELD METAL

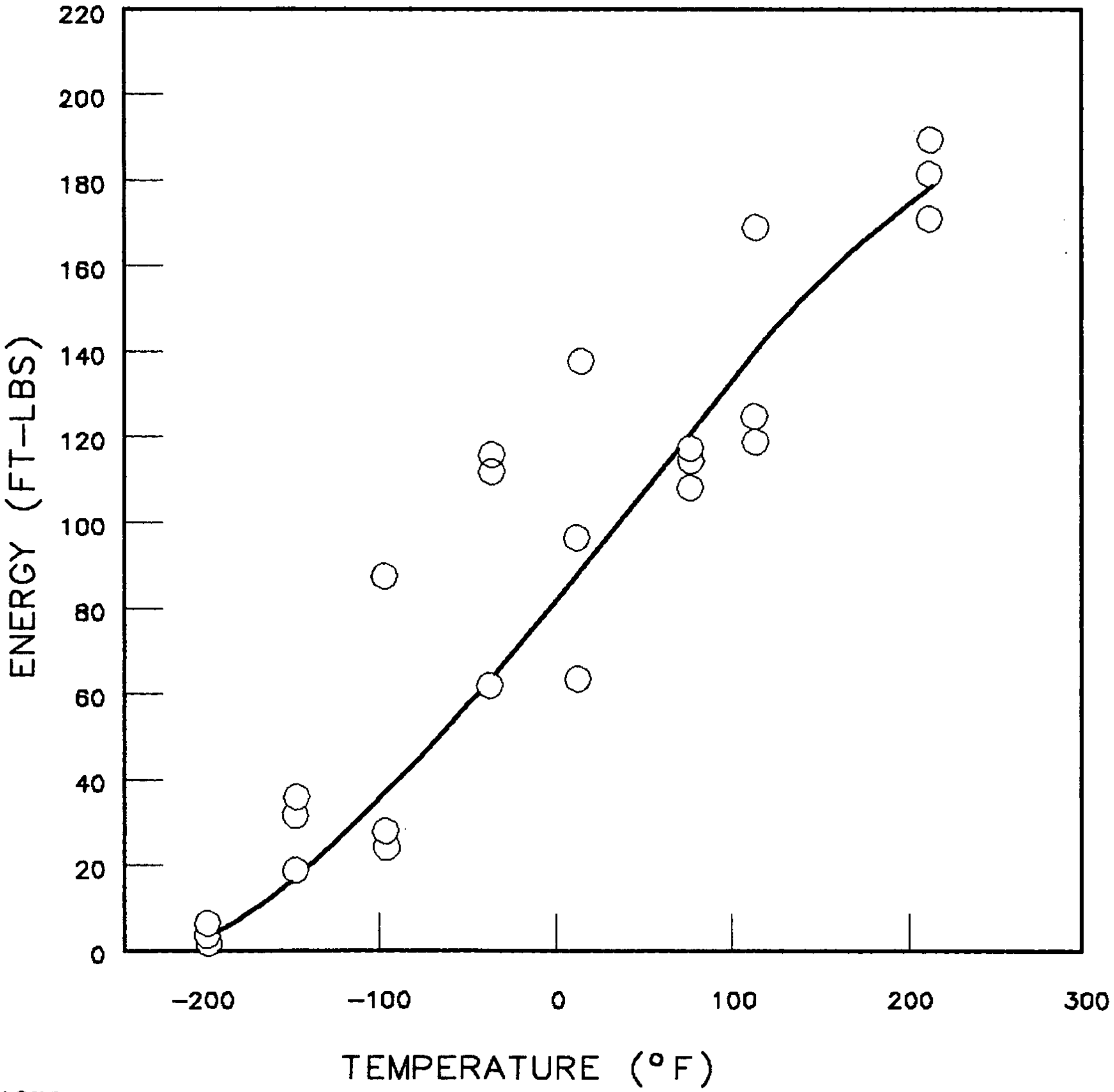
TEST TEMP. (°F)	IMPACT ENERGY (ft-lb)	SHEAR (%)	LATERAL EXPANSION (mils)
−200	3.0	3	3
−200	5.5	5	6
−200	7.0	5	6
−150	36.0	34	34
−150	15.0	17	16
−150	32.0	27	31
−100	11.5	9	13
−100	13.0	13	12
−100	20.0	14	18
−40	40.0	38	37
−40	41.0	43	37
−40	29.5	37	29
10	66.0	55	60
10	67.0	64	62
10	55.5	43	50
40	79.0	81	71
40	83.0	81	72
40	45.0	66	73
75	97.5	90	82
75	92.0	89	77
75	102.5	96	87
210	125.0	100	98
210	126.5	100	98
210	126.0	100	99



(1) DATA TAKEN FROM WCAP-8107, DATED APRIL, 1973

TABLE 7 [1]
PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR KEWAUNEE NUCLEAR POWER PLANT REACTOR PRESSURE VESSEL WELD HEAT AFFECTED ZONE METAL

TEST TEMP. (°F)	IMPACT ENERGY (ft-lb)	SHEAR (%)	LATERAL EXPANSION (mils)
-200	6.0	9	5
-200	4.0	5	8
-200	3.0	3	2
-150	32.0	27	28
-150	35.0	29	27
-150	19.0	18	16
-100	28.0	18	24
-100	25.0	18	22
-100	87.0	45	59
-40	112.0	56	74
-40	115.0	54	82
-40	62.0	51	54
10	96.5	71	77
10	138.0	77	85
10	64.0	58	57
75	109.0	98	87
75	114.0	100	83
75	117.5	100	91
110	169.0	100	93
110	125.0	100	96
110	119.0	100	94
210	181.0	100	90
210	172.0	100	86
210	190.0	100	87



(1) DATA TAKEN FROM WCAP-8107, DATED APRIL, 1973

The reference nil-ductility temperature (RT_{NDT}) is defined as the greater of either the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) or the temperature 60°F less than the 50 ft. lb. (and 35-mil lateral expansion) temperature as determined from Charpy specimens oriented normal (transverse) to the major working direction of the material. The nil-ductility transition temperature (NDTT) was determined from the two shell ring forgings by drop weight tests. The results are reported in Table 8:

TABLE 8[(1)]

PRE-RADIATION NDTT DATA FOR THE KEWAUNEE NUCLEAR POWER PLANT REACTOR PRESSURE VESSEL FORGINGS

Material	NDTT (°F)
122X208VA1	60
123X167VA1	20
(1) Data taken from WCAP-8107 dated April, 1973	

Request 2.b

Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the considerations given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

(2) the heat treatment received by all beltline and surveillance materials;

WPSC Response 2.b.2

The heat treatment received by all the beltline and surveillance materials is presented in Table 9.

TABLE 9(1)

HEAT TREATMENT OF BELTLINE AND SURVEILLANCE MATERIALS

Description		Heat Treatment
Weld	Surveillance Weld	Stress-relieved at 1150°F for 19¼ hours, furnace cooled.
	Intermediate to Lower Shell Forging Weld	Stress relieved at 1150°F ±25°F for 40 hours, furnace cooled.
Forging	Intermediate Shell 122X208VA1	Heated at 1550°F for 8 hr., Water-Quench Tempered at 1230°F for 14 hr., Air-cooled Stress-relieve data 1150°F for 21 hr., Furnace cooled.
	Lower Shell 123X167VA1	Heated at 1550°F for 8 hr., Water-Quench Tempered at 1220°F for 14 hr., Air-cooled Stress-relieved at 1150°F for 21 hr., Furnace cooled.
	Surveillance Intermediate Shell	Same as Intermediate Shell
	Surveillance Lower Shell	Same as Lower Shell

(1) Data taken from letter from E. W. James (WPSC) to A. Schwencer (US NRC) dated February 1, 1978

Request 2.b

Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the considerations given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

(3) the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;

(4) the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld;

WPSC Response to 2.b.3 and 2.b.4

The heat numbers and code numbers for the beltline/surveillance forgings are presented in Table 10.

TABLE 10(1)

HEAT AND CODE NUMBERS FOR BELTLINE/SURVEILLANCE FORGINGS

Description		Heat Number	Code Number
Forging	Intermediate Shell	122X208VA1	B-6306
	Lower Shell	123X167VA1	B-6307
	Surveillance Intermediate Shell	122X208VA1	B-6306
	Surveillance Lower Shell	123X167VA1	B-6307

(1) Data taken from letter from E. W. James (WPSC) to A. Schwencer (US NRC) dated February 1, 1978

The wire heat numbers and batches for the beltline/surveillance welds are presented in Table 11.

TABLE 11[1]

WIRE HEAT NUMBER AND BATCH NUMBER FOR BELTLINE/SURVEILLANCE WELDS

Description		Filler Metal and Heat Number	Type of Flux and Batch Number
Weld	Surveillance Weld	B-4 Mod. 1P3571	Linde 1092 3958
	Intermediate to Upper Shell Forging	B-4 Mod. 1P3571	Linde 1092 3958
	Intermediate to Lower Shell Forging	B-4 Mod. 21935	Linde I092 3869

(1) Data taken from letter from E. W. James (WPSC) to A. Schwencer (US NRC) dated February 1, 1978

Request 2.b

Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the considerations given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

(5) the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulphur for each beltline and surveillance material; and

WPSC Response to 2.b.5

The chemical composition for the beltline forgings/weld and beltline surveillance material is presented in Table 12.

Document Control Desk
July 2, 1992
Attachment 1, Page 16

TABLE 12

BELTLINE FORGING/WELD AND BELTLINE SURVEILLANCE WELD CHEMICAL COMPOSITION

Material	wt. %																			
	C	Mn	P	S	Si	Ni	Mo	Cr	Cu	V	Co	Sn	Ti	Zr	As	Sb	Al	B	Zn	Nz
Intermediate Shell																				
Heat 122 X 208 VA1																				
CTR: Mill Analysis	.23	.68	.009	.011	.27	.70	.59	.36	-	.01	.01	-	-	-	-	-	-	-	-	
Surveillance Material (WCAP-8908)	.21	.69	.01	.011	.25	.71	.58	.40	.06	.02	.011	.01	<.001	.001	.001	<.001	.004	<.003	-	.006
Lower Shell																				
Heat 123 X 167 VA1																				
CTR: Mill Analysis	.21	.73	.007	.009	.22	.74	.61	.36	-	.01	.013	-	-	-	-	-	-	-	-	-
Surveillance Material (WCAP-8908)	.20	.79	.01	.009	.28	.75	.58	.35	.06	.02	.012	.01	<.001	.001	.004	.001	.006	<.003	-	.01
Surveillance Weld Metal (WCAP-8908)	.12	1.37	.016	.011	.20	.745 (1)	.48	.09	.283 (1)	.002	.001	.004	<.001	<.001	.004	.001	.01	<.003	<.001	.012
Intermediate to Lower Shell Forging Weld	.15	2.06	.010	.010	.06	.78	.49	.06	-	-	Wire Analysis Only No Deposit Analysis Available									

(1) % Weight Copper and Nickel values were determined by averaging the chemistry values from eighteen data points (reference WCAP-13257). All other data taken from letter from E. W. James (WPSC) to A. Schwencer (US NRC) dated February 1, 1978.

Request 2.b

(6) the heat number of the wire used for determining the weld and chemical composition if different than Item (3) above.

WPSC Response to 2.b.6

Heat numbers of the wire used for determining the beltline/surveillance weld and chemical composition are as described under WPSC response to requests 2.b.3, 2.b.4, and 2.b.5 above.

Request 3.a

Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:

How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525°F were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and the Charpy upper-shelf energy.

WPSC Response 3.a

The KNPP reactor coolant system is normally operated in a temperature band in which average RCS temperature is maintained between 547 and 562°F between 0 and 100% power. Tcold ranges from 547 to 532°F from 0 to 100% power. Also, the Plant Operating procedures administratively require the reactor coolant system be equal to or greater than 540°F prior to criticality. Operation outside this band is abnormal and any such occurrence would be documented and appropriate parameters recorded in accordance with administrative controls. In an effort to determine if abnormal operation had occurred at power and below 525°F, WPSC performed a review of applicable historical surveillance data. The review focused on Tcold and Tave associated with operation prior to a reactor trip, after going critical from a heatup, prior to going subcritical from a cooldown, and at various plant loadings and unloadings. This evaluation encompassed operation from initial plant startup to present and included review of control room logs, charts from the plant process computer, and control room strip chart records. No occurrences of power operation below 525°F were identified in this review.

WPSC believes that the beltline weld has accumulated very little or no fluence due to operation below 525°F. Based on this evaluation, the total operating time below 525°F is believed to be nonexistent or very small and is not expected to have an impact on predictions of RT_{NDT} or reduction in upper shelf energy. KNPP has surveillance capsules consisting of beltline weld, forging, dosimeter, and correlation monitor material installed in the reactor vessel. This surveillance material will experience equivalent operating conditions as the reactor vessel.

Therefore, any undetected period of operation that may have occurred below 525°F is being monitored and accounted for through the KNPP radiation surveillance program.

Request 3.b

Utilities are requested to provide information on how their surveillance results have been used in response to GL 88-11 and how the predicted amount of embrittlement has been considered.

WPSC Response to 3.b

In 1986, WPSC used draft Revision 2 of Regulatory Guide 1.99 dated February, 1986 to construct pressure-temperature limits for 15 effective full-power years. The final Revision 2 of Regulatory Guide 1.99 became effective May 1988. Generic Letter 88-11 requested WPSC to utilize Regulatory Guide 1.99, Revision 2, to evaluate the effect of neutron irradiation on KNPP reactor vessel materials.

Since the calculational methods presented in the final version of Revision 2 of Regulatory Guide 1.99 varied slightly from the methods presented in the draft version, WPSC generated new adjusted reference temperature (ART) values in response to Generic Letter 88-11. Comparison of the ARTs revealed that the pressure-temperature curves constructed in 1986 are conservative compared to the limits allowed by the final version of Revision 2 of Regulatory Guide 1.99.

Regulatory Guide 1.99 allows the calculation of ART to be performed with either of two methods: using data from the reactor vessel surveillance capsule program or using generic calculational methods. The current KNPP P-T limits were obtained using the generic calculational methods described in draft Revision 2 of Regulatory Guide 1.99.

Since the KNPP surveillance data meets the regulatory guide's credibility criteria, the surveillance data method was used to calculate the ART values in response to Generic Letter 88-11. The results of this calculation are shown in the Table 13 below:

TABLE 13[1]

COMPARISON OF ADJUSTED REFERENCE TEMPERATURE VALUES

Location in Reactor Vessel Wall	**Curreut ART Values Used in Constructiou of PT Limits Based on Geueric Methods in Draft Revision 2**	**ART Values Calculated for Geueric Letter 88-11 and Based on Surveillauce Data and Final Revisiou 2**
1/4 Thickness	226	219
3/4 Thickness	183	173

(1) Letter from C. R. Steinhardt (WPSC) to Document Control Desk (US NRC) dated November 1, 1988

This table shows that the current ART values are higher, resulting in KNPP P-T limits more conservative than those permitted by the final version of Revision 2 of Regulatory Guide 1.99.

Subsequent to our response to Generic Letter 88-11, WPSC has calculated new ART values for heatup and cooldown limit curves applicable for 20 EFPY (1). ART values were calculated using data from both the surveillance capsule program and Revision 2 of Regulatory Guide 1.99. ART values calculated using methods described in Revision 2 of Regulatory Guide 1.99 are higher and were therefore utilized in preparation of the heatup and cooldown limit curves applicable for 20 EFPY.

<u>Request 3.c</u>

If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper-shelf energy for each beltline material as predicted for December 16, 1991 and for its current license.

WPSC Response 3.c

The measured increase in reference temperature and mean value plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, are presented in Table 14. No data from the KNPP surveillance program has exceeded the mean-plus-two standard deviation bound predicted by Regulatory Guide 1.99, Revision 2.

Tahle 14(1)

COMPARISON OF KNPP REACTOR VESSEL SURVEILLANCE CAPSULE CHARPY IMPACT TEST RESULTS WITH REGULATORY GUIDE 1.99 REVISION 2 PREDICTIONS

Material Description	Capsule	ΔRT_{NDT} (30 ft/lb Increase)			ΔUSE	
		Predicted Mean(2) (°F)	Mean(3) Plus 2σ (°F)	Measured (°F)	Predicted(2) Decrease (%)	Measured Decrease (%)
Intermediate Shell 122X208VA1	V	32	66	0	17	0
	R	44	78	15	22.5	0
	P	47	81	25	24	2
Lower Shell 123X167VA1	V	32	66	0	17	0
	R	44	78	20	22.5	2.5
	P	47	81	20	24	0
Circumferential Weld	V	162	218	175	30	35
	R	226	282	235	40	38
	P	242	298	230	44	39.5

(1) Data taken from WCAP-12020 dated November, 1988.

(2) From Reg. Guide 1.99, Rev. 2.

(3) Two sigma is 56°F for welds and 34°F for base metal.

The measured decrease in upper shelf energy and values predicted by Regulatory Guide 1.99, Revision 2 are also presented in Table 14. Note the measured decrease in Charpy upper shelf energy from the first surveillance capsule, i.e. Capsule V, marginally exceeds the value predicted using the guidance in paragraph C.1.2 of Regulatory Guide 1.99, Revision 2. Historically, testing of a 508 CL. 2 material in the nuclear industry has a more rapid decrease in Charpy upper shelf energy for exposures up to 1.0E19 n/cm^2 (E$\geq$ 1.0 MeV); the decrease in Charpy upper shelf energy for exposures beyond 1.0E19 n/cm^2 (E $\geq$ 1.0 MeV) is not as pronounced. Results from analysis of the two more recent surveillance capsules indicate that predicted decrease in Charpy upper shelf energy from Regulatory Guide 1.99, Revision 2, is greater than the observed measured decrease in upper shelf energy.

Neutron exposure projections for December 16, 1991, are not readily available. However, neutron exposure projections for 15 EFPYs have been calculated. KNPP is projected to reach 15 EFPYs in November 1992. Adjusted reference temperature and Charpy upper shelf energy data for the beltline materials for 15 EFPY and for current life as predicted by Regulatory Guide 1.99, Revision 2, are provided in Tables 15 and 16, respectively. This data demonstrates the continued integrity of the KNPP reactor vessel considering the effects of neutron embrittlement.

TABLE 15

RT_{PTS} VALUES FOR KNPP FOR 15, 32, AND 34 EFPY

Material Descriptiou	Cu%	Ni%	Initial RT_{NDT}	Margin	RT_{PTS}		
					15 EFPY	32 EFPY	34 EFPY
Intermediate Shell	0.06	0.71	60	34	138	142.24	142.61
Lower Shell	0.06	0.75	20	34	98	102.24	102.61
Circumferential Weld(1)	0.283	0.745	-56	66	225	284.67	286.79

(1) RT_{PTS} values for circumferential weld using surveillance capsule data were calculated as 236°F, 259.37°F, and 261.29°F for 15, 32, and 34 EFPY, respectively. RT_{PTS} values taken from letter from C. R. Steinhardt (WPSC) to Document Control Desk (US NRC) dated May 27, 1992.

TABLE 16

CHARPY UPPER SHELF ENERGY VALUES FOR KNPP FOR 15, 32, AND 34 EFPY

			15 EFPY			
Component	**Unirradiated NMWD Upper Shelf Euergy (FT LB) (1)**	**Cu% (1)**	**Fluence 1/4 T n/cm^2 (E >1.0 MeV) (6)**	**Corrected USE for L-T Direction**	**Decrease in Shelf Energy Tahle 2, Reg. Guide 1.99, Rev. 2 (4)**	**USE (4)**
Intermediate Shell	---	---	1.31×10^{19}	92(3)	19	75
Lower Shell	---	---	1.31×10^{19}	97(3)	19	79
Circ Weld	---	---	1.31×10^{19}	126	41	74
			32 and 34 EFPY			
Compouent	**Unirradiated NMWD Upper Shelf Energy (FT LB) (1)**	**Cu% (1)**	**Fluence 1/4 T n/cm^2 (E >1.0 MeV) (5)**	**Corrected USE for L-T Direction**	**Decrease in Shelf Energy Table 2, Reg. Guide 1.99, Rev. 2 (4)**	**USE (4)**
Intermediate Shell	141.5	0.06	2.17×10^{19}	92(3)	20	74
Lower	149	0.06	2.17×10^{19}	97(3)	20	78
Circ Weld	126	0.283	2.17×10^{19}	126	42	73

(1) Data taken from WCAP-12020.
(2) Fluence 1/4T = $fluence_{surface}\ e^{-.24x}$. x = .25 Thickness. $Fluence_{surface}$ and thickness taken from WCAP-13229.
(3) Correction to USE for Charpy specimens oriented in longitudinal (L-T) direction by reducing by a factor of .65 per NRC Std. Review Plan, Section 5.3.2.
(4) EOL use determined from Table 2, Reg. Guide 1.99, Rev. 2.
(5) Fluence taken from WCAP-13257.
(6) Fluence taken from WCAP-12333.
(7) USE values for 34 EFPY are essentially equal to values presented herein for 32 EFPY.

REFERENCES:

(a) U.S. Atomic Energy Commission Safety Evaluation of the Kewaunee Nuclear Power Plant Issued July 24, 1972 and Supplemented December 18, 1972 and May 10, 1973.
(b) Letter from E. R. Mathews (WPSC) to Document Management Branch (US NRC) dated August 18, 1981
(c) Letter from C. R. Steinhardt (WPSC) to Document Control Desk (US NRC) dated February 17, 1989
(d) Letter from M. J. Davis (US NRC) to K. H. Evers (WPSC) dated January 25, 1990
(e) Letter from J. G. Giitter (US NRC) to C. R. Steinhardt (WPSC) dated May 20, 1990
(f) Letter from D. L. Wigginton (US NRC) to D. C. Hintz (WPSC) dated July 21, 1987
(g) Letter from M. B. Fairtile (US NRC) to D. C. Hintz (WPSC) dated December 18, 1986
(h) Letter from R. B. A. Licciardo (US NRC) to E. R. Mathews (WPSC) dated April 21, 1982
(i) Letter from A. Schwencer (US NRC) to E. W. James (WPSC) dated December 14, 1977
(j) Letter from A. T. Gody (US NRC) to K. H. Evers (WPSC) dated November 21, 1989
(k) Letter from M. J. Davis (US NRC) to K. H. Evers (WPSC) dated January 25, 1990
(l) Letter from C. R. Steinhardt (WPSC) to Document Control Desk (US NRC) dated May 27, 1992

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