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RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Verifies that info util provided for facility in response to GL 92-01,Rev 1 accurately entered into summary rept attached to NRC 940509 ltr.Requests delay until 940620 for NRC review & approval of equivalent margins analyses performed.

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CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
PO Box 790
Hartsville SC 29551

Robinson File No.: 13510I
Serial: RNP/94-1194

JUN 13 1994

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
RESPONSE TO GENERIC LETTER 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," FOR THE H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

Gentlemen:

The purpose of this letter is to verify that the information we provided for our facility was accurately entered into the summary report attached to your letter dated May 9, 1994. In addition, we are requesting a delay until June 20, 1994, for NRC review and approval of the equivalent margins analyses performed for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, beltline materials to allow for additional management review.

Carolina Power & Light (CP&L) has reviewed the information within the NRC data base tables based on current end of license (EOL) fluence projections. As additional surveillance capsules are withdrawn and/or neutron fluence values are updated (based on projected EOL effective full power years (EFPYs) of operation, etc.), the predicted EOL material properties are also expected to change. These changes will be updated as part of the report submittal required by 10 CFR 50, Appendix H.

CP&L comments are enclosed and corrections to the Pressurized Thermal Shock and Upper Shelf Energy tables contained in Enclosures 1 and 2 of the subject NRC letter, dated May 9, 1994, are also marked-up and attached to the enclosures for this response.

Questions regarding this matter may be referred to Mr. K. R. Jury at (803) 383-1363.

Very truly yours,

R. M. Krich
Manager - Regulatory Affairs

RS:sgk
Enclosures
c: Mr. S. D. Ebneter, Regional Administrator, USNRC, Region II
Ms. B. L. Mozafari, USNRC Project Manager, HBRSEP
Mr. W. T. Orders, USNRC Senior Resident Inspector, HBRSEP

A028

Affidavit

C. S. Hinnant, having been first duly sworn, did depose and say that the information contained in letter RNP/94-1194 is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

CSHinnant
C. S. Hinnant

Deborah W. Martin
Notary (Seal)

My commission expires: 6/23/98

ANALYSIS OF NRC ENCLOSURE 1, "SUMMARY FILE FOR PRESSURIZED THERMAL SHOCK"

(i) "ID Neut. Fluence at EOL/EFPY" should be updated, based on latest available information, as noted in the attached marked-up NRC Enclosure 1. These EOL fluence values are consistent with those provided to the NRC in the Pressurized Thermal Shock (PTS) tables included in Enclosure 6 of CP&L's letter to the NRC dated September 15, 1993[1].

The attached marked-up NRC Enclosure 1 only reports one fluence value for the three axial welds in each shell course. In actuality, each of the three axial welds in each shell course have separate fluence values based on their azimuthal position within the vessel (See Enclosure 6 of CP&L's letter to the NRC dated September 15, 1993[1]). The axial weld fluence values in the attached mark-up of NRC Enclosure 1 represents the maximum fluence value for the three axial welds within each shell course.

(ii) The chemistry factors for Plates W10201-5 and W10201-6 are revised based on Table 2 of 10 CFR 50.61. These chemistry factor numbers are consistent with those provided to the NRC in the PTS tables included in Enclosure 6 of CP&L's letter to the NRC dated September 15, 1993[1].

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[1] "Request for License Amendment - Pressure-Temperature Curves", C. R. Dietz (CP&L) to USNRC, dated September 15, 1993

ANALYSIS OF NRC ENCLOSURE 1, "SUMMARY FILE FOR PRESSURIZED THERMAL SHOCK"

(iii) The initial RT_{NDT} for the "Lower Circ. Weld 11-273" is changed to -80°F based on "sister plant" information as follows.

The HBRSEP lower circumferential weld (Weld 11-273) and the Millstone 1 surveillance weld were each made from the same weld material RACO 3 + Ni 200 (heat # 34B009) and flux (Linde 1092). General Electric (GE) reported tests of the Millstone 1 surveillance weld in Report NEDC-30299[2] in support of Electric Power Research Institute (EPRI) Project Number RP2180-06. Based on this testing, drop weight tests and a full Charpy curve were developed for the Millstone 1 surveillance weld (Figures 9-9 and 9-10 of NEDC-30299). Table 10-1 of NEDC-30299 reported an initial RT_{NDT} of the Millstone 1 surveillance weld to be -80°F. This report did not provide the heat number for the Millstone 1 surveillance weld. However, in December 1984, the Millstone 1 surveillance weld heat number was mistakenly identified in GE Report NEDC-30833[3] as representative of a Millstone 1 reactor vessel longitudinal/axial weld with heat #W5214.

In 1986, while researching Combustion Engineering (CE) fabrication records for HBRSEP reactor vessel data, CP&L identified that the Millstone 1 surveillance weld was actually fabricated of heat #34B009 material. The CE weld inspection report, validated by the CE Quality Assurance organization, identifies the Millstone 1 surveillance weld heat number as heat #34B009. After CP&L notified Northeast Utilities, GE, and EPRI of the discovery; GE issued an "ERRATA and ADDENDUM" to NEDC-30833 in June 1986, correcting the Millstone 1 surveillance weld heat number to heat #34B009, as representative of a Millstone 1 reactor vessel girth weld. Based on CP&L's investigation, the following information is known relative to the similarities between the HBRSEP weld and the Millstone surveillance weld.

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[2] "Fracture Toughness of Reactor Pressure Vessel Steel Welds", General Electric report NEDC-30299, M. T. Wang, dated October 1983

[3] "Millstone Nuclear Power Station, Unit 1, Reactor Pressure Vessel Surveillance Materials Testing and Fracture Toughness Analysis", General Electric report NEDC-30833, T. A. Caine, dated December 1984

ANALYSIS OF NRC ENCLOSURE 1, "SUMMARY FILE FOR PRESSURIZED THERMAL SHOCK"

WELD	MILLSTONE 1 SURVEILLANCE	HBR2 WELD 11-273
VENDOR	CE	CE
FABRICATION	SUBMERGED ARC, TANDEM ELECTRODE	SUBMERGED ARC, TANDEM ELECTRODE
FABRICATION TIME FRAME	APRIL 1967	JULY 1967
MATERIAL SPECIFICATION & HEAT	RACO 3 + Ni 200, LINDE 1092 HEAT # 34B009	RACO 3 + Ni 200, LINDE 1092 HEAT # 34B009

The most recent Millstone 1 surveillance report, GE report number GE-NE-523-165-1292[4], correctly identifies the Millstone surveillance weld as heat #34B009, but reports initial RT_{NDT} of the vessel girth weld as -50°F. A discussion with representatives of both GE and Northeast Utilities suggests that the -50°F was likely based on Charpy test results conducted at +10°F and subtracting 60°F for establishing the -50°F as the RT_{NDT}. This choice by Northeast Utilities in no way challenges the validity of the EPRI-GE program reported in NEDC-30299.

The application of -80°F as initial RT_{NDT} for the HBRSEP Lower Circumferential Weld 11-273 was previously reported to the NRC on December 22, 1988[5].

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[4] "Millstone 1 Nuclear Station Vessel Surveillance Materials Testing Results and Fracture Toughness Analysis", GE report GE-NE-523-165-1292, Revision 1, dated February 1993

[5] "Response to Generic Letter 88-11", R. B. Richey (CP&L) to USNRC, dated December 22, 1988

ANALYSIS OF NRC ENCLOSURE 1, "SUMMARY FILE FOR PRESSURIZED THERMAL SHOCK"

(iv) The %Cu and %Ni values for Upper Circ. Weld 10-273 is being changed to match those surveillance weld measured values reported to NRC in WCAP-10304[6] (March 1983) and CP&L's previously referenced NRC submittal of September 15, 1993. Based on these previous submittals, the chemistry for Weld 10-273 has been established as follows:

%Cu ⇒ 0.34
%Ni ⇒ 0.66
Chemistry Factor (CF) per Table 1 of 10 CFR 50.61 ⇒ 217.7

As noted above, the CF for this weld has been established per Table 1 of 10 CFR 50.61 to be 217.7. Since CP&L has credible surveillance data for this weld, a calculated CF of 216.4 had been reported to the NRC in the previously referenced submittal of December 22, 1988, based on Regulatory Guide 1.99, Revision 1. Since Section (3) of 10 CFR 50.61 only indicates that surveillance data shall be integrated if the RT_{PTS} value changes significantly, the more conservative value obtained from Table 1 is reported in the attached marked-up NRC Enclosure 1. The CF values are very similar, and using the less conservative calculated value of 216.4 does not significantly affect RT_{PTS}, as noted below:

Adjusted EOL RT_{PTS}, based on CF of 217.7 ⇒ 263.78°F.
Adjusted EOL RT_{PTS}, based on CF of 216.4 ⇒ 262.26°F.

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[6] "Analysis Of Capsule T From The H. B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program", Westinghouse report WCAP-10304, dated March 1983

ANALYSIS OF NRC ENCLOSURE 2, "SUMMARY FILE FOR UPPER SHELF ENERGY"

(i) The "1/4T Neutron Fluence at EOL/EFPY" values for each of the plates and axial welds have been changed in the marked-up NRC Enclosure 2 based on the updated "ID Neutron Fluence at EOL/EFPY" values provided in the marked-up NRC Enclosure 1.

(ii) The "Unirrad. USE" values are updated, as appropriate, to match those reported in Table 2-1 of Westinghouse Owners' Group report WCAP-13587 Revision 1 (see above "CP&L RESPONSE" regarding applicability of WCAP-13587 Revision 1 to HBRSEP).

(iii) The "1/4T USE at EOL/EFPY" for the reactor vessel materials have been verified and, as appropriate, changed based on the updated values for "Unirrad. USE", "1/4T Neutron Fluence at EOL/EFPY", and surveillance data, as applicable.

The shell plate USE values reported for 1/4T in the marked-up NRC Enclosure 2 are consistent with those reported in Table 2-1 of WCAP-13587 Revision 1, with the exception of Shell Plate W9807-3. Using the updated 1/4T fluence value for Shell Plate W9807-3, an EOL/EFPY USE of 61 ft-lbs (vs. the 62 ft-lbs reported in WCAP-13587 Revision 1) is established. Since this plate is not limiting, and the value of 61 ft-lbs is well above the USE screening criterion of 50 ft-lbs, this change has no safety significance.

(iv) The "Method of Determin. Unirrad. USE" for Upper Circ. Weld 10-273 is changed to "Surveillance Weld" since the HBRSEP surveillance weld has been firmly established as representative of Weld 10-273. The unirradiated USE of 112 ft-lbs for this weld is based on Charpy test data reported in Table 5-7 of the previously referenced (above) Westinghouse report WCAP-10304 (i.e., Capsule T surveillance report dated March 1983).

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL/EFPY	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Robinson 2 EOL: 7/31/2010	Upper Shell W10201-1	A6623-1	~~1.7E19~~ 1.8E19	69°F	Plant Specific	62.9	Table	0.13	0.11
	Upper Shell W10201-2	A6520-1	~~1.7E19~~ 1.8E19	30°F	Plant Specific	84.75	Table	0.15	0.25
	Upper Shell W10201-3	B1255-1	~~1.7E19~~ 1.8E19	36°F	Plant Specific	51.8	Table	0.11	0.08
	Int. Shell W10201-4	A6604-1	~~4.7E19~~ 4.8E19	20°F	Plant Specific	57.1	Table	0.12	0.09
	Int. Shell W10201-5	B1256-1	~~4.7E19~~ 4.8E19	20°F	Plant Specific	~~43.79~~ 51.2	~~Calculated~~ Table	0.10	0.12
	Int. Shell W10201-6	B1250-1	~~4.7E19~~ 4.8E19	45°F	Plant Specific	~~47.49~~ 44.2	~~Calculated~~ Table	0.09	0.09
	Lower Shell W9807-3	B0650-1	~~1.8E19~~ 2.0E19	50°F	Plant Specific	58	Table	0.12	0.10
	Lower Shell W9807-5	A5891-1	~~1.8E19~~ 2.0E19	33°F	Plant Specific	70.5	Table	0.15	0.10
	Lower Shell W9807-9	P1444-1	~~1.8E19~~ 2.0E19	9°F	Plant Specific	70.5	Table	0.14	0.15
	Upper Shell Axial Welds 1-273ABC	86054B	~~1.8E19~~ 1.3E19	-56°F	Generic	100.75	Table	0.22	0.05
	Int. Shell Axial Welds 2-273ABC	86054B	~~4.7E19~~ 3.9E19	-56°F	Generic	100.75	Table	0.22	0.05
	Lower Shell Axial Welds 3-273ABC	86054B	2.0E19	-56°F	Generic	100.75	Table	0.22	0.05
	Upper Circ. Weld 10-273	W5214	1.8E19	-56°F	Generic	~~213.08~~ 217.7	~~Calculated~~ Table	~~0.2~~ 0.34	~~1.02~~ 0.66
	Lower Circ. Weld 11-273	34B009	2.0E19	~~-56°F~~ -80°F	~~Generic~~ Sister Plant	197.8	Table	0.17	0.92

REFERENCES FOR ROBINSON 2:

~~IRT_{ndt} data are from February 4, 1986, letter from S. R. Zimmerman (CP&L) to L. S. Rubinstein (USNRC), subject: Pressurized Thermal Shock, Correction to Response to Final Rule 10 CFR 50.61.~~

~~Fluence and chemistry data are from July 6, 1992, letter from R. B. Starkey (CP&L) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.~~

~~Chemical composition for welds fabricated using weld wire (heat no. W5214) is reported in a February 23, 1994 letter from D.W. Rogers (Consumer Power) to USNRC. Subject: Palisade Response to GL 92-01.~~

IRT_{ndt}, Fluence, and Chemistry Data are from September 15, 1993, letter from C.R. Dietz to USNRC, subject Pressure-Temperature Curves.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL/EFPY	1/4T Neutron Fluence at EOL/EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Robinson 2	Upper Shell W10201-1	A6623-1	A 302A	42 (EMA)	~~0.97E19~~ 1.03E19	~~52~~ 54	65%
EOL: 7/31/2010	Upper Shell W10201-2	A6520-1	A 302A	~~59~~ 61	~~0.97E19~~ 1.03E19	~~77~~ 80	65%
	Upper Shell W10201-3	B1255-1	A 302A	~~46~~ (EMA) 50	~~0.97E19~~ 1.03E19	~~57~~ 62	65%
	Int. Shell W10201-4	A6604-1	A 302A	46 (EMA)	~~2.69E19~~ 2.75E19	~~59~~ 62	65%
	Int. Shell W10201-5	B1256-1	A 302A	~~56~~ 60	~~2.69E19~~ 2.75E19	~~59~~ 64	65%
	Int. Shell W10201-6	B1250-1	A 302A	~~68~~ 69	~~2.69E19~~ 2.75E19	~~73~~ 74	65%
	Lower Shell W9807-3	B0650-1	A 302A	~~62~~ 61	~~1.03E19~~ 1.14E19	78	65%
	Lower Shell W9807-5	A5891-1	A 302A	~~53~~ 56	~~1.03E19~~ 1.14E19	~~70~~ 74	65%
	Lower Shell W9807-9	P1444-1	A 302A	59	~~1.03E19~~ 1.14E19	~~76~~ 77	65%
	Upper Shell Axial Welds 1-273ABC	86054B RACO3	Arcos B-5, SAW	~~67~~ 69	~~0.97E19~~ 0.74E19	105	Sister Plant
	Int. Shell Axial Welds 2-273ABC	86054B RACO3	Arcos B-5, SAW	~~57~~ 59	~~2.69E19~~ 2.23E19	105	Sister Plant
	Lower Shell Axial Welds 3-273ABC	86054B RACO3	Arcos B-5, SAW	~~65~~ 66	~~1.03E19~~ 1.14E19	105	Sister Plant
	Upper Circ. Weld 10-273	W5214	Linde 1092, SAW	65	1.03E19	112	~~Sister Plant~~ Surveillance Weld
	Lower Circ. Weld 11-273	348009 RACO3+ N:200	Linde 1092, SAW	72	1.14E19	106	Sister Plant

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL/EFPY	1/4T Neutron Fluence at EOL/EFPY	Unirrad. USE	Method of Determin. Unirrad. USE

REFERENCES FOR ROBINSON 2:

~~Fluence, chemical composition, and UUSE data are from July 6, 1992, letter from R. B. Starkey (CP&L) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity~~

~~Applicability of the Equivalent Margins Analysis (EMA) has been addressed in the letters of November 29 and December 21, 1993 from CP&L to USNRC. In accordance with Appendix G, 10 CFR 50, the licensee must request NRC review of this analysis.~~

Fluence data are estimated using Reg. Guide 1.99, Revision 2 from September 15, 1993, Letter from C. R. Dietz to USNRC, subject Pressure - Temperature Curves.

Unirradiated USE data for plate materials from WCAP 13587, Revision 1.